U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

5.4.7 RESIDUAL HEAT REMOVAL (RHR) SYSTEM (mPower™)

REVIEW RESPONSIBILITIES

Primary - Organization responsible for review of reactor thermal hydraulic systems in iPWRs

Secondary - None

I. AREAS OF REVIEW

The mPower™ integral pressurized water reactor (iPWR) designed by Babcock and Wilcox (B&W) makes extensive use of passive systems to meet regulatory requirements. Routine residual heat removal (RHR) for the mPower™ iPWR is provided by the reactor inventory control and purification system (RCI) through the nonregenerative heat exchanger under low pressure conditions. The RCI RHR function is included as a nonsafety-related, risk-significant active system for use during normal plant operation, shutdown and to provide defense in depth to the safety-related passive system. RCI is a Regulatory Treatment of Non-safety Related System (RTNSS) system. The nonsafety-related active system is the first line of defense to reduce challenges to the passive system in the event of transients or plant upsets. The mPower™ iPWR safety-related RHR function is provided by the passive decay heat removal system, an engineered safety feature (ESF) of the mPower™ design, which is evaluated in DSRS Section 6.3.

The RCI is used to cool the reactor coolant system (RCS) during and following normal reactor shutdown. Typically, during a normal RCS shutdown and cooldown, the RCI is used in conjunction with the steam and feedwater systems to remove residual heat. Initially, the steam and feedwater systems remove heat by steaming to the main condenser via the turbine bypass or by steaming through the atmospheric dump valves. Steaming provides fine control of the RCS cooldown rate. The RCI RHR function typically provides shutdown cooling when the RCS temperature is reduced to about 150°C (300°F). Alternatively, since the RCI system is capable of full RCS pressure, RHR during RCS shutdown and cooldown can be performed solely by the RCI system. The Auxiliary Condenser System (CNX, reviewed in DSRS 5.4.14) also provides high pressure decay heat removal from the steam generator via the main steam line by natural circulation. This second method may not provide the same fine control of RCS cooldown rates as the steaming method. Therefore, the review of the RHR function must consider all conditions from shutdown at normal RCS power operating pressure and temperature to the cold depressurized condition. Detailed evaluation of the cooling capacity of the nonregenerative heat exchanger must also be considered for all anticipated RCI operating conditions. The reviewer of this DSRS section will ensure that the design of the RCI for RHR conforms with General Design Criteria (GDCs) 1, 2, 4, 5,14, 19, 34 and 54 in Appendix A to Title 10 of the *Code of Federal Regulations* (CFR), Part 50 or similar requirements in the principle design criteria incorporated into the plant licensing basis.

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To provide active RHR, the mPower™ RCI takes water from the RCS Letdown, cools it via the nonregenerative heat exchanger, and pumps it back to the RCS. The RCI nonregenerative heat exchanger transfers heat to the component cooling water system (reviewed in DSRS 9.2.2).

The RCI is also used to provide makeup and letdown water to maintain the required water inventory and quality in the RCS and the refueling water storage tank (RWST), provide pressurizer auxiliary spray, control the primary water chemistry, and reduce coolant radioactivity level. These functions are reviewed in DSRS Section 9.3.6, which is the primary review for the mechanical aspects of the RCI.

The active mPower™ RCI RHR function is used to cool the core during shutdown operations, including reduced inventory operation. High RCI availability and reliability during shutdown conditions are important to mitigating risk and maintaining defense in depth. DSRS section 9.3.6 covers the review of the methods used to ensure high reliability of the RHR function under these conditions.

NOTE: The RHR function is performed by RCI, and RCI is reviewed under DSRS 9.3.6. Additional review guidance is included in DCRS 9.3.6.

The reviewer will also evaluate the requirements for leakage detection and control identified in NUREG-0737, Item III.D.1.1. The organization responsible for the instrumentation and control systems reviews the hardware and procedures to provide reasonable assurance the leakage requirements are met.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this DSRS section in accordance with Standard Review Plan (SRP) Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this DSRS section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

COL Action Items and Certification Requirements and Restrictions. 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.

Review Interfaces

The mPower™ active RHR function is provided by the RCI. Therefore, the principle RHR interface is:

1. As part of its primary review responsibility for SRP Section 3.12, the organization responsible for the review of materials engineering issues related to flaw evaluation and welding reviews the design of the RHR systems for new light-water reactor designs to

verify, to the extent practical, that low-pressure portions of the RHR that interface with the RCS will withstand full RCS pressure. If designing the RHR with an ultimate rupture strength capable of withstanding full RCS pressure is not possible, the reviewer verifies that appropriate compensating measures have been taken in accordance with the review provided in SRP Section 3.12. Determination of the acceptability of the RCI to support the RHR function for normal and transient operations is coordinated and performed under DSRS Section 9.3.6.

