U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of materials issues related to reactor coolant pressure boundary

Secondary - NONE

INTRODUCTION

The mPower^{TM} design incorporates the reactor core, steam generator and the pressurizer inside the reactor vessel. This limits the size and length of piping connected to the reactor vessel and the number of mechanical components outside the reactor vessel. Mechanical connections to an mPower^{TM} reactor vessel include, but are not limited to, the steam system, the feedwater system, safety relief valves the emergency core cooling system, and the reactor coolant inventory and purification systems.

I. AREAS OF REVIEW

The following areas relating to materials of the reactor coolant pressure boundary (RCPB) other than the reactor pressure vessel - which is covered in Design Specific Review Standard (DSRS) Section 5.3.1, "Reactor Vessel Materials," - are reviewed:

1. Material Specifications. The specifications for pressure-retaining ferritic materials, nonferrous metals and austenitic stainless steels, including weld materials, that are used for each component (e.g., vessels, piping, pumps, and valves) of the reactor coolant pressure boundary, are reviewed by the primary reviewer with support from the secondary reviewers.

   The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for the above applications are determined.

2. Compatibility of Materials with the Reactor Coolant. General corrosion and stress corrosion cracking induced by impurities in the reactor coolant can cause failures of the reactor coolant pressure boundary.

   The chemistry of the reactor coolant and the additives (such as inhibitors) whose function is to control corrosion are reviewed by the organization responsible for chemical engineering issues as part of its primary review responsibility for DSRS Section 9.3.

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The compatibility of the materials of construction employed in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed are reviewed. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant is reviewed.

The possible uses of austenitic stainless steels in the sensitized condition and nickel-chromium-iron alloys are reviewed for potential degradation due to stress corrosion cracking.

3. Fabrication and Processing of Ferritic Materials

A. The fracture toughness properties of ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary are reviewed.

The fracture toughness tests performed on all ferritic materials used for pressure-retaining RCPB components (i.e., vessels, pumps, valves, and piping) are reviewed.

The test procedures used for Charpy V-notch impact and dropweight testing are reviewed.

Fracture toughness of the material is characterized by its reference temperature, RT_{NDT}. This temperature is the higher of the nil ductility temperature (NDT) from the dropweight test or the temperature that is 33°C (60°F) below the temperature at which Charpy V-notch impact test data are 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion.

B. The control of welding in ferritic steels is reviewed.

i. The quality of welds in low alloy steels can be increased significantly by proper controls. In particular, the propensity for cold cracks or reheat cracks to form in areas under the bead and in heat-affected zones (HAZ) can be minimized by maintaining proper preheat temperatures of the base metal concurrent with controls on other welding variables. The minimum preheat temperature and the maximum interpass temperatures are reviewed.

ii. The quality of electroslag welds in low alloy steel components can be increased by maintaining a weld solidification pattern that possesses a strong intergranular bond in the center of the weld. The welding variables, which have a significant effect on the weld solidification pattern, must be controlled. The welding variables, solidification patterns, macro etch tests, and Charpy V-notch impact tests of electroslag welds are reviewed. It should be noted that electroslag welds characteristically exhibit a low degree of fusion between the base metal and such welds. Electroslag welds, where used in the RCPB, are reviewed with respect to regulatory guidance describing acceptable controls for the electroslag weld process.

iii. Experience shows that a welder qualified to weld low-alloy steel or carbon steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted. Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation. The adequacy of accessibility during the welding of ferritic components is reviewed.

iv. Controls can be exercised to limit the occurrence of underclad cracking in low-alloy steel components clad with stainless steel. Welding processes that generate excessive heating and promote base metal coarsening cause underclad cracking of certain steels. These variables are reviewed.

C. The requirements for nondestructive examination of ferritic wrought seamless tubular products used for ASME Class 1 components of nuclear power plants are specified in Paragraphs NB-2550 through NB-2570, ASME Boiler and Pressure Vessel Code (hereafter ”the Code“), Section III, ”Rules for Construction of Nuclear Facility Components.” The methods of examination specified for nondestructive examination are reviewed.

4. <u>Fabrication and Processing of Austenitic Stainless Steel</u>. Austenitic stainless steels in a variety of product forms (including several stabilized product forms) are used for construction of pressure-retaining components in the reactor coolant pressure boundary. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, are more frequently used. Because these compositions are susceptible to stress corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress corrosion cracking.

A. Sensitization is caused by intergranular precipitation of chromium carbide in austenitic stainless steels that are exposed to temperatures in the approximate range of 430°C to 820°C (800°F to 1500°F). Precipitation of the chromium carbide at the grain boundaries increases with increasing carbon content and exposure time. Control of the application and processing of stainless steel is needed to eliminate the occurrences of stress corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data and service experience demonstrate that sensitized stainless steel is significantly more susceptible to stress corrosion cracking than nonsensitized (solution heat treated) stainless steel.

The following areas are reviewed: requirements for solution heat treatment of stainless steel; plans to avoid partial or severe sensitization during welding, including information on welding methods, heat input, and interpass temperatures; and a description of the material inspection program that will be used to verify that unstabilized austenitic stainless steels are not susceptible in service to intergranular attack.

