U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

REVIEW RESPONSIBILITIES

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

Secondary - Organization responsible for the review of instrumentation and control systems

I. AREAS OF REVIEW

An anticipated transient without scram (ATWS) is an anticipated operational occurrence (AOO) as defined in Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion (GDC) 20. Since protection systems (e.g., the reactor trip system) must satisfy the single-failure criterion, multiple failures or a common mode failure must cause the assumed failure of the reactor trip. The probability of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than the probability of any of the other events that are evaluated under DSRS Chapter 15. Therefore, an ATWS event cannot be classified as either an AOO or a design-basis accident. It is designated as a Special Event.

The failure of the reactor to shut down during certain transients can lead to unacceptable reactor coolant system pressures, fuel conditions, and/or containment conditions. Typical AOOs that may result in unacceptable conditions following a pressurized-water reactor (PWR) scram failure are loss of feedwater, loss of load, turbine trip, inadvertent control rod withdrawal, loss of alternating current power, and/or loss of condenser vacuum.

Safety issues associated with an ATWS have been evaluated since the early 1970s. During NRC evaluations of vendor models and analyses addressing ATWS events, the agency formally identified the ATWS as Unresolved Safety Issue (USI) A-9, “Anticipated Transients Without Scram.” The agency presents the U.S. Nuclear Regulatory Commission (NRC) staff’s studies and findings regarding USI A-9 in NUREG-0460. In 1986, the NRC resolved USI A-9 through publication of 10 CFR 50.62, the ATWS rule. SECY 83-293 and the *Federal Register* notice of the final rule in 49 FR 26036 present the bases for current regulatory requirements related to ATWS events, including the associated regulatory evaluation.

The ATWS rule requires that certain light-water-cooled plants have prescribed systems and equipment that have been determined to reduce the risks attributable to ATWS events to an acceptably low level. The rule also requires applicants to submit information sufficient to demonstrate the adequacy of their plants’ prescribed systems and equipment.

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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. Reactivity coefficients and control rod worths are reviewed under DSRS Section 4.3

4. Determination that design and quality assurance criteria specified for instrumentation are consistent with criteria established in conjunction with 10 CFR 50.62 (the ATWS rule) is performed under DSRS Sections 7.1 and 7.8.

5. The functional design of the control rod drive system is reviewed under DSRS 4.6.

6. Determination that the design and reliability of the reactor trip system are acceptable and that required ATWS-related features are independent and diverse from the reactor trip system where required by the ATWS rule is performed under DSRS Section 7.2.

7. Human Factors Engineering review is conducted under DSRS Section 18 to evaluate the manual operator action to activate the borated water delivery system.

II. ACCEPTANCE CRITERIA

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

1. 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy.

2. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry.

3. 10 CFR 52.47(a)(15), as it relates to risk reduction from ATWS events.

4. 10 CFR 52.79(a)(42), as it relates to risk reduction from ATWS events.

5. GDC 12, found in Appendix A to 10 CFR Part 50, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed.

6. GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary.

7. GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents.

8. GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reactor must be limited to negligible amounts.

9. GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment.

10. GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

DSRS Acceptance Criteria

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.

The ATWS rule specifies that light water reactors must have a number of prescribed systems and equipment that are design-dependent, and have been proven to reduce the risk attributable to the ATWS events to an acceptable level. In addition, the applicants must submit information sufficient to demonstrate the adequacy of the implemented ATWS features. Design and quality assurance criteria for the required systems and equipment should meet or exceed the criteria established in conjunction with ATWS rulemaking, as described in DSRS Section 7.1, Appendix A and DSRS Section 7.8 to ensure adequate independence, diversity, and reliability where required by the ATWS rule. The following criteria apply to mPower™:

1. The applicant's scram system, from sensor output to interruption of power to the control rods, should satisfy one of the following options:

   A. A diverse scram system is provided satisfying the design and quality assurance criteria specified in DSRS Section 7.2.

B. In the absence of a diverse scram system the applicant shall demonstrate that the consequences of an ATWS event are within acceptable values based on the criteria in paragraph 3.