2. With respect to the staff review for compliance with Branch Technical Position (BTP) 5-4, the organizations responsible for the review of the steam and feedwater system, RCS, and reactor thermal hydraulic systems divide the evaluation as follows:

   A. The organization responsible for the review of reactor thermal hydraulic systems in pressurized water reactors (PWRs) reviews the approach used to meet the functional requirements of BTP 5-4 with respect to cooling down to the conditions permitting operation of the RHR system. Since an alternate approach to that normally used for cooldown may be specified, the reviewer identifies all components and systems used. The organization responsible for the review of steam and feedwater system, as part of its primary review responsibility for DSRS Sections 10.3 and 5.4.14, reviews the atmospheric dump valves and the source for auxiliary feedwater, respectively, for conformance to BTP 5-4. The organization responsible for the review of reactor thermal-hydraulic systems in pressurized-water reactors reviews the pressurizer relief valves and emergency core cooling system (ECCS), if used. As part of its primary review responsibility for DSRS Section 6.3, the organization responsible for the review of reactor thermal hydraulic systems reviews the PWR depressurization systems used for cooldown. In addition, the organization responsible for the review of reactor thermal hydraulic systems reviews the tests and supporting analysis concerning the mixing of borated water and cooldown under natural circulation as required in BTP 5-4.

   B. The organization responsible for the review of cooling water systems associated with balance of plant reviews the component cooling or service water systems that transfer decay heat from the RHR system to the Ultimate Heat Sink (UHS) as part of its primary review responsibility for DSRS Sections 9.2.1 and 9.2.2.

   C. The organization responsible for the review of reactor thermal hydraulic systems reviews the design and operating characteristics of the RHR system with respect to its shutdown and long-term cooling function. Where the RHR system interfaces with other systems (e.g., service water or component cooling water system), the responsible organization reviews the effect of these systems on the RHR system. The responsible organization also reviews overpressure protection provided by the valving between the RCS and RHR system.

In addition, the organization responsible for the review of reactor thermal hydraulic systems will coordinate evaluations of other reviewers that interface with the overall review of the RHR system as follows:

1. The organization responsible for the review of containment integrity performs the following reviews:

   A. Evaluates the containment heat removal capability and the containment sump

designs as part of its review responsibility for DSRS Section 6.2.2

B. Verifies that portions of the RHR system penetrating the containment barrier are designed with acceptable isolation features to maintain containment integrity for all operating conditions, including accidents, as part of its primary review responsibility for DSRS Section 6.2.4

2. Review of flood protection for the RCI RHR function is performed under DSRS 3.4.1.

3. Review of the effects on the RCI RHR function due to pipe breaks inside and outside of containment, such as pipe whip and jet impingement is performed under DSRS Section 3.6.2. The organizations responsible for the structural analysis reviews and review of seismic/geotechnical issues determine the acceptability of the design analysis, procedures, and criteria used to establish the ability of seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as a safe-shutdown earthquake (SSE), the probable maximum flood, and tornado missiles as part of their primary review responsibility for DSRS Sections 3.3.1, 3.3.2, 3.5.3, 3.7.1 through 3.7.3, 3.8.4, 3.8.5 and SRP 3.7.4. The organization responsible for the review of the inspection, testing, evaluation, and repair of mechanical equipment and components also verifies that inservice inspection requirements are met for system components as part of its primary review responsibility for DSRS Section 6.6.

4. Upon request, the organization responsible for the review of component integrity issues related to engineered safety features verifies the compatibility of the materials of construction with service conditions as part of its primary review responsibility for DSRS Section 6.1.1.

5. The organization responsible for mechanical engineering reviews performs the following reviews:

A. Determines the acceptability of the seismic and quality group classifications for system components as part of its primary review responsibility for DSRS Sections 3.2.1 and 3.2.2. In addition, as part of its primary review responsibility for DSRS Section 3.2.2, if the PWR pressurizer power-operated relief valves (PORVs) and block valves are relied upon to perform a safety-related function, such as plant cooldown in accordance with BTP 5-4, this organization will confirm the classification of the PORVs and block valves.

B. Reviews the effects of pipe breaks inside and outside of containment, such as pipe whip and jet impingement, as part of its primary review responsibilities for DSRS Section 3.6.2.

C. Determines that the components, piping, and structures are designed and tested in accordance with applicable codes and standards as part of its primary review responsibility for DSRS Sections 3.9.1 and SRP Sections 3.9.2 and 3.9.3.

D. Reviews adequacy of the inservice testing program of pumps and valves as part of its primary review responsibility for DSRS Section 3.9.6. The reviewer responsible for the review of reactor thermal hydraulic systems in PWRs should coordinate with the organization responsible for the mechanical engineering reviews to ensure that the RHR system configuration allows for full-flow testing of safety-related pumps and check valves and that provisions are made to allow for

the use of advanced techniques to detect degradation and monitor system performance. Review of the ECCS passive decay heat removal system to ensure that residual heat will be removed during accident conditions is coordinated and performed under DSRS Section 6.3.

6. The organization responsible for quality assurance reviews performs the following reviews:

   A. Evaluates the pre-operational and startup test programs to confirm that they are in conformance with the intent of Regulatory Guide (RG) 1.68 as part of its primary review responsibility for DSRS Section 14.2

   B. Has primary review responsibility for Task Action Plan items I.C.2 and I.C.6 of NUREG-0737 regarding procedures to ensure that system operability status is known, as part of its review responsibility for SRP Section 13.5.1.1

   C. Evaluates quality assurance as part of its primary review responsibility for SRP Chapter 17. Review cooling water systems that transfer decay heat from the RCI RHR to atmosphere is performed under DSRS Sections 9.2.2.

