Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

6.2.1.1.A mPower™ iPWR CONTAINMENT

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 specific areas of review are as follows:

1. The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (LOCAs) (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks.

2. The maximum expected external pressure to which the containment may be subjected.

3. The effect of minimum containment pressure on refueling water storage tank (RWST) gravity drain into the reactor for reactor internal natural convection cooling.

4. The effectiveness of static (passive) and active heat removal mechanisms where besides reactor coolant inventory and purification system the primary components that must be actuated are the opening or closing of direct current-operated or air-operated valves for emergency core cooling system (ECCS) and RWST functions and steam and feedwater lines isolation.

5. The pressure conditions within subcompartments that act on system components and supports due to high energy line breaks.

6. The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.

7. 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 DSRS 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.

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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 DSRS Section 14.3.

8. 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 DSRS sections interface with this section as follows:

1. The risk to containment integrity from explosions or releases from nearby chemical plants using reactor co-generated steam for process heat under DSRS Sections 2.2.1-2.2.2, 2.2.3, and 19.0.

2. The electrical design of the instrumentation provided to monitor and record containment conditions during and following an accident; and the effectiveness of the administrative controls and the instrumentation and control provisions to prevent inadvertent operation of the containment heat removal systems or system trains under DSRS Section 7.5.

3. The design adequacy of the containment and its internal structures under DSRS Sections 3.8.2 and 3.8.3.

4. The design adequacy of mechanical components and their supports under DSRS Section 3.9.3.

5. The proposed technical specifications that pertain to the surveillance requirements for containment isolation valves under DSRS Section 16.0.

6. The environmental qualification of the containment system under DSRS Section 3.11.

7. Offsite and control room dose under DSRS Section 15.0.3.

8. Risk significance of SCCs under DSRS Section 19.0.

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

10. Effects of groundwater on the underground containment structure, including effects of groundwater levels, peizometrichydraulic heads and other hydronamic effects of groundwater on the design bases of subsurface safety-related or risk-significant SSCs, such as the containment isolation system and containment penetrations (DSRS Sections 6.2.2, 6.2.4, and 6.2.6) under DSRS Section 2.4.12.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 4, as it relates to SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

2. GDC 5, as it relates to SSCs important to safety shall not be shared among nuclear power units or modules in a single power unit unless it can be shown that such sharing will not significantly impair their ability to perform their safety or risk-significant functions, including, in the event of an accident in one unit or module, an orderly shutdown and cooldown of the remaining units or modules.

3. GCD 16, as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Since the primary reactor containment is the final barrier of the defense-in-depth concept to protect against the uncontrolled release of radioactivity to the environs, preserving containment integrity under the dynamic conditions imposed by postulated LOCAs is essential.

4. GDC 50, as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin.

5. GDC 38, as it relates to the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

6. GDC 13, as it relates to instrumentation and control, requires instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety.

7. GDC 64, as it relates to monitoring radioactivity releases, requires means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

8. For those applicants subject to Title 10 of the *Code of Federal Regulations* (CFR), Section 50.34(f)[1]: 10 CFR 50.34(f)(3)(v)(A)(1), as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel clad metal-water reaction accompanied by hydrogen burning.

[1] For Part 50 applicants not listed in 10 CFR 50.34(f), the provisions of 50.34(f) will be made a requirement during the licensing review.

9. 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.

10. 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.

<u>DSRS Acceptance Criteria</u>

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. To satisfy the requirements of GDC 4 for an embedded subsurface containment, the containment, its penetrations, its cooling systems, and isolation systems should be designed and constructed to accommodate with no loss of function the effects of internal or external flooding events and the effects of explosion-generated missiles or chemical releases from on-site or nearby plants using reactor-generated steam or process heat.

2. To satisfy the requirements of GDC 5 for multiple module or multiple unit plants, safety-related passive SSCs should not be shared and post-72 hour nonsafety-related, risk-significant SSCs should be shown that such sharing will not significantly impair their ability to perform their risk-significant functions.

3. To satisfy the requirements of GDCs 16 and 50 regarding sufficient design margin, for plants at the construction permit (CP) stage of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a LOCA, or a steam or feedwater line break. For plants at the operating license (OL) stage of review, the peak calculated containment pressure following a LOCA, or a steam or feedwater line break, should be less than the containment design pressure. In general, the peak calculated containment pressure should be approximately the same as at the construction permit or design certification stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin.

