U.S. NUCLEAR REGULATORY COMMISSION

DESIGN-SPECIFIC REVIEW STANDARD FOR mPOWER™ iPWR DESIGN

15.4.10 STARTUP OF AN INACTIVE PUMP OR PUMPS AT AN INCORRECT TEMPERATURE, AND FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN CORE FLOW RATE

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

In the mPower™ integrated pressurized-water reactor(PWR) design, there are eight reactor coolant pumps (RCPs) located around the Pressurizer. Inside the reactor pressure vessel (RPV), there are two flow dividers. The flow divider direct the reactor coolant system (RCS) flow to the RCPs and the RCPs discharge the flow to the bottom of the RPV through the downcomers essentially forming two internal recirculation loops in the RPV. These pumps are expected to be powered from off-site power sources and may be powered from more than a single source. Out of eight total pumps, some may be tripped due to loss of power supply or due to mechanical problems. There is a potential for startup of an inactive pump or pumps at an incorrect temperature and flow controller malfunction causing an increase in core flow.

A number of anticipated operational occurrences (AOOs) may cause either increased core flow or introduction of cooler or de-borated water into the core. These AOOs result in an increase in core reactivity due to decreased moderator temperature or core void fraction. This desigh-specific review standard (DSRS) section is intended to be applicable to all such AOOs. Each of these AOOs should be discussed in individual sections of the applicant's safety analysis report (SAR) or design control document (DCD), as specified in NRC Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" and RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

The specific areas of review are as follows:

1. Startup of an idle reactor coolant pump or pumps.

2. Flow controller malfunction causing increased recirculation flow.

3. Startup of a pump in an inactive loop.

4. 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).

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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.

The review of the core flow increase AOOs considers the sequence of events, the analytical model, the values of parameters used in the analytical model, and the predicted consequences of the AOOs. The reviewer concentrates on the need for the reactor protection system and operator action to secure and maintain the reactor in a safe condition.

The analytical methods are reviewed 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. In addition, the values of all the parameters used in the new analytical model, including the initial conditions of the core and system, are reviewed.

The predicted results of the AOOs are reviewed to ensure that the consequences meet the acceptance criteria given in Subsection II below. Furthermore, the results of the AOOs are reviewed to ascertain that the values of pertinent system parameters are within ranges expected for the type and class of reactor under review.

Review Interfaces

Other DSRS sections interface with this section as follows:

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

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

3. Values of the parameters in the analytical models of the reactor core are reviewed for compliance with plant design and specified operating conditions, acceptance criteria for fuel cladding damage limits are determined, and the core physics, fuel design, and core thermal-hydraulics data in the safety analysis report (SAR) analysis are reviewed under DSRS Sections 4.2, 4.3, and 4.4.

4. Instrumentation and controls aspects of the sequence described in the SAR is 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.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 10 and GDC 20, as they relate to the reactor coolant system being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations and AOOs.

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15 and GDC 28, as they relate to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations and AOOs.

4. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs. This is accomplished by ensuring that appropriate margins for malfunctions, such as stuck rods, is accounted for.

5. The basic objectives of the review of the AOOs described above are:

   A. To identify which of the AOOs are the most limiting.

   B. To verify that, for the most limiting AOOs, the plant responds in such a way that the criteria regarding fuel damage and system pressure are satisfied.

<u>DSRS Acceptance Criteria</u>

Specific DSRS acceptance criteria acceptable to meet the relevant requirements of the U.S. Nuclear Regulatory Commission (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 Title 10 of the *Code of Federal Regulations* (10 CFR) 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.

Using the American Nuclear Society (ANS) standards as guidance, the specific criteria necessary to meet the relevant requirements of the regulations identified above for incidents of moderate frequency are as follows:

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

B. Fuel-cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95% probability (95% confidence DNBR limit based on acceptable correlations) (see DSRS Section 4.4).

C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

D. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of AOOs addressed in this DSRS section.

E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, shall be identified and assumed in the analysis and should satisfy the guidance stated in RG 1.53.

F. The guidance provided in SECY 77-439, SECY 94-084 and RG 1.206 with respect to the consideration of the performance of non-safety related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified.

The applicant's analysis of the most limiting AOOs should be performed using an acceptable model. If analytical methods that have not been approved are proposed by the applicant, they are evaluated by the staff for acceptability.

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

1. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed to be operating, plus an allowance of 2% to account for power measurement uncertainty, unless (a) a lower number can be justified through the measurement uncertainty methodology and evaluation, or (b) unless the uncertainty has otherwise been accounted for (see DSRS Section 4.4). An analysis to determine the effects of a flow increase must be made for each allowed mode of operation (i.e., one, two or three loops initially operating) or the effects referenced to a limiting case.

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

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

4. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 as determined by the organization responsible for instrumentation and controls.

The reviewer shall verify that the protection system (1) automatically initiates the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded for this event, and (2) senses the plant conditions and initiates the operation of structures, systems, and components (SSCs) important to safety.

