Proposed - For Interim Use and Comment

U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.2.8 FEEDWATER SYSTEM PIPE BREAK INSIDE AND OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses for iPWRs

Secondary - None

I. AREAS OF REVIEW

The steam and water release from a postulated feedwater line break results in a loss of secondary coolant which may result in a reactor system cool-down (by excessive energy discharge through the break) or a reactor system heat-up (from the loss of reactor system heat sink). A major feedwater line rupture is defined as a feedwater line break large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell side fluid inventory in the steam generators. If the break is postulated in the feedwater line between the isolation valves and the steam generator, fluid from the steam generator also is discharged from the break. (A break upstream of the feedwater isolation valves would affect the reactor system only as a loss of feedwater. This case is covered by Design-Specific Review Standard (DSRS) Section 15.2.7, “Loss of Normal Feedwater Flow”.)

The specific areas of review are as follows:

1. Evaluation of the applicant's postulated initial core and reactor conditions pertinent to the feedwater line break.

   The results of the analyses are reviewed for whether the values of pertinent system parameters, addressed in subsection II of this DSRS section, are within expected ranges. The parameters of importance for these transients include:

   A. reactor coolant system (RCS) pressure,
   B. steam generator pressure,
   C. fluid temperatures,
   D. fuel and clad temperatures,
   E. break discharge flow rate,
   F. steamline and feedwater flow rates,
   G. safety and relief valve flow rates,
   H. pressurizer and steam generator water levels,
   I. mass and energy transfer within the containment (for breaks inside containment),
   J. reactor power,
   K. total core reactivity,

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L. hot and average channel heat flux,
M. minimum departure from nucleate boiling ratio (DNBR)
N. core flow rate, and
O. decay heat removal rate.

2. Methods of thermal and hydraulic analysis, the postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior and subsequent to initiation of reactor protection system (RPS) actions.

   The analytical thermal/hydraulic methods are reviewed for whether the mathematical modeling and computer codes have been reviewed and accepted by the staff. If a referenced analytical method has not been reviewed, the reviewer requests an evaluation of the new analytical model. The parameter values in the analytical model, the initial conditions of the core, and all nuclear design parameters are reviewed. This review includes:

   A. power level,
   B. power distribution,
   C. Doppler reactivity feedback,
   D. moderator temperature reactivity feedback,
   E. void reactivity feedback,
   F. reactor kinetics,
   G. departure from nucleate boiling correlations, and
   H. control rod worth.

3. The response of the reactor coolant and auxiliary systems, the functional and operational characteristics of RPS effects on the sequence of events, and all operator actions required to secure and maintain the reactor in a safe shutdown condition.

   The sequence of events described in the applicant's safety analysis report (SAR) is reviewed for the performance of the RPS, the engineered safety systems, and operator actions to secure and maintain the reactor in a safe condition.

4. The ECCS, including the depressurization components, is reviewed for whether the natural circulation flow is acceptable for transient control following a feedwater line break.

5. Combined Operating License (COL) Action Items and Certification Requirements and Restrictions. For a Design Certification (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. General information on transient and accident analyses is provided in DSRS Section 15.0.

2. Design basis radiological consequence analyses associated with design basis accidents are reviewed under DSRS Section 15.0.3.

3. Effects of blow-down loads, including jet propulsion piping and component supports and the design bases for safety and relief valves are reviewed under DSRS Sections 3.6.2 and 3.9.1 through 3.9.3. Design bases for safety and relief valves are also reviewed under DSRS Section 3.9.3.

4. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the SAR analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

5. Fracture toughness properties of the reactor coolant pressure boundary and reactor vessel are reviewed under Sections 5.2.3 and 5.3.1.

6. The response of the containment to feedwater line ruptures as to the effects of pressure and temperature on the containment functional capabilities is reviewed under DSRS Section 6.2.1. Analytical methods for deriving mass energy releases exiting the postulated break are reviewed under DSRS Section 6.2.1.3.

7. The sequence of events are reviewed with respect to the reactor system and its interfaces with instrumentation and control systems. Aspects of the sequence described in the SAR are reviewed to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. This review includes the instruments and controls required to ensure automatic and manual emergency core cooling system (ECCS) initiation and flow indication in the control room and is performed under DSRS Sections 7.1 through 7.7. The potential bypass modes and the possibility of manual control by the operator are also reviewed under DSRS Sections 7.1 through 7.7.

