NUREG-0800

U.S. NUCLEAR REGULATORY COMMISSION

# STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

## 5.3.1 REACTOR VESSEL MATERIALS

REVIEW RESPONSIBILITIES

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

Secondary - Organization responsible for review of welding issues

## I. AREAS OF REVIEW

The following areas relating to reactor vessel materials are reviewed:

1. Material Specifications

   The material specifications used for the reactor vessel and applicable attachments and appurtenances, such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings, are reviewed and their adequacy for use in the construction of such components is assessed on the basis of the mechanical and physical properties of the materials, the effects of irradiation on these materials, their corrosion resistance, and their fabric ability. Similarly, the specifications for austenitic steel and nonferrous metals specified for the above applications are reviewed with respect to mechanical properties, stress-corrosion resistance, and fabric ability.

2. Special Processes Used for Manufacture and Fabrication of Components

Information submitted by the applicant for any special process used in the manufacture of the product forms supplied and for their fabrication into the reactor vessel or any of its appurtenances is reviewed, and the capability of these processes to provide components with suitable mechanical and physical properties is assessed. The effects of such special processes on the stress-corrosion characteristics of the material, and any aspect of the process which could cause special requirements for nondestructive examinations, are reviewed.

3. Special Methods for Nondestructive Examination

Nondestructive examination methods differing from those described in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference (Ref.) 24) (hereafter "the Code"), Section III, are reviewed. Attention is directed towards calibration methods, instrumentation, methods of application, sensitivity, reliability, and standards used.

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

Information on special controls and special processes for welding ferritic steels and austenitic stainless steels is reviewed, and their adequacy is assessed. The extent to which the controls and processes deviate from Code rules is reviewed. Information on welding of safe-ends during the fabrication of dissimilar metal joints is given particular attention and details of the methods, processes, and materials used are reviewed. Controls for abrasive work (e.g., grinding) on austenitic stainless steel surfaces are also reviewed with respect to the potential for material contamination and excessive surface cold-working.

5. Fracture Toughness

Fracture toughness of the ferritic materials used for reactor vessels and appurtenances thereto is reviewed to ensure that such components will behave in a non-brittle manner and that the probability of rapidly propagating fracture will be minimized under operating, maintenance, and testing conditions and during anticipated operational occurrences. The review includes the descriptions of the fracture toughness tests performed on all ferritic materials used for the reactor vessel and appurtenances thereto, and includes Charpy V-notch impact test specimens, drop-weight test specimens, and any other test specimens included by the applicant.

The test procedures specified by the applicant are reviewed and their adequacy is confirmed.

The composition of ferritic materials employed for the reactor vessel is reviewed and the amount of residual elements such as copper and phosphorus is checked. The results of impact tests performed on base material, weld metal, and heat-affected zones are reviewed, and the scope of the testing is checked, particularly in the area of the reactor vessel belt-line region, where radiation effects on the material are most significant.

Fracture toughness of the materials employed is characterized by its reference temperature, $RT_{NDT}$. This temperature is the higher value of the nil-ductility temperature (NDT) from the drop-weight test, or the temperature that is 33EC (60EF) below the temperature at which Charpy V-notch impact test data meet a specified toughness level. The information submitted is checked to ensure that the $RT_{NDT}$ of the materials is included with the data and test results for impact testing.

6. Material Surveillance

Reactor vessel material surveillance must be performed to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel belt-line region of water-cooled power reactors resulting from exposure to neutron irradiation and the thermal environment. Under the surveillance programs, fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

For a combined license (COL) application, the review includes the reactor vessel material surveillance program description and the implementation milestones.

7. Reactor Vessel Fasteners

The materials for the stud bolts, washers, and nuts, or other fasteners used to hold the reactor vessel head, are reviewed to determine their adequacy. Mechanical properties, including fracture toughness, are checked to ensure that all requirements are met. Lubricants or surface treatments used are reviewed to ensure that the studs will be resistant to stress-corrosion cracking under the environmental conditions during service and shutdowns. The adequacy of the destructive testing procedures used to ensure initial integrity is reviewed, along with the applicable acceptance criteria.

8. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)

For design certification and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this standard review plan (SRP) section is reviewed in accordance with SRP Section 14.3. It is recognized 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.

Review Interfaces:

The listed 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.

2. The review of the compatibility of the thermal insulation with the austenitic stainless steel of the reactor vessel appurtenances and the acceptability of any nonmetallic thermal insulation that is employed are performed under SRP Section 5.2.3, “Reactor Coolant Pressure Boundary Materials.”

3. The review of the reactor vessel fracture toughness with regard to pressure-temperature limits, including protection from pressurized thermal shock events in accordance with 10 CFR 50.61 is performed under SRP Section 5.3.2, “Pressure-Temperature Limits.”

4. The review of the reactor vessel wall neutron fluence is performed under SRP Section 4.3.

5. The review of the over pressure protection system is performed under SRP Section 5.2.2.

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

7. Operational programs and their implementation milestones are summarized in SRP Section 13.4.

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

## II. ACCEPTANCE CRITERIA

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

A. General Design Criteria (GDC) 1 and 30 found in Appendix A to Part 50, as they relate to quality standards for design, fabrication, erection, and testing of structures, systems and components;

B. GDC 4, as it relates to the compatibility of components with environmental conditions;

C. GDC 14, as it relates to prevention of rapidly propagating fractures of the reactor coolant pressure boundary (RCPB);

D. GDC 31, as it relates to material fracture toughness;

E. GDC 32, as it relates to the requirements for a materials surveillance program;

F. 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness;

G. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G and Appendix H;

H. 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control;

I. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness,

J. 10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness,

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

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

Specific criteria necessary to meet[1] the relevant requirements of the Commission's regulations identified above are as follows for each review described in Subsection I of this SRP section:

1. Materials

   The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for materials, as detailed below:

   a. Acceptable materials for the reactor vessel and its appurtenances and attachments are those identified in the Code, Section III, Appendix I. The materials must also meet the requirements of 10 CFR Part 50, Appendix G.

   b. The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of Code Section III, Appendix IV-1000 and 10 CFR Part 50, Appendix G. These data must include information on mechanical properties, weldability, and physical changes of the material.

