Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

6.2.7 FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The reactor containment pressure boundary relates to the reactor containment system. The reactor containment system design must include the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products attendant to postulated accidents. This design-specific review standard (DSRS) section reviews fracture prevention of the reactor containment pressure boundary materials of the mPower™ integral pressurized-water reactors (iPWR).

The reactor containment pressure boundary, as addressed in the U.S. Nuclear Regulatory Commission (NRC) licensing review process, consists of those ferritic steel parts of the reactor containment system which sustain loading and provide a pressure boundary in the performance of the containment function under the operating, maintenance, testing and postulated accident conditions cited by 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 51, "Fracture Prevention of Containment Pressure Boundary." Within this context, typically reviewed are the ferritic materials of components such as freestanding containment vessels, equipment hatches, personnel airlocks, heads of primary containment drywells, tori, containment penetration sleeves, process pipes, end closure caps and flued heads, and penetrating-piping systems connecting to penetration process pipes and extending to and including the system isolation valves.

The specific areas of review are as follows:

1. The containment vessel and all penetration assemblies or appurtenances attached to the containment vessel; all piping, pumps and valves attached to the containment vessell, or to penetration assemblies out to and including the pressure boundary materials of any valves required to isolate the system and provide a pressure boundary for the containment function.

2. 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 DSRS section in accordance with NUREG-0800, Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that

ADAMS Accession No. ML12278A103

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.

3. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters). For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

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

1. Review of the design of steel containments will be performed under DSRS Section 3.8.2, "Steel Containment."

2. Review of the adequacy of programs for assuring the integrity of bolting and threaded fasteners will be performed under DSRS Section 3.13, "Threaded Fasteners – ASME Code Class 1, 2, and 3."

3. Determination of SSC risk significance under SRP Section 19.0.

II. ACCEPTANCE CRITERIA

Requirements

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

1. GDC 1, found in Appendix A to Part 50, as it relates to the quality standards for design and fabrication.

2. GDC 16, as it relates to the prevention of the release of radioactivity to the environment.

3. GDC 51, as it relates to the reactor containment pressure boundary being designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

4. 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 NRC's regulations.

5. 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 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 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. To meet the requirements of GDCs 1, 16 and 51, ferritic containment pressure boundary materials should meet the fracture toughness criteria and requirements for testing identified in Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the American Society of Mechanical Engineers (ASME) Code or, for materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code.

2. Mandatory fracture toughness testing of ASME Code Section III Class 2 materials was first identified in the Summer 1977 Addenda Code Class 2 rules. As a result, cases exist where Class 2 ferritic materials of the reactor containment pressure boundary were not fracture toughness tested, because the ASME Code Edition and Addenda in effect at the time the components were ordered, did not require that they be tested. The NRC staff's assessment of the fracture toughness of materials that were not fracture toughness tested is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump supports," (ML13106A019) and ASME Code Section III, Summer 1977 Addenda, Subsection NC. The metallurgical characterization of these materials, with respect to their fracture toughness, is developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, provides the technical basis for the staff's evaluation of the compliance with Code Class 2 requirements of the materials which were not fracture toughness tested.

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 requires that SSCs be designed, fabricated, erected and tested commensurate with the importance of the safety functions to be performed. This DSRS section evaluates the fracture toughness of the containment pressure boundary ferritic materials to ensure they are not subject to brittle fracture. ASME Code Section III, Division 1, Class MC (Metal Containment) or Section III, Division 2, Class CC (Concrete Containment) or Class 2 component criteria are used in the performance of this fracture toughness evaluation. The application of Code Class MC or CC component criteria for the evaluation metal containment components and the specific application of Subsection NE-2000 for evaluation of steel containment materials or CC-2520 for concrete containment liners, are established staff practices reflecting that Code Class MC or CC requirements for materials, design, fabrication, and testing are commensurate with the safety function of containment (see SRP Sections 3.8.1 and 3.8.2). The application of Code Class 2 criteria for materials that were not fracture toughness tested is consistent with the methodology for application of quality standards to pressure-retaining components commensurate with the importance of their safety functions as described in Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants." The consistency is developed in that the containment system is addressed in the licensing review process as an engineered safety feature, as is, for example, the emergency core cooling system. RG 1.26 does not explicitly discuss or classify the containment pressure boundary, but does assign a Quality Group B classification to the emergency core cooling system. RG 1.26 assigns correspondence between Quality Group B components and ASME Code Section III, Division 1 requirements for Class 2 components. The containment pressure boundary is one of the barriers that prevent the release of radioactivity to the environment in the event of an accident, and therefore, fulfills a vital safety-related role. Use of appropriate design and fabrication standards in conjunction with Article NE-2300 or CC-2520 fracture toughness testing or evaluation of ferritic containment pressure boundary materials with respect to ASME Code Class 2 fracture toughness requirements provides assurance that containment will not fail due to brittle behavior and will thus be capable of preventing the release of radioactivity to the environment.