8. The ECCS is reviewed to verify its ability to function following a steam line break given a single active component failure with either onsite or offsite power under DSRS Section 6.3.

9. Fission product release assumptions for determining any offsite releases are evaluated and radiological consequences from a feedwater pipe break are verified as within acceptable limits are reviewed under DSRS 15.6.5.

10. The determination of the safety-related and risk significance of structures, systems, and components (SSCs) relied upon to meet required functions during the accidents are

based on the review of the probabilistic risk analysis under Standard Review Plan (SRP) Chapter 19.

II. ACCEPTANCE CRITERIA

Requirements

The basic objective of the review of feedwater system pipe break events is to confirm that the reactor primary system is maintained in a safe status for break sizes up to and including a break equivalent in area to the double-ended rupture of the largest feedwater line.

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

1. General Design Criterion (GDC) 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 17, as to onsite and offsite electric power systems for safety-related structures, systems, and components (SSCs) to function. The safety function for each power system (assuming the other system is not functioning) must be of sufficient capacity and capability so design conditions of the reactor coolant pressure boundary are not exceeded and the core is cooled in postulated accidents.

3. GDCs 27 and 28, as to the RCS design with appropriate margin so acceptable fuel design limits are not exceeded and core cooling capability is maintained.

4. GDC 31, as to RCS design with sufficient margin so the boundary is nonbrittle and the probability of fracture propagation is minimized.

5. GDC 35, as to design of the RCS and its auxiliaries for abundant emergency core cooling.

6. 10 CFR Part 100, as to calculated doses at the site boundary.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission (NRC's) regulations identified above are set forth below. 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." The same approach may be used to meet the requirements of Title 10 of *Code of Federal Regulations* (10 CFR) 10 CFR 52.79(a)(41) for COL applications.

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.

2. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model (see DSRS Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

3. Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 guidelines.

4. The ECCS must be safety grade and automatically initiated when required.

5. Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures:

   A. The power level assumed at the initiation of the transient should correspond to the operating condition which maximizes accident consequences. The assumed initial conditions vary with the particular nuclear steam supply system and sensitivity studies are required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report as references if applicable.

   B. The assumptions as to whether offsite power is lost and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost. A study should determine the most conservative assumption appropriate to the plant design reviewed. The study should take account of the effects that loss of offsite power (LOOP) has on reactor coolant and main feedwater pump trips and on the initiation of the ECCS and the consequent modification of the sequence of events.

   C. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of Branch Technical Positions (BTP) 3-3 and BTP 3-4.

   D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; feedwater pipe breaks should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure. (This position is based upon interpretation of GDC 17 as documented in the final safety evaluation report for the ABB-CE System 80+ DC.)

E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.

F. The core burn-up (time in core life) should be selected to yield the most limiting combination of moderator temperature reactivity feedback, void reactivity feedback, Doppler reactivity feedback, axial power profile, and radial power distribution.

G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin is the result for a feedwater line rupture inside containment; however, this assumption may not be the most conservative. For example, maximum initial core flow increases RCS cool-down and depressurization, decreases shutdown margin, and increases the possibility that the core will become critical and return to power. As it is not clear which initial core flow is most conservative, the applicant's assumption should be justified by appropriate sensitivity studies.

H. During the initial 10 minutes of the transient, if credit for operator action is required (i.e., reactor coolant pump (RCP) trip), an assessment for the limiting consequence must account for operator delay and/or error.

Programmatic Requirements: The NRC regulations require that each operating license contain a technical specification (TS) that define "…the limits, operating conditions, and other requirements imposed upon facility operation for the protection of public health and safety…" The licensee's analysis of DSRS 15.2.8 must be consistent with the information presented in the licensee's TS.

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. 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, 10 CFR Part 100 radiation exposure criteria provide reference values for the site suitability determination based on postulated fission product releases from accidental events.

   10 CFR Part 100 applies to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events like loss of an RCP. For transients with moderate frequencies of occurrence, the calculated doses at the site boundary from any release of radioactive material must be a small fraction, less than 10 percent, of the 10 CFR Part 100 guidelines. For purposes of this review, consideration of the radiological consequences of any feedwater system pipe break must include the containment, confinement, and filtering systems. The applicant's source terms and methodologies as to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequences of events, including automatic actuations of protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. GDC 17 requires onsite and offsite electric power systems to permit functioning of SSCs important to safety. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (A) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences (AOOs) and (B) the core is cooled and containment and other vital functions are maintained in postulated accidents.