2. Provide measures to automatically initiate the auxiliary (or emergency) feedwater or equivalent sytem and a turbine trip under conditions indicative of an ATWS. This equipment shall be independent and diverse from the reactor trip system from sensor output to the final actuation device.

3. These equipment actuation and scram systems and equipment shall be demonstrated to provide reasonable assurance that unacceptable plant conditions do not occur in the event of an anticipated transient, and are based on the following criteria:

   A. Coolable geometry for the reactor core. If fuel and clad damage were to occur following a failure to scram, GDC 35 requires that this condition should not interfere with continued effective core cooling. 10 CFR 50.46 defines three specific core-coolability criteria: (1) Peak clad temperature shall not to exceed 1221°C (2200°F), (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen.

   B. Maintain reactor coolant pressure boundary integrity. Appendix A to WASH-1270 states that in evaluating the reactor coolant system boundary for ATWS events, "the calculated reactor coolant system transient pressure should be limited such that the maximum primary stress anywhere in the system boundary is less than that of the ‘emergency conditions' as defined in the ASME Nuclear Power Plant Components Code, Section III." The acceptance criterion for reactor coolant pressure, based upon the American Society of Mechanical Engineers (ASME) Service Level C limit is approximately 22MPa (3200 psig).

   C. Maintain containment Integrity. Following a failure to scram, the containment pressure and temperature must be maintained at acceptably low levels based on GDC 16 and 38. The containment pressure and temperature limits are design dependent; but to satisfy GDC 50, those limits must ensure that containment design leakage rates are not exceeded when subjected to the calculated pressure and temperature conditions resulting from any ATWS event.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this DSRS section is discussed in the following paragraphs:

1. 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (a) inclusion of prescribed design features and (b) demonstration of their adequacy

   Meeting the ATWS rule (10 CFR 50.62) requires that light-water-cooled plants be equipped with systems and equipment that are designed to reduce risks attributable to

ATWS events to an acceptable level. The rule also requires submission of information sufficient to demonstrate the adequacy of these systems and equipment.

2. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry.

   The applicant's design must satisfy limits derived from regulations in 10 CFR 50.46, which define specific core-coolability criteria, such as peak clad temperature, maximum cladding oxidation and hydrogen generation.

3. GDC 12 as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed.

4. GDC 13, as it relates to instrumentation used for ATWS mitigation systems.

5. GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary;

6. GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents;

7. GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reactor must be limited to negligible amounts;

8. GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment; and

9. GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

III. REVIEW PROCEDURES

The reviewer may perform portions of the evaluation on a generic basis for aspects common to a class of plant designs, or by adopting the results of previous reviews of plants with similar design. The areas to be given attention and emphasis are determined based on whether the information provided in the applicant's safety analysis report is similar to that reviewed for other similar plant designs, previously accepted ATWS evaluations and/or NSSS vendor topical reports are referenced, new or unique features affecting ATWS risk are proposed, and items of special safety significance are involved.

The evaluation may be based upon referenced approved designs, analyses, and/or assessments as applied to the licensee's plant. The categories of referenced approved materials include topical reports, standard design approvals, regulatory analyses associated with the ATWS rulemaking, and designs of systems that the staff has previously reviewed and approved. If any aspect of a design is not identical to that referenced, an evaluation must address the differences, and the applicant's submittal should include the conclusions regarding such differences.

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 (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 (USIs) and medium- and high-priority generic safety issues (GSIs) 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.

3. The reviewer will verify that the applicable required equipment and systems detailed in Subsection II are provided, as follows:

   A. The reviewer will verify that the applicable required instrumentation-related equipment and systems are provided, including automatic emergency core cooling and turbine trip initiation and diverse scram systems.

B. The reviewer will confirm that required equipment and systems are designed for independence and diversity from the reactor trip system where specified in Subsection II. Appendix A to DSRS Sections 7.1, 7.2 and 7.8 describe the criteria and methods for these reviews.

4. The reviewer will verify that sufficient information is provided or referenced to conclude that the required features will satisfy the criteria for acceptable plant conditions specified in Subsection II. The reviewer will place particular emphasis on the identification of plant-design differences with respect to referenced previously accepted materials. The reviewer will also perform the following in connection with this review:

    A. Evaluate the manner in which the applicant credits operator actions, including actions specified in applicable emergency procedure guidelines (EPGs) for the design or in emergency operating procedures (EOPs) or emergency operating instructions (EOIs) for the specific plant under review, as applicable. The reviewer will specifically determine the acceptability of manual activation of the emergency boration system.