2. GDC 16 requires reactor containment and associated systems to be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Containment must be leak-tight and withstand accidents because it is the final barrier against the release of radioactivity to the environment in the event of a loss-of-coolant accident. To ensure leak-tightness, containment must not be subject to brittle fracture even under the most severe postulated conditions. Meeting GDC 16 provides assurance that containment will satisfactorily fulfill its safety role and that significant radioactivity will not be released to the environment.

3. GDC 51 requires, in part, that the reactor containment boundary be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. As the final barrier against the release of radioactivity to the environment, containment must not be subject to brittle failure or rapidly propagating fracture, either of which could cause a breach of containment integrity. Meeting GDC 51 will ensure that the containment pressure

boundary remains intact during the harshest expected conditions, thereby precluding the release of radioactivity to the environment.

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.

The licensing review process assesses the fracture toughness of the materials of the components of the reactor containment pressure boundary identified in Section I, within the context of compliance with the criteria of Article NE-2300 of Section III, Division 1 or Article CC-2520 of Section III, Division 2 of the ASME Code or materials that were not fracture toughness tested as discussed below, the fracture toughness criteria for Class 2 components identified in the Summer 1977 Addenda to Section III, Division 1, Subsection NC of the ASME Code.

The reviewer addresses the information provided by the applicant for the materials of the components of interest. Such information should consist of construction drawings, piping system diagrams and related supplemental information, ASME Code data reports and certified material test reports.

1. In accordance with 10 CFR 52.47(a)(8),(21), and (22), 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 that are identified in the version of NUREG-0933 current on the date 6 months before application and that 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.

2. For those ferritic materials for which fracture toughness data are unavailable, or are inappropriate, the reviewer addresses the applicant's assessment of their fracture toughness based on a metallurgical characterization developed from a review of how these materials were fabricated and the thermal history they experienced during fabrication. The reviewer addresses the applicant's correlation of this information with the fracture toughness data presented in NUREG-0577 and ASME Section III, Summer 1977 Addenda, Subsection NC. The reviewer addresses the applicant's justification of the acceptability of these materials within the context of the criteria for Class 2 materials as stated in the Summer 1977 Addenda, ASME Code Section III. The reviewer verifies that the Class 2 requirements of the Summer 1977 Addenda of ASME Section III Code have been met by the applicant.

3. For review of a DC application, the reviewer should follow the above procedures to verify that that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the design control document, (DCD), meets the acceptance criteria. 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 DCD.

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. The staff concludes that reasonable assurance has been provided that the materials of the reactor containment pressure boundary, under operating, maintenance, testing and postulated accident conditions, will not undergo brittle fracture, and that the probability of rapidly propagating fracture will be minimized, so that the requirements of GDCs 1, 16, and 51 will be met. This conclusion is summarized as follows (provide the finding that applies):

> Based on its review, the staff finds that the ferritic materials of the reactor containment pressure boundary were (or will be where appropriate) acceptably tested and demonstrated to meet the fracture toughness requirements for Class MC components as specified in Article NE-2300 of ASME Code Section III, Division 1 or Class CC components as specified in Article CC-2520 of ASME Code Section III, Division 2.
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> OR
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> For ferritic reactor containment pressure boundary materials that were not fracture toughness tested, based on the licensing process review of the applicant's available fracture toughness data, metallurgical characterizations of the materials of interest developed from their fabrication and thermal histories, and correlations of metallurgical histories with fracture toughness data presented in NUREG-0577 and ASME Code Section III, Summer 1977 Addenda, Subsection NC, the conclusion is made that the fracture toughness of the materials of the reactor containment pressure boundary meet the fracture toughness requirements invoked for ASME Code Section III Class 2 materials effective with the Summer 1977 Addenda.

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^{TM} 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.

VI. REFERENCES

1. 10 CFR Part 50, §50.55a, Codes and Standards.

2. 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records."

3. 10 CFR Part 50, Appendix A, GDC 16, "Containment Design."

4. 10 CFR Part 50, Appendix A, GDC 51, "Fracture Prevention of Containment Pressure Boundary."

5. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

6. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," American Society of Mechanical Engineers.

7. ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," American Society of Mechanical Engineers.

8. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Summer 1977 Addenda, Subsection NC, "Class 2 Components", American Society of Mechanical Engineers.

9. NUREG-0577 Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," USNRC, October 1983. (ML13106A019)

10. RG 1.26, "Quality Group Classifications and Standards for Water-,Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

11. RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."