7. The organization responsible for the review of the mechanical effects of missiles on SSCs performs the following reviews:

   A. Evaluates flood protection as part of its primary review responsibility for DSRS Section 3.4.1.

   B. Identifies the SSCs to be protected against externally generated missiles and reviews the adequacy of protection against such missiles as part of its primary review responsibility for DSRS Sections 3.5.1.4 and 3.5.2.

   C. Reviews protection against internally generated missiles both inside and outside of containment as part of its primary review responsibility for DSRS Sections 3.5.1.1 and 3.5.1.2. Review of steam and feedwater systems that support initiation of normal plant cooldown is performed under DSRS Sections 10.3 and 5.4.14.

8. The organization responsible for the review of proposed preoperational and initial startup test programs ensures that RCI RHR function will remove core residual heat as part of its primary review responsibility for DSRS Section 14.2.

9. The organization responsible for the review of cooling water systems associated with the balance of plant reviews the plant design for protection against postulated piping failures outside containment, as part of its primary review responsibility for SRP Section 3.6.1.

10. The organization responsible for the review of environmental qualification of electrical equipment reviews the acceptability of, and environmental qualification test program for, RHR equipment exposed to a postaccident environment, including consideration of the postaccident environmental design and source term considerations described in NUREG-0737 Task Action Plan Item II.B.2 and NUREG-0718, as part of its review responsibility for DSRS Section 3.11.

11. The organization responsible for the review of fire protection performs a review of fire

protection as part of its primary review responsibility for DSRS Section 9.5.1.

12. The organization responsible for the electrical engineering and power systems reviews identifies the safety-related electrical loads and determines that power systems supplying motive or control power for the RHR system meet acceptable criteria and will perform these intended functions during all plant operating and accident conditions, as part of its primary review responsibility for DSRS Sections 8.1, 8.2, 8.3.1, and 8.3.2. In addition, this organization reviews the capability to withstand or cope with, and recovers from, a station blackout and coordinates with the review of the RHR system if the system is required to ensure adequate core cooling and/or decay heat removal, as part of its review under DSRS Section 8.4.

13. The organization responsible for the review of instrumentation and control systems reviews those systems for the RHR system to determine that it will perform its design function as required and conform to all applicable acceptance criteria, as part of its primary review responsibility for DSRS Sections 7.1 and 7.4.

14. The organization responsible for the review of health physics has primary review responsibility for DSRS Sections 12.1 through 12.5, including conformance with NUREG-0737 Task Action Plan Item II.B.2 and NUREG-0718, which involve a radiation and shielding design review and corrective actions to ensure adequate access to vital areas and protection of safety equipment.

15. The organization responsible for the review of applicable technical specifications evaluates the technical specifications as part of its primary review responsibility for DSRS Section 16.0.

16. Review of the reliability assurance program (RAP) is coordinated and performed under SRP Section 17.4.

17. Review of probabilistic risk assessments evaluating the RTNSS is coordinated and performed under SRP19.3.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations. The specific areas of review for the safety-related functions of RCI are as listed below. Additionally, the nonsafety-related areas of review are shown in italics. RTNSS B and C functions, if they apply, are also shown below. For nonsafety-related (non-risk-significant) functions, nothing applies unless noted below.

1. 10 CFR Part 50, Appendix A, GDC 1, as it relates to system components being assigned quality group classifications and application of quality standards in accordance with the importance of the safety function to be performed.

2. GDC 2, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the RHR system function of the RCI, specifically based on meeting Regulatory Position C-2 of RG 1.29 or its equivalent.

3. GDC 4, as it relates to dynamic effects associated with flow instabilities and loads

(e.g., water hammer).

4. GDC 5, as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

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

6. GDC 19, as it relates to control room requirements for normal operations and shutdown.

7. GDC 34, as it relates to requirements for an RHR system.

8. GDC 54, as it relates to piping systems penetrating primary reactor containment being provided with leak detection, isolation, and containment capabilities.

9. GDC 57, as it relates to closed system isolation valves on piping systems penetrating primary reactor containment.

10. NUREG-0737 Task Action Plan Item III.D.1.1, correlating to 10 CFR 50.34(f)(2)(xxvi), for applicants subject to 10 CFR 50.34(f), as it relates to the provisions for a leakage detection and control program to minimize the leakage from those portions of the RHR system outside of the containment that contain or may contain radioactive material following an accident.

11. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAACs that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations.

    RTNSS B and C apply for the review for ITAAC related to the importance of each defense-in-depth function.

12. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the AEA, and the NRC's regulations.

    RTNSS B and C apply for the review for ITAAC related to the importance of each defense-in-depth function.

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are set forth below. The DSRS is not a substitute for the NRC's regulations, and compliance with it is not required. Identifying the differences between this DSRS section and the design features, analytical techniques, and procedural measures proposed for the facility, and discussing how the proposed alternative provides an acceptable

method of complying with the regulations that underlie the DSRS acceptance criteria, is sufficient to meet the intent of 10 CFR 52.47(a)(9), "Contents of applications; technical information." The same approach may be used to meet the requirements of 10 CFR 52.79(a)(41) for COL applications.

The specific areas of review for the safety-related functions of RCI are as listed below. Additionally, the nonsafety-related areas of review are shown in italics. RTNSS B and C functions, if they apply, are also shown below. For nonsafety-related (non-risk significant) RCI functions, nothing applies unless noted below.