B. Contamination of austenitic stainless steel with halogens and halogen-bearing compounds (e.g., die lubricants, marking compounds, and masking tape) must be avoided to the maximum degree possible to avoid stress corrosion cracking. Plans for cleaning and protecting the material against contaminants capable of causing stress corrosion cracking during fabrication, shipment, storage, construction, testing, and operation of components and systems are reviewed. Controls for abrasive work (e.g., grinding) on austenitic stainless steel surfaces are also reviewed with respect to potential for material contamination and excessive surface cold-working. Any pickling used in processing austenitic stainless steel components and the restrictions placed on pickling sensitized materials are reviewed. The upper limit on the yield strength of austenitic stainless steel materials is reviewed.

C. Whether sensitized or not, austenitic stainless steel is subject to stress corrosion cracking and must be protected from contaminants that can promote cracking. Thermal insulation is often employed adjacent to, or in direct contact with, stainless steel piping and components. The contaminants present in the thermal insulation may be leached by spilled or leaking liquids and deposited on the stainless steel surfaces. The controls on the use of nonmetallic thermal insulation are reviewed.

D. Austenitic stainless steel is subject to hot cracking (microfissuring) during welding if the weld metal composition or the welding procedure is not properly controlled. Because cracks formed in this manner are small and difficult to detect by nondestructive testing methods, welding procedures, weld metal compositions, and delta ferrite percentages that minimize the possibility of hot cracking must be specified. The adequacy of the proposed welding procedures, weld metal compositions, testing of weld metals, and delta ferrite content is reviewed.

The assurance of satisfactory electroslag welds for austenitic stainless steel components can be increased by maintaining a weld solidification pattern with a strong intergranular bond in the center of the weld. The welding variables that have a significant effect on the weld solidification pattern must be controlled.

A number of electroslag welding process variables, such as slag pool depth, electrode feed rate and oscillation, current, voltage, and slag conductivity, have been shown to influence the weld solidification pattern. If the combination of process variables produces a deep pool of molten weld metal, the crystal (dendritic) growth direction from the pool sides will join at an obtuse angle at the center of the weld, and cracks may develop because of the weaker centerline bond between dendrites. A proper combination of process variables promotes a dendritic growth pattern with an acute joining angle, which results in a strong centerline bond. The welding variables, solidification patterns, and macro etch tests used in the electroslag welding of austenitic stainless steel are reviewed.

Experience has shown that a welder qualified to weld stainless steel components under normal fabricating conditions may not produce acceptable welds if the accessibility to the weld area is restricted. Limited accessibility can occur when component parts are joined in the final assembly or at the plant site, where other adjacent components or structures prevent the welder from assuming an advantageous position during the welding operation. The adequacy of

accessibility of field erected structures, for welding austenitic stainless steel components, is reviewed.

E. The requirements for nondestructive examination of wrought seamless tubular products used for components of nuclear power plants are specified in Paragraphs NB-2550 through NB-2570 of the Code, Section III. Nondestructive examination techniques applied to tubular products used for components of the RCPB, or other safety-related ASME Class 1 systems that are designed for pressure in excess of 1.896 MPa (275 psig) or temperatures in excess of 93°C (200°F), must be capable of detecting unacceptable defects regardless of defect shape, orientation, or location in the product.

   The nondestructive examination procedures used for inspection of tubular products are reviewed.

F. Where cast austenitic stainless steel components are proposed for use in the RCPB, the adequacy of material fracture toughness properties and welding controls to resist thermal aging effects over the design life are reviewed. Since welds on such materials are difficult to inspect using ultrasonic techniques, the inspectability is also reviewed.

5. <u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)</u>. For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

6. <u>COL Action Items and Certification Requirements and Restrictions</u>. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

   For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

7. <u>Operational Program Description and Implementation</u>. For a COL application, the staff reviews the Inservice Inspection and Inservice Testing Programs description and the proposed implementation milestones. The staff also reviews final safety analysis report (FSAR) Table 13.x to ensure that the Inservice Inspection and Inservice Testing Programs and associated milestones are included.

<u>Review Interfaces</u>:

The primary reviewer is responsible for ensuring the coordination and interface with secondary reviewers, as necessary.

Other DSRS sections interface with this section as follows:

1. The programs for assuring the integrity of bolting and threaded fasteners are reviewed under DSRS Section 3.13.

2. The reactor coolant chemistry and associated chemistry controls (including additives such as inhibitors) as it relates to corrosion control and compatibility with RCPB materials, is reviewed under DSRS Section 3.6.

3. The design for structural integrity of components and their supports including the adequacy of design fatigue curves for RCPB materials with respect to cumulative reactor service-related environmental and usage factor effects, is reviewed under SRP Section 3.9.3.

4. The inservice inspection requirements specified for the RCPB and the proposed inspection and examination techniques to provide early detection and adequate evaluation of defects in materials and weldments used in the RCPB are reviewed under DSRS Section 5.2.4.

