U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY VALVE, OR AN AUTOMATIC DEPRESSURIZATION VALVE

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

An accidental reactor vessel depressurization and decrease of reactor vessel coolant inventory could be caused by the inadvertent opening of a pressure safety valve, or an automatic depressurization valve (ADV), which could be caused by a spurious electrical signal, hardware malfunction or by an operator error. As this event can occur one or more times during the plant's lifetime, it is an anticipated operational occurrence (AOO), as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 10 CFR Part 50, Appendix A. The event covered in this Design-Specific Review Standard (DSRS) section should be addressed in an individual section of the applicant's technical submittal as specified in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

1. The review of these transients should consider the sequence of events, the analytical model, the values of parameters in the analytical model, and the predicted consequences of the transient. The specific areas of review are as follows:

   The staff reviews the sequence of events described in the applicant's technical submittal. The reviewer focuses on the need for 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 for whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced method has not been reviewed, the reviewer initiates a generic evaluation. The values of all parameters in the new analytical model, including the initial conditions of the core and system, are reviewed. The predicted results of the transient are reviewed for whether the consequences meet the acceptance criteria of DSRS Section 15.0 and subsection II of this DSRS section. The analysis results are reviewed for whether pertinent system parameter values are within ranges expected for the type and class of reactor under review.

ADAMS Accession No. ML12250A318

Reactor protection system functions (e.g., automatic reactor trips) that are identified as available protection for this event, other than the credited function, are evaluated to ascertain whether the specified functions would be effective (i.e., setpoints and response times would lead to timely action, to satisfy the acceptance criteria). For example, the credited protection function might be the low pressurizer pressure reactor trips, and an available protection function might be the overtemperature delta-T trip. The reviewer would verify that the overtemperature delta-T trip would be executed in time to meet the relevant acceptance criteria (e.g., prevention of departure from nucleate boiling (DNB)).

The reviewer verifies whether the applicant's core physics data are appropriate.

2. Combined 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. Verification of the safety analysis sequence described by the applicant for automatic actuation, remote sensing, indication, control, interlocks with auxiliary or shared systems, potential bypass modes, and the possibility of manual control by the operator is performed under DSRS Chapter 7.

4. Verification of whether the equipment necessary to mitigate the event is qualified for the transient and post-transient environments and identification, if requested, of equipment that the failure of which as a result of the initiating event could have adverse consequences performed under applicable DSRS sections.

5. The reviewer verifies whether the control systems power sources needed to mitigate the event are available as required by the applicant's description of the event under DSRS Chapter 8.

6. Plant operating procedures are reviewed to verify they include appropriate actions as to a reactor coolant pump trip after the inadvertent opening of a pressure relief valve as described in Generic Letters 85-12, 86-05, and 86-06 under DSRS Chapter 18.

7. The determination of the risk significance of 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. General Design Criterion (GDC) 10, as it relates to designing the RCS with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs.

2. GDC 13, as it relates to providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation to assure adequate safety, and to providing appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to designing the RCS and associated auxiliary systems with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operations, including AOOs.

4. GDC 26, as it relates to providing a reactivity control system capable of reliably controlling reactivity changes during manual operations and AOOs so the specified fuel design limits are not exceeded.

5. GDC 35, as it relates to providing 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 such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

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.

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity is maintained if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4).

3. An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.

To meet the requirements of GDCs 10, 13, 15, 26 and 35, the positions of RG 1.105, "Instrument Setpoints for Safety-Related Systems," are useful as to their impact on the plant response to the type of transient addressed in this DSRS section.

The most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, should be assumed in the analysis and should satisfy the positions of RG 1.53.

The applicant's analysis of this transient should use an acceptable analytical model. If the applicant proposes to use analytical methods not previously reviewed and approved by the staff, the staff evaluates them for acceptability. For new generic methods, the reviewer initiates an evaluation of the new analytical model.

The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:

1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to operate plus an allowance of 2 percent to account for power measurement uncertainties unless the applicant can justify a lower power level. The operating condition at the initiation of the event should correspond to the operating condition that maximizes the consequences of the event.

