U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 5-2

OVERPRESSURIZATION PROTECTION OF PRESSURIZED-WATER REACTORS WHILE OPERATING AT LOW TEMPERATURES

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of reactor thermal-hydraulic systems in BWRs and PWRs.

Secondary - None

A. BACKGROUND

GDC 15 of Appendix A to 10 CFR Part 50 requires that "the Reactor Coolant System and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

AOOs, as defined in Appendix A to 10 CFR Part 50, are "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power."

Appendix G to 10 CFR Part 50 provides the fracture toughness requirements for reactor pressure vessels under certain conditions. To ensure that the Appendix G limits of the RCPB are not exceeded during any AOOs, technical specification pressure-temperature limits are provided for operating the plant. The primary concern of this position is that, during startup and

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shutdown conditions at low temperature, especially in a water-solid condition, the RCS pressure might exceed the reactor vessel pressure-temperature limitations in the technical specifications established for protection against brittle fracture. Any one of a variety of malfunctions or operator errors could generate this inadvertent overpressurization. Many incidents have occurred in operating plants as described in NUREG-0138.

NUREG-0138 includes additional discussion on the background of this position.

B. BRANCH TECHNICAL POSITION

1. A system should be designed and installed that will prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures. The system should be capable of relieving pressure during all anticipated overpressurization events at a rate sufficient to satisfy the technical specification limits, particularly while the RCS is in a water-solid condition.

2. The low-temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least RT(NDT) + 50°C (90°F) at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations.

3. The system should be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event (e.g., operator error, component malfunction) should not be considered as the single active failure. The analyses should assume the most limiting allowable operating conditions and systems configuration at the time of the postulated cause of the overpressure event.

   All potential overpressurization events should be considered when establishing the worst-case event. Some events may be prevented by using protective interlocks or by locking out power. These events should be identified individually. If the analysis excludes the events, the controls to prevent these events should be in the plant technical specifications.

4. The design of the system should use Institute of Electrical and Electronics Engineers (IEEE) Standard 603 as guidance. The system may be manually enabled; however, an alarm should be provided to alert the operator to enable the system at the correct plant condition during cooldown. Positive indication should be provided to indicate when the system is enabled. An alarm should activate when the protective action is initiated. The reviewer responsible for instrumentation and controls will assist in reviews of the design criteria and the design for the low-temperature overpressure protection system controls and instrumentation, as described in Subsection I of SRP Section 5.2.2.

5. To ensure operational readiness, the overpressure protection system should be testable. Technical specification surveillance requirements should include the following:

   A. A test performed to ensure operability of the system (exclusive of relief valves) before each shutdown.

B. A test for valve operability, as a minimum, to be conducted as specified in the ASME Code Section XI.

6. The system must meet the requirements of Regulatory Guide 1.26 and Section III of the ASME Code.

7. The design of the overpressure protection system should function during an operating-basis earthquake. It should not compromise the design criteria of any other safety-grade system with which it would interface, such that the requirements of Regulatory Guide 1.29 are met.

8. The overpressure protection system should not depend on the availability of offsite power to perform its function. The system should be operable from battery-backed power sources, not necessarily Class 1E buses.

9. Overpressure protection systems that take credit for active component(s) to mitigate the consequences of an overpressurization event should include additional analyses considering inadvertent system initiation/actuation or should provide justification that existing analyses bound such an event.

10. If pressure relief is from a low-pressure system not normally connected to the primary system, interlocks that would isolate the low-pressure system from the primary coolant system should not defeat the overpressure protection function (see Branch Technical Position 7-1).

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 15, “Reactor Coolant System Design.”

2. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

3. NUREG-0138, “Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR, to NRR Staff.”

4. IEEE Standard 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (as endorsed by Regulatory Guide 1.153).

5. ASME Boiler and Pressure Vessel Code, Section II, “Materials Specifications.”

6. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”

7. Branch Technical Position (BTP) 7-1, “Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System.”

8. Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”

9. Regulatory Guide 1.29, “Seismic Design Classification.”