U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.3.3 - 15.3.4 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT SEIZURE AND BREAK ACCIDENTS

REVIEW RESPONSIBILITIES:

Primary - Organization responsible for review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

The mPower™ reactor coolant system comprises a reactor vessel with a single internal steam generator and multiple internal reactor coolant pumps (RCPs). After passing upward through the reactor core, primary coolant flows up through a riser and down through the steam generator tubes, returning to the RCPs. Feedwater and steam lines penetrate the vessel providing the secondary flow to the tube-side of the single steam generator to remove the heat generated in the core.

The events postulated are an instantaneous seizure of the rotor or break of the shaft of one or more reactor coolant pumps. Depending on the number of pumps involved and the associated trip logic, flow through the core may be rapidly reduced, leading to a reactor and turbine trip. A sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The failure of a rotor or break of a shaft leads to a reactor trip and the primary system transitions to natural circulation flow. The mPower™ emergency core cooling system (ECCS) actuates, the primary system depressurizes and heat is transferred to the ultimate heat sink for decay heat removal.

This Design-Specific Review Standard (DSRS) section is intended to cover both (1) the RCP rotor seizure and (2) shaft break accidents.

1. RCP Pump Rotor Seizure Accidents
2. Shaft Seizure or Shaft Break Accidents

The review includes evaluation of the applicant's postulated initial and long-term core and reactor conditions pertinent to the rotor seizure or broken shaft events; the methods of thermal and hydraulic analysis; the postulated sequence of events, including time delays prior to and after protective system actuation; the assumed reactions of reactor system components; the functional and operational characteristics of the reactor protection system in terms of how it affects the sequence of events; and all operator actions required to secure and maintain the reactor in a safe condition.

The results of the applicant's analyses are reviewed to assess fuel damage and to ensure that values of pertinent system parameters are within expected ranges for the type and class of

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reactor under review. Fuel damage is assessed by the methods described in DSRS Section 4.2. System parameters to be reviewed include the following:

A. core flow and flow distribution (including hydraulic instabilities)
B. channel heat flux (average and hot)
C. minimum critical heat flux ratio (or minimum critical power ratio)
D. departure from nucleate boiling ratio
E. fuel centerline temperature
F. vessel two-phase level
G. thermal power
H. vessel pressure

The sequence of events described in the safety analysis report (applicant's technical submittal) is reviewed by the staff. The staff reviews concentrate 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 by the staff to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, the reviewer initiates a generic evaluation of the new analytical model.

The values of all parameters used in a new analytical model, including the initial conditions of the core and system, are reviewed. It is the responsibility of the reviewer to ensure that the appropriate physics and fuel data have been used in any staff calculations.

The staff performs generic reviews of the thermal-hydraulic computer models used for this transient and also performs additional analyses related to these accidents for selected reactor types as part of its primary review responsibility for DSRS Section 4.4.

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

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

<u>Review Interfaces</u>

Other DSRS sections interface with this section as follows:

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

2. The design bases, the design, the test programs and the proposed technical specifications for the ECCS are reviewed under DSRS 6.3.

3. The design of the overpressure protection system is reviewed under DSRS 5.2.2 to gain familiarity with the design and operation of the pressure relief system.

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

5. Instrumentation and controls aspects of the sequence described in the applicant's technical submittal are reviewed to confirm that reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis under DSRS Sections 7.2 through 7.5.

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

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) 17, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

2. GDC 27 and 28, as they relate to the reactor coolant system being designed with appropriate margin to ensure that the capability to cool the core is maintained.

3. GDC 31, as it relates to the reactor coolant system being designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

4. Title 10 of the Code of Federal Regulations (10 CFR Part 100), as it relates to the calculated doses at the site boundary.

The basic objectives of the review of the accident resulting from a rotor seizure or shaft break in a reactor coolant pump are:

1. To identify which of these accidents is the more limiting.

2. To verify that, for the accident, the plant responds in such a way that the regulatory requirements regarding fuel damage, radiological consequences, and system pressure are met.

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.

The specific criteria necessary to meet the relevant requirements of GDC 27, 28, and 31 and 10 CFR Part 100 for the rotor seizure and shaft break event are:

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

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit based on acceptable correlations (see DSRS Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see DSRS Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering departure from nucleate boiling (DNB). The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. Any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

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

5. A rotor seizure or shaft break in a reactor coolant pump should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

6. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component. For new applications, loss of offsite power should not be considered a single failure; reactor coolant pump rotor seizures and shaft breaks should be analyzed with a loss of off-site power (see item 9, below) in combination with a single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Evaluation Report (FSAR) for the ABB-CE System 80+ design certification.)

7. The ability to achieve and maintain long-term core cooling should be verified.

8. This event should be analyzed assuming turbine trip and coincident with loss of offsite power and coastdown of undamaged pumps.