   GDC 17 applies because review under this section covers feedwater system pipe breaks, which can be classed as AOOs or accidents, depending upon severity.

4. GDC 27 requires reactivity control systems designed with a combined capability, with poison added by the ECCS, to control reactivity changes reliably to maintain core cooling capability under postulated accident conditions with appropriate margin for stuck rods.

   GDC 28 requires reactivity control systems designed with appropriate limits on the potential amount and rate of reactivity increase so the effects of postulated reactivity accidents neither (A) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (B) disturb the core, its support structures, or other reactor pressure vessel internals sufficiently to impair the core cooling capability significantly. These postulated reactivity accidents must include consideration of rod ejection (unless prevented by positive means), steam line rupture, reactor temperature and pressure changes, and cold water addition.

   GDCs 27 and 28 apply because this DSRS section is for the review of feedwater system pipe breaks inside and outside containment that can result in transient conditions affecting reactor coolant temperature and pressure with consequent changes in core reactivity. The SAR analyses of these transients must demonstrate that reactivity, pressure, and temperature changes will not be severe enough for an unacceptable impact on the reactor coolant pressure boundary or on core cooling capability. The analyses must be reviewed by the staff independently in accordance with this DSRS section.

5. GDC 31 requires reactor pressure boundary design with sufficient margin to ensure that, when stressed under operation, maintenance, test, and postulated accident conditions, the boundary is nonbrittle and the probability of rapidly propagating fracture is minimal. The design must reflect consideration of service temperatures and other conditions of the boundary material under operation, maintenance, test, and postulated accident conditions and the uncertainties in determining material properties; effects of irradiation on material properties; residual, steady state, and transient stresses; and flaw sizes.

GDC 31 applies because this DSRS section is for the review of feedwater system pipe breaks inside and outside containment that could result in transient reactor coolant temperature and pressure conditions that could affect the reactor coolant pressure boundary adversely. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the steam generator and by the subsequent addition of decay heat could result in undue stress on the RCS pressure boundary. The amount of stress to which the reactor coolant pressure boundary is subjected depends upon AOO severity, which is assessed in the SAR and reviewed by the staff in accordance with this DSRS section.

6. GDC 35 requires a system for abundant emergency core cooling. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts.

   GDC 35 applies because this DSRS section is for the review of feedwater system pipe breaks both inside and outside containment that could result in transient reactor coolant temperature conditions that could challenge the ECCS. A feedwater system pipe break could result in either an RCS cool-down by excessive energy discharge through the break or an RCS heat-up by reduced feedwater flow to the steam generator. Heat-up of the reactor coolant by reduced feedwater flow to the steam generator and by the subsequent addition of decay heat could initiate ECCS reduction of the core coolant temperature to an acceptable level to prevent fuel and clad damage that could interfere with continued effective core cooling and limit fuel clad metal-water reaction to negligible amounts. The severity of this AOO is assessed in the SAR and reviewed by the staff in accordance with this DSRS section.

III. REVIEW PROCEDURES

The review procedures described below 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 procedures are used during reviews of construction permit, operating license, and COL applications. During the construction permit review the values of system parameters and setpoints in the analysis are preliminary in nature and subject to change. At the operating license or COL review stage, final values should be in the analysis, and the reviewer should compare these to the limiting safety system settings in the proposed technical specifications.

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, 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. TS (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 (Regulatory Guide 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). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively. These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding SER section.

3. The values of system parameters and initial core and system conditions as input to the model are reviewed and compared to the initial conditions listed in Subsection I of this DSRS section. Of particular importance are the reactivity feedbacks and control rod worths in the applicant's analysis and the variation of moderator temperature, void, and Doppler reactivity feedback with core life. The applicant’s justification for selection of the core burn-up yielding the minimum margins is evaluated. Reactivity parameter values in the applicant's analysis also are reviewed.

