U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD for NuScale SMR DESIGN

5.3.1 REACTOR VESSEL MATERIALS

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

Integral pressurized-small modular reactors (SMRs) generally incorporate the reactor core, and the pressurizer inside the reactor vessel. One or more steam generators may be inside the reactor vessel or directly connected to the reactor vessel. For the purpose of this review, the applicant should provide an accurate definition of the reactor vessel.

The specific areas of review are as follows:

1. Material Specifications. The material specifications used for the reactor vessel and applicable attachments and appurtenances, such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings, are reviewed and their adequacy for use in the construction of such components is assessed on the basis of the mechanical and physical properties of the materials, the effects of irradiation on these materials, their corrosion resistance, and their 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter "the Code"), Section III, are reviewed. Attention is directed towards calibration methods, instrumentation, methods of application, sensitivity, reliability, and standards used.

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

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

   The testing 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, nickel, and phosphorus, is checked. The results of impact tests performed on base material, weld metal, and heat-affected zones are reviewed, and the scope of the testing is checked, particularly in the area of the reactor vessel belt-line region, where radiation effects on the material are most significant.

   Fracture toughness of the materials employed is characterized by its reference temperature, RT_{NDT}. This temperature is the higher value of the nil-ductility temperature (NDT) from the drop-weight test, or the temperature that is 33°C (60°F) below the temperature at which Charpy V-notch impact test data meet a specified toughness level in accordance with 10 CFR Part 50, Appendix G. The information submitted is checked to ensure that the RT_{NDT} of the materials is included with the data and test results for impact testing.

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

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 used to ensure initial integrity is reviewed, along with the applicable acceptance criteria.

8. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this Design-Specific Review Standard (DSRS) section in accordance with Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria," and SRP Section 14.3.4, "Reactor Systems - 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 and 14.3.4.

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

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

10. Operational Program Description and Implementation. For a COL application, the staff reviews the Reactor Vessel Material Surveillance Program description and the proposed implementation milestones. The staff also reviews final safety analysis report (FSAR) Table 13.x to ensure that the Reactor Vessel Material Surveillance Program and associated milestones are included.

Review Interfaces

Other SRP and DSRS sections interface with this section as follows:

1. The review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners is performed under SRP Section 3.13.

2. The review of the reactor vessel fracture toughness with regard to pressure-temperature limits, including protection from pressurized thermal shock events in accordance with Title 10 of the *Code of Federal Regulations* (CFR), Section 50.61 is performed under DSRS Section 5.3.2.

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

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

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

6. The review of the quality assurance program is performed under SRP Section 17.5.

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 (GDCs) 1 and 30 found in Appendix A to Part 50, as they relate to quality standards for design, fabrication, erection, and testing of SSCs;

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

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

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

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

6. 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness;

7. 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G and Appendix H;

8. 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control;

9. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness,

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

11. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations;

12. 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 COL, the provisions of the AEA, 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 set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. As an alternative, and as described in more detail below, an applicant may identify the differences between a DSRS section and the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and discuss how the proposed alternative provides an acceptable method of complying with the NRC regulations that underlie the DSRS acceptance criteria.

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

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

   B. The acceptability of materials not specified in the Code are considered on an individual basis. Their suitability is evaluated on the basis of data submitted in accordance with the requirements of Code Section III, Appendix IV-1000 and where applicable, 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 GDCs 1 and 30 and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for fabrication of

components. The reactor vessel and its appurtenances are fabricated and installed in accordance with Code Section III, Paragraph NB-4100. The manufacturer or installer of such components is required to certify, by application of the appropriate Code Symbol and completion of an appropriate data report in accordance with Code Section III, Article NCA-8000, that the materials used comply with the requirements of NB-2000, and that the fabrication or installation comply with the requirements of NB-4000.

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

4. Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels. The acceptance criteria for special controls and processes in welding austenitic or ferritic steel components are based upon the following regulatory guides (RGs), ASME Code provisions, and other regulatory documents necessary to satisfy the relevant requirements of GDCs 1, 4, 14, and 30; Appendix B; and 10 CFR 50.55a.

