U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER^{TM} iPWR DESIGN

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of pressurized-water reactor systems

Secondary - Organization responsible for the review of Containment

I. AREAS OF REVIEW

The specific areas of review are as follows:

1. Loss-of-coolant accidents (LOCAs) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the normal reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. An additional safety concern is the potential for build-up of boric acid due to coolant vaporization in a pressurized water reactor (PWR). If left uncontrolled, the boron concentration may reach precipitation limits and block the coolant channels in the core, preventing heat removal for any size break. While mPower does not use soluble boron as a reactivity control element during normal operation, a boron solution may be injected from the nitrogen-pressurized emergency boron tank into the reactor vessel during a LOCA.

In mPower^{TM}, the emergency core cooling safety functions are divided into four functional categories: Reactor Coolant System (RCS) Automatic Depressurization, Passive Core Cooling Function, Emergency Decay Heat Removal, and Long Term Core Cooling.

General Design Criterion (GDC) 35 requires each iPWR to be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS given in 10 CFR 50.46 and Appendix K to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 and the applicable general design requirements discussed in Design-Specific Review Standard (DSRS) Section 6.3. The analysis of ECCS performance has an impact on the design of the piping and support structures for the reactor coolant system, the design of the steam generator, the containment design, and the possible need for pump overspeed protection.

The review of the applicant's analysis of the spectrum of postulated loss-of-coolant accidents is closely associated with the review of the ECCS, as described in DSRS Section 6.3. As a

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portion of the review effort described in this DSRS section and in DSRS Section 6.3, the reviewer evaluates whether the entire break spectrum (break size and location) has been addressed; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on the ECCS performance have been provided. For postulated break sizes and locations, the staff reviews the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protection and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

A spectrum small break LOCAs is to be evaluated and the limiting break identified through sufficient analyses to determine the worst break peak clad temperature (PCT), the worst local clad oxidation, and the highest core wide oxidation percentage. The small break spectrum should have sufficient resolution to locate these limiting conditions. In the analysis of small breaks, evaluating integer diameter break sizes (i.e., 1, 2, 3, 4-inch, etc.) may be insufficient to determine the worst break because the break areas associated with these integer diameters may be too coarse to adequately identify the highest PCT. The analyses must also be carried out until the top of the active fuel has been recovered with a two-phase mixture and the cladding temperatures have been reduced to temperatures near the saturation temperature. The analyses must also consider the case with a severed ECC injection line, along with the degraded ECC injection into the reactor vessel. Break locations should include various locations around the letdown and charging piping to and from the reactor coolant inventory and purification system (RCI). If operator action is required to maintain conditions within 10 CFR 50.46 limits, then the equipment and operator action times to achieve a successful core cooling condition should also be identified.

An evaluation of post-LOCA long term cooling should also be performed to identify the operator actions to successfully control and prevent boric acid precipitation. Analyses of small break LOCAs should be performed to identify the timing for boric acid precipitation. The timing for the switch to simultaneous injection for breaks should be identified using acceptable analysis methods. A spectrum of small breaks should also be analyzed to identify other means to control boric acid precipitation when RCS pressure remains too high to enable flushing of the core through a simultaneous injection line-up during the long term. All equipment and operator action times should also be clearly identified in the analyses.

The calculational framework used for the evaluation of the ECCS system in terms of core short term behavior and long term cooling performance are referred to as an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS given in 10 CFR 50.46. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, reactor coolant system letdown and charging (RCI) piping and ECCS piping and support structure design), all differences should be identified and described by the applicant. The reviewer evaluates these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability. The reviewer initiates a generic computer program review as required.

The staff review of this DSRS section covers the following areas:

A. The failure mode analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided in conjunction with the effort described in DSRS Section 6.3.

B. The analytical techniques and computer programs used by the applicant for the blowdown, depressurization, gravity-drain refill, and reflood portions of the loss-of-coolant transient.

C. The analytical techniques and computer programs used by the applicant for power transient calculations (including moderator temperature, void and fuel temperature reactivity feedback effects, and decay heat) and for the cladding temperature, cladding rupture and swelling calculations.

D. Independent audit calculations of the blowdown, depressurization, gravity-drain phases of the accident, and cladding heatup calculations, as required to verify the applicant's conclusions.

E. Verification that the core physics data used by the applicant, or by the staff in independent audit analyses, is the appropriate data to be used.

