U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (mPower™ iPWR)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of transient and accident analyses for mPower™ iPWR

Secondary - None

I. AREAS OF REVIEW

The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system (RCS) and results in a reduction of coolant temperature and pressure. The negative moderator temperature reactivity feedback and the cooldown of the reactor system cause an increase in core reactivity. The core reactivity increase may cause a loss of reactor core shutdown margin and a resulting increase in reactor power. If the plant is at power, the reactor and the reactor coolant pumps (RCPs) are automatically tripped and the steam generator is isolated. The emergency core cooling (ECC) system automatically activates and decay heat is removed from the RCS to the ultimate heat sink.

Analysis of the transient following a steam line break is sensitive to the fluid discharge rate at the break so that a range of break sizes must be considered both inside and outside containment to determine the acceptability of the system response. The course that the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (e.g., hot shutdown; full power; number of RCPs in operation). The most limiting scenario needs to be evaluated regarding the number of RCPs in operation. Evaluation with various assumed initial conditions is required to verify that the condition leading to the severest consequences has been identified.

The specific areas of review are as follows:

1. Postulated initial core and reactor conditions pertinent to the steam line break accident;

2. Methods of thermal and hydraulic analyses, including the effects of hydraulic instabilities;

3. Postulated sequence of events, including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system;

4. Assumed responses of the reactor coolant and auxiliary systems;

5. Functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events;

ADAMS Accession No. ML12207A108

6. Operator actions required to secure and maintain the reactor in a safe-shutdown condition;

7. Core power excursion due to power demand created by excessive steam flow out of the break; and

8. Variables influencing neutronics.

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include:

A. RCS pressure,
B. Steam generator pressure,
C. Fluid temperatures,
D. Clad temperatures,
E. Discharge flow rates,
F. Steam line and feedwater flow rates,
G. Safety and relief valve flow rates,
H. Pressurizer and steam generator water levels,
I. Reactor power,
J. Total core reactivity,
K. Hot and average channel heat flux, and
L. Minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's technical submittal is reviewed by the branches that review reactor systems and instrumentation and control. The reviewer from the branch that reviews reactor systems concentrates on the capability of the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed by the branch that reviews reactor systems to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, Reactor Systems initiates an evaluation of the new analytical model.

9. <u>Combined License (COL) Action Items and Certification Requirements and Restrictions</u>. 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.

<u>Review Interfaces</u>

Other design-specific review standard (DSRS) sections interface with this section as follows:

1. DSRS Section 15.0. General information on transient and accident analyses is provided in DSRS Section 15.0.

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

3. DSRS Sections 4.2, 4.3 and 4.4. 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 technical submittal analysis are reviewed under DSRS Sections 4.2, 4.3 and 4.4.

4. DSRS Section 3.9.3. The ECC, CNX and reactor coolant inventory (RCI) systems are reviewed to verify their ability to function following a steam line break given a single active component failure with either onsite or offsite power under DSRS Section 10.4.9. 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.

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

6. DSRS Section 6.2.1.3. The response of the containment to ruptures of steam lines with regard 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. DSRS Sections 7.1 through 7.7. Aspects of the sequence described in the technical submittal 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 ECC, CNX or RCI 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.

II. ACCEPTANCE CRITERIA

Requirements

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

The general objective of the review of steam line rupture events is to verify that short-term and long-term ability to cool the core has been achieved by confirming that the primary RCS is maintained in a safe status for a break equivalent in area to the double-ended rupture of the largest steam line. The acceptance criteria are based on meeting the relevant requirements of the following 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 it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of structures, systems, and components (SSCs) important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence (AOO) and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

3. GDCs 27 and 28, as they relate to the RCS being designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

4. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

5. GDC 35, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant ECC.

Requirements for ensuring adequate decay heat removal and RCP Integrity and operation are specified in Title of the *Code of Federal Regulations* (CFR), Section 50.34(f)(2)(xii)[1] and 10 CFR 50.34(f)(1)(iii), respectively.

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the 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 10 CFR 52.79(a)(41) for COL applications.

Specific criteria necessary to meet the relevant requirements of the above regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle, as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for pressurized-water reactors (PWRs) 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 that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see DSRS Section 4.2), which includes the potential adverse effects of hydraulic

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

instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in DSRS Section 15.0.3.

4. The integrity of the RCPs should be maintained such that loss of alternating current (ac) power and containment isolation will not result in pump seal damage.

5. System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. In the case of mPower™, the active CNX and RCI systems, the passive ECC system, provides the safety-related means of decay heat removal.

6. Tripping of the RCPs should be consistent with the resolution to Task Action Plan, Item II.K.3.5.

There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

1. The reactor power level, number of operating RCPs and other plant operating mode parameters assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. Sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the technical submittal.