5. The quality assurance program is reviewed under SRP Chapter 17.

6. For COL reviews of operational programs, the review of the applicant's implementation plan is performed under SRP Section 13.4, "Operational Programs."

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. 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;

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

3. GDC 14 and 31, as they relate to minimizing the probability of rapidly propagating fracture and gross rupture of the RCPB;

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

5. Appendix G to Part 50, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB;

6. Section 50.55a, as it relates to quality standards applicable to the RCPB;

7. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

8. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in this DSRS section. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information."

1. Material Specifications. The requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of materials Code Cases as described in Regulatory Guide (RG) 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

   The materials specified for use in RCPB components must conform to Section III of the Code and to Parts A, B, C and D of Section II of the Code. RG 1.84 describes acceptable materials Code Cases and guidelines for their application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications.

2. Compatibility of Materials with the Reactor Coolant. The requirements of GDC 4 relative to compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME Code and by compliance with the positions of RG 1.44, "Control of the Use of Sensitized Stainless Steel."

   Ferritic low alloy steels and carbon steels, which are used in many principal pressure-retaining components, are clad with a layer of austenitic stainless steel. If cladding is not used, conservative corrosion allowances must be indicated for all exposed surfaces of carbon and low alloy steels, as indicated in the ASME Code, Section III, NB-3121, "Corrosion."

   RG 1.44 contains staff positions related to unstabilized austenitic stainless steel of the AISI Type 3XX series used for components of the RCPB.

3. Fabrication and Processing of Ferritic Materials

A. The acceptance criteria for fracture toughness are the requirements of Appendix G, "Fracture Toughness Requirements," of 10 CFR Part 50. These criteria satisfy the requirements of GDC 14 and GDC 31 regarding prevention of fracture of the RCPB.

Appendix G requires that the pressure-retaining components of the RCPB that are made of ferritic materials shall meet the requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. With respect to absorbed energy in J (ft-lbs) and lateral expansion as shown by Charpy V-notch (C_v) impact tests, all materials shall meet the acceptance standards of Article NB-2300 of the Code, Section III, and the requirements of Section IV of Appendix G, 10 CFR Part 50, as follows:

i. Materials for piping (i.e., pipes, tubes, and fittings), pumps, and valves, excluding bolting materials, shall meet the requirements of the Code, Section III, Paragraph NB-2331 or NB-2332 (as applicable based upon thickness), and Appendix G, Paragraph G-3100 to the Code, Section III. The required C_v values for piping, pumps, and valves are specified in Table NB-2332(a)-1 of the Code, Section III.

ii. Materials for bolting for which impact tests are required shall meet the requirements of the Code, Section III, Paragraph NB-2333.

iii. Calibration of instruments and equipment shall meet the requirements of the Code, Section III, Paragraph NB-2360.

The special acceptance requirements and staff positions for fracture toughness of reactor vessels are covered by DSRS Section 5.3.1.

B. The acceptance criteria for control of ferritic steel welding are based upon the following regulatory guides and ASME Code provisions to satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a:

i. The amount of specified preheat must be in accordance with the requirements of the Code, Section III, Appendix D, Paragraph D-1210. These requirements are supplemented by positions described in RG 1.50, "Control of Preheat Temperature for Welding of Low Alloy Steel."

The supplemental acceptance criteria for control of preheat temperature are as follows:

- According to the welding procedure qualification minimum preheat and maximum interpass temperatures should be specified and the welding procedure should be qualified at the minimum preheat temperature. For production welds, the preheat temperature should be maintained until a final post weld heat treatment or a

post weld hydrogen bakeout is performed prior to the performance of the final post weld heat treatment.

- Production welding should be monitored to verify that the limits on preheat and interpass temperatures are maintained. In the event that the above criteria are not met, the weld is subject to rejection.

ii. The acceptance criteria for electroslag welds are presented in RG 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns and impact test limits (for qualification of welds in Class 1 and Class 2 components) and the criteria for verifying conformance during production welding.

iii. RG 1.71, "Welder Qualification for Areas of Limited Accessibility," provides guidance for the qualification of welders and welding operators performing welds with limited accessibility. Performance qualification should provide testing the welder or welding operator under simulated access, and visibility limitations when physical conditions restrict the welder's access to a production weld to less than 30 cm (12 inches) in any direction from the joint and which would affect electrode manipulation, or bead progression, or require indirect means of weld pool observation (such as a morror). Requalification should be necessary when (a) the use of an indirect means is required to view the weld pool (such as a mirror) during qualification welding and the welder or welding operator did not qualify for the welding in areas of limited accessibility using that indirect means of weld pool observation, or (b) any of the essential welding variables for welders or welding operators as listed in ASME Code Section IX "Welding and Brazing Qualifications" change, or (c) the qualification expires per ASME Code Section IX.