The applicant's analysis should be performed using an acceptable analytical model. The equations, sensitivity studies, and models described in References 8 through 12 are acceptable. References 15 through 19 were found to be acceptable computer codes for transient analyses (i.e., except for loss-of-coolant accidents, or (LOCAs) for the Combustion Engineering System 80+ FSAR staff review. In addition, NUREG-1465 contains guidance on accident source terms for light-water nuclear power plants. When conducting transient analyses, the NUREG-1465 guidance is particularly important for reviewing fractions of relevant isotopes (noble gases, iodine, cesium, and rubidium) and chemical species of iodine assumed to exist within the gap between fuel pellets and cladding. If other analytical methods are proposed by the applicant, these methods are evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation. There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

1. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating, plus an allowance to account for power measurement uncertainties. The number of reactor coolant pumps and the number of feedwater/steam outlet trains operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

2. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N-1 reactor coolant pumps (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution will change as more pumps begin to coastdown following turbine trip and coincident loss of offsite power.

3. Conservative scram characteristics are assumed, i.e., a maximum time delay with the most reactive rod held out of the core.

4. The core burnup is selected to yield the most limiting combination of moderator temperature reactivity, void reactivity, Doppler reactivity, axial power profile, and radial power distribution.

5. 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 15.3.3-15.3.4 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:

The technical rationale for application of these acceptance criteria to reviewing analyses of transients initiated by RCP rotor seizure and shaft break is discussed in the following paragraphs:

1. GDC 17 requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of SSCs important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that: (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. GDC 17 is applicable to DSRS Section 15.3.3-15.3.4 because this section reviews the analysis of events that are classified as abnormal operating occurrences or postulated accidents, depending on the severity of the results. Meeting the requirements of GDC 17 provides assurance that the design conditions of the reactor coolant pressure boundary are not exceeded as a result of reactor coolant pump rotor seizures and shaft breaks and that the core is cooled and containment and other vital functions are maintained.

2. Compliance with GDC 27 requires that reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes, to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

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

   GDC 27 and GDC 28 are applicable to this section because the reviewer evaluates two events (i.e., RCP rotor seizure and shaft break) that will result in transient conditions having the potential to affect reactor coolant temperature and pressure, which in turn could result in complex changes in core reactivity. The applicant's analyses of these transients in the applicant’s technical submittal must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for core cooling. The analyses must be independently reviewed by the staff in accordance with this DSRS section.

   Meeting the requirements of GDC 27 and GDC 28 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in (a) unacceptable stress on the reactor coolant pressure boundary or (b) a reduction in the capability of the core cooling or reactivity control systems to perform their design safety functions.

3. Compliance with GDC 31 requires that the reactor coolant pressure boundary be desaigned with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of

the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertrainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

GDC 31 is applicable to this section because the reviewer evaluates two events (RCP rotor seizure and shaft break) that could result in transient conditions having the potential for adversely affecting the reactor coolant pressure boundary. Loss of a reactor cooling pump will cause a rapid reduction in coolant flow through the core and, consequently, an increase in temperature and pressure. The amount of stress to which the reactor coolant pressure boundary is subjected depends on the severity of the transient. The severity of the transient is assessed by the applicant in the applicant's technical submittal and reviewed by the staff in accordance with this DSRS section.

Meeting the requirements of GDC 31 provides a level of assurance that a transient initiated by an RCP rotor seizure or shaft break will not result in an unacceptable stress on the reactor coolant pressure boundary or on the ability to cool the reactor core.

4. To establish the suitability of a nuclear power plant site, 10 CFR Part 100 specifies how the exclusion area, low population zone, and population center distance should be determined. Further, radiation exposure criteria stipulated in 10 CFR Part 100 provide reference values to be used in the site suitability determination based on postulated fission product releases associated with accidental events.

   10 CFR Part 100 is applicable to this section because it specifies the methodology for calculating radiation exposures at the site boundary for postulated accidents or events such as loss of a reactor coolant pump. For transients having a moderate frequency of occurrence, any release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of a RCP rotor seizure or shaft break must include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies such as those contained in NUREG-1465.

   Meeting this requirement provides a level of assurance that, in the event of a transient initiated by a reactor coolant pump rotor seizure or shaft break, radiation exposures at the site boundary will not exceed a small fraction of the reference values specified in 10 CFR Part 100.

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 for the design certification (DC) application review, and combined license (COL) reviews. At the COL review stage, final values should be used in the

analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

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

    A. Maintenance Rule 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. 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 SER section.

The applicant's analyses of the rotor seizure and shaft break events are reviewed by the staff regarding the occurrences leading to the initiating event. The sequence of events, from initiation until a stabilized condition is reached, 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 extent to which credit is 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 and time at which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods (see II.3.b), are accounted for.

If the applicant's technical submittal states that one of the accidents is not as limiting as the other, the reviewer evaluates the justification presented by the applicant. The applicant is to present a quantitative analysis in the applicant's technical submittal of the accident that is determined to be more limiting. For the accident that is found more limiting, the reviewer confirms that the effects of the accident are determined for each mode of operation (e.g., number of RCPs; number of feedwater/steam lines) allowed by the technical specifications. Either a separate analysis should be presented or each mode of operation or the effects of each mode should be referenced to the limiting case.

For the more limiting accident, the reviewer, with the aid of the staff reviews the timing of the initiation of those protection, engineered safety, and other systems needed to limit the consequences of the accident to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip and system initiation setpoints. The staff review of Chapter 7 of the applicant's technical submittal confirms that the instrumentation and control systems design is consistent with the requirements for safety systems actions for these events.