F. The results of the small break post-LOCA long term cooling analyses and assures that an acceptable model has been employed to identify the timing for boric acid precipitation for breaks and an adequate procedure has been devised to control boric acid precipitation for all small breaks that cannot successfully employ simultaneous injection to assure long term cooling.

The reviewer provides an evaluation of fission product releases and radiological consequences. For applications under 10 CFR Part 52, this effort is described in DSRS Section 15.0.3,"Radiological Consequences of Design Basis Accidents - for early site permit (ESP), design certification (DC) and combined license (COL) Applications."

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

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

<u>Review Interfaces</u>

Other DSRS sections interface with this section as follows:

1. General information on transient and accident analyses is provided in DSRS Section 5.0.

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

3. Fuel failure modes and burst correlations are evaluated for compliance with 10 CFR 50.46 as part of its fuel design review under DSRS Section 4.2.

4. Evaluation of the functional capability of the containment for the spectrum of loss-of-coolant events is performed under DSRS Section 6.2.1. The reviewer verifies that the assumptions used for the containment response analysis have been selected in a conservative manner for the LOCA analysis performed, the containment pressure calculations utilized by the applicant, or by the staff in an audit analysis, for the reflood portion of the ECCS performance analyses.

5. Aspects of the transient sequences described in the applicant’s technical submittal are evaluated to determine 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 under DSRS Sections 7.2 and 7.3. The reviewer evaluates the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS instrumentation and controls equipment and the effect of the failure modes of that equipment on the ECCS performance has been provided.

6. Staff evaluates the emergency onsite power functional capabilities described in DSRS Sections 8.3.1 and 8.3.2 Staff also verifies that the control system power sources needed to function to mitigate the event are available as required by the applicant's description of the event. The reviewer evaluates the failure modes analysis of the ECCS to verify that an adequate analysis of possible failure modes of ECCS equipment and the effect of the failure modes on the ECCS performance has been provided.

7. Evaluation of auxiliary systems (e.g. RCI, component cooling water system, chilled water system, ultimate heat sink and condensate storage facility) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a LOCA is reviewed in the applicable DSRS Sections in Chapter 9 and 10. Integrity of the reactor coolant pump seals is performed under DSRS Section 9.2.2.

8. Effects of the combined blowdown and seismic loads on core support structures and on control rod guide structures under DSRS Sections 3.6.2, 3.9.2, 3.9.3, 3.9.4, and 3.9.5. The reviewer verifies that the core remains in a coolable geometry following a LOCA and that the control rods can also be inserted for breaks crediting this function. Analyses of the deformed bundle in the core should be performed to show that the acceptance criteria of 10 CFR 50.46 are met. The staff verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures of the reactor coolant pressure boundary and engineered safety feature equipment in the event of a LOCA.

9. Plant operating procedures are reviewed under DSRS Section 13.5.2.1. The procedures are reviewed to verify that they include actions relative to reactor coolant pump trip following LOCAs that are based on plant-specific safety evaluations.

10. The determination of the 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

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

1. 10 CFR 50.46 as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

2. 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.

3. GDC 35 as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts. The analyses should reflect that the ECCS has suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities available such that the safety functions could be accomplished assuming a single failure. In addition, consideration should be given to the availability of onsite power (assuming offsite electric power is not available with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available).

4. 10 CFR 100 or 10 CFR 50.67 as they relate to mitigating the radiological consequences of an accident.

DSRS Acceptance Criteria

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission's (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 regulations identified above and necessary to meet the Three Mile Island (TMI) Action Plan requirements are as follows:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46. RG 1.157 and Section I of Appendix K to 10 CFR Part 50 provide guidance on acceptable evaluation models. This also includes analyses of a spectrum of small break LOCAs to assure boric acid precipitation is precluded for all break sizes and locations.

The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

A. The calculated maximum fuel element cladding temperature does not exceed 1200^{o}C (2200^{o}F).

B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occur during the course of the accident.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

If the mPower LOCA analyses demonstrate that there is no core uncovery or heatup for any design-basis LOCA, the PCT is expected to be within the acceptance criterion of 1,204 ^{o}C (2,200 ^{o}F). The parameter of interest will be reactor level rather than the PCT. There is no additional oxidation of the cladding as a result of a LOCA. There is no additional hydrogen generated from the chemical reaction of the cladding with water or steam, because the temperatures are not high enough to create this chemical reaction. There are no changes in core geometry resulting from a LOCA that would prevent the core from being amenable to cooling.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR 100 or 10 CFR 50.67. For applications under 10 CFR Part 52, reviewers should use DSRS Section 15.0.3,"Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications."

