NUREG-0800
(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

5.3.1 REACTOR VESSEL MATERIALS

REVIEW RESPONSIBILITIES

Primary - Materials and Chemical Engineering Branch (~~MTEB~~ EMCB[1])

Secondary - None

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 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 fabricability. 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 fabricability.

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 ASME Boiler and Pressure Vessel Code (Reference 24)[2] (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.[3]

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 nonbrittle 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, dropweight 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 beltline 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 dropweight test, or the temperature that is 33°C (60°F)[4] 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 beltline 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.

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.

Review Interfaces:[5]

The EMCB also performs the following reviews under the SRP sections indicated:[6]

1. The EMCB evaluates the adequacy of programs for assuring the integrity of bolting and threaded fasteners as part of its primary review responsibility for SRP Section 3.13 (proposed).[7]

2. The EMCB determines the compatibility of the thermal insulation with the austenitic stainless steel of the reactor vessel appurtenances and determines the acceptability of any nonmetallic thermal insulation that is employed, as part of its review responsibility for SRP Section 5.2.3.[8]

3. The EMCB reviews 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, as part of its review responsibility for SRP Section 5.3.2.[9]

In addition, the ~~MTEB~~ EMCB[10] will coordinate evaluations, ~~of~~by[11] other branches, that interface with the overall review of the reactor vessel materials as follows:

~~The Chemical Engineering Branch (CMEB) determines the compatibility of the thermal insulation with the austenitic stainless steel of the reactor vessel appurtenances and determines the acceptability of any nonmetallic thermal insulation that is employed, as part of its secondary review responsibility for SRP Section 5.2.3.~~

1. The Reactor Systems Branch (SRXB) reviews the reactor vessel wall neutron fluence as part of its primary review responsibilities for SRP Section 4.3. The SRXB also reviews the overpressure protection system as part of its primary review responsibility for SRP Section 5.2.2.[12]

2. The review for Quality Assurance is coordinated and performed by the Quality Assurance and Maintenance Branch (~~QAB~~ HQMB[13]) as part of its primary review responsibility for SRP Sections 17.1 and 17.2.

3. For new plant applicants, the Probabilistic Safety Assessment Branch (SPSB) coordinates and performs shutdown risk assessment reviews, including the ability of materials and methods used to seal the reactor vessel in-core instrument seal table (PWRs) during shutdown operations, as part of its primary review responsibility for SRP Section 19.1 (Proposed).[14]

For those areas of review identified above as part of the review in other SRP sections ~~responsibility of other branches~~, the acceptance criteria necessary for the review and their method of application are contained in the referenced SRP section ~~of the corresponding branch~~.[15]

II. ACCEPTANCE CRITERIA

~~MTEB~~ EMCB[16] acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

~~2~~A.[17] General Design Criteria (GDC) 1 and 30, as they relate to quality standards for design, fabrication, erection, and testing of structures, systems and components;

~~3~~B. General Design Criterion 4, as it relates to compatibility of components with environmental conditions;

~~4~~C. General Design Criterion 14, as it relates to prevention of rapidly propagating fractures of the RCPB;

~~5~~D. General Design Criterion 31, as it relates to material fracture toughness;

~~6~~E. General Design Criterion 32, as it relates to the requirements for a materials surveillance program;

~~1~~F. ~~Section~~10 CFR Part 50, §[18] 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness;

G. 10 CFR Part 50, §50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," as it relates to reactor coolant pressure boundary fracture toughness and material surveillance requirements of 10 CFR 50, Appendix G and Appendix H;[19]

~~7~~H. 10 CFR Part 50,[20] Appendix B, Criterion XIII,[21] as it relates to onsite material cleaning control;

~~8~~I. 10 CFR Part 50,[22] Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness; and

~~9~~J. 10 CFR Part 50,[23] Appendix H, as it relates to the determination and monitoring of fracture toughness.

Specific criteria necessary to meet the relevant requirements of ~~10 CFR Part 50, § 50.55a; GDC 1, 4, 14, 30, 31, and 32; and Appendices B, G, and H~~ the Commissions regulations identified above[24] are as follows for each review described in subsection I of this SRP section:

1. Material Specifications

   The requirements of GDC 1 and 30 and ~~§~~10 CFR[25] 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for material specifications, as detailed below:

   a. Acceptable material specifications for the reactor vessel and its appurtenances are those listed in the Code, Section III, Appendix I, and are presented in detail in Code Section II, Parts A, B, and C. The materials must also meet the specifications 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-1400 and 10 CFR Part 50, Appendix G. These data must include information on mechanical properties, weldability, and physical changes of the material.

2. Special Processes Used for Manufacture and Fabrication of Components

   The requirements of GDC 1 and 30 and ~~§~~10 CFR[26] 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, Paragraph NA-8000[27], 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[28] 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, for normal methods of examination. When special techniques or procedures are developed, they must be equivalent or superior to the techniques described in Appendix IX-6000[29] of Code Section III, and must be proven so, by demonstration on the specific type of component part.

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, ~~and~~ ASME Code provisions, and other regulatory documents[30] necessary to satisfy the relevant requirements of GDC 1, 4, 14, and 30; Appendix B; and ~~§~~10 CFR[31] 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 1.50, "Control of Preheat Temperature for Welding Low-Alloy Steel," and 1.34, "Control of Electroslag Weld Properties." Westinghouse Topical Report, WCAP-8678, "Effects of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," (Reference 28) provides acceptable alternatives to the recommendations in Regulatory Guide 1.50, Position C.2.[32]

c. The regulatory positions of Regulatory Guide 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 Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and Regulatory Guide 1.34. For the BWR austenitic stainless steel reactor vessel attachments and appurtenences specified in Generic Letter 88-01 (Reference 23), the weld metal ferrite content should be controlled as described in the positions of Attachment A to Generic Letter 88-01 or the recommendations of NUREG-0313 (Reference 21).[33]

e. The regulatory positions of Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37[34], "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.

Regulatory Guide 1.44 states that non-sensitization should be verified using ASTM A 262[35] (Reference 25) Practices A or E, or another method that can be demonstrated to show non-sensitization of austenitic stainless steel. Alternative tests to those in ASTM A-262[36] that has been previously accepted include ASTM A 708 (Reference 26).[37] For BWRs, the control of sensitized steel per Regulatory Guide 1.44 should be modified as necessary to conform with the positions in Attachment A to Generic Letter 88-01 or the recommendations of NUREG-0313.[38]

The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in Regulatory Guide 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.[39]

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 appurtenences. The additional controls are described in Attachment A to Generic Letter 88-01 and in NUREG-0313. 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.[40]

The referenced regulatory guides are described in detail in the acceptance criteria of SRP Section 5.2.3, which is reviewed by ~~MTEB~~ EMCB[41].

5. Fracture Toughness

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 ~~§ 50.55a(i)~~10 CFR 50.60[42] 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, during conditions of normal operation, and during anticipated operational occurrences:

a. The ferritic materials shall be tested in accordance with the ASME Code ~~section~~paragraph[43] 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} + 33°C)[(T_{NDT} + 60°F)][44] each Charpy C_v specimen tested shall exhibit at least 0.89 mm (35 mils)[45] lateral expansion and not less than 68 J (50 ft-lbs)[46] of absorbed energy.

(3) When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}. 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 $RT_{NDT} = T_{cv}$ - 33°C ($RT_{NDT} = T_{cv}$- 60°F)[47]. Thus the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{cv} - 33°C) [(T_{cv} - 60°F)][48]

(4) When a C_v impact test has not been performed at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)][49], or when the C_v impact test at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)][50] does not exhibit a minimum of 68 J (50 ft-lbs)[51] and 0.89 mm (35 mils)[52] lateral expansion, a temperature representing a minimum of 68 J (50 ft-lbs)[53] and 0.89 mm (35 mils)[54] 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,[55] IV.A.2, ~~and~~ IV.A.3[56] and IV.B of Appendix G of 10 CFR Part 50 shall be met.

(1) Standard Review Plan Section 5.3.2, "Pressure-Temperature Limits," discusses the requirements of paragraphs IV.A.2 and IV.A.3[57] of Appendix G in detail.

(2) The acceptance criteria discussed in paragraph IV.~~B~~A.1[58] of Appendix G states that reactor vessel beltline materials shall have a minimum uppershelf energy as determined from Charpy V-notch impact tests on unirradiated specimens in accordance with paragraph NB-2322a[59] of the Code, Section III, of 102 J (75 ft-lbs)[60], and must maintain an upper shelf energy no less than 68 J (50 ft-lb) throughout the life of the vessel[61] unless it is demonstrated to the Commission by appropriate data and analyses based on other types of tests that lower values of uppershelf fracture energy are adequate.

c. The neutron radiation embrittlement effects on reactor vessel materials shall be determined in accordance with 10 CFR 50, Appendix G, Section V, and Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

Reactor vessel beltline materials for which either the predicted Charpy upper shelf energy at end of life is below 68 J (50 ft-lb), or the predicted adjusted reference temperature at end of life exceeds 93°C (200°F), must be designed to allow thermal annealing for the recovery of the material toughness properties. Alternatively, the requirements of 10 CFR 50, Appendix G, paragraph V.C may be met.[62]

6. Material Surveillance

The material surveillance acceptance criteria are the requirements of Section II 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 II 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 that have been verified by 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 these conditions shall have their beltline regions monitored by a surveillance program complying with the American Society for Testing and Materials (ASTM) standard ~~"Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,"~~ ASTM ~~Designation~~ E-185 (Reference 27)[63], 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.A of Appendix G, except that dropweight specimens are not required.

(2) Surveillance capsules containing the surveillance specimens shall be located near the inside vessel wall in the beltline 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 II.~~C~~B.~~3~~2[64] 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 II.~~C~~B.3[65] 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, depending on the degree of commonality and the predicted severity of irradiation in accordance with the requirements of paragraph II.C of Appendix H[66].

The material surveillance program criteria of ASTM E-185 cited in 10 CFR 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.[67]

7. <u>Reactor Vessel Fasteners</u>

The acceptance criteria for the reactor vessel bolting material are given by paragraph IV.A~~.3~~[68] of Appendix G to 10 CFR Part 50 and by the recommendations of Regulatory Guide 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[69] 50.55a, and meet the requirements of GDC 31 regarding prevention of fracture of the RCPB.

Regulatory Positions C.1 and C.2 of Regulatory Guide 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)[70], and the fracture toughness tests and acceptance levels of paragraph IV.A~~.3~~[71] of Appendix G to 10 CFR Part 50 must be met as detailed in paragraph NB-2333 of Section III of the Code.

c. Surface treatments, plating, or thread lubricants used must 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, subarticle NB-2580, and including additional recommendations given in Regulatory Position C.2 of Regulatory Guide 1.65.

Technical Rationale:[72]

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 (SSC) important to safety. GDC 1 and 10 CFR 50.55a establish that the quality assurance standards to be applied to SSC 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 reactor coolant pressure boundary (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. Regulatory Guides 1.31, 1.34, 1.43, 1.44, 1.50, and 1.65 provide regulatory positions applicable to compliance with GDCs 1 and 30. Compliance with GDCs 1 and 30, 10 CFR 50.55a, and the positions of the Regulatory Guides, 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 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. Regulatory Guide 1.31 provides regulatory positions regarding the control of ferrite content in stainless steel welds and that are relevant to compliance with GDC 14. Application of GDC 14 and Regulatory Guide 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 nonbrittle 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 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) 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 beltline 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 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 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 of 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 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 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. Regulatory Guide 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 50, Appendix G, establishes requirements for the fracture toughness of RCPB 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 GDCs 14 and 31 related to fracture prevention. The rationale for these requirements is as discussed in Items 3 and 4 above.