Technical Rationale

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

1. 1Compliance with GDC 10 requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

   GDC 10 is applicable to this DSRS section because the reviewer evaluates the consequences of events associated with startup of an inactive loop or recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in core flow rate. This section, DSRS Sections 4.2 through 4.4, and 7.2 through 7.5, and RG 1.53 and 1.105, provide guidance for ensuring that the reactor core, coolant, control and protection systems are designed with appropriate margin.

   Meeting the requirements of GDC 10 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

2. GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to assure adequate safety, and of controls that can maintain these variables and systems within prescribed operating ranges.

   GDC 13 applies to this section because the reviewer evaluates the sequences of events, including automatic actuations or protection systems, and manual actions, and determines whether the sequence of events is justified, based upon the expected values of the relevant monitored parameters and instrument indications.

3. Compliance with GDC 15 requires that the reactor coolant system and associated auxiliary, control and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including AOOs.

   GDC 15 is applicable to this DSRS section because the reviewer evaluates the consequences of events associated with startup of an inactive internal loop or internal recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in core flow rate. This section, DSRS Sections 4.2 through 4.4, and 7.2 through 7.5, and RG 1.53 and 1.105, provide guidance ensuring that the reactor coolant system and associated auxiliary, control and protection systems are designed with appropriate margin.

   Meeting the requirements of GDC 15 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.

4. Compliance with GDC 20 requires that the protection system be designed (a) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs, and (b) to sense accident conditions and to initiate the operation of SSCs important to safety.

   GDC 20 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase core flow rate. This section, DSRS Sections 4.2 through 4.4, and 7.2 through 7.5, and RG 1.53 and 1.105, provide guidance for ensuring that the reactor coolant system is designed with appropriate margin. Thus, when the reactor protection system senses an

accident condition, it will initiate the operation of SSCs important to safety to ensure that SAFDLs are not exceeded.

Meeting the requirements of GDC 20 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

5. Compliance with GDC 26 requires that one of the reactivity control systems shall use control rods capable of reliably controlling reactivity changes to ensure that under conditions of normal operation including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded.

   GDC 26 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive internal loop or a internal recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in core flow rate. This section, DSRS Sections 15.4.4, 4.2 through 4.4, and 7.2 through 7.5, and RG 1.53 and 1.105, provide guidance for ensuring that the reactivity control system (control rods) is capable of reliably controlling reactivity changes with appropriate margin for malfunctions such as stuck rods.

   Meeting the requirements of GDC 26 provides assurance that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

6. Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure 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) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and cold water addition. In addition, the introduction of hot water into the core of a PWR with a positive moderator temperature coefficient should be considered.

   GDC 28 is applicable to this section because the reviewer evaluates the consequences of the events associated with startup of an inactive loop or a recirculation loop at an incorrect temperature and with a flow controller malfunction causing an increase in core flow rate. This section, DSRS Sections 4.2 through 4.4, and 7.2 through 7.5, and RG 1.53 and 1.105, provide guidance for ensuring that the reactor coolant system and associated auxiliary, control and protection systems are designed with appropriate margin to ensure that the reactor coolant pressure boundary will not be breached.

   Meeting the requirements of GDC 28 provides assurance that the reactor coolant pressure boundary will not be breached during any condition of normal operation, including the effects of AOOs.

III. REVIEW PROCEDURES

For each area of review specified in Subsection I of this DSRS section, the review procedure is identified below. These review procedures are based on the identified DSRS acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the

applicant's evaluation of how the proposed alternatives to the DSRS criteria 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, the construction permit (CP), operating license (OL) and COL reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL or 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 Standard Review Plan (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 (RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants, "DSRS Section 14.2, and DSRS Section 13.4, Table 13.4, Item 19).

   F. Inspections, tests, analyses, and acceptance criteria (DSRS Chapter 14).

2. In accordance with 10 CFR 52.47(a)(8),(21), and (22), for new reactor license applications submitted under Part 52, the applicant is required to (1) address the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues that are identified in the version of NUREG-0933 current on the date six months before application and that are technically relevant to the design; (2) demonstrate how the operating experience insights have been incorporated into the plant design; and, (3) provide information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island (TMI) requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Reference: 10 CFR 52.47(a)(21), 10 CFR 52.47(a)(22) , and 10 CFR 52.47(a)(8), respectively.

These cross-cutting review areas should be addressed by the reviewer for each technical subsection and relevant conclusions documented in the corresponding safety evaluation report section.

The sequence of core flow increase 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 credit taken for the functioning of normally operating plant systems.

4. The operation of engineered safety systems that are required.

5. The extent to which operator actions are required.

6. That appropriate margin for malfunctions, such as stuck rods, is accounted for.

7. That instrumentation uncertainties of system and operating parameters are appropriately accounted for.

If the SAR or DCD states that a particular core flow AOO is not as limiting as some other similar AOO, the reviewer evaluates the justification presented by the applicant. The applicant should present a quantitative analysis in the SAR or DCD of the increase in flow AOO that is determined to be most limiting. For this AOO, the reactor systems reviewer, with the aid of the instrumentation and controls reviewer, reviews the timing of the initiation of protection, engineered safety feature, and other systems needed to limit the consequences of the core flow increase AOO to acceptable levels. The reviewer compares the predicted variation of system parameters with various trip setpoints. The review of Chapter 7 of the SAR or the corresponding chapter of the DCD confirms that the instrumentation and control system design is consistent with the requirements for safety system actions for these events.