[1]Note: 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 an acceptable method of complying with the NRC regulations.

2. Special Processes Used for Manufacture and Fabrication of Components

The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for fabrication of components. The reactor vessel and its appurtenances are fabricated and installed in accordance with Code Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with Code Section III, Article NCA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.

3. Special Methods for Nondestructive Examination

The requirements of GDC 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the ASME Code, Section III, for fabrication nondestructive testing. The acceptance criteria for examination of the reactor vessel and its appurtenances by nondestructive examination are those specified in Code Section III, NB-5000.

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

The acceptance criteria for special controls and processes in welding austenitic or ferritic steel components are based upon the following regulatory guides, ASME Code provisions, and other regulatory documents necessary to satisfy the relevant requirements of GDC 1, 4, 14, and 30; Appendix B; and 10 CFR 50.55a.

a. Only those welding processes capable of producing welds in accordance with the welding procedure qualification requirements of Code Sections III and IX may be used. Any process used shall be such that the records required by NB-4300 of Section III can be made, with the exception of stud welding, which is acceptable only for minor nonpressure attachments.

b. ASME Code Sections III and IX criteria for welding ferritic steel are supplemented by the regulatory positions in Regulatory Guides (RGs) 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” and 1.34, “Control of Electroslag Weld Properties.”

c. The regulatory positions of RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” provide the acceptance criteria to avoid underclad cracking of stainless steel clad ferritic components.

d. ASME Code Sections III and IX criteria for welding austenitic stainless steels are supplemented by the regulatory positions in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and RG 1.34. For the BWR austenitic stainless steel reactor vessel attachments and appurtenances specified in Generic Letter (GL) 88-01 (Ref. 23), the weld metal ferrite content should be controlled as described in the positions of Attachment A to GL 88-01 or the recommendations of NUREG-0313, Revision 2 (Ref. 22).

e. The regulatory positions of RGs 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," provide the acceptance criteria to avoid sensitization and contamination of stainless steel.

RG 1.44 states that non-sensitization should be verified using ASTM A-262 (Ref. 25) Practices A or E, or another method that can be demonstrated to show nonsensitization of austenitic stainless steel. Alternative tests to those in ASTM A-262 that have been previously accepted include ASTM A 708 (Ref. 26). For BWRs, the control of sensitized steel per RG 1.44 should be modified as necessary to conform with the positions in Attachment A to GL 88-01 or the recommendations of NUREG-0313.

The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in RG 1.37 position C.5 to prevent contamination which promotes stress corrosion cracking. Tools which contain materials that could contribute to intergranular or stress-corrosion cracking or which, because of previous usage, may have become contaminated with such materials, should not be used on austenitic stainless steel surfaces.

f. Additional controls, beyond those described above, are considered necessary to avoid intergranular stress corrosion cracking (IGSCC) in and near welds in BWR austenitic stainless steel reactor vessel attachments and appurtenances. The additional controls are described in Attachment A to GL 88-01 and in NUREG-0313, Revision 2. These controls include material and weldment specifications for IGSCC resistant materials, processing techniques, categorization of the IGSCC resistance of installations based upon material properties, treatment history, and post-weld treatments. The technical bases for these controls are described in NUREG-0313, Revision 2.

The referenced regulatory guides are described in detail in the acceptance criteria of SRP Section 5.2.3.

5. <u>Fracture Toughness</u>

The acceptance criteria for this area of review are the requirements of Appendix G of 10 CFR Part 50. These criteria satisfy the requirements of GDC 31 and 10 CFR 50.60 regarding materials testing and acceptance standards for fracture toughness.

Appendix G requires that the reactor vessel and appurtenances thereto which are made of ferritic materials shall meet the following minimum requirements for fracture toughness during system hydrostatic tests, conditions of normal operation, and anticipated operational occurrences:

a. The ferritic materials shall be tested in accordance with the ASME Code paragraph NB-2300 including:

(1) $T_{NDT}$ shall be determined for each material by means of a drop weight test.

(2) The materials shall meet the acceptance standards of paragraph NB-2330 of the Code, which states that at a temperature not greater than ($T_{NDT}$ + 33EC)[($T_{NDT}$ + 60EF)] each Charpy $C_v$ specimen tested shall exhibit at least 0.89 mm (35 mils) lateral expansion and not less than 68 J (50 ft-lbs) of absorbed energy. When these requirements are met, $T_{NDT}$ is defined as the reference temperature, $RT_{NDT}$.