    B. Verify that the applicant demonstrates assured capability for long-term shutdown and cooling following an ATWS event using systems identified in EPGs, EOPs, or EOIs for ATWS events.

    C. Evaluate the applicant's assumptions regarding the moderator temperature coefficient (MTC) and determine them to be either consistent with MTC modeling assumptions serving as bases for the Rule or adequately justified. Appendix C to NUREG-0460 discusses the MTC values used by PWR vendors for ATWS analyses before promulgation of the Rule. Enclosure D, Section 5.5, of 49 FR 29036 discusses the probabilities assumed during the Rulemaking of "unfavorable" MTC values resulting in unacceptable plant conditions.

    D. Review instrumentation parameters such as setpoints, tolerances, time delays, ranges, and channel response times specified in the design are acceptable with respect to any critical values assumed in the applicant's demonstration of the adequacy of required features.

    E. For events in which the analysis results predict fuel damage and offsite releases, the reviewer will evaluate the predicted extent of fuel damage and resulting offsite consequences with respect to those predicted in the relevant topical reports that the staff has previously accepted, and with acceptance criteria for postulated accidents.

5. Where new methods for the evaluation of ATWS events, risk reduction features, and/or consequences are proposed, based upon the unique features or ATWS sequences of a specific design, the reviewer will initiate a generic evaluation of affected portions of the applicant's assumptions, plant behavior criteria and models, data, and/or methods for the determination of offsite consequences. The staff may accept such new evaluations, consistent with the principles inherent in acceptable evaluation techniques and the basic approach to determining acceptable ATWS risks and consequences outlined in this DSRS section.

6. The reviewer will ensure that the staff has reviewed and approved all analysis methodologies, including the treatment of uncertainties, used in the submittal.

7. The reviewer will ensure that all restrictions and limitations specified in safety evaluations approving a licensing topical report are met, especially when operating under expanded operating domains.

8. The reviewer will ensure that the actual component and actuation setpoint testing supports the technical specification values used in the analyses. The reviewer will evaluate plant testing data to ensure that component performance supports the plant-specific technical specification values used in the analyses. For example, the Pressurizer safety valve upper lift setpoint tolerance may drift, which might affect the ATWS peak pressure.

9. The reviewer will ensure that if the technical specifications allow for equipment out of service, the ATWS analysis assumes the most conservative configuration. For example, if the technical specifications allow Pressurizer safety valve out of service, then the ATWS analysis should include this configuration irrespective of the basis for the technical specification section.

10. The reviewer will evaluate the need for staff confirmatory calculations if the design changes deviate significantly from established practice.

11. 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 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 submittal document.

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

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 plant design adequately addresses ATWS events and meets the requirements of 10 CFR 50.62. This conclusion is based on the following:
>
> 1. The applicant's plant design includes the ATWS risk reduction features prescribed by the Rule.

2. The ATWS risk reduction features are independent and diverse from the reactor trip system and are designed to be reliable, as required under the Rule.

3. The applicant has provided or referenced information, analyses, and/or evaluations that demonstrate that limiting ATWS and event sequences have been considered and that features included in the design pursuant to the ATWS rule result in reasonable assurance, that unacceptable plant conditions, as defined during the Rulemaking, will not occur because of ATWS events.

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 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^{TM} 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 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

2. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

3. 10 CFR 50.55a, “Codes and Standards.”

4. 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”

5. NUREG-0460, “Anticipated Transients Without Scram for Light Water Reactors,” (Vol. 1) April 1978, (Vol. 2) April 1978, (Vol. 3) December 1978, (Vol. 4) March 1980.

6. SECY 83-293, “Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events,” July 19, 1983.

7. U.S. Nuclear Regulatory Commission, “10 CFR Part 50.62, Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants,” 49 FR 26036, June 26, 1984.

8. WASH-1270, “Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors”, U.S. Atomic Energy Commission, September 1973.