U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

5.2.2 OVERPRESSURE PROTECTION

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of reactor systems at boiling-water reactors and pressurized-water reactors

Secondary - Organization responsible for environmental qualification, human factors, instrumentation and controls, materials and chemical engineering, mechanical engineering, plant systems, quality assurance, and technical specifications

I. AREAS OF REVIEW

The application of safety and relief valves (SRVs) and the reactor protection system ensures overpressure protection for the reactor coolant pressure boundary (RCPB) during operation at power. The application of pressure-relieving systems that function during low-temperature operation ensures overpressure protection for the RCPB during low-temperature operation of the plant (startup, shutdown).

The pressure-retaining portions and supports of mechanical equipment shall be Safety Class 1 if they form part of the RCPB and have requirements that fall within the scope of Section III of Division 1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). Different allowable stress limits (service limits) for pressure-retaining components provide different margins of failure and hence different reliability levels for the pressure-retaining function of the RCPB. Plant conditions and the operating status of the reactor determine the allowable ASME Code service limits. For normal plant operation, the design pressure limit (stress limit) of the RCPB shall not be exceeded. However, the ASME Code allows the design pressure to be exceeded by 10 percent for anticipated operational occurrences (AOOs) such as an inadvertent emergency core cooling system actuation. The ASME Code also permits the design pressure to be exceeded by 20 percent for infrequent events such as the loss of offsite alternating current power. These transient pressure load allowances are based on the frequency and duration of the event and should be reviewed and approved by the NRC staff. American National Standards Institute (ANSI)/American Nuclear Society (ANS) 51.1 and ANSI/ANS 52.1, respectively, include useful discussions of overall plant design requirements.

Additional consideration should be given to License Amendment Requests (LARs) which may impact analyzed accident scenarios that result in RCPB overpressure. The validity of the acceptance criteria specified in this SRP section should be confirmed with regards to a LAR.

The specific areas of review are as follows:

1. System Design

The staff assesses information on the design of the overpressure protection system, including any subsystems and supporting systems, to gain familiarity with the design and operation of the system. Such information includes the following:

A. For boiling water reactors (BWRs), the area of review for operation at power includes SRVs on the main steamlines and piping from these valves to the suppression pool. The BWR design also may incorporate interfacing systems, such as an isolation condenser, to prevent challenges to the SRVs during normal operations. The BWR description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for power operation.

B. For BWRs the areas of review do not inlcude any special area of review for low-temperature operation of the plant (startup, shutdown) because BWRs do not operate or perform maintenance in water-solid conditions. However, a cold overpressure incident occurred at a foreign BWR, of U.S. design, during test conditions, as outlined in Section 4 of NUREG-1511, Supplement 2. Therefore, the reviewer should confirm (1) that operation of the condensate and feedwater pumps, high-pressure injection control rod drive pumps, low-pressure coolant spray or injection system, high-pressure coolant spray system, or the standby liquid control system is prevented administratively or (2) that system operation cannot generate sufficient pressure head during low-temperature operation leading to a possible cold overpressure event provided that the applicable pressure-temperature curves, the vessel temperature and the pressure rating of the available means of injecting water into the vessel.

C. For pressurized water reactors (PWRs), the area of review for operation at power includes pressurizer, SRVs, and the piping from these valves to a quench tank or to containment atmosphere on the primary side, as well as steam generator SRVs on the secondary side. The PWR description of the basic design concept;

the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for power operation.

D. For PWRs, the area of review for low-temperature operation of the plant (startup, shutdown) includes relief valves with piping to a quench tank or the containment sump, the makeup and letdown system, and the residual heat removal (RHR) system that may be operating when the primary system is water solid. The PWR description of the basic design concept; the systems, subsystems, and support systems providing overpressure protection to the RCPB; the components and instrumentation employed in these systems; and process and instrumentation diagrams should be reviewed for low temperature operation.

2. Testing and Inspections. The areas of review include an examination of the adequacy of the proposed pre-operational and initial startup test programs.

3. Technical Specifications. The areas of review include technical specifications to assure that they are adequate with regard to limiting conditions of operation and periodic surveillance testing.