Qualification of the welder or welding operators for limited accessibility may be waived provided that 100% radiographic and/or ultrasonic examination of the completed welded joint is performed. Examination procedures and acceptance standards should meet the requirements of the ASME Section III of the Code. Records of the examination reports and radiographs should be retained and made part of the Quality Assurance Documentation for the completed weld.

iv. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," provides criteria to limit the occurrence of underclad cracking in low-alloy steel safety-related components clad with stainless steel. According to these criteria, material known to have susceptibility to underclad cracking should not be weld clad by high-heat-input welding processes and should be qualified for use to demonstrate that underclad cracking is not induced.

C. For nondestructive examination of ferritic steel tubular products, the requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards are met by compliance with the applicable provisions of the ASME Code.
The acceptance criteria are given in Section III of the Code, Paragraphs NB-2550 through NB-2570.

4. Fabrication and Processing of Austenitic Stainless Steel

A. The requirements of GDC 4 relative to compatibility of components with environmental conditions are met with measures to avoid sensitization in austenitic stainless steels. The acceptance criteria for testing, alloy compositions, and heat treatment, to avoid sensitization in austenitic stainless steels, are covered in RG 1.44.

RG 1.44 also identifies acceptable methods for verification of non-sensitization of austenitic stainless steel materials and qualification of welding processes employed in production including testing using ASTM A-262 Practice A or E or another method which can be demonstrated to show non-sensitization.

B. The requirements of GDC 4 relative to compatibility of components with environmental conditions are met with additional controls to avoid stress corrosion cracking in austenitic stainless steels. These controls consist of acceptance criteria on prevention of contamination, cleaning, and upper limit on yield strength.

Controls to avoid stress corrosion cracking in austenitic stainless steels are also covered in RG 1.44. This regulatory guide provides acceptance criteria on the cleaning and protection of the material against contaminants capable of causing stress corrosion cracking. Acid pickling is to be avoided on fabricated stainless steels. Necessary pickling is to be done only with appropriate controls. Pickling should not be performed upon sensitized stainless steels.

The quality of water used for final cleaning or flushing of finished surfaces during installation should be in accordance with RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." Vented tanks with deionized or demineralized water are an acceptable source of water for final cleaning or flushing of finished surfaces. The oxygen content of the water need not be controlled.

The controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in ASME Nuclear Quality Assurance (NQA) standard NQA-1-1994, "Quality Assurance Requirements for Nuclear Facilities Applications," ASME NQA-1-1994, which is referenced in RG 1.37, 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.

Laboratory stress corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 620 MPa (90,000 psi).

C. The acceptance criteria for compatibility of austenitic stainless steel with thermal insulation are based on RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," to satisfy GDC 14 and 31 relative to prevention of failure of the

RCPB. The compatibility of austenitic stainless steel materials with thermal insulation is dependent upon the type of insulation. The thermal insulation is acceptable if either reflective metal insulation is employed or a nonmetallic insulation which meets the criteria of RG 1.36 is used. The acceptance criteria for nonmetallic insulation for stainless steel are based on the levels of leachable contaminants in the material and are presented in position C.2.b and Figure 1 of RG 1.36.

D. The acceptance criteria for control of welding of austenitic stainless steels are based on RGs 1.31, 1.34, 1.44 and 1.71, to satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

The acceptance criteria for delta ferrite in austenitic stainless steel welds are given in RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." These acceptance criteria address (1) verification of delta ferrite content of filler metals, (2) ferrite measurement, (3) instrumentation, (4) acceptability of test results, and (5) documentation of weld pad verification tests.

The acceptance criteria for electroslag welds in austenitic stainless steel are given in RG 1.34, "Control of Electroslag Weld Properties." These criteria specify acceptable solidification patterns for qualification of austenitic stainless steel welds and the basis for verifying conformance during production welding.

The acceptance criteria for welder qualification for areas of limted accessibility are given in RG 1.71. RG 1.71 provides guidance for the qualification of welders and welding operators performing welds with limited accessibility. Performance qualification should provide for testing the welder or welding operator under simulated access, and visibility limitations when physical conditions restrict the welder's access to a production weld to less than 30 cm (12 inches) in any direction from the joint and which would affect electrode manipulation, or bead progression, or require indirect means of weld pool observation (such as a morror). Requalification should be necessary when (a) the use of an indirect means is required to view the weld pool (such as a mirror) during qualification welding and the welder or welding operator did not qualify for the welding in areas of limited accessibility using that indirect means of weld pool observation, or (b) any of the essential welding variables for welders or welding operators as listed in ASME Code Section IX "Welding and Brazing Qualifications" change, or (c) the qualification expires per ASME Code Section IX.

E. For nondestructive examination of austenitic stainless steel tubular products, the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a are met by compliance with the applicable provisions of the ASME Code. The acceptance criteria are given in Section III of the Code, Paragraphs NB-2550 through NB-2570.

F. Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestones for the Inservice Inspection and Inservice Testing Programs are reviewed in accordance with 10 CFR 50.55a(g), 10 CFR 50.55a(f) and 10 CFR 50, Appendix A. The implementation milestones in the Inservice Inspection and Inservice Testing Programs are identified under DSRS Sections 5.2.4 and 3.9.6, respectively.

5. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

Technical Rationale

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

1. GDC 1 and 10 CFR 50.55a require that structures, systems, and components (SSCs) be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. 10 CFR 50.55a also incorporates by reference applicable editions and addenda of the ASME Boiler and Pressure Vessel Code. GDC 30 requires that components which are part of the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. The reactor coolant pressure boundary provides the following safety functions: a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of 10 CFR 50.55a, GDC 1, and GDC 30 to the RCPB materials provides assurance that established standard practices of proven or demonstrated effectiveness are used to achieve a high likelihood that these safety functions will be performed.

2. GDC 4 requires 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 RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of GDC 4 to the RCPB materials provides assurance that degradation and/or failure of the RCPB resulting from environmental service conditions that could cause substantial reduction in capability to contain reactor coolant inventoryor to confine fission products, or cause 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 RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of GDC 14 to the RCPB materials assures that they are selected, fabricated, installed, and tested to provide a low probability of significant degradation (which in the case of extreme degradation could cause gross failure of the RCPB resulting in substantial reduction in capability to contain reactor coolant inventory), and reduction in capability to confine fission products, or interference with core cooling.

4. GDC 31 requires 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. 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 RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of GDC 31 to the RCPB 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. The probability of substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, and interference with core cooling is thereby minimized.

5. Appendix G of 10 CFR Part 50 requires that the fracture toughness of RCPB ferritic materials be tested in accordance with the requirements of the ASME Code and that the pressure-retaining components of the RCPB that are made of ferritic materials meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. 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 discussed in Items 3 and 4 above.

6. Appendix B of 10 CFR Part 50 requires, in Criterion XIII, that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. The RCPB provides a fission product barrier, a confined volume for the inventory of reactor coolant, and flow paths to facilitate core cooling. Application of cleaning requirements to the RCPB 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 of degradation and/or failure of the RCPB that could cause substantial reduction in capability to contain reactor coolant inventory or to confine fission products, or cause interference with core cooling.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

In accordance with 10 CFR 52.47(a)(8),(21), and (22), and 10 CFR 52.79(a)(17) and (20), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report (SER) section.

1. <u>Material Specifications</u>. The material specifications for each major pressure-retaining component or part used in the RCPB are compared with the acceptable specifications listed in the Code, Sections II, Part D and/or acceptable material Code Cases as identified in RG 1.84. Exceptions to the material specifications of the Code are clearly identified, and the basis evaluated. The reviewer judges the significance of the exceptions and, taking into account precedents set in earlier cases, determines the acceptability of the proposed exceptions. In those instances where the primary reviewer takes 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 for what reason.

   Operating experience has indicated that certain nickel-chromium-iron alloys (e.g. Alloy 600 and associated weld materials, Alloy 82 and 182) are susceptible to stress corrosion cracking as documented in NUREG-1823 and NRC Generic Letter 97-01. The NRC Order EA-03-009 was issued to provide interim inspection requirements for reactor pressure vessel heads (including penetration nozzles) at pressurized water reactors. Alloy 690 and associated weld materials Alloy 52 and 152 have improved stress-corrosion-cracking resistance in comparison to Alloy 600 used in PWR RCPB applications. Where nickel-chromium-iron alloys are proposed for use in the PWR RCPB, use of Alloy 690 materials is preferred. If use of Alloy 600 materials is proposed, the reviewer verifies that an acceptable technical basis is either identified (based upon demonstrated satisfactory use in similar applications) or presented by the applicant to support use of the material under the expected environmental conditions (e.g., exposure to the reactor coolant). In addition the reviewer verifies that acceptable augmented inspection requirements have been proposed based on operating experience and service conditions. For all RCPB environments, particular review emphasis is placed upon the corrosion resistance and stress corrosion cracking resistance properties of the proposed nickel-chromium-iron alloy(s).

   Where cast austenitic stainless steels are proposed for use in the RCPB, the reviewer verifies that the material specifications ensure adequate fracture toughness over the design life to support use of the material under the expected environmental conditions (e.g., exposure to the reactor coolant operating temperatures).

2. <u>Compatibility of Materials with the Reactor Coolant</u>. The reviewer verifies that the following information is provided at each respective stage of the review process:

   A. At the construction permit stage of review:

      i. A list of the materials of construction of the components of the reactor coolant pressure boundary that are exposed to the reactor coolant, including a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed in service.

      ii. A list of the materials of construction of the RCPB, and a description of material compatibility with external insulation and with the environment in the event of reactor coolant leakage.

      iii. The fabrication and cleaning controls imposed on stainless steel components to minimize contamination with chloride and fluoride ions.

B. At the operating license stage of the review process:

   i. The items listed under Subsection III.2.A above are reviewed to provide assurance that any changes are noted that may have occurred during the period between the submittal of the preliminary and final safety analysis reports.

C. For design certification and COL applications under 10 CFR Part 52, the reviewer verifies the information identified for a construction permit in Section 2.A, above.