2. Applicant should conservatively assume the maximum time delay and the most reactive rod held out of the core.

3. The core burn-up is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105.

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. GDC 10 requires design of the reactor core and its coolant, control, and protection systems with appropriate margin so specified acceptable fuel design limits are not exceeded during any conditions of normal operation, including the effects of AOOs.

GDC 10 applies to this section because the reviewer evaluates the consequences of an inadvertent opening of a pressurizer pressure relief valve. These AOOs could exceed allowable thermal design criteria for fuel cladding integrity. RG 1.53 provides guidance for applying the single-failure criterion to the design and analysis of nuclear power plant protection systems. RG 1.105 provides guidance for ensuring that instrument setpoints remain within the technical specification limits.

GDC 10 requirements provide assurance that specified acceptable fuel design limits are not exceeded in the inadvertent opening of a pressurizer pressure relief valve.

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. GDC 15 requires design of the RCS and its auxiliary, control, and protection systems with sufficient margin so reactor coolant pressure boundary design conditions are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 applies to this section because the reviewer evaluates the consequences of the inadvertent opening of a pressurizer pressure relief valve.

   As part of the reactor coolant pressure boundary, the pressurizer relief and safety valves must be able to reseat properly after actuation. As an AOO, the inadvertent opening of a pressurizer relief valve should not prevent the plant from returning to power a short time after shutdown.

4. DC 26 requires that reactivity control systems at nuclear power plants include control rods that can control reactivity changes so specified acceptable fuel design limits are not exceeded under conditions of normal operation, including AOOs. This system design must have an appropriate margin to accommodate malfunctions (*e.g.*, stuck rods).

   GDC 26 applies to this section because the transient analyzed by the reviewer may require the responsive movement of control rods. In such instances, rod misalignment, including stuck rods, can result in more severe thermal-hydraulic conditions. GDC 26 requires that the thermal margin be sufficient to accommodate these conditions. DSRS Section 15.6.1 examines this margin for whether thermal criteria are satisfied. Compliance with GDC 26 is best demonstrated by showing that adequate thermal margin is maintained, during the event, as the result of automatic protective action (*e.g.*, a reactor trip) actuated by the monitoring of parameters related directly to thermal margin (*e.g.*, the the low thermal margin trip or low DNBR trip). The review should encompass claims that automatic reactor protection is available from specified trip

signals in addition to the trip signal credited in the licensing basis analysis. The review should verify whether such signals can provide adequate, timely protection.

GDC 26 requirements provide assurance of appropriate margins to accommodate malfunctions of the reactivity control system, including stuck rods, minimizing the possibility that specified acceptable fuel design limits would be exceeded.

5. GDC 35 requires providing abundant emergency core cooling. The system safety function must be satisfied to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible.

6. 10 CFR 50.34(f), Additional TMI-Related Requirements, applies to this section because the TMI incident involved a stuck open power-operated relief valve. For plants licensed under 10 CFR Part 52, the requirements of 10 CFR 50.34 are incorporated under 10 CFR 52.47 and 10 CFR 52.79.

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.

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. 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 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 safety evaluation report section.

The applicant's description of the inadvertent pressure relief valve opening transient is reviewed for the occurrences leading to the initiating event. The sequence of events from initiation until stabilization is reviewed to ascertain:

1. The extent to which normally operating plant instrumentation and controls are assumed to function.

2. The extent to which plant and reactor protection systems are required to function.

3. The credit taken for the functioning of normally operating plant systems.

4. The extent to which the operation of engineered safety systems is required.

5. The extent to which operator actions are required.

6. If the applicant's technical submittal states that the inadvertent pressure relief valve opening transient is not as limiting as some other similar transient, the reviewer evaluates the applicant's justification. If the applicant's technical submittal presents a quantitative analysis of the transient, the timing of the initiation of those protection, engineered safety, and other systems needed to limit transient consequence to acceptable levels is reviewed. The reviewer compares the predicted variation of system parameters to various trip and system initiation setpoints.