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

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

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

   D. ASME Code Sections III and IX criteria for welding austenitic stainless steels are supplemented by the regulatory positions in RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and RG 1.34.

   E. The regulatory positions of RG 1.44, “Control of the Use of Sensitized Stainless Steel,” and RG 1.28, “Quality Assurance Program Criteria (Design and Construction),” provide the acceptance criteria to avoid sensitization and contamination of stainless steel.

      RG 1.44 states that non-sensitization should be verified using ASTM A-262 Practices A or E, or another method that can be demonstrated to show nonsensitization of austenitic stainless steel.

RG 1.28 endorses the provisions and recommendations included in ASME NQA-1-1994, Part II, Subpart 2.1 for onsite cleaning of materials and components, cleanness control, and preoperational cleaning and layup of water-cooled nuclear power plant fluid systems. As such, the controls for abrasive work on austenitic stainless steel surfaces should, as a minimum, be equivalent to the controls described in ASME NQA-1-1994, Part II, Subpart 2.1, 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.

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

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 10 CFR 50.60 regarding materials testing and acceptance standards for fracture toughness.

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

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

      i. T_{NDT} shall be determined for each material by means of a drop weight test.

      ii. 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)] each Charpy C_v specimen tested shall exhibit at least 0.89 mm (35 mils) lateral expansion and not less than 68 J (50 ft-lbs) of absorbed energy. When these requirements are met, T_{NDT} is defined as the reference temperature, RT_{NDT}.

      iii. 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). Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{cv} - 33°C) [(T_{cv} - 60°F)]

      iv. When a C_v impact test has not been performed at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)], or when the C_v impact test at (T_{NDT} + 33°C) [(T_{NDT} + 60°F)] does not exhibit a minimum of 68 J (50 ft-lbs) and 0.89 mm (35 mils) lateral expansion, a temperature representing a minimum of 68 J (50 ft-lbs) and

0.89 mm (35 mils) lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v impact tests performed.

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

i. DSRS Section 5.3.2 discusses the requirements of paragraphs IV.A.2 and of Appendix G in detail.

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

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

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

Section III of Appendix H requirements are:

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

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

C. The surveillance program shall meet the following requirements:

i. 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.

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

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

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

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

Operational Programs. For COL reviews, the description of the operational program and proposed implementation milestone(s) for the Reactor Vessel Material Surveillance Program are reviewed in accordance with 10 CFR 50.60 and 10 CFR Part 50, Appendix H. The Reactor Vessel Material Surveillance Program and associated implementation milestone(s) are included within the license condition on operational program implementation.

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

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

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

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

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

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

8. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the AEA, 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. GDCs 1 and 30 and 10 CFR 50.55a, establish quality assurance requirements for the design, fabrication, erection, and testing of SSCs important to safety. GDCs 1 and 10 CFR 50.55a establish that the quality assurance standards to be applied to SSCs shall be commensurate with the importance of the safety functions to be performed and will be established and implemented through the development of a quality assurance program. 10 CFR 50.55a also incorporates by reference applicable editions and addenda of the ASME Boiler and Pressure Vessel Code. GDC 30 establishes that RCPB components shall meet the highest quality standards practical. The safety functions of the reactor vessel are to provide (1) a support structure for the internal reactor components, (2) reactor coolant confinement as part of the reactor coolant flow path, and (3) a containment barrier to the release of fission products as part of the RCPB. RGs 1.31, 1.34, 1.43, 1.44, 1.50, and 1.65 provide regulatory positions applicable to compliance with GDCs 1 and 30. Compliance with GDCs 1 and 30 and 10 CFR 50.55a provides assurance that the reactor vessel will be designed, fabricated, erected, and tested to established and proven standards thereby reducing the likelihood of reactor vessel failure.

