U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of Containment Integrity

Secondary - None

I. <u>AREAS OF REVIEW</u>

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 responsible staff reviews information regarding the functional capability of the reactor containment presented in Section 6.2.1 of the applicant's application. The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents. The containment structure must also maintain functional integrity in the long term following a postulated accident; i.e., it must remain a low leakage barrier against the release of fission products for as long as postulated accident conditions require. The containment structure must be able to withstand postulated hydraulic forces caused by groundwater, flooding, and tsunami hazards.

The design and sizing of containment systems are largely based on the pressure and temperature conditions which result from release of the reactor coolant in the event of a loss-of-coolant accident (LOCA). The containment design basis includes the effects of stored energy in the reactor coolant system, decay energy, and energy from other sources such as the secondary system, and metal-water reactions including the recombination of hydrogen and oxygen. The containment system is not required to be a complete and independent safeguard against a LOCA by itself, but functions to contain any fission products released while the emergency core cooling system cools the reactor core.

The evaluation of a containment functional design includes calculation of the various effects associated with the postulated rupture in the primary or secondary coolant system piping. The subsequent thermodynamic effects in the containment resulting from the release of the coolant mass and energy are determined from a solution of the incremental space and time-dependent energy, mass, and momentum conservation equations. The basic functional design requirements for containment are given in General Design Criteria (GDC) 4, 16 and 50 in Appendix A to Title 10 of *Code of Federal Regulations* (10 CFR) Part 50 and in 10 CFR Part 50, Appendix K. GDC 4 provides the basic environmental and dynamic effects design requirements for all structures, systems, and components important to safety including leak-before-break.

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GDC 16 establishes the fundamental requirement to design a containment that is essentially a leak-tight barrier against the uncontrolled release of radioactivity to the environment. GDC 50, among other things, requires that consideration be given to the potential consequences of degraded engineered safety features, such as the containment heat removal system and the emergency core cooling system, the limitations in defining accident phenomena, and the conservatism of calculational models and input parameters in assessing containment design margins.

The various aspects to be reviewed under this design-specific review standard (DSRS) section have been separated and assigned to a set of other standard review plan (SRP) and DSRS sections as follows:

1 Pressurized water reactor (PWR) dry containments, including sub-atmospheric containments (DSRS Section 6.2.1.1).

2. Subcompartment analysis (DSRS Section 6.2.1.2).

3. Mass and energy release analysis for postulated loss-of-coolant accidents (DSRS Section 6.2.1.3).

4. Mass and energy release analysis for postulated secondary system pipe ruptures (DSRS Section 6.2.1.4).

5. Minimum containment pressure analysis for emergency core cooling system (ECCS) performance capability studies (SRP Section 6.2.1.5).

Areas related to the evaluation of the containment functional capability are treated in other DSRS sections; e.g., Containment Heat Removal (DSRS Section 6.2.2), Containment Isolation System (DSRS Section 6.2.4), Combustible Gas Control (SRP Section 6.2.5), and Containment Leakage Testing (DSRS Section 6.2.6). In addition, the evaluation of the secondary containment functional design capability is reviewed in DSRS Section 6.2.3.

The specific areas of review are described in the "Areas of Review" subsections of the five SRP and DSRS sections listed above.

Review Interfaces:

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

1. The review of effects of static and dynamic hydraulic forces on containment caused by tsunami hazards under DSRS Section 2.4.6.

2. The review of flooding protection measures under DSRS Section 2.4.10.

3. The review of effects of groundwater on the underground containment structure , including effects of groundwater levels, piezometric/hydraulic heads and other hydrodynamic effects of groundwater on the design bases of subsurface safety-related SSCs under DSRS Section 2.4.12.

4. The review of areas relating to concrete containments or to concrete portions of steel/concrete containments under SRP Section 3.8.1.

5. The review of areas relating to steel containments or to other Class MC steel portions of steel/concrete containments under SRP Section 3.8.2.

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

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are given in the "Acceptance Criteria" subsections of the SRP and DSRS sections listed above.

SRP and DSRS Acceptance Criteria

Specific SRP and DSRS acceptance criteria are provided in the referenced SRP and DSRS Sections. The SRP and DSRS are not substitutes for the NRC's regulations, and compliance with them is not required. Identifying the differences between this SRP or 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 SRP or DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information." The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

III. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, 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 safety evaluation report. The reviewer also states the bases for those conclusions.

1. Containment Functional Design

The scope of review of the functional design of the containment for the mPower™ nuclear power plant design has included a review of plant arrangement drawings, system drawings, and descriptive information for the containment building, subcompartments, and associated systems, components, and structures that are essential to the functional capability and integrity of the containment. The review has included the applicant's proposed design bases for the containment building and internal structures, and associated structures and systems upon which the containment function depends, and the applicant's analysis of postulated accidents and operational occurrences which support the adequacy of the design bases.