To the extent deemed necessary, the reviewer evaluates the effect of single active failures of safety systems and components which may alter the course of the accident. For new applications, the loss of off-site power is not considered a single active failure, but considered in addition to a single active failure as discussed in subsection II.7. This phase of the review uses the system review procedures described in the DSRS sections for Chapters 5, 6, 7, and 8 of the applicant's technical submittal.

The mathematical models used by the applicant to evaluate core performance and to predict system pressure in the reactor coolant system and main steam lines are reviewed by the staff to determine if these models have been previously reviewed and found acceptable by the staff. If not, the staff initiates a generic review of the model proposed by the applicant.

The values of system parameters and initial core and system conditions used as input to the model are reviewed by the staff. Of particular importance are the reactivities and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void, and Doppler reactivities with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated.

The results of the applicant's analysis are reviewed and compared to the acceptance criteria presented in subsection II regarding the maximum pressure in the reactor coolant and main steam systems. Time-related variations of the following parameters are reviewed:

- reactor power
- heat fluxes (average and maximum)
- fuel centerline temperature
- reactor coolant system pressure
- minimum DNBR
- core coolant flow rates
- coolant conditions (inlet temperature, core average temperature, average exit and hot channel exit temperatures, and steam fractions)
- steam line pressure
- containment pressure
- pressure relief valve flow rate
- flow rate from the reactor coolant system to the containment system (if applicable)

The more important of these parameters (as listed in Subsection I of this DSRS section) are compared to those predicted for other similar plants to confirm that they are within the expected range. The percentage of fuel rods that experience failure is reviewed and the staff is notified regarding the extent of fuel failures predicted by the analysis.

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. DCs have referred to the applicant's technical submittal as the "applicant's technical submittal". 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 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 early site permit or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the staff's technical review and analysis, as augmented by the application of programmatic requirements in accordance with the staff's technical review approach in the DSRS Introduction, support

conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

The staff concludes that the consequences of postulated rotor seizure or broken shaft events meet the requirements set forth in the GDC 17, 27, 28, and 31 regarding the ability to insert control rods and to cool the core, and 10 CFR Part 100 guidelines regarding radiological dose at the site boundary. This conclusion is based upon the following:

1. The applicant has demonstrated that the resultant fuel damage was limited such that the ability to insert control rods would be maintained, and that no loss of core cooling capability resulted. The minimum DNBR experienced by any fuel rod was _____, resulting in ____% of the rods experiencing cladding perforation.

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

3. The analyses and effects of pump rotor seizure and shaft breaks, during various modes of operation and with and without offsite power, have been reviewed.

4. The accidents analyzed were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

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

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this DSRS section.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower^{TM}-specific design certification (DC), or combined license (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 (SMR) 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 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 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, GDC 17, "Electric Power Systems."

2. 10 CFR Part 50, Appendix A, GDC 27, "Combined Reactivity Control Systems Capability."

3. 10 CFR Part 50, Appendix A, GDC 28, "Reactivity Limits."

4. 10 CFR Part 50, Appendix A, GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

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

6. "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 1), U.S. Nuclear Regulatory Commission, NUREG-0318, November 1976.

7. "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from the Director, NRR, to NRR Staff" (Issue No. 22), U.S. Nuclear Regulatory Commission, NUREG-0158, December 1976.

8. F. M. Bordelon, "Calculation of Flow Coastdown after Loss of Reactor Coolant Pump," WCAP-7973, Westinghouse Electric Corporation, August 1970.

9. C. D. Morgan, H. C. Cheatwood, and J. R. Glandermans, "RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown," BAW-10069, Babcock and Wilcox Company, July 1973.

10. R. H. Stoudt and J. E. Busby, "CADD - Computer Applications to Direct Simulation of Transient Events on Water Reactors," BAW-10080 (nonproprietary) and BAW-10076 (proprietary), Babcock and Wilcox Company, July 1973.

11. "System 80 Standard Safety Analysis Report (CESSAR)," Combustion Engineering, Inc., August 1973.

12. CESEC-III (CENPD-107; LD-82-001). (Calculates system parameters such as core power, flow, pressure, temperature, and valve actions during a transient.)

13. TORC (CENPD-161) and CETOP (CENPD-206-P-A). (TORC is used to simulate the three-dimensional fluid conditions within the reactor core. Results from TORC include the core radial distribution of the relative channel axial flow that is used to calibrate CETOP. TORC or CETOP calculations for the DNBR use the CE-1 critical heat flux correlation.)

14. HERMITE (CENPD-188-A). (HERMITE is used to determine short-term response of the reactor core during the postulated reactor coolant pump rotor-seizure event and total loss-of-flow event.)

15. COAST (SSAR; CENPD-98). (Calculates the time-dependent reactor coolant mass flow rate in each loop during reactor coolant pump coastdown transients.)

16. STRIKIN-II (CENPD-133; CENPD-135 Supps. 2 and 4). (Calculates the cladding and fuel temperatures for an average or hot fuel rod.)

17 NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995.

18. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License."

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