2. Assumptions as to the loss of offsite power (LOOP) and the time of loss should be made to study their effects on the consequences of the accident. A LOOP may occur simultaneously with the pipe break or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The reviewer should note that the assumption that offsite power is not lost may maximize heat removal from the core and reactor system and thereby maximize containment pressure and reactivity feedback within the core. The analyses should take account of the effect that LOOP has on RCP and main feedwater pump trips on the initiation of the decay heat removal system such as CNX, RCI and ECC system, and the effects on the sequence of events for these accidents. For new applications, LOOP should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Analysis Report (FSAR) for the ABB-CE System 80+ DC.)

3. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Position (BTP) 3-3 and BTP 3-4.

4. The worst single active component failure should be assumed to occur. For new applications, LOOP should not be considered as a single failure, (see Assumption 2 above). The assumed single failure may cause the steam generator to blow down, failure of main feedwater to isolate, or may be in any of the systems required to control the transient.

6. The maximum-worth control rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used. Local power peaking at the location of the stuck out control rod should be considered.

Local power peaking will affect the DNBR analysis in the initial period as the safety rods are entering the core and during any subsequent return to power resulting from reactivity addition to the core from the cooldown.

6. The core burnup (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.

7. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results; however, for the analysis of steam line break accidents, this may not be the most conservative assumption. For example, maximum initial core flow results in increased RCS cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified.

8. Failure of a steam line may cause asymmetric temperatures within the reactor core. Asymmetric core temperatures will affect the local power distribution and the DNBR analysis. Assumptions for mixing in the downcomer and the reactor vessel lower plenum will affect the predicted core temperature distributions, reactivity feedback and local power. Assumptions for mixing should be chosen so as to be conservative for predicting maximum local core power and DNBR.

9. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the main steam isolation valves (MSIVs) due to a seismically initiated event, only safety-related equipment should be assumed operative to mitigate the consequences of the break.

10. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety-related equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup nonsafety-related component to mitigate the consequences of the break.

11. During the initial 10 minutes of the transient, should credit for operator action be required (e.g., RCP trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.

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. Compliance with 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 sequence 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.

2. Compliance with GDC 17 requires (in part) that onsite and offsite electrical power systems be provided to ensure the functioning of SSCs important to safety. The safety function for each power system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (1) the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs and (2) the core is cooled and containment integrity and other vital functions are maintained.

   GDC 17 is applicable to this section because it requires that the LOOP be considered not as a single failure event, but assumed in the analyses for each event without changing the event category. Thus, the applicant is required to consider a LOOP concurrent with a single failure in the analysis of steam system piping failures.

3. Compliance with GDC 27 requires that reactivity control systems be designed to have a combined capability (in conjunction with poison added by the ECC system) of reliably controlling reactivity changes, thereby ensuring that the capability for core cooling is maintained under postulated accident conditions and with appropriate margin for stuck rods.

   Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and addition of cold water.

   GDCs 27 and 28 are applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment that could cause transient conditions affecting reactor coolant temperature and pressure, including complex changes in core reactivity. The applicant's analyses of these transients in the technical submittal must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for cooling the core. These analyses must be independently reviewed by the staff in accordance with this DSRS section.

4. Compliance with GDC 31 requires that, under the stress of operation, maintenance, testing, and postulated accidents, the reactor pressure boundary shall be designed with sufficient margin to ensure that (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other boundary material variables under a full-range of conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

   GDC 31 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment that could cause transient conditions with a potentially harmful effect on the reactor coolant pressure boundary. A steam system piping break can result in a rapid decrease in reactor coolant temperature and steam generator pressure, placing undue stress on the reactor coolant pressure boundary. This potential problem could be aggravated by a pressurization of the primary system when the ECC system is activated. The amount of stress to the reactor coolant pressure boundary depends on the severity of the transient. The severity of the transient is assessed by the applicant in the technical submittal and is reviewed by the staff in accordance with this DSRS section.

5. Compliance with GDC 35 requires a system that will provide abundant ECC. The system safety function shall be to transfer heat from the reactor core after any loss of reactor coolant at a rate ensuring that (1) fuel and clad damage interfering with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

   GDC 35 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment that could cause transient conditions with the potential to challenge the ECC system. During a steam system piping break, excessive steam loss will result in a rapid reduction of reactor coolant temperature and steam generator pressure. A subsequent reactor trip can further reduce the primary system pressure, producing a void within the pressure vessel and creating the need for ECC.

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 procedures below are used during the construction permit (CP), operating license (OL), and COL reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL or COL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included 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, Standard Review Plan (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 (TMI) 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 safety evaluation report (SER) section.

3. The reviewer determines the acceptability of the analytical models and assumptions, as follows:

A. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by the branch that reviews reactor systems. Of particular importance is (1) the reactivity feedback and control rod worths used in the analysis and (2) the variation of moderator temperature, void and Doppler reactivity feedback with core life. The reviewer will evaluate the justification provided by the applicant to show that the core burnup yielding the minimum margins has been selected. The branch reviewing core performance reviews core-related parameters such as DNBR correlations and the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating the methods and assumptions, by comparing these results with those of an acceptable analysis performed on another plant of similar design, or by comparing the results with staff calculations.