4. To satisfy the requirements of GDC 38 to rapidly reduce the containment pressure, the containment pressure should be reduced to less than 50% of the peak calculated pressure for the design-basis LOCA within 24 hours after the postulated accident. If analysis shows that the calculated containment pressure may not be reduced to 50% of

the peak calculated pressure within 24 hours, the organization responsible for DSRS Section 15.0.3 should be notified.

5. To satisfy the requirements of GDCs 38 and 50 with respect to the containment heat removal capability and design margin, the LOCA analysis should be based on the assumption of loss of offsite power and the most severe single failure in the emergency power system (e.g., a diesel generator failure), the containment heat removal systems (e.g., a fan, pump, or valve failure), or the core cooling systems (e.g., a pump or valve failure). The selection made should result in the highest calculated containment pressure.

6. To satisfy the requirements of GDCs 38 and 50 with respect to the containment heat removal capability and design margin, the containment response analysis for postulated secondary system pipe ruptures should be based on the most severe single failure in the containment heat removal systems (e.g., no fan, pump, or valve failure) or the secondary system isolation provisions (e.g., main steam isolation valve failure or feedwater line isolation valve failure). The analysis should also be based on a spectrum of pipe break sizes and reactor power levels. The accident conditions selected should result in the highest calculated containment pressure or temperature depending on the purpose of the analysis. Acceptable methods for the calculation of the containment environmental response to main steam line break accidents are found in NUREG-0588.

7. To satisfy the requirements of GDCs 38 and 50 with respect to the functional capability of the containment heat removal systems and containment structure under LOCA conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result, for example, from inadvertent operation of containment heat removal systems. The provisions made should include conservative structural design to assure that the containment structure is capable of withstanding the maximum expected external pressure; or interlocks in the plant protection system and administrative controls to preclude inadvertent operation of the systems. If the containment is designed to withstand the maximum expected external pressure, the external design pressure of the containment should provide an adequate margin above the maximum expected external pressure to account for uncertainties in the analysis of the postulated event.

8. In accordance with the requirements of GDCs 13 and 64, and 10 CFR 50.34(f)(2)(xvii) (for those applicants subject to 10 CFR 50.34(f)), instrumentation capable of operating in the post-accident environment should be provided to monitor the containment atmosphere pressure and temperature and the sump water level and temperature following an accident. The instrumentation should have adequate range, accuracy, and response to assure that the above parameters can be tracked and recorded throughout the course of an accident. See Item II.F.1 of NUREG-0737 and NUREG-0718, and Branch Technical Position (BTP) 7-10, Guidance on Application of Regulatory Guide (RG) 1.97.

9. In accordance with 10 CFR 50.46 Appendix K, Item I.D.2, the minimum calculated containment pressure should not be less than that used in the analysis of the emergency core cooling system capability (See DSRS Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies").

10. In accordance with GDC 4, containment internal structures and system components (e.g., reactor vessel, pressurizer, steam generators) and supports should be designed to

withstand the differential pressure loadings that may be imposed as a result of pipe breaks within the containment subcompartments (See DSRS Section 6.2.1.2, "Subcompartment Analysis").

11. In meeting the requirements of 10 CFR 50.34(f)(3)(v)(A)(1), applicants subject to this section should evaluate an accident that releases hydrogen generated from a 100% fuel clad metal-water reaction. The evaluation should demonstrate that the appropriate article for service Level C limits (considering pressure and dead load only), for either concrete or steel containments, from American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code, Section III, are met. In addition to the containment pressurization caused directly by this accident, the increase in pressure from hydrogen burning in containment should be analyzed.

<u>Technical Rationale</u>

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 4 requires that the embedded subsurface containment, its penetrations, its cooling systems, and its isolation system must accommodate with no loss of function the effects of internal or external flooding events and the effects of explosion-generated missiles or chemical releases from onsite or nearby plants using reactor-generated steam or process heat.

2. GDC 5 requires that for multiple module or multiple unit plants, safety-related passive SSCs should not be shared and post-72 hour nonsafety-related, risk-significant SSCs should be demonstrated that such sharing will not significantly impair their ability to perform their risk-significant functions. These requirements apply to the containment, its penetrations, its cooling system, and its isolation system.

3. GDC 16 requires containment to be designed as a leak tight barrier that will withstand the most extreme accident conditions for the duration of any postulated accident. This DSRS section evaluates the peak pressure and temperature conditions for which the containment must be designed. The containment must be leak tight and withstand accidents because it is the final barrier against the release of radioactivity to the environment. Meeting GDC 16 provides assurance that radioactivity will not be released to the environment.