(3) In the event that the above requirements are not met, additional $C_v$ notch impact tests are performed (in groups of three specimens) to determine the temperature $T_{cv}$ at which they are met. In this case the reference temperature $RT_{NDT}$ = $T_{cv}$ - 33EC ($RT_{NDT}$ = $T_{cv}$- 60EF). Thus the reference temperature $RT_{NDT}$ is the higher of $T_{NDT}$ and ($T_{cv}$ - 33EC) [($T_{cv}$ - 60EF)]

(4) When a $C_v$ impact test has not been performed at ($T_{NDT}$ + 33EC) [($T_{NDT}$ + 60EF)], or when the $C_v$ impact test at ($T_{NDT}$ + 33EC) [($T_{NDT}$ + 60EF)] does not exhibit a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion, a temperature representing a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion may be obtained from a full $C_v$ impact curve developed from the minimum data points of all the $C_v$ impact tests performed.

b. In addition to the above criteria, the requirements of paragraphs IV.A.1, IV.A.2, and IV.B of Appendix G of 10 CFR Part 50 and 10 CFR 50.61(b)(2) (for PWRs) shall be met.

(1) SRP Section 5.3.2 discusses the requirements of paragraphs IV.A.2 and of Appendix G in detail.

(2) The acceptance criteria discussed in paragraph IV.A.1 of Appendix G states that reactor vessel belt-line materials shall have a minimum upper shelf energy of 102 J (75 ft-lbs) as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with paragraph NB-2331(a) of the Code, Section III. Reactor vessel belt-line materials must also maintain an upper shelf energy no less than 68 J (50 ft-lb) throughout the life of the vessel. These two requirements do not apply, however, if it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of upper shelf fracture energy are adequate.

c. The neutron radiation embrittlement effects on reactor vessel materials shall be determined in accordance with 10 CFR Part 50, Appendix G, Section III, and RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

6. Material Surveillance

The material surveillance acceptance criteria are the requirements of Section III of Appendix H of 10 CFR Part 50. Complying with the acceptance criteria satisfies the requirements of GDC 32 regarding an appropriate material surveillance program for the reactor vessel.

Section III of Appendix H requirements are:

a. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence (E > 1 MeV) at the end of the design life of the vessel will not exceed $10^{17}$ n/cm$^2$.

b. Reactor vessels constructed of ferritic materials which do not meet the conditions in paragraph a. shall have their belt-line regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) standard ASTM E-185 (Ref. 27), except as modified by Appendix H to 10 CFR Part 50.

c. The surveillance program shall meet the following requirements:

(1) Surveillance specimens shall be taken from locations alongside the fracture toughness test specimens required by Section III of Appendix G of 10 CFR Part 50. The specimen types shall comply with the requirements of Section III.B of Appendix H, except that drop-weight specimens are not required.

(2) Surveillance capsules containing the surveillance specimens shall be located near the inside vessel wall in the belt-line region, so that the neutron flux received by the specimens approximates that received by the vessel inner surface, and the thermal environment is as close as practical to that of the vessel inner surface. If the capsule holders are attached to the vessel wall or cladding, inspection shall be done according to the requirements for permanent structural attachments as given in ASME Code Sections III and XI. The design and location of the capsules shall permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules specified in paragraph III.B.1 of Appendix H.

(3) The required number of capsules, which will vary from three to five depending upon the adjusted reference temperature at the end of the service lifetime of the reactor vessel, and their withdrawal schedules, shall be in accordance with the requirements of paragraph III.B.2 of Appendix H.

(4) For multiple reactors located at a single site, an integrated surveillance program may be authorized by the Commission on an individual case basis in accordance with the requirements of paragraph III.C of Appendix H.

The material surveillance program criteria of ASTM E-185 cited in 10 CFR Part 50, Appendix H, is predicated on an assumed 40-year reactor vessel design life. For those applicants proposing a facility with greater than a 40-year design life, the criteria of ASTM E-185 must be supplemented to provide for monitoring of the reactor vessel materials for the entire reactor vessel design life.

For a COL application, the program implementation milestones are:

(1) When the surveillance capsules will have been fabricated and the capsule holders are installed in the vessel.

(2) The program is required to be implemented (e.g., surveillance capsules installed) prior to plant startup.

7. Reactor Vessel Fasteners

The acceptance criteria for the reactor vessel bolting material are given by paragraph IV.A of Appendix G to 10 CFR Part 50 and by the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." These acceptance criteria satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a, and meet the requirements of GDC 31 regarding prevention of fracture of the RCPB.

Regulatory Positions C.1 and C.2 of RG 1.65 recommend the following:

a. Materials for reactor vessel studs (and other fasteners) that are considered suitable are SA-540 Grades B-23 and B-24, SA-193 Grade B-7, SA-194 Grade 7, and SA-320 Grade L-43, as presented in Section II of the ASME Code.

b. The fastener material should not have an ultimate tensile strength over 1170 MPa (170 ksi), and the fracture toughness tests and acceptance levels of NB-2333 of Section III of the Code must be met as required by paragraph IV.A of Appendix G to 10 CFR Part 50.

c. Surface treatments, plating, or thread lubricants used should be shown to be compatible with the materials, and stable at operating temperatures.

d. Nondestructive examination should be performed according to Section III of the Code, subsubarticle NB-2580 including additional recommendations given in Regulatory Position C.2 of RG 1.65.

Technical Rationale:

The technical rationale for application of the above acceptance criteria to reactor vessel materials is discussed in the following paragraphs.

1. General Design Criteria (GDC) 1 and 30 and 10 CFR 50.55a, establish quality assurance requirements for the design, fabrication, erection, and testing of structures, systems and components (SSCs) important to safety. GDC 1 and 10 CFR 50.55a establish that the quality assurance standards to be applied to SSCs shall be commensurate with the importance of the safety functions to be performed and will be established and implemented through the development of a quality assurance program. 10 CFR 50.55a also incorporates by reference applicable editions and addenda of the ASME Boiler and Pressure Vessel Code. GDC 30 establishes that RCPB components shall meet the highest quality standards practical. The safety functions of the reactor vessel are to provide 1) a support structure for the internal reactor components, 2) reactor coolant confinement as part of the reactor coolant flow path, and 3) a containment barrier to the release of fission products as part of the RCPB. RGs 1.31, 1.34, 1.43, 1.44, 1.50, and 1.65 provide regulatory positions applicable to compliance with GDC 1 and 30. Compliance with GDC 1 and 30 and 10 CFR 50.55a provides assurance that the reactor vessel will be designed, fabricated, erected, and tested to established and proven standards thereby reducing the likelihood of reactor vessel failure.

2. GDC 4 establishes that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The reactor vessel provides support for internal reactor components, a fission product barrier, and confinement of the reactor coolant. Application of GDC 4 to the reactor vessel materials provides assurance that degradation and/or failure of the reactor vessel resulting from environmental conditions that could cause substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling are not likely to occur.

3. GDC 14 requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The reactor vessel is an integral part of the RCPB. RG 1.31 provides regulatory positions regarding the control of ferrite content in stainless steel welds that are relevant to compliance with GDC 14. Application of GDC 14 and RG 1.31 to the reactor vessel materials assures that they are selected, fabricated, installed, and tested to provide a low probability of significant degradation or gross failure of the reactor vessel that could cause substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling.

4. GDC 31 and 10 CFR 50.55a establish fracture toughness requirements and the applicable ASME standards respectively. GDC 31 establishes that the RCPB be designed with sufficient margin to assure that when stressed under operating,

maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. 10 CFR 50.55a incorporates the applicable editions and addenda of the ASME Boiler and Pressure Vessel Code that are relevant to the fracture toughness requirements of GDC 31 and 10 CFR Part 50, Appendix G. The design is required to reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) the size of flaws. The reactor vessel is an integral part of the RCPB and is fabricated of thick section materials subjected to stresses including those from full reactor coolant pressure and thermal gradients. Application of GDC 31 to the reactor vessel materials assures that they are selected to provide sufficient design margin to account for uncertainties associated with flaws and the effects of service and operating conditions, and thereby to provide a minimum probability of material degradation leading to rapid failure of the vessel and loss of reactor coolant.

5. GDC 32 requires that RCPB components shall be designed to allow periodic inspection and testing to assess their structural and leak-tight integrity, and a material surveillance program for the reactor pressure vessel. The reactor vessel material surveillance program monitors the reactor vessel belt-line materials for changes in fracture toughness resulting from exposure to neutron irradiation and the thermal environment. The specific material surveillance program requirements are established in 10 CFR Part 50, Appendix H and the data is utilized to determine compliance of the irradiated material with the fracture toughness requirements and criteria of 10 CFR Part 50, Appendix G. Compliance with GDC 32 provides assurance that degradation potentially affecting RCPB integrity is detected prior to fracture. Further, a materials surveillance program assures that the reactor vessel materials maintain sufficient toughness, thereby reducing the probability of reactor vessel failures.

6. 10 CFR 50.60 establishes that all light-water nuclear power reactors must meet the fracture toughness and material surveillance requirements set forth in 10 CFR Part 50, Appendix G and Appendix H. Compliance with the requirements of this rule and the associated appendices provide assurance regarding the structural integrity of the RCPB and specifically the reactor vessel. The rationale for compliance with this rule is discussed in Technical Rationale items 3, 4, 8 and 9 of this subsection.

7. 10 CFR Part 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.37 provides regulatory positions relevant to compliance with Appendix B. Application of cleaning requirements to the reactor vessel materials provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking. This reduces the likelihood that degradation and/or failure of the reactor vessel could cause substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling.

8. 10 CFR Part 50, Appendix G, establishes requirements for the fracture toughness of pressure-retaining components of the RCPB made of ferritic materials. The reactor vessel is an integral part of the RCPB. Application of these requirements to the RCPB materials provides a method of satisfying the requirements of GDC 14 and 31 related to fracture prevention. The rationale for these requirements is as discussed in Items 3 and 4 above.

9. 10 CFR Part 50, Appendix H, establishes the reactor vessel material surveillance program requirements. The surveillance program monitors the changes in fracture toughness properties of ferritic materials in the reactor vessel belt-line, resulting from exposure to neutron irradiation and the thermal environment. Data from the surveillance program is utilized in complying with 10 CFR Part 50, Appendix G requirements for establishing pressure-temperature limits and corrective actions (such as vessel annealing) if fracture toughness criteria can not be met. The structural integrity of the reactor vessel material is essential in assuring support of internal reactor components, confinement of reactor coolant, and a barrier to the release of fission products. Compliance with 10 CFR Part 50, Appendix H, provides assurance that changes to the reactor vessel materials resulting from the operational environment will be monitored, and that appropriate actions will be taken if significant changes occur in the material fracture toughness that may affect the integrity of the reactor vessel, and thus its ability to accomplish the safety functions under all anticipated and postulated conditions.

10. In Staff Requirements - SECY-05-0197 - Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria, dated February 22, 2006, the Commission approved the use of a license condition for operational program implementation milestones that are fully described or referenced in the final safety analysis report.

## 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 Subsection II.

1. <u>Materials</u>

   The materials for the reactor vessel and its appurtenances and attachments are compared with the acceptable materials identified in the Code, Section III, Appendix I.

   Materials not listed in the Code are clearly identified. A study of the suitability of the material and comparisons with precedents set in earlier cases enable the reviewer to determine the acceptability of the proposed exceptions. In those instances where the reviewer has taken exception to the use of a specific material the applicant is advised which material is not acceptable, and the reason for disapproval.

2. Special Processes Used for Manufacture and Fabrication of Components

Information on special processes used for manufacture and fabrication of the reactor vessel and its appurtenances is reviewed to (1) identify each special process, (2) determine whether there are any Code restrictions on its use, (3) establish the adequacy of the process in providing components with suitable mechanical and physical properties, (4) establish the effects of such processes on the stress-corrosion characteristics of the material, and (5) identify whether special requirements for nondestructive examination are needed if the process is used.

Since there are no specific Code requirements on the use of special processes, the suitability of a process is assessed on the basis of service experience with similar parts fabricated by the process being reviewed.

3. Special Methods for Nondestructive Examination

Section V of the Code includes methods for performing nondestructive examinations to detect surface and internal discontinuities when these methods are referenced by Section III of the Code. They include the following methods: radiographic, magnetic particle, liquid penetrants, and ultrasonic. The methods as described are applicable to most geometric configurations and materials encountered in fabrication, and are applied for normal conditions. However, special configurations and materials may warrant modified methods and techniques. If such special procedures are developed, the reviewer must determine that they are equivalent or superior to the techniques described in Section V of the Code, and are capable of producing meaningful results under the special conditions.

Such special procedures may be modifications or combinations of methods described in Section V, or may be entirely different, but the reviewer verifies that they have been proven by demonstration to result in an examination capable of detecting discontinuities under the special conditions to the same extent that applicable normal techniques which are included in Section V would result in detection of discontinuities under normal conditions.

Such special procedures are submitted to the authorized inspector or inspecting agency for review and approval prior to use.

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steel

The controls on welding of ferritic steels and austenitic stainless steels discussed in SRP Section 5.2.3 are considered applicable to welding of the reactor vessel and its components. The reviewer verifies that any special welding control or special welding process is able to conform to the qualification requirements of the Code, Section IX, or that justification is made for this deviation.

The reviewer also reviews the controls (before, during, and after welding of austenitic stainless steel) to avoid contamination and sensitization that could increase the possibility of stress corrosion cracking in austenitic stainless steel. Additionally, controls to avoid underclad cracking during weld cladding of the reactor vessel are reviewed.

The abrasive work controls for austenitic stainless steel surfaces are reviewed and are verified as adequate to minimize the introduction of contaminants and surface cold-working which may promote stress corrosion cracking.

For BWRs, the reactor vessel attachments and appurtenances are reviewed for conformance with the staff positions of GL 88-01 or the recommendations of NUREG-0313, Revision 2 with regard to protection against IGSSC in or near weldment to the reactor vessel.

5. <u>Fracture Toughness</u>

The information submitted by the applicant relative to tests for fracture toughness is reviewed for conformance with the Code, Section III, paragraph NB-2300, and Appendix G of 10 CFR Part 50.

These tests include Charpy V-notch impact tests and drop-weight tests. A description of the tests is reviewed, and the location of the test specimens and their orientation are verified.

Information regarding calibration of instruments and equipment is reviewed for conformance to Code Section III, paragraph NB-2300.

In the event that none of the fracture toughness tests has been performed, the preliminary safety analysis report (PSAR) must contain a statement of the applicant's intention to perform this work in accordance with Code Section III, NB-2300 and Appendix G of 10 CFR Part 50.

The final safety analysis report (FSAR) is reviewed to ensure that all the impact tests shown in NB-2300 have been performed. The results of the tests shall be in accordance with the acceptance criteria shown in Subsection II.5 of this SRP section.

The reviewer evaluates the initial Charpy upper shelf energy for the reactor vessel materials in accordance with the acceptance criterion specified in 10 CFR Part 50, Appendix G, paragraph IV.A.1.a. Reactor vessel materials that do not meet the specified initial Charpy upper shelf energy acceptance criterion shall be evaluated in accordance with the provisions for additional analysis also specified paragraph in IV.A.1.a. In addition to the ASME Code, RG 1.161 (Ref. 22) provides an acceptable methodology for the performance of analyses intended to meet the provisions for the additional analysis in paragraph IV.A.1.a.

The reviewer also evaluates the end-of-license Charpy upper shelf energy for the reactor vessel materials in accordance with the acceptance criterion specified in 10 CFR Part 50, Appendix G, paragraph IV.A.1.a. Reactor vessel materials that do not meet the specified initial Charpy upper shelf energy acceptance criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In accordance with paragraph IV.A.1.c., this analysis must be submitted to the staff for review and approval at least three years prior to the date on which the predicted Charpy upper shelf energy will no longer satisfy the requirements of paragraph IV.A.1.a. In addition to the ASME Code, RG 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in paragraph IV.A.1.a.

For those plants that were designed and constructed prior to the effective date of Appendix G, 10 CFR Part 50, some of the fracture toughness requirements of Appendix G may not be explicitly met. The detailed procedure for conducting the review of such cases is found in SRP Section 5.3.2 and in MTEB Branch Technical Position 5-2, “Fracture Toughness Requirements.” Exemptions from the regulation can only be granted when the applicant has demonstrated equivalence to the required margin of safety.

6. <u>Material Surveillance</u>

The reviewer verifies that the information contained in the SAR and the Technical Specifications is complete enough to determine that the surveillance program will comply with Appendix H, 10 CFR Part 50. The following information must be provided as a minimum:

a. The reviewer verifies that the PSAR states the end-of-life fluence calculated for the vessel belt-line, the maximum predicted shift in reference transition temperature ($RT_{NDT}$), the number of capsules, and the number and types of specimens to be placed in the capsules, and that the program is in compliance with ASTM E-185 and Appendix H, 10 CFR Part 50. For plants with a proposed design life that exceeds 40 years, the reviewer verifies that the requirements of 10 CFR Part 50, Appendix H and ASTM E-185 have been supplemented as necessary to provide for surveillance of the reactor vessel materials over the entire design life of the facility.

b. For design certification applications, a combined license action item, and associated ITAAC (e.g., as to material samples), must be included to verify that the plant specific surveillance program is in accordance with the assumptions in the certified design material and the requirements of Appendix H of 10 CFR Part 50.

c. For review of a COL application, the staff reviews the description of the reactor material surveillance program and its implementation, and verifies the adequacy of implementation milestones. In addition, the staff identifies this program and its implementation milestones within the license condition on operational program implementation described in SRP Section 13.4.

d. The reviewer verifies that the FSAR provides the information listed above and, in addition, includes results of all fracture toughness tests, chemical analyses of all materials in the belt-line region, and provides the information needed by the reviewer to evaluate the adequacy of the program.

7. Reactor Vessel Fasteners

The reviewer verifies that the information in the SAR covers all requirements for reactor vessel studs and other fasteners, as described in the previous section. For FSARs, the results of tensile and fracture toughness tests performed on the fastener materials are checked to ensure that all requirements are met.

For reviews of 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 design certification applications, the reviewer should identify necessary combined license action items. Following 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. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a design certification, an early site permit (ESP), or other NRC-approved material.

## IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to conforms to the guidance of this SRP section and that this evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report.

For the reasons set forth in detail above:

1. The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Further, the applicant's special measures for control of residual elements in ferritic materials have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a. In addition, the use of austenitic stainless steel in BWR reactor vessel attachments and appurtenances conforms with the staff positions in GL 88-01 or the recommendations of NUREG-0313, Revision 2.

2. Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code have been submitted by the applicant. Since certification has been made by the applicant that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes used are considered acceptable. Compliance with these Code provisions meets the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a. In addition, the use of austenitic stainless steel in BWR reactor vessel attachments and appurtenances conforms with the staff positions in GL 88-01 or the recommendations of NUREG-0313, Revision 2.

3. Special methods used for nondestructive examination of the reactor vessel and its appurtenances have been identified and have been found equivalent or superior to the techniques described in Appendix X of Code Section III. Demonstrations have been made using these special techniques and have satisfied all requirements of the Code. The special methods of nondestructive examination are deemed acceptable. This acceptability based on the Code provisions satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

4. Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of Code Sections III and IX. Qualification in accordance with the Code provisions meets the requirements of GDC 1, GDC 30, and 10 CFR 50.55a concerning quality standards.

5. When welding components of ferritic steels as identified in Item 4 above, Code controls are supplemented by conformance with the recommendations of regulatory guides as follows:

   a. The controls imposed on welding preheat temperatures are in conformance with the recommendations of RG 1.50 since these controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the potential for subsequent cracking. These controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

   b. The controls imposed on electroslag welding of ferritic steels are in conformance with the recommendations of RG 1.34 because the welds fabricated by the process will ensure high integrity and will have a sufficient degree of toughness to furnish adequate safety margins. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

   c. The controls imposed during weld cladding of ferritic steel components are in conformance with the recommendations of RG 1.43 because the process used provides reasonable assurance that under-clad cracking will not occur during the weld cladding process. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

6. When welding components of austenitic stainless steels, Code controls are supplemented by conformance with the recommendations of regulatory guides and other regulatory positions as follows:

   a. The controls imposed on delta ferrite in austenitic stainless steel welds are in conformance with the recommendations of RG 1.31, as supplemented (for BWRs only) by the positions of GL 88-01 or the recommendations of NUREG-0313, Revision 2, because the controls used provide reasonable assurance that the welds will not contain micro cracks. These controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR

50.55a and the requirements of GDC 14 regarding fabrication to prevent rapid propagating failure of the RCPB.

b. The controls imposed on electroslag welding of austenitic stainless steels are in conformance with the recommendations of RG 1.34, for the same reason as stated in Item 5b discussed above.

7. The controls (during, all stages of welding) employed that avoid contamination and sensitization that could cause stress-corrosion cracking in austenitic stainless steels conform with the recommendations of regulatory guides and other regulatory positions as follows:

a. The controls employed that avoid contamination and sensitization of austenitic stainless steel are in conformance with the recommendations of RG 1.44, or an acceptable alternative, as supplemented (for BWRs only) by the positions of GL 88-01 or the recommendations of NUREG-0313, Revision 2, because the controls used provide assurance that welded components will not be contaminated nor sensitized prior to and during the welding process. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a and the GDC 4 requirement relative to material compatibility.

b. The controls regarding onsite cleaning and cleanliness control of austenitic stainless steel are in conformance with the recommendations of RG 1.37 because the controls used provide assurance that austenitic stainless steel components will be properly cleaned onsite. The controls satisfy Appendix B of 10 CFR Part 50 regarding controls for onsite cleaning of materials and components.