9. 10 CFR 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 beltline, resulting from exposure to neutron irradiation and the thermal environment. Data from the surveillance program is utilized in complying with 10 CFR 50, Appendix G requirements for establishing pressure-temperature limits, inservice inspection requirements, and corrective actions (such as vessel annealing) if fracture toughness criteria can not 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 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 effect the integrity of the reactor vessel, and thus its ability to accomplish the safety functions under all anticipated and postulated conditions.

III. REVIEW PROCEDURES

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 as follows:

1. Material Specifications

The material specifications for the reactor vessel and its appurtenances are compared with the acceptable specifications listed in the Code, Section III, Appendix I, and Section II, Parts A, B, C.

Materials not listed in the Code, or deviations in a listed specification, are clearly identified, and the bases for deviation or nonconformance evaluated. 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 ~~Materials Engineering Branch~~ EMCB[73] has taken exception to the use of a specific material, or questions certain aspects of a specification, 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. <u>Special Methods for Nondestructive Examination</u>

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 penetrant, 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 require 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. <u>Special Controls and Special Processes Used for Ferritic Steels and austenitic stainless steel.</u>

The controls on welding of ferritic steels and austenitic stainless steels discussed in Standard Review Plan Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," 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 stress corrosion cracking promoting contaminants and the cold-working of surfaces.[74]

For BWRs, the reactor vessel attachments and appurtenances are reviewed for conformance with the staff positions of Generic Letter 88-01 or the recommendations of NUREG-0313 with regard to protection against IGSSC in or near weldments to the reactor vessel.[75]

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 dropweight 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 predicted end-of-life Charpy upper shelf energy and adjusted reference temperature for the reactor vessel materials in accordance with 10 CFR 50, Appendix G, paragraph IV.B. Reactor vessel materials that do not meet the specified end-of-life acceptance criteria are reviewed in accordance with paragraphs V.C and V.D of 10 CFR 50, Appendix G.[76] NUREG-0744 (Reference 22) provides an acceptable methodology for performance of fracture analysis for demonstrating adequate margins of safety for continued operation in accordance with 10 CFR 50, Appendix G, paragraph V.C.3.[77]

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, "Pressure-Temperature Limits," 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. Material Surveillance

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 beltline, 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[78] 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 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.[79]

b. For design certification applications, a Combined License action item, and associated ITAAC, 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.[80]

~~b~~c.[81] 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 beltline 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 standard design certification reviews under 10 CFR 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[82]

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient and adequate information has been provided to satisfy the requirements of the SRP section, and that ~~his~~the[83] evaluation supports the conclusions of the following type, to be included in the staff's safety evaluation report.

The staff concludes that the reactor vessel materials are acceptable and meet the requirements of General Design Criteria 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR Part 50; the material testing and monitoring requirements of Appendices B, G, and H of 10 CFR Part 50; and the requirements of ~~§~~10 CFR 50.55a~~ of 10 CFR Part 50~~[84] and 10 CFR 50.60.[85] This conclusion is based on the following:

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. Special requirements of the applicant with regard to 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[86] 50.55a. In addition, the use of austenitic stainless steel in BWR reactor vessel attachments and appurtenances conforms with the staff positions in Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2.[87]

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[88] 50.55a. In addition, the use of austenitic stainless steel in BWR reactor vessel attachments and appurtenances conforms with the staff positions in Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2.[89]

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[90] 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[91] 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 Regulatory Guide 1.50, or an acceptable alternative,[92] "Control of Preheat Temperature for Welding of Low-Alloy Steel," 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[93] 50.55a.

b. The controls imposed on electroslag welding of ferritic steels are in conformance with the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties," 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[94] 50.55a.

c. The controls imposed during weld cladding of ferritic steel components are in conformance with the recommendations of Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," because the process used provides reasonable assurance that under-clad[95] cracking will not occur during the weld cladding process. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and ~~§~~10 CFR[96] 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[97] as follows:

a. The controls imposed on delta ferrite in austenitic stainless steel welds are in conformance with the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," as supplemented (for BWRs only) by the positions of Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2,[98] because the controls used provide reasonable assurance that the welds will not contain micro cracks. These controls also satisfy the quality standards requirement of GDC 1, GDC 30, and ~~§~~10 CFR[99] 50.55a and the requirements of GDC 14 regarding fabrication to prevent RCPB rapid propagating failure.

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

7. The controls (during, all stages of welding) to 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[100] as follows:

a. The controls to avoid contamination and sensitization of austenitic stainless steel are in conformance with the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," or an acceptable alternative,[101] as supplemented (for BWRs only) by the positions of Generic Letter 88-01 or the recommendations of NUREG-0313, Revision 2,[102] 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 requirement of GDC 1, GDC 30, and ~~§~~10 CFR[103] 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 Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," 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,[104] which specifies ASME Code provisions and supplementary requirements of Appendix G~~,~~ to 10 CFR ~~Part~~ 50.[105] The fracture toughness tests required by the ASME Code and by Appendix G ~~of~~ to 10 CFR ~~Part~~ 50[106] 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, ~~and~~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 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.[107] Compliance with the provisions of Appendix G~~,~~ to 10 CFR ~~Part~~ 50,[108] satisfies the requirements of GDC 14, GDC 31, ~~and §~~10 CFR[109] 50.55a, and 10 CFR 50.60[110] regarding prevention of fracture of the reactor coolant pressure boundary.