1. The mPower™ RCI RHR function should satisfy the functional, isolation, pressure relief, pump protection, and test requirements specified in Branch Technical Position (BTP) 5-4 and GDC 1.

2. The RCI and the RCI RHR function should be capable of withstanding the effects of natural phenomena like earthquakes, tornadoes, hurricanes, and floods to meet the requirements of GDC 2.

   Note: RTNSS B and C SSCs are designed to withstand the effects of natural phenomena without loss of function. SRP 19.3 provides further guidance related to the reliability and availability missions of RTNSS B and C SSCs.

3. To meet the requirements of GDC 4, Design features and operating procedures should be provided for the RCI RHR function to prevent damaging water hammer caused by such mechanisms as voided lines.

4. To meet the requirements of GDC 5, SSCs performing a safety function must not be shared between units of a multi-unit site, unless it can be shown that such sharing will not significantly impair the ability of the SSCs to perform their safety functions.

5. Extremely low failure probability of the reactor coolant pressure boundary must be demonstrated to meet the requirements of GDC 14.

6. Controls should be available to the operator to provide for RHR from the control room and an alternate shutdown location outside the control room during normal and shutdown operations as required by GDC 19.

7. Operation of the risk-significant RCI RHR function should be administratively controlled by procedure and coordinated with the operation of the ECCS passive decay heat removal system to provide for RHR from the reactor core at a rate such that specified fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as required by GDC 34.

8. Piping systems penetrating primary reactor containment must be provided with leak detection, isolation, and containment capabilities to meet the requirements of GDC 54.

9. Interfaces between the RHR system and component or service water systems should be designed so that operation of one does not interfere with, and provides proper support (where required) for, the other. In relation to these and other shared systems (e.g., emergency core cooling and containment heat removal systems), the RHR system must conform to GDC 5.

10. The guidelines of RG 1.82 regarding water sources for long term recirculation cooling following a loss-of-coolant accident should be considered in the design.

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 specific areas of review for the safety-related RCI functions are as listed below. Additionally, the nonsafety-related areas of review are shown in italics. RTNSS B and C functions, if they apply, are also shown below. For nonsafety-related (non-risk-significant) RCI, nothing applies unless noted below.

1. GDC 1 requires that SSCs important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. The RCI may be important to safety in that: 1) the RCI is relied upon to control RCS water chemistry to maintain the integrity of the RCS pressure boundary; 2) through connections to the RCS a RCI failure could adversely affect the integrity of the RCS or containment systems; and 3) portions of the RCI contain radioactive material. Meeting the requirements of GDC 1 (and the guidance of RG 1.26) ensures that the RCI will be designed, fabricated, erected and tested to generally accepted and recognized codes and standards that are sufficient to assure a quality system in keeping with the importance of the designated safety functions.

2. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without the loss of capability to perform their safety functions. The mPower™ RCI RHR system function is relied upon to provide defense in depth to the ECCS passive decay heat removal system for the removal of residual heat from the reactor core to maintain the reactor in a safe-shutdown condition. This avoids reliance on an mPower™ ESF design feature for normal plant operations. In addition, the RHR system may be capable of cooling the spent fuel pool. RG 1.29 provides guidance for determining which systems should be designated Seismic Category I; Regulatory Position C.1 provides guidance for safety-related portions and Regulatory Position C.2 addresses nonsafety-related systems and components. Meeting the requirements of GDC 2 will enhance plant safety by ensuring that the RHR system function of the RCI will be available to cool the core and/or the spent fuel pool during and following a seismic event.

   Note: RTNSS B and C SSCs are designed to withstand the effects of natural phenomena without loss of function. SRP 19.3 provides further guidance related to the reliability and availability missions of RTNSS B and C SSCs.

3. GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement. The safety risk-significant RHR function of the mPower™ RCI RHR system is to transfer heat from the reactor to the environment during and after plant shutdown instead of relying solely on the passive ECCS decay heat removal system. To ensure the availability of the decay heat removal function, the RHR system must be capable of performing heat transfer under the expected operational and postulated accident conditions for the plant. Heat transfer during postulated accident conditions for the plant is a function of the passive ECCS decay heat removal system, evaluated in DSRS Section 6.3. These conditions include

consideration of the dynamic effects of flow instabilities and the loadings caused by water hammer events. Compliance with GDC 4 enhances plant safety by providing assurance that the dynamic effects of events such as flow instabilities and water hammer will not affect the capability of the mPower™ RCI RHR system function to remove decay heat during operational and transient conditions.

4. GDC 5 prohibits the sharing of SSCs among nuclear power units unless it can be shown that such sharing will not significantly impair the ability of the SSCs to perform their safety functions, including, in the event of an accident in one unit, and orderly shutdown and cooldown of the remaining units. The RHR systems are relied upon to transfer decay heat from the reactor to the environment after a reactor shutdown. The RHR system must be designed such that the ability to perform this and other designated safety-related functions is not compromised for each unit regardless of equipment failures or other events that may occur in another unit. Meeting the requirements of GDC 5 enhances plant safety by providing assurance that the unacceptable effects of equipment failures or other events occurring in one unit of a multiunit site will not prevent an orderly shutdown and cooldown of the unaffected unit(s).