3. Fabrication and Processing of Ferritic Materials

A. The information submitted by the applicant relative to tests for fracture toughness is reviewed for conformance with the acceptance criteria stated in Subsection II.3.A. These tests include Charpy V-notch impact and dropweight tests. A description of the tests is reviewed, and the locations of the test specimens and their orientation are verified. Information regarding calibration of instruments and equipment is reviewed for conformance with the acceptance criteria stated in Subsection II.3.A.(3) of this DSRS section.

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

   The SAR is reviewed to assure that all the impact tests required by Appendix G to 10 CFR Part 50, as detailed in NB-2300, have been performed.

B. The control of welding in ferritic steels is reviewed as described below:

   i. The information submitted by the applicant regarding the control of preheat temperatures for welding low alloy steel is reviewed for conformance with the acceptance criteria stated in Subsection II.3.B.(1) of this DSRS section.

   ii. The electroslag weld information submitted by the applicant is reviewed for conformance to the acceptance criteria discussed in Subsection II.3.B.(2) of this DSRS section. The information in the SAR is reviewed to verify that macroetch tests have been conducted (to assure that an acceptable weld solidification pattern is obtained) and that impact tests specified in RG 1.34 meet the acceptance criteria discussed previously in Subsection II.3.B.(2) of this DSRS section.

   iii. The ASME Code, Section III, requires adherence to the requirements of Section IX, "Welding and Brazing Qualifications of the Code." One of the requirements is welder qualification for production welds. However, there is a need for supplementing this section of the Code because the assurance of providing satisfactory welds in locations of restricted direct physical and visual accessibility can be increased significantly by

qualifying the welder under conditions simulating the space limitations under which the actual welds will be made.

RG 1.71 provides the necessary supplement to the Code, Section IX, in this respect. The information submitted by the applicant is reviewed for conformance with acceptance criteria discussed in Subsection II.3.B.(3) of this DSRS section.

iv. The information submitted by the applicant regarding controls to limit the occurrence of underclad cracking in low alloy steel components when weld cladding with austenitic stainless steel is reviewed for conformance with acceptance criteria given in Subsection II.3.B.(4) of this DSRS section.

C. The reviewer verifies that acceptable methods specified in the ASME Code, Section III, paragraphs NB-2550 through NB-2570 are proposed by the applicant for examination of ferritic steel tubular products.

4. <u>Fabrication and Processing of Austenitic Stainless Steels</u>

A. The information submitted by the applicant in the following areas is reviewed for conformance with the acceptance criteria stated in Subsection II.4.A of this DSRS section regarding:

i. The desirable stage in the sequence of processing for solution heat treatment, including the rates of cooling and the quenching media.

ii. Controls to prevent sensitization during welding, as described in RG 1.44.

iii. Controls to verify non-sensitization and to qualify welding processes employed in production, as described in Subsection II.4.A of this DSRS section.

In the event that information in the above areas is not supplied, sufficient justification for the deviation must be presented.

B. The information submitted by the applicant is reviewed for conformance with the acceptance criteria discussed in Subsection II.4.B of this DSRS section as follows:

Verification is sought that process controls are exercised during all stages of component manufacture and reactor construction to minimize the exposure of austenitic stainless steels to contaminants that could lead to stress corrosion cracking.

Information is also checked to assure that precautions have been taken to require removal of all cleaning solutions, processing compounds, degreasing agents, and any other foreign material from the surfaces of the component at any stage of processing prior to any elevated temperature treatment and prior to hydrotests. The reviewer verifies that a statement is contained in the SAR that pickling of sensitized austenitic stainless is avoided and that the quality of water used for

final cleaning or flushing of finished surfaces during installation is in accordance with acceptance criteria discussed in Subsection II.4.B of this DSRS section.

The applicant's description of abrasive work controls for austenitic stainless steel surfaces is reviewed and is verified adequate to minimize the introduction of stress corrosion cracking promoting contaminants and the cold-working of surfaces.

Because excessive cold work in austenitic stainless steel can render this material susceptible to stress corrosion cracking, control must be exerted by the applicant, by placing an upper limit on the yield strength, in accordance with the acceptance criteria discussed in Subsection II.4.B of this DSRS section. Verification is obtained that the applicant has such a control measure.

C. The information submitted by the applicant is reviewed to determine the type of insulation used and to determine its compatibility with the austenitic stainless steel used in construction of the component.

There are no compatibility concerns with the use of reflective metal insulation; the chief compatibility concern is with the use of nonmetallic insulation. A review is performed to assure that any such material specified by the applicant is in conformance with the acceptance criteria stated in Subsection II.4.C of this DSRS section. Verification is obtained that the material has been chemically analyzed by methods equivalent to those prescribed in RG 1.36 and that evidence is obtained that the levels of leachable contaminants are such that stress corrosion of stainless steel will not result from use of the insulation.

D. The information submitted by the applicant regarding control of delta ferrite in austenitic stainless steel welds is reviewed to determine its conformance with the acceptance criteria stated in Subsection II.4.D of this DSRS section. The reviewer verifies that appropriate filler metal acceptance tests have been conducted and that a certified materials test report has been received. The reviewer also verifies that the applicant's program is in compliance with the staff positions in RG 1.31 and the more stringent criteria specified in II.4.D where applicable.

