U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

6.2.1.3 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

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 analyses of the mass and energy release are reviewed to assure that the data used to evaluate the containment and subcompartment functional design are acceptable for that purpose.

The specific areas of review are as follows:

1. The energy sources that are available for release to the containment.

2. The mass and energy release rate calculations for the initial blowdown phase of the accident.

3. Because of the additional steam generator stored energy available for release, the mass and energy release rate calculations for the core reflood and post-reflood phases of the accident.

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.

ADAMS Accession No. ML12230A034

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

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

<u>Review Interfaces</u>

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

1. Review of the acceptability of piping design criteria, selected break locations and break sizes based on the provisions made to limit pipe motion, for breaks postulated to occur within subcompartments is performed under DSRS Section 3.6.2.

2. Determination of SSC risk significance is performed under SRP Chapter 19.

The specific acceptance criteria and review procedures are contained in the referenced DSRS and SRP sections.

II. <u>ACCEPTANCE CRITERIA</u>

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

1. General Design Criterion (GDC) 50, as it relates to the containment and subcompartments being designed with sufficient margin, requires that the containment and its associated systems can accommodate, without exceeding the design leakage rate, and the containment and subcompartment design can withstand the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA).

2. Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix K, as it relates to sources of energy during the LOCA, provides requirements to assure that all the energy sources have been considered.

3. 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 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.

4. 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. GDC 50 and Appendix K to 10 CFR Part 50.

   A. Sources of Energy. The sources of stored and generated energy that should be considered in analyses of LOCAs include: reactor power; decay heat; stored energy in the core; stored energy in the reactor coolant system (RCS) metal, including the reactor vessel and reactor vessel internals; metal-water reaction energy; and stored energy in the secondary system including the steam generator tubing and secondary water.

      Calculations of the energy available for release from the above sources should be done in general accordance with the requirements of 10 CFR Part 50, Appendix K, paragraph I.A. However, additional conservatism should be included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA. An example of this would be accomplished by maximizing the sensible heat stored in the RCS and steam generator metal and increasing the RCS and steam generator secondary mass to account for uncertainties and thermal expansion.

      The requirements of paragraph I.B in Appendix K to 10 CFR Part 50, concerning the prediction of fuel clad swelling and rupture should not be considered. This will maximize the energy available for release from the core.

   B. Break Size and Location

      i. The staff's review of the applicant's choice of break locations and types is discussed in DSRS Section 3.6.2.

      ii. Of several breaks postulated on the basis of A., above, the break selected as the reference case for subcompartment analysis should yield the highest mass and energy release rates, consistent with the criteria for establishing the break location and area.

      iii. Containment design-basis calculations should be performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified.

   C. Calculations. In general, calculations of the mass and energy release rates for a LOCA should be performed in a manner that conservatively establishes the containment internal design pressure (i.e., maximizes the post-accident containment pressure and the containment subcompartment response). The criteria given below for each phase of the accident indicate the conservatism that should exist.

i. <u>Subcompartment Analysis</u>

The analytical approach used to compute the mass and energy release profile will be accepted if both the computer program and volume noding of the piping system are similar to those of an approved emergency core cooling system (ECCS) analysis. The computer programs that are currently acceptable include CRAFT-2, and RELAP5, when a flow multiplier of 1.0 is used with the applicable choked flow correlation. An alternate approach, which is also acceptable, is to assume a constant blowdown profile using the initial conditions with an acceptable choked flow correlation.

ii. <u>Initial Blowdown Phase Containment Design-Basis</u>

The initial mass of water in the RCS should be based on the RCS volume calculated for the temperature and pressure conditions assuming that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error). An assumed power level lower than the level specified (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.

Mass release rates should be calculated using a model that has been demonstrated to be conservative by comparison to experimental data.

Calculations of heat transfer from surfaces exposed to the primary coolant should be based on nucleate boiling heat transfer. For surfaces exposed to steam, heat transfer calculations should be based on forced convection.

Calculations of heat transfer from the secondary coolant to the steam generator tubes should be based on natural convection heat transfer for tube surfaces immersed in water and condensing heat transfer for the tube surfaces exposed to steam.

iii. <u>Core Reflood Phase (Cold Leg Reactor Pressure Vessel Penetration Breaks Only)</u>

Following initial blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state full power operation condition) to the time that the RCS equalizes to the containment pressure, the water remaining in the reactor vessel should be assumed to be saturated. Justification should be provided for the refill period, which is the time from the end of the blowdown to the time when the ECCS refills the vessel lower plenum by gravity drain from the refueling water storage tanks (RWSTs). An acceptable approach is to assume a water level at the bottom of the active core at the end of blowdown so there is no refill time.

Calculations of the core flooding rate should be based on the ECCS operating condition during the core reflood phase, which begins when the water starts to flood the core and continues until the core is completely quenched, or the post-reflood phase, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the steam generators, that maximizes the containment pressure.