4. Analytical models should be of sufficient detail to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The applicant’s equations, sensitivity studies, proposed models, and justification for methods as conservative compared to appropriate test data are reviewed. The pressurizer is of particular importance in the modeling of the over-pressure transient, the likely result of a large

feedwater line break. Assumptions for pressurizer spray performance if credited should be reviewed as well as heat transfer by condensation within the pressurizer steam space. Test data examples which might be useful in validation of pressurizer models are in ”The Pressure Response of a pressurized-water reactor Pressurizer During an Insurge Transient,” Transactions of the American Nuclear Society, 1983 Annual Meeting, Detroit, MI, June 12-16, 1983.

5. Credit taken for a reactor trip signal or for ECCS actuation should be reviewed for the ability of the instrumentation and control systems to respond as assumed under accident conditions.

6. The ECCS ability to supply adequate feedwater flow to the unaffected steam generators during the accident and subsequent shutdown is evaluated as to availability and capability to affect an orderly shutdown. As ECCS designs are diverse and may require both automatic and manual actuation, pre-operational tests should be specified for any necessary operator actions and for the maximum times for their completion.

   To the extent necessary, the reviewer evaluates the effect of system and component single, active failures that may alter the course of the accident. For new applications, the LOOP is not a single, active failure but an addition to a single, active failure as addressed in subsection II.5.D of this DSRS section. This phase of the review uses the system review procedures described in the DSRS sections for SAR Chapters 5, 6, 7, 8, and 10. During the transient the variations with time of parameter listed in Sections 15.X.X.3(C) and 15.X.X.4(C) of the Standard Format, Regulatory Guide 1.70, are reviewed. The more important of these parameters for the feedwater line break accident (as listed in Subsection I of this DSRS section) are compared to those predicted for other similar plants for whether they are within the expected range.

7. The reviewer confirms that the amount of secondary coolant expelled from the system is calculated conservatively by evaluation of the applicant's methods and assumptions, by comparison with an acceptable analysis on another plant of similar design, or by comparison with staff calculations.

` The reviewer confirms an SAR commitment to conduct pre-operational tests to verify that valve discharge rates and response times (*e.g.*, opening and closing times (delay times) for main feedwater, the ECCS, turbine and main steam isolation, and steam generator, pressurizer relief, and safety valves) are modeled conservatively in the accident analyses. In addition, pre-operational testing should include verification of reactor trip delay times, startup delay times for ECCS actuation, safety injection signal delay time, and delay times for delivery of any high-concentration boron injection required to bring the plant to a safe shutdown condition.

8. Using the information developed in the review, the reviewer evaluates the radiological consequences of the design-basis feedwater line break. This evaluation is based on a qualitative comparison to the results of the design-basis steam line break or on a detailed analysis using the approach described in the DSRS Section 15.0.3.

8. 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 final safety analysis report (FSAR) meets the

acceptance criteria. DCs have referred to the FSAR as 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 DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

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 safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the applicant''s analysis of consequences of postulated feedwater line breaks meets the requirements of GDC 13, 17, 27, 28, 31, and 35 for ability to insert control rods and ability to cool the core, 10 CFR Part 100 guidelines for radiological doses at the site boundary, and applicable Three Mile Island Action Plan Items. This conclusion is based upon the following findings:

1. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

2. The applicant meets GDC 27 and 28 requirements by demonstrating minimal fuel damage, maintained ability to insert the control rod, and no loss of core cooling capability. The minimum DNBR for any fuel rod was _______ with the result of __ percent of the rods experiencing clad perforation.

3. The applicant meets GDC 31 requirements for demonstrating primary system boundary capability to withstand the postulated accident.

4. The applicant meets GDC 35 requirements for demonstrating emergency cooling system adequacy for abundant core cooling and reactivity control (via boron injection).

5. The analyses of effects of feedwater line break accidents inside and outside containment during various modes of operation with and without offsite power have been reviewed and evaluated by a mathematical model previously reviewed and found acceptable by the staff.

6. The input parameters for this model were reviewed and found suitably conservative.

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.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific DC, or 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™ 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 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, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application." The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ 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), and COL applications.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

    A. GDC 13, "Instrumentation and Control."
    B. GDC 17, "Electric Power Systems."
    C. GDC 27, "Combined Reactivity Control System Capability."
    D. GDC 28, "Reactivity Limits."
    E. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."
    F. GDC 35, "Emergency Core Cooling."

2. 10 CFR Part 100, "Reactor Site Criteria."

3. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

4. BTP 3-3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

5. BTP 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."

6. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," Article NB-7000, "Protection Against Over pressure," American Society of Mechanical Engineers.