The information submitted by the applicant regarding control of electroslag weld properties for austenitic stainless steel materials is reviewed for conformance with the acceptance criteria discussed in Subsection II.4.D of this DSRS section.

The review of information on the control of electroslag weld properties in austenitic stainless steels is essentially the same as that discussed previously for ferritic steels. However, because electroslag-welded austenitic stainless steels have very high impact resistance, the checks are: (1) a macroetch test is used to provide assurance that the solidification pattern is in accordance with the requirement of the acceptance criteria shown in Subsection II.4.D of this DSRS section, and (2) wrought stainless steel parts are solution heat treated after welding.

The review procedure for information submitted on welder qualification for limited accessibility areas, applicable to austenitic stainless steels, is the

same as that for ferritic steels, which has been discussed previously under Subsection III.3.B.(3) of this DSRS section.

E. The procedures for review of nondestructive examination of tubular products fabricated from austenitic stainless steel are the same as those discussed for similar ferritic products in Subsection III.3.C of this DSRS section, and the acceptance criteria are as shown in Subsection II.4.E of this DSRS section.

F. Cast austenitic stainless steel is susceptible to thermal aging at reactor coolant temperatures. The reviewer verifies that the applicant has considered alternative materials to cast stainless steel and has limited use of cast stainless steel in the RCPB to those specific applications where demonstrated to be the best material selection alternative. Where cast material is used, the range of temperatures to which the material will be exposed and the ferrite content of the material receive particular review emphasis. The reviewer verifies that the applicant's proposed material specifications and fabrication controls ensure adequate fracture toughness over the design life of the plant.

Where cast austenitic stainless steel components with welded joints requiring preservice and inservice inspection are proposed, the reviewer confirms the inspectability of the welded joints using ultrasonic techniques.

5. General. If the information contained in the safety analysis reports or the plant Technical Specifications does not comply with the appropriate acceptance criteria, or if the information provided is inadequate to establish such compliance, a request for additional information is prepared and transmitted. Such requests identify not only the necessary additional information but also the changes needed in the SAR or the Technical Specifications. Subsequent amendments received in response to these requests are reviewed for compliance with the applicable acceptance criteria.

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

Implementation of this program will be inspected in accordance with NRC Inspection Manual Chapter IMC-2504, "Construction Inspection Program - Non-ITAAC Inspections."

For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions

> In sum, the materials used for construction of components of the reactor coolant pressure boundary (RCPB) have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above provisions for material specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.
>
> The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which is chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with the positions of RG 1.44, "Control of Sensitized Stainless Steel." The cast austenitic stainless steels and nickel-chromium-iron alloys to be used as RCPB materials have also been demonstrated to be compatible with reactor coolant under the anticipated environmental conditions of RCPB service. General corrosion of all materials, except unclad carbon and low alloy steel, will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of the Code, Section III. Accordingly, all RCPB materials are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, as required by GDC 4.
>
> The materials of construction for the RCPB are compatible with the thermal insulation used in these areas and are in conformance with the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations satisfies the requirements of GDC 14 and GDC 31 that the probability of rapidly propagating failure of RCPB materials be extremely low, and the probability of rapidly propagating fracture of RCPB materials be minimized.
>
> The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with the provisions of the ASME Code, Section III. Compliance with these Code requirements satisfies the quality standards requirements of GDC 1, GDC 30 and 10 CFR 50.55a for these materials.

The fracture toughness tests required by the ASME Code, augmented by Appendix G to 10 CFR Part 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the reactor coolant pressure boundary. The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations satisfies the requirements of GDC 14 and GDC 31 that the probability of rapidly propagating failure of RCPB materials be extremely low, and the probability of rapidly propagating fracture of RCPB materials be minimized.

The controls imposed on welding preheat temperatures for welding ferritic steels are in conformance with the recommendations of RG 1.50, “Control of Preheat Temperature for Welding Low-Alloy Steel.” These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a for these materials under the specified conditions.

The controls imposed on electroslag welding of ferritic steels are in accordance with the recommendations of RG 1.34, “Control of Electroslag Weld Properties,” and provide assurance that welds fabricated by the process will have high integrity and will have a sufficient degree of toughness to furnish adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Conformance with the recommendations of RG 1.34 also satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a for these materials under the specified conditions.

The controls imposed on welding ferritic steels under conditions of limited accessibility are in accordance with the recommendations of RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” and provide assurance that proper requalification of welders will be required in accordance with the welding conditions. These controls also satisfy the quality standards requirements of GDC 1, GDC 50, and 10 CFR 50.55a for welding of ferrite materials under limited accessibility. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are in accordance with the recommendations of RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.” These controls provide assurance that practices that could result in underclad cracking will be restricted. The controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a for weld cladding of low alloy steel components by austenitic stainless steel.