Calculations of liquid entrainment, i.e., the carryout rate fraction (CRF), which is the mass ratio of liquid exiting the core to the liquid entering the core, should be based on the pressurized-water reactor full length emergency cooling heat transfer experiments

The assumption of steam quenching should be justified by comparison with applicable experimental data. Liquid entrainment calculations should consider the effect on the CRF of the increased core inlet water temperature caused by steam quenching assumed to occur from mixing with the ECCS water.

Steam leaving the steam generators should be assumed to be superheated to the temperature of the secondary coolant.

iv. Post-Reflood Phase

All remaining stored energy in the primary and secondary systems should be removed during the post-reflood phase.

Steam quenching should be justified by comparison with applicable experimental data.

The results of post-reflood analytical models should be compared to applicable experimental data.

v. Decay Heat Phase

The dissipation of core decay heat should be considered during this phase of the accident. The fission product decay energy model is acceptable if it is equal to or more conservative than the decay energy model given in DSRS Section 9.2.5.

Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with ECCS injection water, such as from the pipe break occurring in one of the two RWST drain lines to the reactor.

The following methods and computer models are acceptable for calculating the mass and energy releases for containment design-basis calculations:

Babcock & Wilcox / Framatome ANP: CRAFT, CRAFT-2, RELAP5/MOD2-B&W, Revision 1 and RELAP5/MOD2-B&W, Revision 4. Methods for calculating the mass and energy releases for containment design-basis calculations should be conservative for these calculations.

2. 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 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. 10 CFR 52.47(a)(1)(vi) provides the requirement for ITAAC for DC reviews.

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 system to be designed with margin to accommodate any LOCA such that the containment design leak rate is not exceeded. A LOCA potentially causes the greatest pressure surge and release of fission products when compared to any other accident. Since it is the most severe challenge expected, containment must be designed to definitively withstand this accident. Following GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment.

2. Appendix K to 10 CFR Part 50 provides required and acceptable features of evaluation models used to analyze various circumstances applicable to the ECCS. Section I.A of Appendix K provides a comprehensive list of LOCA heat (energy) sources and the reactor operating history assumptions associated with those heat sources. Since the mass and energy release analysis for postulated LOCAs is used to design containment and containment subcompartments such that they will withstand the worst case LOCA, it is critical that all potential energy sources are taken into account. Following 10 CFR Part 50 Appendix K will ensure that containment and containment subcompartments are designed to accommodate all energy sources for the worst case LOCA, thus precluding the potential release of radioactivity to the environment following such a LOCA.

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 reviewer confirms, with the lead reviewer for DSRS Section 3.6.2, the validity of the applicant's analysis of pipe break size, type and locations for subcompartments containing high energy lines by using elevation and plan drawings of the containment showing the routing of lines containing high energy fluids. The reviewer determines that an appropriate reference case for subcompartment analysis has been identified. In the event a pipe break other than a double-ended pipe rupture is postulated by the applicant, the lead reviewer for DSRS Section 3.6.2 will evaluate the applicant's justification for assuming a limited displacement pipe break.

The reviewer compares the sources of energy considered in the loss-of-coolant analysis and the methods and assumptions used to calculate the energy available for release from the various sources with the acceptance criteria listed in Section II, above. The reviewer determines the acceptability of the analytical models and the assumptions used to calculate the rates of mass and energy release during the initial blowdown, core reflood, and post-reflood phases of a LOCA. The reviewer also compares energy inventories at various times during a LOCA to ensure that the energy from the various sources has been accounted for and has been transferred to the containment on an appropriate time scale.

The reviewer reviews comparisons made by the applicant to experimental data and makes comparisons to other available experimental data to determine the amount of conservatism in the mass and energy release models.

The reviewer may perform confirmatory analyses of the mass and energy profiles. The purpose of the analysis is to confirm the predictions of the mass and energy release rates appearing in the safety analysis report, and to confirm that an appropriate break location has been considered in these analyses.

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.

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 section.

IV. EVALUATION FINDINGS

The conclusions reached on completion of the review of this DSRS section are presented in DSRS Section 6.2.1.1.

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^{TM}-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^{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 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. BAW-10092, "CRAFT-2 Fortran Program for Digital Simulation of a Multinode Reactor Plant During a Loss-of-Coolant Accident," Babcock and Wilcox Company.

7. W. H. Retting, G. A. Jayne, K. Y. Moore, C. E. Slater, and M. L. Uptmor, "RELAP3 - A Computer Program for Reactor Slowdown Analysts," IN-1321, Idaho Nuclear Corporation.

8. BAW-10030, "CRAFT - Description of Model for Equilibrium LOCA Analysis Program," Babcock and Wilcox, Lynchburg, VA.

9. B&W-10164P, Revision 1, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," Babcock and Wilcox.

10. BAW-10164P-A, Revision 4, "RELAP5-MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."