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

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

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

5. GDC 32 requires that RCPB components shall be designed to allow periodic inspection and testing to assess their structural and leak-tight integrity, and a material surveillance program for the reactor pressure vessel. The reactor vessel material surveillance program monitors the reactor vessel belt-line materials for changes in fracture toughness resulting from exposure to neutron irradiation and the thermal environment. The specific

material surveillance program requirements are established in 10 CFR Part 50, Appendix H and the data is utilized to determine compliance of the irradiated material with the fracture toughness requirements and criteria of 10 CFR Part 50, Appendix G. Compliance with GDC 32 provides assurance that degradation potentially affecting RCPB integrity is detected prior to fracture. Further, a materials surveillance program assures that the reactor vessel materials maintain sufficient toughness, thereby reducing the probability of reactor vessel failures.

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

7. 10 CFR Part 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.28 provides regulatory positions relevant to compliance with Appendix B. Application of cleaning requirements to the reactor vessel materials provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking. This reduces the likelihood that degradation and/or failure of the reactor vessel could cause substantial reduction in capability to contain reactor coolant inventory, reduction in capability to confine fission products, or interference with core cooling.

8. 10 CFR Part 50, Appendix G, establishes requirements for the fracture toughness of pressure-retaining components of the RCPB made of ferritic materials. The reactor vessel is an integral part of the RCPB. Application of these requirements to the RCPB materials provides a method of satisfying the requirements of 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 Part 50, Appendix H, establishes the reactor vessel material surveillance program requirements. The surveillance program monitors the changes in fracture toughness properties of ferritic materials in the reactor vessel belt-line, resulting from exposure to neutron irradiation and the thermal environment. Data from the surveillance program is utilized in complying with 10 CFR Part 50, Appendix G requirements for establishing pressure-temperature limits and corrective actions (such as vessel annealing) if fracture toughness criteria can not be met. The structural integrity of the reactor vessel material is essential in assuring support of internal reactor components, confinement of reactor coolant, and a barrier to the release of fission products. Compliance with 10 CFR Part 50, Appendix H, provides assurance that changes to the reactor vessel materials resulting from the operational environment will be monitored, and that appropriate actions will be taken if significant changes occur in the material fracture toughness that may affect the integrity of the reactor vessel, and thus its ability to accomplish the safety functions under all anticipated and postulated conditions.

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

    Implementation of this program will be inspected in accordance with NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Inspection of Construction and Operational Programs.”

III. <u>REVIEW PROCEDURES</u>

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.

1. Selected Programs and Guidance—In accordance with the guidance in NUREG–0800, “Introduction – Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light-Water Small Modular Reactor Edition” (NUREG-0800, Intro Part 2), as applied to this DSRS Section, the staff will review the information proposed by the applicant to evaluate whether it meets the acceptance criteria described in Subsection II of this DSRS. As noted in NUREG-0800, Intro Part 2, the NRC requirements that must be met by an SSC do not change under the small modular reactor (SMR) framework. Using the graded approach described in NUREG-0800, Intro Part 2, the NRC staff may determine that, for certain SSCs, the applicant’s basis for compliance with other selected NRC requirements may help demonstrate satisfaction of the applicable acceptance criteria for that SSC in lieu of detailed independent analyses. The design-basis capabilities of specific SSCs would be verified, where applicable, as part of completing the applicable ITAAC. The use of the selected programs to augment or replace traditional review procedures is shown in Figure 1 of NUREG-0800, Intro Part 2. Examples of such programs that may be relevant to the graded approach for these SSCs include:

    - 10 CFR Part 50, Appendix A, GDC, Overall Requirements, Criteria 1–5
    - 10 CFR Part 50, Appendix B, Quality Assurance (QA) Program
    - 10 CFR 50.49, Environmental Qualification of Electrical Equipment (EQ) Program
    - 10 CFR 50.55a, Code Design, Inservice Inspection, and Inservice Testing (ISI/IST) Programs
    - 10 CFR 50.65, Maintenance Rule requirements

- Reliability Assurance Program (RAP)
- 10 CFR 50.36, “Technical Specifications”
- Availability Controls for SSCs Subject to Regulatory Treatment of Nonsafety Systems (RTNSS)
- Initial Test Program (ITP)
- Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

This list of examples is not intended to be all inclusive. It is the responsibility of the technical reviewers to determine whether the information in the application, including the degree to which the applicant seeks to rely on such selected programs and guidance, demonstrates that all acceptance criteria have been met to support the safety finding for a particular SSC.