9. Changes in the fracture toughness of material in the reactor vessel beltline 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-[111]185 and Appendix H~~,~~ to 10 CFR ~~Part~~ 50,[112] are met. Compliance with these documents 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[113] 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 Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." Compliance with these recommendations satisfies the quality standards requirements of GDC 1, GDC 30, and ~~§~~10 CFR[114] 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G~~,~~ to 10 CFR ~~Part~~ 50,[115] as detailed in the provisions of the ASME Code, Sections II and III.

For design certification 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 inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[116]

V. IMPLEMENTATION

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

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52. [117] 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 six months or more after the date of issuance of this SRP section.[118]

Implementation schedules for conformance to parts of the methods discussed herein are contained in the referenced regulations and[119] 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 appurtenences in new BWR applications.[120]

VI. REFERENCES

1. 10 CFR ~~Part 50, Section~~ [121]50.55a, "Codes and Standards."

2. 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation."[122]

3. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."[123]

~~2~~4.[124] 10 CFR Part 50, Appendix A, ~~"~~General Design ~~Criteria for Nuclear Plants. (~~Criterion 1, "Quality Standards and Records."~~; Criterion 4, "Environmental and Missile Design Bases"; Criterion 14, "Reactor Coolant Pressure Boundary"; Criterion 30, "Quality of Reactor Coolant Pressure Boundary"; Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"; and Criterion 32, "Inspection of Reactor Coolant Pressure Boundary.")~~

5. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Dynamic Effects Design Bases."[125]

6. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

7. 10 CFR Part 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

8. 10 CFR Part 50, Appendix A, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

9. 10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

10~~3~~.[126] 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants[127]."

11~~4~~. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

12~~5~~. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

13~~8~~. Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal."

14~~9~~. Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

15~~10~~. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."[128]

16~~11~~. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

17~~12~~. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

18~~13~~. Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel."

19~~14~~. Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs."

20. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."[129]

21. 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.[130]

22. NUREG-0744, Revision 1, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Johnson, R., Division of Safety Technology, October 1982.[131]

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.[132]

24~~6~~.[133] ASME Boiler and Pressure Vessel Code, Sections II, "Materials," III, "Rules for Construction of Nuclear Power Plant 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.[135] [136]

26. ASTM A-708-1974[137], "Detection of Susceptibility to Intergranular Corrosion in Severely Sensitized Austenitic Stainless Steel," Annual Book of ASTM Standards, American Society for Testing and Materials.[138]

27~~7~~.[139] ASTM E-185-1982[140], "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, American Society for Testing and Materials.[141]

28. WCAP-8678, "Effects of Preheat and Post Weld Heat Treat on Hydrogen-Induced Cracking in Pressure Vessel Steels," Westinghouse Electric Corporation Topical Report, September 1975.[142]

SRP Draft Section 5.3.1

Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 5.3.1.
2.	SRP-UDP format item, Reformat References	Added parenthetical reference identification for the citation of the ASME B&PV Code.
3.	**Integrated Impact 823**	Added area of review covering grinding controls.
4.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 33°C for the cited temperature difference of 60°F. See attached metrication documentation.
5.	SRP-UDP format item, Reformat Areas of Review.	Added "Review Interfaces" heading to Areas of Review. Reformatted existing description of review interfaces in numbered format to describe how SRXB reviews aspects of the RCIC under other SRP sections and how other branches support the review.
6.	SRP-UDP format item, Editorial	Editorial change to add the typical lead-in sentence for those SRP sections that are review interfaces with SRP Section 5.3.1, and are also the responsibility of the primary PRB for SRP Section 5.3.1.
7.	SRP-UDP Integration of Bolting Issues, Potential Impacts 983 and 21614	Added a review interface reflecting reviews of bolting and threaded fastener programs under new SRP Section 3.13.
8.	SRP-UPD format item, Reformat Areas of Review.	Moved this paragraph from reviews performed by other branches to a review interface within the same PRB for SRP Section 5.3.1. Also deleted reference to "secondary review" from the original paragraph to address current PRB responsibilities.
9.	Editorial, **Integrated Impact 820.**	This review interface was adapted from text in existing SRP Section 5.3.1, specific criteria 5.b.(1) and Review Procedure III.5, and also incorporates a specific interface for review of pressurized thermal shock in accordance with 10 CFR 50.61.
10.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 5.3.1.
11.	Editorial	Replaced "of" with "by".

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
12.	**PI # 401, PI # 404**	Added a review interface with SRP Sections 4.3 and 5.2.2. The neutron fluence for the reactor vessel wall is reviewed in SRP Section 4.3 and is utilized by reviewers in SRP Section 5.3.1 in evaluating the conformance with 10 CFR 50, Appendix G and Appendix H requirements regarding reactor vessel material embrittlement and associated changes in fracture toughness properties. SRP Section 5.2.2 is the implementing section for review of overpressure protection systems that maintain the RCPB within the 10 CFR 50, Appendix G pressure-temperature limits.
13.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP sections 17.1 and 17.2.
14.	From NUREG-1449, This is Review Interface so no PIs or IIs were necessary	This review interface identifies reviews conducted to satisfy NUREG-1449 guidance on Shutdown and Low Power Operations. The staff requested that design certification applicants complete an assessment of shutdown and low-power risk. The shutdown and low-power risk assessment must identify design-specific vulnerabilities and weaknesses and document consideration and incorporation of design features that minimize such vulnerabilities. The materials and methods for ensuring a proper in-core instrument seal table seal to prevent reactor vessel draining during shutdown operations is one aspect of the shutdown and low-power risk evaluation that will require an interface with reactor vessel materials. Consideration of this issue in the shutdown and low-power risk assessment is the responsibility of the SPSB and will be included in the proposed SRP Section 19.1 on risk assessments.
15.	Editorial	Revised the paragraph to be consistent with the review interfaces that involve both the PRB responsible for SRP Section 5.3.1 and PRBs responsible for other SRP sections.
16.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 5.3.1.
17.	Editorial	Existing Acceptance Criteria 1 related to 10 CFR 50.55a was repositioned following the listing of GDCs in accordance with SRP-UDP guidance for ordering of Acceptance Criteria and to allow grouping of 10 CFR references, both new and existing. The Acceptance Criteria number designations were changed to letters to differentiate the list of Acceptance Criteria from the list of specific criteria.