5. GDC 14 requires assurance that the reactor coolant pressure boundary (RCPB) will have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture. Failure of the RCPB may be postulated where the mechanisms of general corrosion and/or stress corrosion cracking induced by impurities in the reactor coolant are present. The RCI maintains acceptable purity levels in the reactor coolant through the removal of insoluble corrosion products and dissolved ionic material by filtration and ion exchange. In addition, the RCI maintains proper RCS chemistry by controlling total dissolved solids, pH, oxygen concentration, and halide concentrations within the acceptable ranges. Meeting the requirements of GDC 14 enhances facility safety by providing assurance that the probability of corrosion-induced failure of the RCPB will be minimized, thereby maintaining the integrity of the RCPB.

6. GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit during both normal operating and accident conditions, including the loss-of-coolant accident (LOCA). BTP 5-4 provides guidance for compliance with GDC 19 with regard to achieving cold shutdown from the control room using only safety-grade equipment. The mPower™ RCI RHR system function is required for safe shutdown and cooldown of the reactor during normal and accident transient conditions. Compliance with GDC 19 enhances plant safety by ensuring the availability of adequate instrumentation and controls in the control room to perform the required safety risk -significant functions of the RCI RHR system function under all anticipated conditions.

7. GDC 34 requires a system to remove fission product decay heat—residual heat—from the reactor core at a rate such that specified fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. The RCI serves as a defense in depth decay heat removal system. Meeting the requirements of GDC 34 enhances plant safety by ensuring sufficient heat removal from the core to maintain fuel design and reactor coolant pressure boundary design conditions. In addition, the system shall be designed with suitable redundancy and isolation capability to ensure that the safety function can be accomplished assuming a single failure of an active component either with or without a coincident loss of offsite power. The RHR system transfers the fission product decay and other residual heat from the reactor core. Removal of decay and residual heat is necessary to prevent core damage under both normal and accident

shutdown conditions. BTP 5-4 provides an acceptable approach to ensure compliance with GDC 34 with regard to accomplishing the RHR system safety functions assuming a single failure. Compliance with GDC 34 enhances plant safety by providing assurance that decay and RHR will be accomplished and the RCS pressure boundary and fuel cladding integrity will be maintained, thereby minimizing the potential for the release of fission products to the environment.

8. GDC 54 requires that piping systems that penetrate primary reactor containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems must be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. Piping in the RCI passes through the containment boundary and is provided with isolation valves and integrity verification capabilities. Compliance with GDC 54's containment isolation and leak detection requirements provides a high level of assurance that the containment will perform its safety function in the event of a postulated accident and will maintain the capability to prevent a significant uncontrolled release of radioactivity.

9. GDC 57 requires that a closed system shall have at least one containment isolation valve and that valve shall be located as close to the containment as practical.

III. REVIEW PROCEDURES

These review procedures are based on the identified DSRS acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

For design control document (DCD) and COL reviews, the reviewer uses the procedures to verify that the final design appropriately implements the initial design criteria and bases as set forth in the final safety analysis report (FSAR) technical submittal. The COL review also covers the proposed technical specifications to ensure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

The review includes all of the systems used to transfer residual heat from the reactor over the entire range of potential reactor coolant temperatures and pressures.

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 (SRP 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 (SRP 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 SRP 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 (SER) section.

3. Using the description given in the applicant's FSAR technical submittal, including component lists and performance specifications, the reviewer determines that the system RCI piping and instrumentation provide reasonable assurance that the system(s) will operate as intended, with or without available offsite power and given any single active component failure. To do this, the reviewer evaluates the piping and instrumentation diagrams (P&IDs)system description and schematics to confirm that piping arrangements permit the achievement of the required flowpaths and that sufficient process sensors are available to measure and transmit required information. The reviewer uses a failure modes and effects analysis (or similar system safety analysis) provided in the FSAR to determine conformance to the single failure criterion.

4. Where possible, comparisons should be made with actual performance data from similar systems in operating plants. The reviewer can use previously reviewed RHR designs as a guide, but must verify that any differences are justified.

5. From the system description and P&IDs, the reviewer determines that the isolation requirements of BTP 5-4 are satisfied.

6. The reviewer determines that the RCI RHR system design has provisions to prevent damage to the RCI RHR pumps in accordance with BTP 5-4. The reviewer checks the isolation valves in the suction line for potential closure, net positive suction head requirements, pump runout, and potential loss of miniflow line during pump testing. If

operator action is required to protect the pumps, the reviewer evaluates the instrumentation that will alert the operator and the adequacy of the timeframe for operator action.

The reviewer verifies that the applicant has considered the following guidance regarding the design of the RCI RHR miniflow systems necessary to ensure safety-related RCI RHR pump protection (see NRC Generic Letter 89-04 and Bulletins 86-01 and 88-04):

A. Ensure that the minimum cooling flow provided for the RCI RHR pumps is adequate under all conditions, including verification that the system configuration precludes pump-to-pump interaction during miniflow operation that could result in dead-heading one or more of the pumps. The miniflow must be sufficient to prevent damage to the pump(s) under all conditions.

B. The miniflow system should be designed such that the miniflow function can be performed assuming a single failure. A single failure should not result in conditions causing no flow through the RCI RHR pumps.