The basis for the staff's acceptance has been conformance of designs and design bases for the containment building, internal structures, and associated systems, components, and structures to the Commission's regulations as set forth in the general design criteria, and to applicable regulatory guides, branch technical positions, and industry codes and standards. (Special problems or exceptions that the staff takes to the design or functional capability of containment structures, systems, and components should be discussed.)

To support the basis for the staff's acceptance of the containment system, the reviewer of the containment system should include in the staff's safety evaluation report, as necessary, the results of the reviews for the five SRP and DSRS sections above. The SER writeup should demonstrate conformance with the Commission regulations in the manner indicated. The staff concludes that the containment functional design is acceptable and meets the requirements of General Design Criteria 4, 16, 50 and 10 CFR Part 50 Appendix K. The conclusion is based on the following: [The reviewer should discuss each item of the regulations or related set of regulations as indicated.]

1. The applicant has met the requirements of (cite regulation) with respect to (state limits of review in relation to regulation) by (for each item that is applicable to the review, state how it was met and why it is acceptable with respect to regulation being discussed):

   A. meeting the regulatory positions in Regulatory Guide ________ or Guides;

   B. providing and meeting an alternative method to regulatory positions in Regulatory Guide ________, that the staff has reviewed and found to be acceptable because ________;

   C. meeting the regulatory position in the branch technical position (BTP);

   D. using calculational methods for (state what was evaluated) that have previously been reviewed by the staff and found acceptable; the staff has reviewed the impact parameters in this case and found them to be suitably conservative or performed independent calculations to verify acceptability of their analysis; and/or

   E. meeting the provisions of (industry standard number and title) that has been reviewed by the staff and determined to be appropriate for this application.

2. Repeat discussion for each regulation cited above.

3. The temperature/pressure profiles provided in the applicant's technical submittal for the spectrum of LOCA and main steam line break accidents are acceptable for use in equipment qualification, i.e., there is reasonable assurance that the actual temperatures and pressures for the postulated accidents will not exceed these profiles anywhere within the specified environmental zones, except in the break zone.

IV. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower^{TM}-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 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 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.

V. REFERENCES

1. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

2. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects19 Design Bases."

3. 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and Control."

4. 10 CFR Part 50, Appendix A, GDC 16, "Containment Design."

5. 10 CFR Part 50, Appendix A, GDC 38, "Containment Heat Removal."

6. 10 CFR Part 50, Appendix A, GDC 39, "Inspection of Containment Heat Removal System."

7. 10 CFR Part 50, Appendix A, General Design Criterion 40, "Testing of Containment Heat Removal System."

8. 10 CFR Part 50, Appendix A, GDC 50, "Containment Design Basis."

9. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

10. RELAP4 MOD5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems Users Manual, ANCR-NUREG-1335, September 1976.

11. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment."

12. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981.

13. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," March 1981.

14. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

15. F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Jour. of Heat Transfer, Trans. Am. Soc. of Mechanical Engineers, Vol. 87, No. 1, February 1965.

16. RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

17. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."

18. NRC Safety Evaluation Report, Babcock and Wilcox Company, Reference Safety Analysis Report, B-SAR-205, May 1978.

19. "NRC Safety Evaluation Report - Standard Reference System, CESSAR System 80," Combustion Engineering Inc., December 1975.

20. BTP 6-2, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

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

22. RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

23. T. Tagami, "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," prepared for the National Reactor Testing Station, February 28, 1966 (unpublished work).

24. H. Uchida, A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proc. Third International Conference on the Peaceful Uses of Atomic Energy, Volume 13, Session 3.9, United Nations, Geneva (1964).

25. "CRAFT-2 Fortran Program for Digital Simulation of a Multinode Reactor Plant during a Loss-of-Coolant Accident," BAW-10092, Babcock and Wilcox Company, December 1974.

26. Code Manual for CONTAIN 2.0: A Computer Code for Nuclear reactor Containment Analysis, K.K. Murata, et al., Sandia National Laboratories, NUREG/CR-6533, December 1997.

27. GOTHIC: Containment Analysis Package User Manual, Qualification Report and Technical manual, NAI 8907.

28. Letter from Anthony C. McMurtray, USNRC, to Thomas Coutu, Site Vice President, Kewaunee Nuclear Power Plant, September 29, 2003 [ADAMS Accession Number ML0326810500].

29. Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design, Volumes 1, 2 and 3, NUREG-1793, US NRC, September 2004 (ML043450284).

30. Final Safety Evaluation Report Related to Certification of the AP600 Standard Design, Volumes 1, 2 and 3, NUREG-1512, US NRC, September 1998.

31. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ML003716792).

32. RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design basis Accidents at Light Water Nuclear Power Reactors," May 2003 (ML020160023).

33. DSRS Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

34. NRC Generic Letter 88-17, "Loss of Decay Heat Removal," US NRC, October 17, 1988.