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Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
18.	SRP-UDP format item, Reformat References	The existing reference cites "section 50.55a" as Acceptance Criteria. The reference was reformatted to cite "10 CFR Part 50, § 50.55a" to provide a more complete reference to the listed criteria, consistent with SRP-UDP guidance for citation of Acceptance Criteria.
19.	**Integrated Impact 819.**	Added 10 CFR 50.60 to the Acceptance Criteria. 10 CFR 50.60 provides requirements for compliance with the provisions of 10 CFR 50 Appendices G and H regarding fracture toughness and material surveillance.
20.	SRP-UDP format item, Reformat References	Added "10 CFR Part 50" to Acceptance Criteria references to Appendix B to clarify the source document for the subject Appendix.
21.	SRP-UDP format item, Verification of References, Editorial	Revised citation to state a more precise location of the applicable requirements.
22.	SRP-UDP format item, Reformat References	Added "10 CFR Part 50" to Acceptance Criteria references to Appendix G to clarify the source document for the subject Appendix.
23.	SRP-UDP format item, Reformat References	Added "10 CFR Part 50" to Acceptance Criteria references to Appendix H to clarify the source document for the subject Appendix.
24.	Editorial	The lead-in sentence for the discussion of specific criteria was revised to simplify the sentence and to eliminate the redundant listing of the Acceptance Criteria. The modified sentence is consistent with the corresponding sentence in related SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials."
25.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent with SRP-UDP guidance throughout the SRP Section.
26.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent with SRP-UDP guidance throughout the SRP Section.
27.	**Integrated Impact 975.**	This is a placeholder integrated impact. Citations of ASME Article NA-8000 is not consistent with the latest version of the ASME B&PV Code.
28.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent with SRP-UDP guidance throughout the SRP Section.
29.	**Integrated Impact 975.**	This is a placeholder integrated impact. The citation of ASME B&PV Code, Appendix IX-6000 is out of date. The latest version of the Code no longer contains this Appendix.

Item	Source	Description
30.	Editorial	Revised the lead-in sentence to accommodate the addition of Generic Letter 88-01 and NUREG-0313.
31.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent with SRP-UDP guidance throughout the SRP Section.
32.	**Integrated Impact 974.**	Added WCAP-8678 as an acceptable alternative to Regulatory Guide 1.50.
33.	**Integrated Impact 815.**	Incorporated NUREG-0313 as providing supplemental acceptance criteria to Regulatory Guide 1.31 for ferrite content in welds.
34.	**Integrated Impact 814.**	This is a placeholder integrated impact. Regulatory Guide 1.37 cites ANSI N45.2.1 with regard to cleanliness and cleanliness controls. In the CE FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1. The FSER further states that the staff reviewed NQA-2-1983 and found it acceptable. The staff is in the process of reviewing the NQA standards and developing regulatory guidance for implementation.
35.	**Integrated Impact 975.**	This is a placeholder integrated impact. The latest version of ASTM A-262 is dated 1993.
36.	**Integrated Impact 975.**	This is a placeholder integrated impact. The latest version of ASTM A-262 is dated 1993.
37.	**Integrated Impact 973.**	Incorporated position from CE 80+ FSER that the staff accepts ASTM A 708 as an alternative to ASTM A 262 as endorsed by Regulatory Guide 1.44. This staff position is also documented in SRP Section 4.5.1, paragraph III.2.
38.	**Integrated Impact 816.**	Incorporated NUREG-0313 as providing supplemental acceptance criteria to Regulatory Guide 1.44 for control of sensitization of austenitic stainless steel.
39.	**Integrated Impact 823**	Added description of criteria for abrasive work controls, based on RG 1.37 position C.5.
40.	**Integrated Impact 812.**	Incorporated Generic Letter 88-01 and NUREG-0313 as providing acceptance criteria for control of intergranular stress corrosion cracking in BWRs.
41.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 5.3.1.
42.	**Integrated Impact 819.**	Paragraph 50.55a(i) was deleted and replaced by new rule 10 CFR 50.60 (48 FR 24009). This change was included in the 1983 amendment to 10 CFR 50, Appendix G and Appendix H.

SRP Draft Section 5.3.1
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
43.	SRP-UDP format item, Reformat References, Editorial	"section" was changed to "paragraph" to be consistent with other references to NB-2300 contained throughout SRP Section 5.3.1.
44.	SRP-UDP format item, Metrication Policy Implementation	Provided the SI equivalent equation for the cited equation (T_{NDT} + 60 °F). See attached metrication documentation.
45.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 0.89 mm for the cited lateral expansion value of 35 mils. See attached metrication documentation.
46.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 68 J for the cited absorbed energy value of 50 ft-lbs. See attached metrication documentation.
47.	SRP-UDP format item, Metrication Policy Implementation	Provided the SI equivalent equation for the cited equation $RT_{NDT} = T_{cv}$ - 60°F. See attached metrication documentation.
48.	SRP-UDP format item, Metrication Policy Implementation	Provided the SI equivalent equation for the cited equation (T_{cv} - 60°F). See attached metrication documentation.
49.	SRP-UDP format item, Metrication Policy Implementation	Provided the SI equivalent equation for the cited equation (T_{NDT} + 60 °F). See attached metrication documentation.
50.	SRP-UDP format item, Metrication Policy Implementation	Provided the SI equivalent equation for the cited equation (T_{NDT} + 60 °F). See attached metrication documentation.
51.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 68 J for the cited absorbed energy value of 50 ft-lbs. See attached metrication documentation.
52.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 0.89 mm for the cited lateral expansion value of 35 mils. See attached metrication documentation.
53.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 68 J for the cited absorbed energy value of 50 ft-lbs. See attached metrication documentation.
54.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 0.89 mm for the cited lateral expansion value of 35 mils. See attached metrication documentation.
55.	**Integrated Impact 811,** Reference Verification	Added reference to paragraph IV.A.1 which contains the requirements related to existing paragraph 5.b.(2).
56.	Editorial	Revised the paragraph number to provide clarification and consistency with regard to the associated paragraph citations.