8. Fracture toughness of the reactor vessel and its appurtenances is controlled by conformance with Appendix G to 10 CFR Part 50, which specifies ASME Code provisions and supplementary requirements of Appendix G to 10 CFR Part 50. The fracture toughness tests required by the ASME Code and by Appendix G to 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G of the Code as a guide in establishing safe operating procedures, the use of the results of the fracture toughness tests performed in accordance with the Code and NRC regulations, and the implementation of the material surveillance program in accordance with 10 CFR Part 50 Appendix G and Appendix H, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions for the service life of the reactor vessel. Compliance with the provisions of Appendix G to 10 CFR Part 50, satisfies the requirements of GDC 14, GDC 31, 10 CFR 50.55a, and 10 CFR 50.60 regarding prevention of fracture of the RCPB. For COL applications, the implementation milestones for the reactor vessel surveillance program are included in Table [__] on operating programs in SRP Section 13.4 and the associated license condition on operational program implementation.

9. Changes in the fracture toughness of material in the reactor vessel belt-line caused by exposure to neutron radiation have been assessed properly, and adequate safety margins against the possibility of vessel failure are provided as the material surveillance requirements of ASTM E-185 and Appendix H to 10 CFR Part 50, are met. Compliance with these requirements assures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of GDC 32 and 10 CFR 50.60 regarding an appropriate material surveillance program for the reactor vessel.

10. Integrity of the reactor vessel studs and fasteners is assured by conformance with the recommendations of RG 1.65. Compliance with these recommendations satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a; the prevention of RCPB fracture requirement of GDC 31; and the requirements of Appendix G to 10 CFR Part 50, as detailed in the provisions of the ASME Code, Sections II and III.

    Accordingly, the staff concludes that the plant design is acceptable and meets the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A of 10 CFR Part 50; the requirements of Appendices B and G of 10 CFR Part 50; and the requirements of § 50.55a of 10 CFR Part 50.

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

## V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

The staff will use this SRP section in performing safety evaluations of design certifications and license applications submitted by applicants pursuant to 10 CFR Parts 50 or 52. Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed 6 months or more after the date of issuance of this SRP section, unless superceded by a later revision.

Implementation schedules for conformance to parts of the methods discussed herein are contained in the referenced regulations and regulatory guides. Acceptable repairs and upgrades are described in the referenced generic letter for previously accepted materials and welds that do not meet NUREG-0313, Revision 2 recommendations related to material specifications and post-weld treatments for stress corrosion cracking resistant installations. NUREG-0313, Revision 2 recommendations for stress corrosion cracking resistant installations will be used by the staff for evaluation of reactor vessel attachments and appurtenances in new BWR applications.

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, General Design Criterion (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. Regulatory Guide (RG) 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

14. RG 1.34, “Control of Electroslag Weld Properties.”

15. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.”

16. RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.”

17. RG 1.44, “Control of the Use of Sensitized Stainless Steel.”

18. RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.”

19. RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs.”

20. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

21. RG 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-lb."

22. NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Hazelton, W.S., Koo, W.H., Division of Engineering and Systems Technology, January 1988.

23. NRC Letter to All Licensees of Operating Boiling Water Reactors (BWRs), and Holders of Construction Permits for BWRs, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter No. 88-01)," January 25, 1988.

24. ASME Boiler and Pressure Vessel Code, Sections II, "Materials," III, "Rules for Construction of Nuclear Facility Components," V, "Nondestructive Examination," IX, "Welding and Brazing Qualifications," XI, "Rules for Inservice Inspection of Nuclear Power Plant Components,"[134] American Society of Mechanical Engineers.

25. ASTM A-262 1970, "Detecting Susceptibility to Intergranular Attack in Stainless Steels," Practice A, "Oxalic Acid Etch Test for Classification of Etch Structures of Stainless Steels," Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, American Society for Testing and Materials.

26. ASTM A-708-1974, "Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel," Annual Book of ASTM Standards, American Society for Testing and Materials.

27. ASTM E-185-1982, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials.

## PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the draft standard review plan are covered by the requirements of 10 CFR 50.54, which were approved by the Office of Management and Budget, approval number 3150 - 0011.

## 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.

**SRP Section 5.3.1**
Description of Changes

The following summarizes the changes in Revision 2, dated xxxxxx 2006.

1. General changes included editorial and formatting changes. Note: minor editorial and formatting changes are not identified by side bars.

2. Standard language was added throughout the SRP section to extend the applicability to licensing and design certification reviews submitted under 10 CFR Part 52, including the applicability of the Combined License Applications for Nuclear Power Plants (LWR Edition) - Regulatory Guide DG-1145 as superceded by the final guide expected December 2006.

3. Language was added to the boilerplate on the front page, acceptance criteria and review procedures to clarify that the SRP represents an acceptable approach for meeting the Commission's regulations and that applicants are required to identify deviations from this criteria and evaluate how the alternative approaches meet the Commission's regulations.

4. Specific changes identified by section of the SRP:

<u>REVIEW RESPONSIBILITIES</u> - Editorial revision to reflect change in primary review branch resulting from office reorganization – identified by function. This change is reflected throughout the SRP section.

I. <u>AREAS OF REVIEW</u>

3. <u>Special Methods for Nondestructive Examination</u>

Updated SRP format: added ref. to ASME B&PV Code.

4. <u>Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels</u>

Introduced review area covering grinding controls.

8. <u>Inspection, Test, Analysis, and Acceptance Criteria (ITAAC)</u>

This subsection was added for the purpose of addressing design certification and combined license reviews performed pursuant to 10 CFR Part 52.

Added Review Interfaces - captures related reviews and associated SRP section interfaces.

## II. ACCEPTANCE CRITERIA

Citations of 10 CFR regulations are listed alphabetically. Revisions reflect new and updated regulations covering reactor vessel materials, including 10 CFR 50.60. A citation of 10 CFR 52.97(b)(1) was added in reference to ITAAC requirements for design certification and combined license reviews.

### 1. Materials

Editorial changes reflect either new code editions or updated formatting. Similar editorial changes are found in item nos. 2. and 3.

### 4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

Text deleted form paragraph b. describes the alternative weld heat treatment controls of Westinghouse Topical Report WCAP-8678, published in September 1975. This text was deleted due to the expectation that future applicants will consider such alternative methods obsolete.

Paragraph d. was expanded to include citations of Generic Letter (GL) 88-01 and NUREG-0313, Rev. 2, concerning BWR materials. These citations represent the current standards which the staff has been using to evaluate the issue of sensitized, stainless steels in BWR environments since issuance of GL 88-01. This change is reflected throughout SRP Section 5.3.1.

Paragraph e. was expanded to provide references to test standards to be used to meet acceptance criteria, including criteria for BWRs. Additional criteria in Regulatory Guide (RG) 1.37 regarding abrasive work controls are also referenced.

Introduced Paragraph f. to reference additional controls described in GL 88-01 and in NUREG-0313, considered necessary to avoid weld IGSCC in BWRs.

### 5. Fracture Toughness

First paragraph edited to reflect change in the regulations: Appendix G to 10 CFR Part 50 is invoked by 10 CFR 50.60, not 10 CFR 50.55a(I).

Paragraphs b., b.(1), b.(2) provide updated citations of Appendix G to 10 CFR Part 50.

Paragraph c. was added regarding regulatory acceptance criteria and guidance concerning neutron embrittlement of reactor vessel materials. References to Appendix G to 10 CFR Part 50, the ASME Code, Section 3, and RG 1.99 are included. The sentences addressing the design of reactor vessel belt-line materials to allow for thermal annealing were removed, as they are no longer considered relevant to the review of license applications.

6. Material Surveillance

Paragraph b. - title for ASTM E-185 deleted, replacing it with reference number.

Subparagraph c.(1) revised regulatory citation from Appendix G to Appendix H to Part 50.

Subparagraph c.(4) was revised to include a regulatory citation.

Last paragraph added to reflect supplemental material surveillance program requirements for a proposed facility design life in excess of 40 years.

7. Reactor Vessel Fasteners

First paragraph and paragraph b. contain minor editorial changes.

Technical Rational section introduced as part of the SRP updated format.

III. REVIEW PROCEDURES

1. Materials

Several editorial changes incorporated as part of SRP updated format.

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steel

Third and fourth paragraphs, respectively, address abrasive work controls and positions in GL 88-01 related to BWR reactor vessel attachments.

5. Fracture Toughness

The sixth and seventh paragraphs address review and acceptance criteria for the predicted initial and end-of-life reactor vessel material properties.

6. Material Surveillance

Expanded paragraph a. to address material surveillance programs for applicants proposing a facility design life in excess of 40 years.

Paragraph b. was introduced to address design certification reviews of material surveillance programs applicable to 10 CFR Part 52.

Final paragraph of review procedures subsection was introduced based on its applicability to standard design certification reviews and combined license reviews under 10 CFR Part 52.

IV. EVALUATION FINDINGS

Revisions to the first two paragraphs include an added 10 CFR 50.60 citation.

Additional language in paragraphs 1., 2., 6. and 7. captures previous revisions concerning staff positions in GL 88-01 regarding BWR materials.

Paragraph 8. now includes additional references to material surveillance program requirements in 10 CFR Part 50, Appendices G and H, and 10 CFR 50.60.

Editorial revisions in paragraph 9. include added reference to 10 CFR 50.60.

Final paragraph of evaluation findings subsection was introduced based on its applicability to standard design certification reviews and combined license reviews under 10 CFR Part 52.

V. IMPLEMENTATION

Second paragraph now includes language addressing SRP implementation for applications submitted under either 10 CFR Part 50 or 10 CFR Part 52.

Added third paragraph concerning SRP applicability to new applications.

Fourth paragraph expanded to address implementation of staff positions and resolution of issues associated with GL 88-01.

VI. REFERENCES

References were updated and renumbered to reflect applicable regulations and guidance.

## IV. EVALUATION FINDINGS

Revisions to the first two paragraphs include an added 10 CFR 50.60 citation.

Additional language in paragraphs 1., 2., 6. and 7. captures previous revisions concerning staff positions in GL 88-01 regarding BWR materials.

Paragraph 8. now includes additional references to material surveillance program requirements in 10 CFR Part 50, Appendices G and H, and 10 CFR 50.60.

Editorial revisions in paragraph 9. include added reference to 10 CFR 50.60.

Final paragraph of evaluation findings subsection was introduced based on its applicability to standard design certification reviews and combined license reviews under 10 CFR Part 52.

## V. IMPLEMENTATION

Second paragraph now includes language addressing SRP implementation for applications submitted under either 10 CFR Part 50 or 10 CFR Part 52.

Added third paragraph concerning SRP applicability to new applications.

Fourth paragraph expanded to address implementation of staff positions and resolution of issues associated with GL 88-01.

## VI. REFERENCES

References were updated and renumbered to reflect applicable regulations and guidance.

| OFFICE | EMCB:DE | NRPB | DCI/CVIB | OGC (NLO w comments) | DCI |
|---|---|---|---|---|---|
| NAME | CSydnor | SKoenick | MMitchell | RWeisman | JGrobe |
| DATE | 06/20/2005 | 02/13/06 | 01/09/06 | 05/30/2006 | 06/23/2006 |