To the extent deemed necessary, the reviewer evaluates the effects of system and component single active failures which may alter the course of the transient. In this phase of the review the system reviews are as described in the DSRS sections for technical submittal Chapters 5, 6, 7,

and 8. The reviewer considers possible single failures in systems that replenish or maintain the reactor coolant inventory.

The applicant's mathematical models to evaluate core performance and to predict system pressure in the RCS and main steam line are reviewed for whether they have been reviewed and found acceptable by the staff. If not, the reviewer initiates a generic review of the applicant's proposed model.

The values of system parameters and initial core and system conditions as input to the model also are reviewed. Of particular importance are the reactivity coefficients and control rod worths in the applicant's analysis and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The applicant's justification showing that the selected core burn-up yields the minimum margins is evaluated.

The results of the analysis are reviewed and compared to the acceptance criteria of subsection II for the maximum pressure in the reactor coolant and main steam systems. The following transient parameters are reviewed: reactor power, heat fluxes (average and maximum), RCS pressure, pressurizer water volume, minimum DNBR, core coolant flow rate, coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions), steamline pressure, containment pressure, pressure relief valve flow rate, and flow rate from the RCS to the containment system (if applicable).

Values of the more important of these parameters for the transient caused by the inadvertent pressure relief valve opening are compared to those predicted for other similar plants for whether they are within the expected range.

Upon request from the reviewer, other responsible organizations provide input for the areas of review stated in subsection I of this DSRS section. The reviewer uses the input requested as required to complete the review procedure.

7. 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 DC final safety analysis report (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 applicant evaluated this transient using a mathematical model previously reviewed and found acceptable by the staff. The input parameters for this model were reviewed and found suitably conservative. The results showed specified acceptable fuel design limits maintained by a minimum DNBR not below ____ and a maximum pressure within the reactor coolant and main steam systems not in excess of 110 percent of the design pressures.

The staff concludes that the analysis or evaluation of this AOO is acceptable and meets the relevant requirements of GDC 10, 13, 15, 26 and 35 and the applicable paragraphs of 10 CFR 50.34(f)(1). This conclusion is based on the following findings:

1. The applicant meets the requirements of GDCs 10, 26 and 35 by demonstrating that resultant fuel integrity is maintained because the specified acceptable fuel design limits were not exceeded for the event.

2. 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.

3. The applicant meets GDC 15 requirements by demonstrating that the reactor coolant pressure boundary limits were not exceeded by the event and that resultant leakage is within acceptable limits. This requirement is met because the maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressure.

4. The applicant meets GDC 26 requirements for the capability of the reactivity control system to control reactivity adequately during the event with appropriate margin for stuck rods because the specified acceptable fuel design limits were not exceeded.

5. The applicant meets GDC 35 requirement for removal of heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

6. The staff has determined that, in meeting GDC 10, 13, 15, 26 and 35, the analysis used a mathematical model previously reviewed and accepted by the staff. The input parameters for this model were reviewed and found suitably conservative. In addition, we have determined further that the positions of RG 1.53 on single failure criterion and RG 1.105 for instrument setpoints also are satisfied.

7. The applicant has shown that this AOO would not develop into a postulated accident without other faults occurring independently.

8. 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^{TM} -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^{TM} and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (ML102510405), to develop risk-informed licensing review plans for each of the small modular reactor reviews including the associated pre-application activities, the staff has developed the content of this DSRS section as an alternative method for mPower^{TM} -specific DC, or COL submitted pursuant to 10 CFR Part 52 to comply with 10 CFR 52.47(a)(9), "Contents of applications; technical information."

This regulation states, in part, that the application must contain "an evaluation of the standard plant design against the SRP revision in effect six 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^{TM} 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."

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

3. NUREG-0737, "Clarification of TMI Action Plan Requirements."

4. NRC Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Westinghouse-Designed Nuclear Steam Supply Systems."

5. NRC Generic Letter 86-05, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Babcock and Wilcox-Designed Nuclear Steam Supply Systems."

6. NRC Generic Letter 86-06, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' for Combustion Engineering-Designed Nuclear Steam Supply Systems."

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

8. RG 1.105, "Instrument Setpoints for Safety-Related Systems."

9. RG 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems."