U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

6.2.1.4 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY SYSTEM PIPE RUPTURES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of containment integrity

Secondary - None

I. AREAS OF REVIEW

Babcock & Wilcox Nuclear Energy mPower™ is an integral pressurized-water reactor with the reactor, steam generator, pressurizer, and control rod drives all located in a single pressure vessel. The mPower™ reactor containment is a free-standing carbon steel structure that is located below grade level.

The mass and energy release analysis for secondary system pipe ruptures is reviewed to ensure the acceptability of the data used to evaluate the containment and subcompartment functional design.

The specific areas of review are as follows:

1. Sources of Energy: All of the energy sources from steam and feedwater line break accidents that are available for release to the containment are reviewed.

2. Mass and Energy Release Rate: The mass and energy release rate calculations are reviewed.

3. Single-Failure Analyses: The single-failure analyses performed for steam and feedwater line isolation provisions that would limit the flow of steam or feedwater to the assumed pipe rupture are reviewed.

4. 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." 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.

5. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and

ADAMS Accession No. ML12230A037

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 various types and aspects of the containment design are identified in DSRS Section 6.2.1.

2. The seismic classification and system quality group classification of steam and feedwater line isolation valves are reviewed under DSRS Sections 3.2.1 and 3.2.2 to determine the acceptability of these valves in limiting the mass and energy releases from the steam and feedwater systems.

3. Postulated pipe break locations and sizes are reviewed under DSRS Section 3.6.2.

4. Risk significance of SCCs is reviewed 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) 50, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated secondary system pipe ruptures to ensure the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA).

2. Title of 10 of the *Code of Federal Regulations* (CFR), Section 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 plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission’s (NRC's) regulations.

3. 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 as follows for the 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. Sources of Energy. The sources of energy that should be considered in the analyses of steam and feedwater line break accidents include the stored energy in the steam generator's metal, including the vessel tubing, feedwater line, and steam line; stored energy in the water contained within the steam generator; stored energy in the feedwater transferred to the steam generator before closure of the isolation valves in the feedwater line; stored energy in the steam from the steam generator before the closure of the isolation valves in the steam generator outlet line; and energy transferred from the primary coolant to the water in the steam generator during blowdown.

   The steam line break accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102 percent of full power. The applicant need only analyze the 102-percent power condition if it can demonstrate that the feedwater flows and fluid inventory are greatest at full power.

2. Mass and Energy Release Rate. In general, calculations of the mass and energy release rates during a steam or feedwater line break accident should be performed in a conservative manner from a containment response standpoint (i.e., the post-accident containment pressure and temperature are maximized). The following criteria indicate the degree of conservatism that is desired:

   A. Mass release rates should be calculated using the Moody model (Reference 1) for saturated conditions or a model that is demonstrated to be equally conservative.

   B. Calculations of heat transfer to the water in the steam generator should be based on nucleate boiling heat transfer.

   C. Calculations of mass release should consider the water in the steam generator and feedwater line, feedwater transferred to the steam generator before the closure of the isolation valves in the feedwater lines, and steam in the steam generator.

   D. If liquid entrainment is assumed in the steam line breaks, experimental data should support the predictions of the liquid entrainment model. The effect on the entrained liquid of steam separators located upstream from the break should be taken into account. A spectrum of steam line breaks should be analyzed, beginning with the double-ended break and decreasing in area until no entrainment is calculated to occur. This will allow selection of the maximum release case.

      If no liquid entrainment is assumed, a spectrum of the steam line breaks should be analyzed beginning with the double-ended break and decreasing in area until it has been demonstrated that the maximum release rate has been considered.

E. Feedwater flow to the steam generator should be calculated considering the diversion of flow between the two feedwater pipes to the common header with inlets to the steam generator on opposite sides of the reactor vessel, feedwater flashing, and increased feedwater pump flow caused by the reduction in steam generator pressure. An acceptable method for computing feedwater flow is to assume all feedwater travels to the steam generator at the pump run-out rate before isolation. After isolation, the unisolated feedwater mass should be added to the available inventory in the steam generator.

Any general-purpose thermal-hydraulics computer codes that the responsible reviewing organization for the subject application finds acceptable may be used to compute mass and energy releases from steam and feedwater line break accidents.

3. Single-Failure Analyses. Steam and feedwater line break analyses should assume a single active failure in the steam or feedwater line isolation provisions or feedwater pumps to maximize the containment peak pressure and temperature.

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 50 requires the containment structure and associated heat removal systems be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant. DSRS Section 6.2.1.4 applies the requirements of this GDC to postulated secondary system pipe ruptures to assure that mass and energy inputs are appropriately conservative. A secondary system pipe rupture releases a significant amount of energy which potentially could damage the containment structure or associated systems. Containment, therefore, must be designed to definitively withstand this accident. Meeting the requirements of GDC 50 will ensure that containment integrity is maintained under the most severe secondary system pipe rupture, thus 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.

1. Programmatic Requirements – In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant SSCs, but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

A. Maintenance rule, SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guide (RG) 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

B. Quality Assurance Program, SRP Sections 17.3 and 17.5 (DSRS Section 13.4, Table 13.4, Item 16).

C. Technical Specifications (DSRS Section 16.0 and SRP Section 16.1) – including brackets value for DC and COL. Brackets are used to identify information or characteristics that are plant specific or are based on preliminary design information.

D. Reliability Assurance Program (SRP Section 17.4).

E. Initial Plant Test Program (RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

F. ITAAC (DSRS Chapter 14).

2. 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.

3. <u>Sources of Energy</u>. The reviewer evaluates the sources of energy identified by the applicant in the analyses of steam and feedwater line break accidents to ensure that the sources listed in Subsection II of this DSRS section have been considered.

   The reviewer also examines the assumptions of the secondary coolant system pipe break analysis to determine whether the applicant has identified the worst case pipe break accident and completed the analysis in a conservative manner from the standpoint of containment pressure and temperature. This review involves the proposed methods and models used for blowdown analyses. The reviewer will evaluate the acceptability of the approach used by the applicant based on the acceptance criteria in Subsection II of this DSRS section.

4. <u>Mass and Energy Release Rate</u>. The reviewer evaluates the applicant’s calculations for main feedwater flow into the steam generator to determine whether the flow rate is conservatively maximized.

   If the applicant’s steam line break model calculates liquid entrainment, the reviewer determines the validity of the experimental data provided to support the entrainment calculation. The reviewer will also ascertain whether the analysis considered the effect of steam separators located upstream from the postulated steam line break. The reviewer evaluates comparisons to experimental data made by the applicant and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

The reviewer examines the results of a spectrum of steam line breaks, beginning with the double-ended break and decreasing in area until no entrainment occurs, to ensure that the applicant has identified the steam line break size producing the highest containment temperature and pressure.

The reviewer may perform confirmatory analyses of the containment pressure and temperature response to steam and feedwater line breaks inside the containment using thermal-hydraulic computer codes that the responsible reviewing organization for the subject application finds acceptable.

5. Single-Failure Analyses. The reviewer reviews analyses of postulated single failures of active components in the secondary systems, such as steam and feedwater line isolation valves and feedwater pumps, and determines whether the single failure that maximizes containment pressure and temperature has been selected.

   The reviewer requests the review of DSRS Sections 3.2.1, 3.2.2, and 3.6.2 by the responsible organization as to the acceptability of nonsafety valves in limiting the mass and energy releases from the steam and feedwater systems.

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 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 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 (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, as augmented by the application of programmatic requirements in accordance with the staff’s technical review approach in the DSRS Introduction, support conclusions of the following type to be included in the staff's SER. The reviewer also states the bases for those conclusions.

The evaluation findings will follow the format provided in DSRS Section 6.2.1 and conclude that the applicant followed the DSRS acceptance criteria identified above [or identified deviations from the DSRS acceptance criteria with appropriate justification] and meets GDC 50, as it relates to providing sufficient conservatism in the mass and energy release analysis for postulated secondary system pipe ruptures for the containment design-basis.

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 DC, COL, or ESP 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 (Agencywide Documents Access and Management System Accession No. ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews, including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™-specific DC, COL, or ESP applications 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™ 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 revise the DSRS section in order to address new design assumptions. The same approach may be used to meet the requirements of 10 CFR 52.17 (a)(1)(xii) and 10 CFR 52.79 (a)(41), for ESP and COL applications, respectively.

VI. REFERENCES

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

2. RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

3. RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants."

4. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

5. RG 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52."

6. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," with discussion comments and Authors' Closure, Jour. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.