4. 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. Meeting GDC 50 will ensure that containment integrity is maintained under the most severe accident conditions thus precluding the release of radioactivity to the environment.

5. GDC 38 requires the establishment of a containment heat removal system that will rapidly reduce containment pressure and temperature following any LOCA. The containment heat removal system supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a LOCA thus lessening the challenge to containment integrity. Meeting GDC 38 will help ensure that the containment can fulfill its role as the final barrier against the release of radioactivity to the environment.

5. GDC 13 requires that instrumentation be provided to monitor all expected parameters of normal operation, anticipated operational occurrences, and accidents to assure adequate reactor safety is maintained. Since containment plays a vital safety role, appropriate instrumentation, such as temperature and pressure, must be provided so that operators can verify containment is properly fulfilling its function. RG 1.97 provides specific criteria for the design of containment instrumentation which have been found acceptable by the NRC as fulfilling GDC 13. Meeting GDC 13 and the specific guidance of RG 1.97 will help ensure that containment accomplishes its mission of precluding the release of radioactivity to the environment. BTP 7-10, “Guidance on Application of Regulatory Guide 1.97," provides the specific acceptance criteria to satisfy RG 1.97.

6. GDC 64 requires that the containment atmosphere be monitored for the release of radioactivity from normal operations, anticipated operational occurrences, and accidents. In order to ensure that the containment functions properly, operators must be aware of any radioactive releases within containment so that they can take appropriate manual action or monitor automatic action. RG 1.97 provides specific criteria for the design of containment instrumentation which have been found acceptable by the NRC as fulfilling GDC 64. BTP 7-10, “Guidance on Application of Regulatory Guide 1.97," provides the specific acceptance criteria to satisfy RG 1.97. Meeting GDC 64 and the specific guidance of RG 1.97 will assist operators in ensuring that containment meets its safety function of preventing the release of radioactivity to the environment.

7. 10 CFR 50.34(f)(3)(v)(A)(1) requires that the containment be designed to withstand either hydrogen burning or initiation of the post-accident inerting system, if installed, during an accident that releases hydrogen from a 100% fuel clad metal-water reaction. During the accident at Three Mile Island (TMI) Unit 2, metal-water reactions generated hydrogen in excess of the amounts originally anticipated. As a result of this finding, the Commission issued requirements on hydrogen control in 10 CFR 50.34(f). Other criteria require the containment to be designed to withstand postulated accidents. If such a postulated accident releases or generates hydrogen, an added containment pressurization effect beyond the initial accident may be experienced due to burning of hydrogen. The containment must be designed to withstand this additional pressure to ensure that its integrity is maintained, 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, Standard Review Plan (SRP) Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, 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 TMI 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. Upon request from the primary reviewer, the reviewer of an interfacing DSRS Section may provide input to address an area of review stated in Subsection I of this DSRS section. The primary reviewer obtains and uses such input as required to assure that this review is complete. These reviews include addressing GDC 4 in the interface reviews for an embedded subsurface containment.

4. The primary review organization reviews the containment response analyses to determine the acceptability of the calculated containment design pressure and temperature, and in addition, the containment depressurization time. The organization responsible for DSRS Section 15.0.3 must be notified if the containment depressurization time does not meet the acceptance criterion. The primary review organization for this DSRS section reviews the assumptions made in the analyses to maximize the calculated containment pressure and temperature. The primary review organization for this DSRS section determines the conservatism of the respective containment response analyses by comparing the analytical models, and the assumptions made, with the acceptance criteria in Subsection II of this DSRS section and by performing appropriate confirmatory analyses. It is not necessary to perform accident pressure calculations for every plant. The primary review organization for this DSRS section will ascertain; however, whether the adequacy of the applicant's

calculational model has been demonstrated. The primary review organization for this DSRS section determines whether the applicant has identified the pipe break(s) resulting in the highest containment pressure and temperature. Hot leg, cold leg (pump suction), and cold leg (pump discharge) interfacing systems pipe breaks that can release the reactor coolant system water, and secondary system steam and feedwater line breaks, should be analyzed by the applicant. The primary review organization for this DSRS section reviews the assumptions used to determine whether the analyses are acceptably conservative. DC applicants should meet the CP containment design pressure margin criterion of Item 1 of the DSRS Acceptance Criteria (above).