To the extent required, the reviewer evaluates the effect of single active failures of safety systems and components that may alter the course of the AOO. This phase of the review uses the system review procedures described in the DSRS sections for Chapters 5, 6, 7, and 8 of the SAR or the corresponding chapters of the DCD. The reviewer considers and evaluates the possibility of a single failure that would permit the loop operation prior to startup of a pump in an internal idle loop . If this could occur, the resulting rate of reactivity insertion could be greater than for other AOOs of this group.

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 to determine if these models have been previously reviewed and found acceptable by the staff. If not, a generic review of the model proposed by the applicant is initiated.

The values of system parameters and initial core conditions, including fuel data and system conditions used as input to the model, are reviewed. Of particular importance are the reactivity coefficients and control rod worths used in the applicant's analysis, and the variation of moderator temperature, void and Doppler coefficients of reactivity with core life. The

justification provided by the applicant to show that the selected core burnup yields the minimum margins is evaluated.

The reviewer should review SAR Section 4.4 or the corresponding chapter of the DCD to evaluate how uncertainties of the input parameters are applied in the DNBR analyses.

The results of the analysis are reviewed and compared with the acceptance criteria presented in Subsection II of this DSRS section regarding the maximum pressure in the reactor coolant and main steam systems, as well as minimum DNBR. Time-related variations of the following parameters should be reviewed for consistency:

- reactor power
- heat fluxes (average and maximum)
- reactor coolant system pressure
- core and internal recirculation loop coolant flow rates
- coolant conditions (inlet temperature, core average temperature, core average steam volume fraction, average exit and hot channel exit temperatures, and steam fractions)
- steam line pressure
- pressure relief valve flow rate and quality
- flow rate from the reactor coolant system to the containment system (if applicable).

The values of the more important of these parameters for the core flow increase AOOs are compared with those predicted for other similar plants to see that they are within the range expected.

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 final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the DCD. 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).

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.

A number of plant AOOs can result in a core flow increase. Those that might be expected to occur with moderate frequency are the startup of an idle internal recirculation pump , flow controller malfunction causing increasing core flow, startup of a pump in an inactive internal reactor coolant loop, and startup of a pump in an initially isolated inactive reactor coolant pump loop. All these postulated AOOs have been reviewed. It was found that the most limiting with regard to core thermal margins and pressure within the reactor coolant and main steam systems was the AOO. This AOO was analyzed by the applicant using a mathematical model that has been previously reviewed and found acceptable by the staff. The parameters used as input to this model were reviewed and found to be suitably conservative.

The staff concludes that the plant design with regard to AOOs that result in an increase in coolant flow through the reactor core is acceptable and meets the relevant requirements of GDCs 10, 15, 20, 26 and 28. This conclusion is based on the following:

1. The applicant has met the requirements of GDCs 10, 20 and 26 with respect to demonstrating that SAFDLs are not exceeded for this event.

2. The applicant meets GDC 13 requirements by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instruments' prescribed operating ranges.

3. The applicant has met the requirements of GDCs 15 and 28 with respect to ensuring that the design conditions of the reactor coolant pressure boundary are not exceeded because the protection system operates to maintain the maximum pressure within the reactor coolant and main steam system pressures below 110% of the design values.

4. The applicant has met the positions of RG 1.53, SECY 77-439, SECY 94-084 and RG 1.206 as related to the single-failure criterion and RG as related to instrument actuations of SSCs important to safety.

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

V. IMPLEMENTATION

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

This regulation states, in part, that the application must contain “an evaluation of the standard plant design against the SRP revision in effect six months before the docket date of the application.” The content of this DSRS section has been accepted as an alternative method for complying with 10 CFR 52.47(a)(9) as long as the mPower™ 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. RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

2. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

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

4. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. GDC 10, "Reactor Design."

7. GDC 13, “Instrumentation and Control.”

8. GDC 15, "Reactor Coolant System Design."

9. GDC 20, "Protection System Functions."

10. GDC 26, "Reactivity Control System Redundancy and Capability."

11. GDC 28, "Reactivity Limits."

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

13. RG 1.105, "Instrument Spans and Setpoints."

14. ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," (replaced ANSI N18.2-1974, reaffirmed 1988, withdrawn 1998).

15. ANSI/ANS 52.1-1983, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants," (replaced ANS Trial Use Standard N212-1974, reaffirmed 1988, withdrawn 1998).

16. NUREG-1801, "Generic Aging Lessons Learned Report," Rev. 1, v.1-2.

17. SECY-77-439, "Single Failure Criterion."

18. SECY-94-084, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs."

19. NUREG-0933, "A Prioritization of Generic Safety Issues."

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