Item	Source	Description
57.	Editorial	Revised the paragraph number to provide clarification and consistency with regard to the associated paragraph citations.
58.	**Integrated Impact 811,** Reference Verification	The citation of paragraph IV.B of 10 CFR 50, Appendix G, was corrected to cite paragraph IV.A.1, which contains the reference to the 75 ft-lb initial Charpy upper shelf energy. Paragraph IV.B provides thermal annealing requirements for vessels that do not meet minimum Charpy upper shelf energy requirements or the maximum allowable adjusted reference temperature.
59.	**Integrated Impact 975,** SRP-UDP standards citation update	Consideration should be given to updating the citation of ASME Boiler and Pressure Vessel Code, Division I, Section III, Subsection NB, paragraph NB2322a. No standard comparison is currently planned for this portion of the ASME Code.
60.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 102 J for the cited upper shelf energy value of 75 ft-lbs. See attached metrication documentation.
61.	**Integrated Impact 811.**	Added the minimum upper shelf energy criteria from 10 CFR 50, Appendix G, paragraph IV.A.1.
62.	**Integrated Impact 811.**	Added paragraph regarding acceptance criteria for the limiting adjusted reference temperature and Charpy upper shelf energy. Regulatory Guide 1.99 was added with regard to criteria for determining the adjusted reference temperature.
63.	SRP-UDP format item, Reformat References	The reference to ASTM E-185 was revised to delete the title and add a parenthetical reference identification. The use of the title in the text is redundant to reference in the Reference section of the SRP and its replacement by the parenthetical reference identification simplifies the text.
64.	SRP-UDP format item, Reference Verification	The referenced paragraph from 10 CFR 50, Appendix H was revised to cite the current paragraph number that is associated with the subject matter of the SRP.
65.	SRP-UDP format item, Reference Verification	The referenced paragraph from 10 CFR 50, Appendix H was revised to cite the current paragraph number that is associated with the subject matter of the SRP.
66.	Editorial	Added the regulatory reference for the existing requirement to be consistent with the other items listed under paragraph II.6.c of SRP Section 5.3.1.
67.	**Integrated Impact 813.**	Added a paragraph under specific criteria, paragraph II.6, to address the need to supplement the material surveillance program requirements for facilities with a design life that exceeds 40 years.

Item	Source	Description
68.	**Integrated Impact 811,** Reference Verification	The paragraph number was revised to reference the general applicability clause of Appendix G, Section IV. The reason for this change is that Paragraph IV.A.3 of the latest amendment of Appendix G does not correspond with the subject matter (i.e., reactor vessel fasteners) for which the paragraph is referenced in the existing SRP.
69.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
70.	SRP-UDP format item, Metrication Policy Implementation	Added the SI equivalent of 1170 MPa for the cited ultimate strength value of 170 ksi. See attached metrication documentation.
71.	**Integrated Impact 811,** Reference Verification	The paragraph number was revised to reference the general applicability clause of Appendix G, Section IV. The reason for this change is that Paragraph IV.A.3 of the latest amendment of Appendix G does not correspond with the subject matter (i.e., reactor vessel fastners) for which the paragraph is referenced in the existing SRP.
72.	SRP-UDP format item, Develop Technical Rationale.	Technical Rationale were developed for GDCs 1, 4, 14, 30, 31, and 32, as well as for 10 CFR 50.55a, 50.60, and 10 CFR 50, Appendices B, G, and H, in accordance with SRP-UDP guidance.
73.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 5.3.1.
74.	**Integrated Impact 823**	Added procedure for review of abrasive work controls.
75.	**Integrated Impact 812.**	Added a Review Procedure regarding the review of BWRs for IGSSC in accordance with Generic Letter 88-01 and NUREG-0313.
76.	**Integrated Impact 811.**	Added Review Procedure to address review of the reactor vessel materials for predicted end-of-life conditions.
77.	**Integrated Impact 819.**	Added Review Procedure to incorporate NUREG-0744 related to acceptable methods of meeting the requirements of 10 CFR 50, Appendix G, paragraph V.C.3.
78.	Editorial	A hyphen was added to ASTM E-185 for consistency with the initial and Reference section citations of the standard.
79.	**Integrated Impact 813.**	Revised Review Procedure III.6.a to address 60-year design life review regarding the material surveillance program for evolutionary plants.