5. The primary review organization verifies that the containment is designed to withstand hydrogen burning during an accident that releases hydrogen from a 100% fuel clad metal-water reaction as described in Item 10 of the DSRS Acceptance Criteria (above).

6. The primary review organization performs confirmatory containment response analyses when necessary. The purpose of these analyses is to confirm the applicant's predictions of the response of the containment to LOCAs and main steam and feedwater line breaks. In general, only the limiting pipe breaks, i.e., the pipe breaks which establish the containment design pressure and containment depressurization time, are analyzed. However, if in the reviewer's judgment the worst break has not been identified, other pipe breaks will be analyzed. The review includes the role of nonsafety-related, risk-significant systems in post-72 hour accident containment response for multiple module units.

7. The primary review organization reviews analyses of the external pressure of the containment structure caused by pressure and temperature changes inside the containment due to inadvertent operation of containment heat removal systems. The primary review organization determines whether the most severe condition has been identified and whether the analysis was done in a conservative manner. For plants at the CP or DC stage of review the external design pressure margin should be at least 10%. For plants at the OL stage of review, the maximum expected external pressure should be less than the containment external design pressure. In general, the maximum expected external pressure should be approximately the same as at the construction permit or design certification stage of review. However, revised or upgraded analytical models or minor changes in the as-built design of the plant may result in a decrease in the margin. If the primary containment is not designed to withstand the maximum external pressure, the primary review organization will evaluate the acceptability of the provisions made in the plant design to mitigate or withstand the consequences of the above postulated events, and will evaluate in conjunction with the primary reviewer for DSRS Section 7.5, the administrative controls and instrumentation and control provisions to preclude these events.

8. The primary review organization for this DSRS section reviews the accuracy and range of the instrumentation provided to monitor the post-accident environment. The primary review organization for DSRS Section 7.5 and the primary review organization for DSRS Section 3.11 have review responsibility for the acceptability of, and the qualification test program for the sensing and actuation instrumentation of the plant protection system and the post-accident monitoring instrumentation and recording equipment.

9. For new plant applicants, the containment analyses should also consider shutdown conditions and the operator actions in multiple module units, when appropriate, to ensure that a basis is provided for procedures, instrumentation, operator response, equipment interactions, and equipment response during shutdown operations. The analyses should encompass shutdown thermodynamic states and physical

configurations to which the plant can be subjected during shutdown conditions (such as containment closure time, temperature and time to uncover the core during loss of decay heat removal).

10. 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).

11. For review of both DC and COL applications, DSRS 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 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 SER. The reviewer also states the bases for those conclusions. The conclusions reached on completion of the review of this DSRS section are presented under DSRS Section 6.2.1.

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 (ADAMS) 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. 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 Effects Design Bases."

3. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of structures, systems, and components."

4. 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and Control."

5. 10 CFR Part 50, Appendix A, GDC 16, "Containment Design."

6. 10 CFR Part 50, Appendix A, GDC 38, "Containment Heat Removal."

7. 10 CFR Part 50, Appendix A, GDC 39, "Inspection of Containment Heat Removal System."

8. 10 CFR Part 50, Appendix A, GDC 40, "Testing of Containment Heat Removal System."

9. 10 CFR Part 50, Appendix A, GDC 50, "Containment Design Basis."

10. 10 CFR Part 50, Appendix A, GDC 64, "Monitoring Radioactivity Releases."

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

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

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

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

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

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

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

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

19. "COMPARE: A Computer Program for the Transient Calculation of a System of Volumes Connected by Flowing Vents," LA NUREG-6488 MS, September 1976.

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

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

22. 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.

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

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

25. NRC SER, Babcock & Wilcox Company, Reference Safety Analysis Report, B SAR 205, May 1978.

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

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

28. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components," ASME.

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

30. 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).

31. "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.

32. "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.

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

34. Letter from Anthony C. McMurtray, USNRC, to Thomas Coutu, Site Vice President, Kewaunee Nuclear Power Plant, September 29, 2003 (ADAMS Accession No. ML0326810500).

35. "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," Volumes 1, 2 and 3, NUREG-1793, USNRC, September 2004 (ADAMS Accession No. ML043450284).

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

37. RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

38. RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design basis Accidents at Light Water Nuclear Power Reactors," May 2003 (ADAMS Accession No. ML020160023).

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

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