C. In cases where only the miniflow return line is available for pump testing, flow instrumentation must be installed on the miniflow return line. This instrumentation is necessary to provide flow rate measurements during pump testing so these data can be evaluated with the measured pump differential pressure to monitor for pump hydraulic degradation.

7. The staff reviews the RCI RHR system function to evaluate the adequacy of design features and procedures that have been provided to prevent damaging water hammer and degradation or loss of RHR pumps because of such mechanisms as voided lines. NUREG-0927 provides guidance for water hammer prevention and mitigation and Generic Letter 88-17 provides guidance for shutdown operation.

8. Using the system process diagrams, piping and instrumentation diagrams (P&IDs), system descriptions and schematics, failure modes and effects analysis, and component performance specifications, the reviewer determines that the RCI system(s) has the capacity to bring the reactor to conditions permitting operation of the RCI RHR system function in a reasonable period of time, assuming a single failure of an active component with only either onsite or offsite electric power available. For the purposes of this review, the NRC considers 36 hours a reasonable time period. The organization responsible for the review of steam and feedwater systems evaluates the initial cooldown phase for the mPower™ iPWR, so this review effort should be coordinated with that reviewer. For PWRs, if the PORVs are relied upon in the performance of a safety-related function such as plant cooldown for compliance with BTP 5-4, the PORVs must meet the guidance contained in Generic Letter 90-06 (see also NUREG-1316), as reviewed under DSRS Section 6.3. For new PWRs that use PORVs, the valves should be safety related.

9. The staff reviews the cooldown function to determine whether it can be performed from the control room assuming a single failure of an active component, with only either onsite or offsite electric power available. The applicant must justify any operation required outside of the control room. As with Item 6-4 above, the organization responsible for the review of steam and feedwater systems will evaluate the initial cooldown for the mPower™ iPWR.

10. By reviewing the system description and the P&IDs, the reviewer confirms that the RCI

RHR system function satisfies the pressure relief requirements of BTP 5-4.

11. By reviewing the piping arrangement and system description of the RCI RHR system, the reviewer confirms that the RCI RHR system meets the requirements of GDC 5 concerning shared systems.

12. The reviewer of reactor thermal hydraulic systems contacts the reviewer of steam and feedwater systems in conjunction with the review of the RCI RHR system function heat sink and refueling system interaction to exchange information and ensure that the reviews consider the interfacing parameters consistently. For example, the organization responsible for the review of cooling water systems associated with balance of plant review determines the maximum service or component cooling water temperature. The reviewer of reactor thermal hydraulic systems then evaluates the RCI RHR system function description to determine that the RCI RHR system design allows for this maximum temperature. Specifically, the reviewer should verify the maximum cooling capacity of the RCI nonregenerative heat exchanger using maximum component cooling water temperature.

13. The reviewer of reactor thermal hydraulic systems contacts the reviewer of instrumentation and control systems to obtain any needed information from that review. Specifically, the reviewer of instrumentation and control systems confirms that automatic actuation and remote-manual valve controls are capable of performing the functions required, and that sensor and monitoring provisions are adequate. The instrumentation and controls of the RCI RHR system must have sufficient redundancy to satisfy the single failure criterion.

14. The reviewer of reactor thermal hydraulic systems contacts the reviewer of containment integrity to exchange information related to their reviews.

15. The reviewer of reactor thermal hydraulic systems contacts the reviewer of the quality assurance and maintenance to discuss any special test requirements and to confirm that the proposed preoperational test program for the RCI RHR system function to demonstrate the capability of all systems and components associated with the removal of decay heat.

16. The reviewer evaluates the proposed plant technical specifications as follows:

    A. Confirm the suitability of the limiting conditions of operation, including the proposed time limits and reactor operating restrictions for periods when system equipment is inoperable because of repairs and maintenance.

    B. Verify that the frequency and scope of periodic surveillance testing is adequate.

17. The reviewer contacts the reviewers of structural analysis and seismic/geotechnical issues to confirm that the systems employed to remove residual heat are housed in a structure whose design and design criteria provide adequate protection against wind, tornadoes, floods, and missiles, as appropriate.

18. The reviewer provides information to other reviewers in those areas where the organization responsible for the review of reactor thermal hydraulic systems has a review responsibility that is not explicitly covered in steps 1– 15 above. These additional areas of review responsibility include :

A. Identification of engineered safety features and safe-shutdown electrical loads, and verification that the minimum time intervals for the connection of the engineered safety features to the standby power systems are satisfactory

B. Identification of vital auxiliary systems associated with the RCI RHR system function and determination of cooling load functional requirements and minimum time intervals

C. Identification of essential components associated with the main steam supply and the auxiliary feedwater system that are required to operate during and following shutdown

19. The reviewer considers compliance with acceptance criteria requirement II.9 of this document by verifying that a leakage control program includes those portions of the RCI RHR systems located outside of containment that contain or may contain radioactive material following an accident. The leakage control program should include periodic leak testing and measures to minimize leakage from the RHR systems.

20. As necessary, the reviewer verifies that actions have been taken to ensure the continued availability and high reliability of the decay heat removal systems during shutdown operations.