2. In accordance with 10 CFR 52.47(a)(8), (21), and (22), and 10 CFR 52.79(a)(17), (20), and (37), for DC or COL applications submitted under 10 CFR 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, “Resolution of Generic Safety Issues,” 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), for a DC application, and except paragraphs (f)(1)(xii), (f)(2)(ix), (f)(2)(xxv), and (f)(3)(v),for a COL application. 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.

3. <u>Materials</u>. The materials for the reactor vessel and its appurtenances and attachments are compared with the acceptable materials identified in the Code, Section III, Appendix I.

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

4. <u>Special Processes Used for Manufacture and Fabrication of Components</u>. 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.

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

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

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

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

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

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

7. <u>Fracture Toughness</u>. The information submitted by the applicant relative to tests for fracture toughness is reviewed for conformance with the Code, Section III, Paragraph NB-2300, and Appendix G of 10 CFR Part 50.

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

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

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

The FSAR is reviewed to ensure that all the impact tests shown in Paragraph 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 DSRS section.

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

The reviewer also evaluates the end-of-license Charpy upper shelf energy for the reactor vessel materials in accordance with the acceptance criterion specified in 10 CFR Part 50, Appendix G, paragraph IV.A.1.a. Reactor vessel materials that do not meet the specified initial Charpy upper shelf energy acceptance criterion shall be evaluated in accordance with the provisions for additional analysis also specified in paragraph IV.A.1.a. In accordance with paragraph IV.A.1.c., this analysis must be submitted to the staff for review and approval at least 3 years prior to the date on which the predicted Charpy upper shelf energy will no longer satisfy the requirements of paragraph IV.A.1.a. In addition to the ASME Code, RG 1.161 provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in paragraph IV.A.1.a.

8. <u>Material Surveillance</u>. The reviewer verifies that the information contained in the safety analysis report (SAR) and the Technical Specifications is complete enough to determine that the surveillance program will comply with Appendix H, 10 CFR Part 50. The following information must be provided as a minimum:

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

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

C. Operational Programs. The reviewer verifies that the Reactor Vessel Material Surveillance Program is 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.

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

9. 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 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 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 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 SER. The reviewer also states the bases for those conclusions.

For the reasons set forth in detail below:

1. The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Further, the applicant’s special measures for control of residual elements in ferritic materials have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

2. Special processes used for manufacture or fabrication of the reactor vessel and its appurtenances have been identified, and appropriate data reports on each process as required by Section III of the ASME Code have been submitted by the applicant. Since certification has been made by the applicant that the materials and fabrication requirements of Section III of the Code have been complied with, the special processes used are considered acceptable. Compliance with these Code provisions meets the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a.

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

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

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

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

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

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

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

   A. The controls imposed on delta ferrite in austenitic stainless steel welds are in conformance with the recommendations of RG 1.31 because the controls used provide reasonable assurance that the welds will not contain micro cracks. These controls also satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a and the requirements of GDC 14 regarding fabrication to prevent rapid propagating failure of the RCPB.

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

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

   A. The controls employed that avoid contamination and sensitization of austenitic stainless steel are in conformance with the recommendations of RG 1.44, or an acceptable alternative because the controls used provide assurance that welded components will not be contaminated nor sensitized prior to and during the welding process. These controls satisfy the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a and the GDC 4 requirement relative to material compatibility.

   B. The controls regarding onsite cleaning and cleanliness control of austenitic stainless steel are in conformance with the recommendations of RG 1.28 because the controls used provide assurance that austenitic stainless steel components will be properly cleaned onsite. The controls satisfy Appendix B of 10 CFR Part 50 regarding controls for onsite cleaning of materials and components.