SRP Draft Section 5.3.1

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
80.	10 CFR 52 Applicability	Added a paragraph regarding the review of material surveillance programs performed in accordance with the licensing requirements of 10 CFR 52.
81.	Editorial	Revise paragraph numbering to accommodate the new paragraph regarding design certification reviews.
82.	SRP integration format item	Added boiler-plate statement regarding reviews performed in accordance with 10 CFR 52.
83.	Editorial	Changed "his" to "the" to make the text gender neutral.
84.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
85.	**Integrated Impact 819.**	Revised the Evaluation Findings to include reference to 10 CFR 50.60 which was added as Acceptance Criteria related to implementation of 10 CFR 50, Appendices G and H.
86.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
87.	**Integrated Impact 812.**	Added an Evaluation Finding to incorporate Generic Letter 88-01 and NUREG-0313 regarding IGSSC of reactor vessel weldments associated with vessel attachments and appurtenances.
88.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
89.	**Integrated Impact 812.**	Added an Evaluation Finding to incorporate Generic Letter 88-01 and NUREG-0313 regarding IGSSC of reactor vessel weldments associated with vessel attachments and appurtenances.
90.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
91.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
92.	**Integrated Impact 974.**	Revised text to accommodate alternatives to conformance with Regulatory Guide 1.50 as described in Acceptance Criteria II.4.b.
93.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
94.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
95.	Editorial	Added a hyphen between "under" and "clad" to be specific with wording in II.4.c which uses "underclad."

SRP Draft Section 5.3.1
Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
96.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
97.	Editorial	Revised the lead-in sentence to accommodate the addition of Generic Letter 88-01 and NUREG-0313 pursuant to ROC #815.
98.	**Integrated Impact 815.**	Revised the Evaluation Findings to address Generic Letter 88-01 and NUREG-0313 regarding ferrite limits in welds for austenitic stainless steel.
99.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
100.	Editorial	Revised the lead-in sentence to accommodate the addition of Generic Letter 88-01 and NUREG-0313 pursuant to ROC #816.
101.	**Integrated Impact 973.**	Revised text to accommodate alternatives to conformance with Regulatory Guide 1.44 as described in Acceptance Criteria II.4.e.
102.	**Integrated Impact 816.**	Revised the Evaluation Findings to address Generic Letter 88-01 and NUREG-0313 regarding sensitization of austenitic stainless steels.
103.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
104.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix G, to be consistent SRP-UDP guidance and other changes to the SRP section.
105.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix G, to be consistent SRP-UDP guidance and other changes to the SRP section.
106.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix G, to be consistent SRP-UDP guidance and other changes to the SRP section.
107.	**Integrated Impact 811, 813.**	Incorporated the material surveillance program requirements from Appendix G and H into the evaluation findings and added that compliance with the subject requirements provides the necessary margins of safety for the service life of the reactor vessel.
108.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix G, to be consistent SRP-UDP guidance and other changes to the SRP section.
109.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.

SRP Draft Section 5.3.1

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
110.	**Integrated Impact 819.**	Added 10 CFR 50.60 to the Evaluation Findings related to fracture toughness and compliance with 10 CFR 50 Appendix G.
111.	Editorial	A hyphen was added to ASTM E-185 for consistency with the initial and Reference section citations of the standard.
112.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix H, to be consistent SRP-UDP guidance and other changes to the SRP section.
113.	**Integrated Impact 819.**	Added 10 CFR 50.60 to the Evaluation Findings related to fracture toughness and compliance with 10 CFR 50 Appendix H.
114.	SRP-UDP format item, Reformat References	Citations of 10 CFR 50.55a were revised to be consistent throughout the SRP Section.
115.	SRP-UDP format item, Reformat References	Revised references to 10 CFR 50, Appendix G, to be consistent SRP-UDP guidance and other changes to the SRP section.
116.	SRP-UDP format item,10 CFR 52 Applicability	Added standard paragraph regarding the evaluation findings related to design certification per the requirements of 10 CFR 52.
117.	SRP-UDP Format Item	Added boiler-plate statement indicating the applicability of the SRP to 10 CFR 52 license applications.
118.	SRP-UDP Format Item	Added boiler-plate statement describing the applicability of the SRP to existing and new applications.
119.	Editorial	Added "regulations and" to accommodate the citation of regulatory requirements in SRP Section 5.3.1, including the addition of 10 CFR 50.60 and 50.61.
120.	**Integrated Impact 812.**	Added text to the last paragraph in the Implementation subsection to address implementation of staff positions and resolution of issues associated with Generic Letter 88-01.
121.	SRP-UDP format item, Reformat References	The reference citation of 10 CFR 50.55a was revised to be consistent with the other citations of 10 CFR 50 that have been added to the section.
122.	**Integrated Impact 819.**	Revised the references to include 10 CFR 50.60 which was incorporated into Acceptance Criteria II.5.
123.	**Integrated Impact 820.**	Added a reference to 10 CFR 50.61 which was incorporated into the Review Interface subsection of SRP Section 5.3.1.

Item	Source	Description
124.	SRP-UDP format item, Reformat References	The reference was renumbered to accommodate the addition of new references. An editorial change was made to separate and individually number the references to General Design Criterion 1, 4, 14, 30, 31, and 32.
125.	SRP-UDP format item, Verification of References, PI 21742	The title of GDC 4 is revised to reflect the current amendment which replaced "Missile" with "Dynamic Effects."
126.	Editorial	Renumbered the references to accommodate the addition of individual references for the GDCs, which were previously included under a single reference.
127.	SRP-UDP format item, Reference Verification	Updated the title for Appendix B to 10 CFR 50.
128.	**Integrated Impact 814.**	This is a placeholder integrated impact. Regulatory Guide 1.37 cites ANSI N45.2.1 with regard to cleanliness and cleanliness controls. In the CE FSER, the staff indicated that ANSI/ASME NQA-2 supersedes ANSI N45.2.1. The FSER further states that the staff reviewed NQA-2-1983 and found it acceptable. The staff is in the process of reviewing the NQA standards and developing regulatory guidance for implementation.
129.	**Integrated Impact 811.**	Added reference to Regulatory Guide 1.99 which was incorporated into Acceptance Criteria II.5.
130.	**Integrated Impact 812, 815, 816.**	Added reference to NUREG-0313, which was incorporated in the Acceptance Criteria and Review Procedures sections of the SRP.
131.	**Integrated Impact 819.**	Added reference to NUREG-0744 which was incorporated into the Review Procedures.
132.	**Integrated Impact 812, 815, 816.**	Added reference to Generic Letter 88-01, which was incorporated in the Acceptance Criteria and Review Procedures sections of the SRP.
133.	Editorial	Moved and renumbered the reference for the ASME standard to be consistent with SRP-UDP guidance on the ordering of references.
134.	SRP-UDP format item, Verification of References, Editorial	Added titles associated with the referenced sections of the ASME Boiler and Pressure Vessel Code.
135.	**Integrated Impact 973.**	Added reference to ASTM A-262 which was incorporated into Acceptance Criteria II.4.
136.	**Integrated Impact 975,** SRP-UDP standards citation update	Consideration should be given to updating the citation of ASTM A262-1970 pending the review and approval of the associated standard comparison.