3. The TMI Action Plan requirements for II.E.2.3, II.K.3.5, II.K.3.25, II.K.3.30,II.K.3.31 and II.K.3.40 have been met.

4. 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 15.6.5 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. Compliance with 10 CFR 50.46 requires that light water cooled nuclear power reactors be equipped with an emergency core cooling system designed so that core performance following postulated loss-of-coolant accidents conforms to specified criteria related to limiting core damage.

   The requirements specified in 10 CFR 50.46 provide an acceptable and conservative means of calculation of the consequences of LOCAs from a spectrum of pipe break sizes and locations that have been subject to careful review and experimental verification. If the calculations of the performance of the emergency core cooling system are conducted in accordance with these methods, there is a high level of probability that the acceptance criteria on core performance will not be exceeded and damage to the core and offsite consequences will be minimized. RG 1.157, "Best Estimate Calculations of Emergency Core Cooling System Performance," and Appendix K to 10 CFR Part 50, provide guidance and requirements on evaluation models needed to demonstrate compliance with the acceptance criteria. Appendix K also specifies documentation required for evaluation models.

   Meeting the requirements outlined in the references provides assurance that following a LOCA the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR 100 or 10 CFR 50.67.

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 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.

3. Compliance with GDC 35 requires that a means of providing abundant emergency core cooling be provided that will transfer heat from the reactor core in the event of a LOCA, and that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure. GDC 35 specifies that an emergency core cooling system be installed in all nuclear power reactors. DSRS Section 15.6.5 specifies the analytical procedures that are to be followed to establish that the ECCS will function to meet acceptance criteria specified in 10 CFR 50.46. 10 CFR Part 50, Appendix K and RG 1.157 provide guidance on calculational procedures needed to demonstrate compliance with the acceptance criteria.

   Meeting the requirements of GDC 35 will provide assurance that following a LOCA that the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR 100 or 10 CFR 50.67.

4. 10 CFR 100 or 10 CFR 50.67, Reactor Site Criteria, describe criteria that guide the Commission in its evaluation of the suitability of proposed sites for nuclear power and testing reactors. 10 CFR 100 or 10 CFR 50.67 specify radiation dose guidelines that should not be exceeded in the event of postulated accidents including LOCAs.

   In order to satisfy the requirements of 10 CFR 100 or 10 CFR 50.67, the applicant must demonstrate that the offsite doses resulting from various accidents presented in the applicant's technical submittal are within the guideline values. Meeting the guideline doses is achieved by a combination of engineered safety features installed in the nuclear facility, an effective emergency core cooling system, and siting the nuclear plant in an area that does not exceed population density requirements.

   Meeting the nuclear power plant siting criteria provides a level of assurance that the plant will pose no undue risk to the public as a result of the consequences of LOCAs.

III. <u>REVIEW PROCEDURES</u>

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), standard design certification, COL, and operating license (OL) reviews, as appropriate. During the CP or the standard design certification review, the values of system parameters setpoints used in the analysis are considered preliminary in nature and subject to change. At the OL or COL review, final values should be used in the analysis and the reviewer compares these to the limiting safety system settings included in the proposed technical specifications.

For the review of the ECCS performance analysis, as presented in the applicant's Technical Submittal, the reviewer verifies the following:

1. Programmatic Requirements - In accordance with the guidance in NUREG-0800 "Introduction," Part 2 as applied to this DSRS Section, the staff will review the programs proposed by the applicant to satisfy the following programmatic requirements. If any of the proposed programs satisfies the acceptance criteria described in Subsection II, it can be used to augment or replace some of the review procedures. It should be noted that the wording of "to augment or replace" applies to nonsafety-related risk-significant SSCs, but "to replace" applies to nonsafety-related nonrisk-significant SSCs according to the "graded approach" discussion in NUREG-0800 "Introduction," Part 2. Commission regulations and policy mandate programs applicable to SSCs that include:

   A. Maintenance Rule SRP Section 17.6 (DSRS Section 13.4, Table 13.4, Item 17, Regulatory Guides 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 (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. Inspections, tests, analyses, and acceptance criteria (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 six 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). 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 calculations were performed using an evaluation model as specified in 10 CFR 50.46 following the guidance of Appendix K, Section I, or RG 1.157. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required. Evaluation models pertain to both the short term behavior following both large and small break LOCAs as well as post-LOCA long term cooling evaluations that properly address boric acid precipitation and prevention for both large and small break LOCAs.

4. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under DSRS Section 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

5. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, sensitivity studies on the important parameters should have been made by the applicant. If such sensitivity studies are not presented in the applicant’s technical submittal, the reviewer requests that they be made.

6. If core uncovery is not expected during the entire period of a LOCA staff should ensure that a significant number of fuel rods will not be damaged due to local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the critical heat flux (CHF) at a given pressure after depressurization has taken place. If, however, the heat flux exceeds the CHF, further analyses should be performed to estimate the amount of fuel damage expected from ”burn-out“ while the bulk of the core

remains covered with water during the LOCA. Fuel damage and potential for radioactivity release to the environment must be consistent with 10 CFR 100 or 10 CFR 50.67. If such evaluations are not provided in the applicant's technical submittal, the reviewer requests that they be made.

7. The parameters and assumptions used for the calculations were conservatively chosen, including the following points:

   A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of 2 percent to account for power measurement uncertainties, unless a lower level of uncertainty can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

   B. The maximum linear heat generation rate used should be based on the proposed licensed core thermal power as discussed in Item A and the technical specification limit on peaking factors, or on the technical specification limits on maximum linear heat generation rate. For many plants these limits may also be found in Technical Specification 5.6.5, Core Operating Limits Report (COLR).

   C. All permitted axial power shapes, as given in Section 4.3 of the DSRS, should be addressed by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

   D. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in DSRS Section 4.4, similar calculations performed by the staff, or calculations done for similar plants by previous applicants.

   E. Appropriate analyses are presented to support any credit taken for control rod insertion.

   F. The applicant's analysis conservatively addresses the operation of the reactor coolant pump including requirements for reactor coolant pump trip during small break LOCAs as required by Generic Letters 85-012, 86-005, and 86-006.

   G. The analysis of boric acid precipitation should include a justified mixing volume which is computed as a function of time as ECC injection enters the core region. Since the size of the mixing volume is controlled by the external loop resistance and the balance of hydrostatic heads between the downcomer and inner vessel regions containing voids, the model must account for these effects. The precipitation limit must also be justified in the evaluation model. If the system design includes high concentrate boric acid tanks and/or sources, then these systems must be assumed to be operating at the time of the break initiation.

   H. The containment pressure response used during the ECCS performance evaluation reflects a conservatively low minimum containment pressure.

l. Debris wash-down into the reactor cavity has been adequately treated so as to result in a conservative recirculation flow rate after sufficient water accumulates in the reactor cavity and containment.

8. Reactor protection system actions and safety injection actuation and delivery are consistent with the set points and the associated uncertainties and delay times listed in the applicant's technical submittal. The ECCS flow rates should be checked against the applicant's data on natural circulation flow in piping networks that are sufficiently similar to the mPower ECCS system. The Regional Offices may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

   In the case of mPower which uses passive rather than active systems to provide ECCS to the reactor vessel, pressure drop test results should be reviewed to determine that the passive ECCS flow rate is consistent with that in the analyses of the system performance.

9. The results of the applicant's calculations are reasonable based on the selected input parameters. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in the core, and in the downcomer; reactor coolant temperature and quality at core inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, depressurization, refill, and reflood; heat flux transients from piping and vessel walls; primary-secondary heat transfer; timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested); predicted "end-of-bypass" time compared to calculated downcomer flow and to staff calculations for typical plants; pump speed transients; containment pressure transients; and carryover fraction (if it is not an input to the calculations). The boric acid concentration should be shown as a function of time for the limiting large and small breaks. For small breaks where simultaneous injection is unable to flush the core, other procedures must be employed to show that the boric acid concentration does not achieve precipitation limits.

   mPower may base their ECCS and reactor coolant system designs on prevention of core uncovery. Should that be the case, the reviewer should compare the applicant's analysis with the staff independent analysis to determine if the predicted level of core coverage is consistent.

10. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS given in 10 CFR 50.46(b). Boric acid concentration should be shown to be controlled prior to reaching the precipitation limit and all equipment and operator action times identified for inclusion in the EOPs.

11. The applicant's analysis addresses the full LOCA sequence of events, for the full spectrum of break sizes and locations, to the point where the plant is in the long-term cooling mode and removal of decay heat has been well established for both large and small breaks. The reviewer checks the assumed sources of coolant water, redundancy of delivery routes, alignment of valves, control of boron concentration and all required operator actions.

12. TMI action Item II.K.3.5 "Automatic reactor coolant pump (RCP) Trip during a LOCA", requires reactor coolant pump trip following all small breaks. While II.K.3.5 requires trip, restart of the RCPs should only be attempted based on explicit guidance in the EOPs dealing with a safe restart of the pumps. This will ensure small-break LOCA ECCS performance is not later degraded and pump damage is also avoided.