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

9. The applicant described the Reactor Vessel Material Surveillance Program and its implementation in conformance with 10 CFR 50.60 and 10 CFR Part 50, Appendix H. The reviewer ensures the Reactor Vessel Material Surveillance Program and associated implementation milestones are included within the license condition on operational program implementations.

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

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

    Accordingly, the staff concludes that the plant design is acceptable and meets the requirements of GDCs 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.

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 regulations in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), and 10 CFR 52.79(a)(41) establish requirements for applications for ESPs, DCs, and COLs, respectively. These regulations require the application to include an evaluation of the site (ESP), standard plant design (DC), or facility (COL) against the SRP revision in effect 6 months before the docket date of the application. While the SRP provides generic guidance, the staff developed the SRP guidance based on the staff's experience in reviewing applications for construction permits and operating licenses for large light-water nuclear power reactors. The proposed SMR designs, however, differ significantly from large light-water nuclear power plant designs.

In view of the differences between the designs of SMRs and the designs of large light-water power reactors, the Commission issued Staff Requirements Memorandum (SRM)-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights To Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010. In the SRM, the Commission directed the staff to develop risk-informed licensing review plans for each of the SMR design reviews, including plans for the associated preapplication activities. Accordingly, the staff has developed the content of the DSRS as an alternative method for evaluating a NuScale-specific application submitted pursuant to 10 CFR Part 52, and the staff has determined that each application may address the DSRS in lieu of addressing the SRP, with specified exceptions. These exceptions include particular review areas in which the DSRS directs reviewers to consult the SRP and others in which the SRP is used for the review. If an applicant chooses to address the DSRS, the application should identify and describe all differences between the design features (DC and COL applications only), analytical techniques, and procedural measures proposed in an application and the guidance of the applicable DSRS section (or SRP section, as specified in the DSRS), and discuss how the proposed alternative provides an acceptable method of complying with the regulations that underlie the DSRS acceptance criteria.

The staff has accepted the content of the DSRS as an alternative method for evaluating whether an application complies with NRC regulations for NuScale SMR applications, provided that the application does not deviate significantly from the design and siting assumptions made by the NRC staff while preparing the DSRS. If the design or siting assumptions in a NuScale application deviate significantly from the design and siting assumptions the staff used in preparing the DSRS, the staff will use the more general guidance in the SRP, as specified in 10 CFR 52.17(a)(1)(xii), 10 CFR 52.47(a)(9), or 10 CFR 52.79(a)(41), depending on the type of application. Alternatively, the staff may supplement the DSRS section by adding appropriate criteria to address new design or siting assumptions.

VI. REFERENCES

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

2. 10 CFR 50.60, “Acceptance Criteria for Fracture Prevention Measures for Light water Nuclear Power Reactors for Normal Operation.”

3. 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

4. 10 CFR Part 50, Appendix A, GDC 1, “Quality Standards and Records.”

5. 10 CFR Part 50, Appendix A, GDC 4, “Environmental and Dynamic Effects Design Bases.”

6. 10 CFR Part 50, Appendix A, GDC 14, “Reactor Coolant Pressure Boundary.”

7. 10 CFR Part 50, Appendix A, GDC 30, “Quality of Reactor Coolant Pressure Boundary.”

8. 10 CFR Part 50, Appendix A, GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary.”

9. 10 CFR Part 50, Appendix A, GDC 32, “Inspection of Reactor Coolant Pressure Boundary.”

10. 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.”

11. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

12. 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.”

13. 10CFR Part 52, Subpart B, “Standard Design Certifications,” Section 52.47, “Contents of Applications: Technical Information” and Subpart C, “Combined Licenses,” Sections 52.80, “Contents of Applications: Additional Technical Information.”

14. RG 1.28, “Quality Assurance Program Criteria (Design and Construction)”

15. RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

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

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

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

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

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

21. RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

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

23. RG 1.215, “Guidance for ITAAC Closure Under 10 CFR Part 52.”

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

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

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

27. NRC Inspection Manual Chapter IMC-2504, “Construction Inspection Program - Inspection of Construction and Operational Programs,” issued September 15, 2009.