    For PWRs, design features should be incorporated to prevent a loss of RHR functions under reduced inventory and mid-Loop operations. The reviewer should verify that the RHR-specific guidance and measures contained in Generic Letter 88-17 and summarized as follows are satisfied:

    A. The reviewer verifies that the applicant/licensee will have measures in place to ensure that the RCS will remain stable and controlled while in a reduced inventory condition. These measures include both prevention of a loss of active RHR, and enhanced monitoring requirements to ensure timely response to a loss of active RHR, should such a loss occur, and the capability to switch to the passive decay heat removal system.

    B. The reviewer verifies that the applicant/licensee has the capability of continuously monitoring RCI RHR system performance and RCS characteristics important for core cooling whenever a the active RCI RHR system function is being used for cooling the RCS.

    C. The reviewer verifies that the RCI RHR system has visible and audible indications of abnormal conditions in temperature, level, and RHR system function performance parameters.

21. The reviewer verifies that new light-water reactor applicants have ensured high reliability of the shutdown decay heat removal system as follows (see SECY-90-016 and the associated staff requirements memorandum (SRM) dated June 26, 1990, SECY-93-087 and the associated SRM dated July 21, 1993, and NUREG-1449):

    A. The reviewer verifies that design provisions exist to reasonably ensure the continuity of flow through the core and RCI RHR system with low-liquid levels at the junction of the RHR system function suction lines and the RCS.

B. The reviewer verifies that provisions exist to ensure the availability of reliable systems for decay heat removal.

C. The reviewer verifies that the applicant has provided reliable measurements of liquid levels at the junction of the RCI RHR system function suction lines and the RCS.

D. The reviewer verifies that automatic closure interlocks for the RCI RHR suction isolation valves, if provided, are designed in such a manner as to minimize inadvertent valve closure during system operation.

22. The reviewer verifies that the applicant has reviewed its RCI RHR system design configurations to identify any piping connected to the RCS that could be subjected to temperature distributions that could result in unacceptable thermal stresses. This review should consider the potential for thermal stratification, thermal cycling, and thermal fatigue, given the RCI RHR system configuration. The reviewer verifies that appropriate action has been taken, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses (see NRC Bulletin 88-08). This review should focus on RCI RHR system function configurations; the organization responsible for mechanical engineering reviews under SRP Section 3.9.3 reviews the stress analysis and ensures that it conforms to the American Society of Mechanical Engineers (ASCE) Boiler and Pressure Vessel Code.

23. The reviewer will verify that, to the extent practical, the mPower™ RCI RHR system for advanced or evolutionary light-water reactors is designed to an ultimate rupture strength at least equal to the normal RCS operating pressure. The NRC states its regulatory position with respect to minimizing the potential for an intersystem LOCA in advanced or evolutionary light-water reactors in SECY-90-016 and SECY-93-087 and their associated SRM. All elements of the RCI RHR system are to be considered (e.g., instrument lines, pump seals, heat exchanger tubes, valve bonnets). The licensee should provide justification for elements not designed to an ultimate rupture strength at least equal to the normal RCS operating pressure.

24. When reviewing the mPower™ design, the reviewer should use SECY-94-084 to determine the amount of regulatory oversight necessary. In SECY-94-084, the staff developed new regulatory and review guidance for a reliability assurance program to establish the regulatory treatment of nonsafety systems. The mPower™ iPWR advanced light-water reactor (ALWR) design makes extensive use of passive systems to meet regulatory requirements. The mPower™ design has used the design philosophy of safety-related passive systems and nonsafety-related active systems for RHR. The RCI RHR function is included as a nonsafety-related, risk-significant active system for use during normal plant operation and to provide defense in depth to the safety-related passive system. The nonsafety-related active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. The extensive use of safety-related passive systems and the nonsafety-related active system design philosophy presents a departure from previous licensing practices.

25. 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 FSAR 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 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).

For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

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.

1. For iPWRs

The RCI RHR function for normal plant cooldown is accomplished by one of two methods. The traditional method provides for fine control of the RCS cooldown rate and is conducted in two phases: the initial cooldown phase and the RHR system operation phase. In the event of a loss of offsite power, the initial phase of cooldown is accomplished through the auxiliary feedwater system and the atmospheric dump valves using the steam and feedwater system. This equipment is used to reduce the reactor coolant system temperature and pressure to values that permit operation of the RCI RHR system equipment where the steam plant no longer provides for sufficient RHR. Subsequently, Section ___ of the SER discusses the review of the initial cooldown phase. The review of the RHR system operational phase is discussed below. The RCI RHR system removes core decay heat and provides long-term core cooling following the initial phase of a normal reactor cooldown. Alternately, because the RCI is designed for full RCS pressure, the entire RCS cooldown and long-term cooling requirements can be handled by the RCI RHR function, which rejects heat to the atmosphere using mechanical coolers. The scope of review of the mPower™ RCI RHR system for the plantfunction included piping and instrumentation diagrams, system descriptions and schematics, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential risk-significant components. The review included the applicant's proposed design criteria and design bases for the RCI RHR system function, analysis of the adequacy of those criteria and bases, and conformance of the design to those criteria and bases.