    The following steps shall be included in PWR emergency operating procedures as a condition for reactor coolant pump startup after a small break LOCA:

    A. Verify adequate single phase natural circulation,

    B. If single phase natural circulation cannot be established, verify adequate two phase natural circulation,

    C. Determine if reactor coolant pump restart is needed and desired, and

    D. Verify that all reactor coolant pump restart criteria are met.

13. The TMI Action Plans items are reviewed to assure compliance with the acceptance criteria.

    A. The reviewer evaluates the uncertainty analyses performed by the applicant to assure that the modeling assumptions and phenomena for small-break LOCA calculations are properly accounted for to determine the acceptability of the ECCS performance pursuant to 10 CFR 50.46 and RG 1.157 or Appendix K of 10 CFR Part 50 (Item II.E.2.3).

    B. The reviewer evaluates the assumptions made regarding RCP trip to assure that they are consistent and conservatively modeled with respect to the final pump trip criteria which result from resolution of TMI action plan (Item II.K.3.5).

    C. If, as a result of a LOCA, or as a result of loss of alternating current (A/C) power, containment isolation is indicated to occur, the RCP component cooling water may be lost. The reviewer evaluates the applicant's submittal to determine that the reactor coolant pump seal integrity is not lost. If it cannot be established that seal integrity is assured, the reviewer assures that the evaluation of this event correctly accounts for seal failure (Item II.K.3.25 and II.K.3.40).

    D. The reviewer evaluates the small-break LOCA model verification performed by the applicant and assures that any modifications required are incorporated into the specific plant analyses (Item II.K.3.30 and II.K.3.31).

14. Reviewers representing all relevant technical disciplines will provide input for the areas of review stated in Subsection I. The staff review uses such input as required to assure that this review procedure is complete.

15. The review of fission product releases and radiological consequences of design basis (most severe) LOCA is performed by the emergency preparedness and radiation protection in Section 15.0.3, based on the licensing basis or application, as appropriate.

16. 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 applicant's technical submittal meets the acceptance criteria. 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 applicant's technical submittal.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an ESP 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 loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary is acceptable and meets the relevant requirements of 10 CFR 50.46, GDC 13, GDC 35, and 10 CFR 100 or 10 CFR 50.67. This conclusion is based on 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 meets GDC 35 requirements by demonstrating that a means of providing abundant emergency core cooling is provided that will transfer heat from the reactor core in the event of a LOCA, and that suitable redundancy of components and features is provided so that the safety function can be accomplished assuming a single failure. Meeting the requirements of GDC 35 will provide assurance that following a LOCA that the reactor core will remain in a coolable geometry and offsite consequences will be within the guidelines specified in 10 CFR 100 or 10 CFR 50.67.

3. The applicant has performed analyses of the performance of the ECCS in accordance with the Commission's regulations (10 CFR 50.46). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that follows the guidance contained in RG 1.157 or Section I of Appendix K to 10 CFR Part 50 and meets the requirements of 10 CFR 50.46. The results of the analyses show that the ECCS satisfies the following criteria:

    A. The calculated maximum fuel rod cladding temperature does not exceed 1200 ^{o}C (2200 ^{o}F).

    B. The calculated total maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

    C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

F. The applicant has met the requirements of TMI Action Plan items.

G. Boric acid precipitation can be prevented for all break sizes and locations during post-LOCA long term cooling.

4. The radiological consequences meet 10 CFR 100 or 10 CFR 50.67 requirements for the postulated spectrum of LOCA which were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS.

The staff concludes that the calculated performance of the emergency core cooling system following a postulated LOCA and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable.

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 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 final safety analysis report 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 Criterion 35, "Emergency Core Cooling."

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to 10 CFR Part 50, "ECCS Evaluation Models."

3. SRP Section 6.3, "Emergency Core Cooling System."

4. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems Resolution of Generic Task Action Plan A-2."

5. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

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

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

8. Generic Letter 85-012, "Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Westinghouse Designed Nuclear Steam Supply Systems."

9. Generic Letter 86-005, “Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Babcock and Wilcox Designed Nuclear Steam Supply Systems."

10. Generic Letter 86-006, "Implementation of TMI Action Item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps" for Combustion Engineering Designed Nuclear Steam Supply Systems."

11. Technical Desk Reference, TMI items: II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40.