The controls to avoid stress corrosion cracking in reactor coolant pressure boundary components constructed of austenitic stainless steels limit yield strength of cold-worked austenitic stainless steels to 620 MPa (90,000 psi) maximum and conform to the recommendations of RG 1.44, “Control of the Use of Sensitized Stainless Steel” and Regulatory Guide 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.” The controls in accordance with these recommendations are followed during material selection, fabrication,

examination, and protection, in order to prevent excessive yield strength, sensitization, and contamination. These controls provide reasonable assurance that the RCPB components of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. Accordingly, these controls ensure that austenitic stainless steel RCPB components are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, as required by GDC 4. For the same reasons, these controls meet the requirements of GDC 14 that the RCPB have an extremely low probability of abnormal leakage and rapidly propagating failure.

The controls imposed during welding of austenitic stainless steels in the RCPB are in accordance with the recommendations of RG 1.31, ”Control of Ferrite Content in Stainless Steel Weld Metal”; RG 1.34, “Control of Electroslag Weld Properties”; and RG 1.71, “Welder Qualification of Areas of Limited Accessibility.” These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. Accordingly, these controls meet the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a for welding of austenitic stainless steels in the RCPB, and, with respect to such welding, satisfy the requirements of GDC 14 that the RCPB have an extremely low probability of abnormal leakage and rapidly propagating failure.

The fabrication controls for cast austenitic stainless steel components, in conjunction with acceptable base material and weld metal specifications, provide for welded joint inspectability and adequate fracture toughness to resist thermal aging for the design life. Accordingly, these controls therefore satisfy the applicable requirements of GDC 1, GDC 4, GDC 14, GDC 30, and 10 CFR 50.55a for these RCPB materials.

Accordingly, the staff concludes that the RCPB materials are acceptable and meet the requirements of General Design Criteria 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 10 CFR 50.55a of 10 CFR Part 50.

The applicant described the Inservice Inspection and Inservice Testing Programs and its implementation in conformance with 10 CFR 50.55a(g) and 10 CFR 50 Appendix A.

For DC and COL reviews, the findings will also summarize the staff’s evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific design certification (DC), or combined license (COL), applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower™ and large light-water nuclear

reactor power plants, and in accordance with the direction given by the Commission in SRM-COMGBJ-10-0004/COMGEA-10-0001, “Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews,” dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor (SMR) reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), “Contents of applications; technical information.”

This regulation states, in part, that the application must contain “an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application.” The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower^{TM} DCD FSAR does not deviate significantly from the design assumptions made by the NRC staff while preparing this DSRS section. The application must identify and describe all differences between the standard plant design and this DSRS section, and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria. If the design assumptions in the DC application deviate significantly from the DSRS, the staff will use the SRP as specified in 10 CFR 52.47(a)(9). Alternatively, the staff may supplement the DSRS section by adding appropriate criteria in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced RGs.

VI. REFERENCES

1. 10 CFR Part 50, Section 50.55a, “Codes and Standards.”

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

3. 10 CFR Part 50, Appendix A, Criterion 4, “Environmental and Dynamic Effects Design Bases.”

4. 10 CFR Part 50, Appendix A, Criterion 14, “Reactor Coolant Pressure Boundary.”

5. 10 CFR Part 50, Appendix A, Criterion 30, “Quality of Reactor Coolant Pressure Boundary.”

6. 10 CFR Part 50, Appendix A, Criterion 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.”

7. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” Criterion XIII, “Handling, Storage and Shipping.”

8. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

9. Regulatory Guide 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

10. Regulatory Guide 1.34, “Control of Electroslag Weld Properties.”

11. Regulatory Guide 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel.”

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

13. Regulatory Guide 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.”

14. Regulatory Guide 1.44, “Control of the Use of Sensitized Stainless Steel.”

15. Regulatory Guide 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.”

16. Regulatory Guide 1.71, “Welder Qualification for Areas of Limited Accessibility.”

17. Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.”

18. NUREG/CR-4513 Revision 1 (ANL-93/2), “Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems,”.

19. ASME Boiler and Pressure Vessel Code, Section II, “Materials,” Parts A, B, and C; Section III, ”Rules for Construction of Nuclear Facility Components“; and Section IX, “Welding and Brazing Qualifications”; American Society of Mechanical Engineers.

20. ASTM, A-262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels.” American Society for Testing and Materials, West Conshohocken, PA

21. ASME Nuclear Quality Assurance (NQA) standard NQA-1-1994, "Quality Assurance Requirements for Nuclear Facilities Applications," ASME NQA-1-1994

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

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

24. NRC Order EA-03-009, “Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, Washington, DC, February 11, 2003.

25. NRC Order EA-03-009, Revision 1: “Issuance of First Revised NRC Order (EA-03-009) Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, Washington, DC, February 20, 2004.

26. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Non-ITAAC Inspections,” issued April 25, 2006.

27 10 CFR Part 52, Subpart B “Standard Design Certifications”, 10 CFR Section 52.47 “Contents of Applications; Technical Information” and 10 CFR Part 52, Subpart C “Combined Licenses”, Section 52.80 “Contents of Applications; Additional Technical Information.”