B. The acceptability of the methods equations, sensitivity studies, and models proposed by the applicant are evaluated.

C. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

i. Reactor trip signal: credit taken for any reactor trip signal is reviewed by the branch that reviews instrumentation and control to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

ii. ECC system: credit taken for actuation of the ECC is reviewed by the branch that reviews instrument and control systems to verify the ability of the instrumentation and control systems to respond as assumed.

iii. ECC system: the availability of the ECCS to remove heat from the primary system and to supply adequate makeup flow to the reactor vessel during the accident and the subsequent shutdown condition is evaluated. This is done by the branch reviewing plant systems as to availability of the system and by the branch reviewing reactor systems as to capability to affect an orderly shutdown. Since ECCS designs may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion. In the case of mPower™ the safety-related functions of decay heat removal and reactor vessel makeup flow are performed by the RCI, CNX and ECC.

iv. Auxiliary Condenser System (CNX) and Reactor Coolant Inventory and Purification System (RCI): the credit taken for actuation of the CNX and RCI is reviewed by the branch that reviews instrument and control systems to verify the ability of the instrumentation and control systems to respond as assumed. Since these systems may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion.

D. Time-related variations of the following parameters are reviewed:

i. reactor power;

ii. heat fluxes (average and maximum);

iii. total core reactivity;

iv. RCS pressure;

v. minimum DNBR;

vi. coolant conditions (inlet temperature core average temperature and average exit and hot channel exit temperatures;

vii. fuel rod conditions (maximum fuel center–line temperature, maximum clad temperature, or maximum fuel enthalpy);

viii. steam generator pressure;

ix. containment pressure;

x. relief and/or safety valve flow rates;

xi. discharge flow rate;

xii. steam line and feedwater flow rates; and

xiii. pressurizer and steam generator water levels.

The values of the more important of these parameters for the steam line break accident (as listed in Subsection I) are compared with those predicted for other similar plants to see that they are within the range expected.

1. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. For new applications, LOOP should not be treated as a single active failure, as discussed under Subsection II, Assumptions 2 and 4. This phase of the review is done using the system review procedures described in the DSRS sections for Chapters 5, 6, 7, 8, and 10 of the technical submittal. The reviewer also considers single failures that may cause more than one steam generator to blow down or failure of main feedwater to isolate, thus increasing the reactivity addition to the core.

2. The reviewer confirms that a commitment has been made in the technical submittal to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for CNX, RCI or ECC actuation, safety injection signal delay time, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe-shutdown condition. The reviewer should note that for mPower the safety-related means of decay heat removal is the ECC rather than the auxiliary feedwater system.

3. Based on the above information, the branch that reviews radiological consequences of design basis accidents evaluates the radiological consequences of the design-basis steam line break accident as described in the DSRS Section 15.0.3.

4. Upon request from the primary reviewer, other secondary reviewers will provide input for the areas of review stated in Subsection I of this DSRS section. The primary reviewer obtains and uses such input as required to ensure that this review procedure is complete.

5. The reliability and integrity of the RCP seals during loss of ac power and loss of coolant to the seals (e.g., resulting from containment isolation) are reviewed to ensure compliance with the requirements of 10 CFR 50.34(f)(1)(iii).

6. 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 technical submittal meets the acceptance criteria. DCs have referred to the technical submittal 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 technical submittal.

   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 SER. The reviewer also states the bases for those conclusions.

The staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in the GDCs 13, 17, 27, 28, 31, and 35 regarding (1) the ability to insert the control rods and to cool the core and (2) TMI Action Plan items. This conclusion is based upon the following:

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 has met the requirements of GDCs 27 and 28 by demonstrating that the resultant fuel damage was limited such that the ability to insert control rods would be maintained and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was _________, resulting in __% of the rods experiencing cladding perforation.

3. The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

4. The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

5 The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power (as required by GDC 17), have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

6. The parameters used as input to this model were reviewed and found to be suitably conservative.

7. The applicant has met the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) with respect to demonstrating the adequacy of the design of the ECC or other qualified systems to remove decay heat following steam system piping failures.

8. The applicant has met the requirements of 10 CFR 50.34(f)(1)(iii) with respect to demonstrating the integrity and operation of the RCPs to withstand the postulated accident.

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 (Agencywide Documents Access and Management System Accession No. ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews, including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower™-specific DC, 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, GDC 13, "Instrumentation and Control."

2. 10 CFR Part 50, GDC 17, "Electric Power Systems."

3. 10 CFR Part 50, GDC 27, "Combined Reactivity Control Systems Capability."

4. 10 CFR Part 50, GDC 28, "Reactivity Limits."

5. 10 CFR Part 50, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

6. 10 CFR Part 50, GDC 35, "Emergency Core Cooling."

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

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