Based on the following, the staff concludes that the design of the RCI RHR system function is acceptable and meets the requirements of GDCs 1, 2, 4, 5, 14, 19, 34, 54, 57, and 10 CFR 50.34(f)(2)(xxvi):

A. The applicant has met GDC 1 requirements for the RCI RHR function. Acceptance is based on the structures, systems, and components important to safety as being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Recognized codes and standards shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or

modified as necessary to assure a quality product. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

B. The applicant has met GDC 2 requirements with respect to Regulatory Position C-2 of RG 1.29 concerning the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the RHR system.

C. The applicant has met GDC 4 requirements with respect to dynamic effects associated with flow instabilities and loads (e.g., water hammer).

D. The applicant has met the requirements of GDC 5 with respect to the sharing of SSCs by demonstrating that such sharing does not significantly impair the ability of the RHR system to perform its safety function, including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

E. The applicant has met the requirements of GDC 14 with respect to reactor coolant pressure boundary design, fabrication, erection and testing so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

F. The applicant has met GDC 19, with respect to the main control room requirements for normal operations and shutdown, and GDC 34, which specifies requirements for the RHR by meeting the regulatory positions in BTP 5-4.

G. The applicant has met GDC 34 requirements for RHR. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor at a rate such that specified acceptable fuel design limits and the design of the reactor coolant pressure boundary are not exceeded. The applicant has demonstrated that the RCI can perform its function of RHR under normal operating conditions. The applicant has demonstrated that the RCI in conjunction with the safety related ECC and non-safety-related systems can perform the function of RHR during accident conditions, assuming loss of offsite power and a single failure, and that portions of the system can be isolated so system safety functions are not compromised. The design has been evaluated for adequate margins related to heat exchanger heat removal performance during normal and accident conditions.

H. The applicant has met GDC 54 requirements, which provides reasonable assurance that the containment isolation system will isolate piping systems penetrating containment reliably as required.

I. The applicant has met GDC 57 requirements for automatic containment isolation for those portions of the system which penetrate primary reactor containment and are not part of the reactor coolant pressure boundary.

J. The applicant has met the parameters in Item III.D.1.1 of NUREG-0737, equivalent to 10 CFR 50.34(f)(2)(xxvi) for applicants subject to 10 CFR 50.34(f), with respect to leakage detection and control in the design of RHR systems outside containment that contain (or may contain) radioactive material following an accident.

V. IMPLEMENTATION

The staff will use this DSRS section in performing safety evaluations of mPower™-specific DC, or COL, applications submitted by applicants pursuant to 10 CFR Part 52. The staff will use the method described herein to evaluate conformance with Commission regulations.

Because of the numerous design differences between the mPower™ and large light-water nuclear reactor power plants, and in accordance with the direction given by the Commission in SRM- COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (Agencywide Document Access and Management System (ADAMS) Accession Number 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™-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 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 50.34(f), "Additional TMI-Related Requirements."

2. 10 CFR Part 50, Appendix A, GDC 2, "Design Bases for Protection Against Natural Phenomena."

3. 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

4. 10 CFR Part 50, Appendix A, GDC 5, "Sharing of Structures, Systems and Components."

5. 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary"

6. 10 CFR Part 50, Appendix A, GDC 19, "Control Room."

7. 10 CFR Part 50, Appendix A, GDC 34, "Residual Heat Removal."

8. 10 CFR Part 50, Appendix A, GDC 54, "Primary Systems Penetrating Containment"

9. BTP 5-4, “Design Requirements of the Residual Heat Removal System.”

10. RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.”

11. RG 1.29, “Seismic Design Classification.”

12. RG 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.”

13. RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.”

14. RG 1.182, “Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.”

15. RG 1.206, “Combined License Applications for Nuclear Power Plants (LWR Edition).”

16. RG 1.215, “Guidance for ITAAC Closure under 10 CFR Part 52.”

17. SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” January 12, 1990.

18. SRM, “SECY 90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements,” June 26, 1990.

19. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 2, 1993.

20. SRM, “SECY 93-087 ’Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” July 21, 1993.

21. Generic Letter 88-17, “Loss of Decay Heat Removal” October 17, 1988.

22. Generic Letter 89-04, “Guidance on Developing Acceptable Inservice Testing Programs,” April 3, 1989.

23. Generic Letter 90-06, “Resolution of Generic Issue 70, ‘Power-Operated Relief-Valve and Block Valve Reliability,’ and Generic Issue 94, ‘Additional Low-Temperature Overpressure Protection for Light-Water Reactors,” June 25, 1990.

24. Generic Letter 92-02, “Resolution of Generic Issue 79, ‘Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown’, March 6, 1992.

25. NRC Bulletin 86-01, “Minimum Flow Logic Problems That Could Disable RHR Pumps,” May 23, 1986.

26. NRC Bulletin 88-04, “Potential Safety-Related Pump Loss,” May 5, 1988.

27. NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems,” June 22, 1988, and Supplements 1 through 3.

28. NUREG-0660, “NRC Action Plan Developed as a Result of the TMI-2 Accident.”

29. NUREG-0718, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License.”

30. NUREG-0737, “Clarification of TMI Action Plan Requirements.”

31. NUREG-1316, “Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants.”

32. NUREG-1449, “Shutdown and Low-Power Operation at Nuclear Power Plants in the United States.”

33. NUREG-0927, Revision 1, “Evaluation of Water Hammer Occurrences in Nuclear Power Plants,” March 1984.

34. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994.

35. SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” May 22, 1995.