Item	Source	Description
137.	**Integrated Impact 1413.**	Added the applicable date to the reference for standard ASTM A708.
138.	**Integrated Impact 973.**	Added reference to ASTM A-708 which was incorporated into Acceptance Criteria II.4.
139.	Editorial	Moved and renumbered the reference for ASTM standard E-185 to be consistent with SRP-UDP guidance on the ordering of references and to accommodate the addition of other references.
140.	**Integrated Impact 1412**	Revised the reference citation of ASTM E185 to cite the applicable version of the standard.
141.	**Integrated Impact 825.**	This is a placeholder integrated impact for ASTM E-185. The cited version of this standard is not current. The latest version of this standard is ASTM E-185 1994. There are currently no plans to perform a comparison for this standard.
142.	**Integrated Impact 974.**	Added reference to WCAP-8678 which was incorporated into Acceptance Criteria II.4

Integrated Impact No.	Issue	SRP Subsections Affected
811	Revise Acceptance Criteria and Review Procedures to reflect requirements of 10 CFR 50, Appendix G, and to use Regulatory Guide 1.99 as guidance relative to the effects of irradiation on reactor vessel material fracture toughness.	II, III, IV, and VI
812	Revise Review Procedures for review of the adequacy of austenitic stainless materials in BWRs to resist intergranular stress corrosion cracking (IGSCC).	II, III, IV, and VI
813	Provide Acceptance Criteria and Review Procedures for the reactor vessel material surveillance program beyond a 40 year design lifetime.	II, III, and IV
814	Revise Review Procedures for the reactor vessel materials cleanliness and cleanliness controls.	Placeholder II. No changes were made to the SRP.
815	Revise Review Procedures to address staff positions that are more restrictive than Regulatory Guide 1.31.	II, and IV
816	Revise Review Procedures for austenitic stainless steel to address staff positions more restrictive than Regulatory Guide 1.44.	II and IV
817	Revise Review Procedures to include a review of controls provided for reactor vessel materials to limit impurity levels.	TYPE II change. No changes were made to the existing SRP to address this issue.
818	Modify Acceptance Criteria and Review Procedures to reflect staff positions related to radiation embrittlement of reactor vessel materials.	Redundant to II No. 811 and therefore was not processed further.
819	Revise Acceptance Criteria and Review Procedures to incorporate 10 CFR 50.60 and NUREG-0744.	II, III, IV, and VI
820	Develop Acceptance Criteria, Review Procedures and Evaluation Findings for determining PWR reactor vessel acceptability under pressurized thermal shock (PTS) conditions.	I, and VI
822	Perform detailed comparisons between the cited versions of AWS A4.2 and AWS A5.4 to the current versions to allow SRP reviewers to use the more current version of the standards.	Placeholder II. No changes were made to the SRP.
823	Revise Review Procedures to address staff positions on grinding that are more restrictive than Regulatory Guide 1.37.	I, II, and III
824	Revise RGs 1.44 and 1.37 as future work items. The future work recommendations will be tracked with IPD 7.0 Forms number 4.5.1-1 (RG 1.44) and 4.5.1-4 (RG 1.37).	No changes to the SRP are associated with this Integrated Impact.

825	Revise the reference to ASTM-185 to specify the current 1982 version pending completion and review of the side-by-side comparison.	Placeholder II. No changes were made to the SRP.
973	Revise Acceptance Criteria and Review Procedures to allow for the use of alternatives to recommendations in Regulatory Guide 1.44 for verifying the non-sensitization of austenitic stainless steels.	II, IV, and VI
974	Revise the Acceptance Criteria and Review Procedures to incorporate accepted alternatives to Regulatory Guide 1.50, Position C.2	II, IV, and VI
975	Update references to Codes and Standards	Placeholder II. No changes were made to the SRP.
1159	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1025.	Placeholder II. No changes were made to the SRP.
1162	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1027.	Placeholder II. No changes were made to the SRP.
1205	Revise the Acceptance Criteria, Review Procedures and Evaluation Findings to incorporate the requirements from proposed rulemaking 59 FR 50513.	Placeholder II. No changes were made to the SRP.
1289	Revise the Acceptance Criteria, Review Procedures, and Evaluation Findings as necessary to incorporate the guidance of the proposed draft Regulatory Guide DG-1023.	Placeholder II. No changes were made to the SRP.
1319	Revise Regulatory Guide 1.70 to be consistent with SRP Section 5.3.1. This action is being tracked as future work by IPD 7.0 form 5.3.1-3.	Placeholder II. No changes were made to the SRP.
1412	Consider updating the citation of ASTM E185 to cite the 1982 version.	VI
1413	Consider updating the citation of ASTM A708 to cite the 1974 version.	VI