4. Inspection, Test, Analysis, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the applicant's proposed information on the ITAAC associated with the systems, structures, and components (SSCs) related to this SRP section is reviewed in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria - Design Certification." The staff recognizes that the review of ITAAC is performed after review of the rest of this portion of the application against acceptance criteria contained in this SRP section. Furthermore, the ITAAC are reviewed to assure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

5. COL Action Items and Certification Requirements and Restrictions. COL action items may be identified in the NRC staff's final safety evaluation report (FSER) for each certified design to identify information that COL applicants must address in the application. Additionally, DCs contain requirements and restrictions (e.g., interface requirements) that COL applicants must address in the application. For COL applications referencing a DC, the review performed under this SRP section includes information provided in response to COL action items and certification requirements and restrictions pertaining to this SRP section, as identified in the FSER for the referenced certified design.

Review Interfaces

The listed SRP sections interface with this section as follows:

1. Review of seismic design criteria for components of the overpressure protection system (under SRP Sections 3.2.1 and 3.2.2).

2. Review of proposed inservice testing of pumps and valves to ensure that overpressure components will perform their safety functions (under SRP Section 3.9.6).

3. Review of seismic and dynamic qualification for components of the overpressure protection system (under SRP Section 3.10).

4. Review of environmental qualification for components of the overpressure protection system (under SRP Section 3.11).

5. Review of the design of systems that interface with the reactor coolant system (RCS) with regard to the capability of the interfacing system to withstand full RCS pressure (under SRP Section 3.12).

6. Review of the fracture toughness of the RCPB and reactor vessel and the pressure-temperature limits and pressurized thermal shock analysis (under SRP Sections 5.2.3, 5.2.4, 5.3.1, and 5.3.2).

7. For PWRs, review of the pressurizer relief tank (quench tank) capability to condense and cool the discharge from the pressurizer SRVs (under SRP Section 5.4.11).

8. For BWRs, review of the isolation condenser for sufficient capacity to preclude actuation of the overpressure protection system (under SRP Section 5.4.13).

9. For BWRs, review of the suppression pool capability to condense and cool the discharge from the safety valves (under SRP Section 6.2.1.1.C).

10. For PWRs, review of the design of power-operated relief valves (PORVs) and PORV block-valves to ensure that these valves will function to protect the RCPB against overpressurization in plant designs if these valves are associated with a reactor coolant depressurization system (under SRP Section 6.3).

11. Review of the adequacy of controls and instrumentation for the automatic and manual actuation of overpressure protection components (under SRP Section 7.6).

12. Review of proposed preoperational and initial startup test programs to ensure that overpressure components will perform their safety functions (under SRP Section 14.2).

13. Review of proposed ITAAC associated with SSCs for RCPB overpressure protection (under SRP Section 14.3).

14. Review of technical specifications (under SRP Section 16.0).

15. Review of quality assurance requirements (under SRP Sections 17.1 and 17.2).

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

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

1. General Design Criterion (GDC) 15, as it relates to designing the RCS and associated auxiliary, control, and protection systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

2. GDC 31, as it relates to designing the RCPB with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fractures is minimized.

3. 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that RCS SRVs meet Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

4. 10 CFR 52.47(a)(8)provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

5. 10 CFR 52.79(a)(17) provides the requirement for COL applications to comply with the technically relevant information in 10 CFR 50.34. This includes the TMI-related requirements specified by 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi).

6. 10 CFR 52.47(b)(1), as it relates to ITAAC (for design certification) sufficient to assure that the SSCs in this area of review will operate in accordance with the certification.

7. 10 CFR 52.80(a)(1), as it relates to ITAAC (for combined licenses) sufficient to assure that the SSCs in this area of review have been constructed and will be operated in conformity with the license, provisions of the Atomic Energy Act and the Commission's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for review described in Subsection I of this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. Material Specifications

The requirements of GDC 1, GDC 30, and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of the ASME Code and by acceptable application of material code cases, as described in Regulatory Guide 1.84.

The specifications for permitted materials are identified in Appendix I to Section III of the ASME Code or described in detail in Parts A, B, and C of Section II of the ASME Code. Regulatory Guide 1.84 describes acceptable material code cases and guidelines for application in light-water-cooled nuclear power plants that may be used in conjunction with the above specifications.

2. Design Requirements for BWRs Operating at Power

A. For overpressure protection during power operation of the BWR reactor, the designs of the pilot-operated relief valves with auxiliary actuation devices, isolation condensers, or other pressure dissipation systems should have sufficient capacity to preclude actuation of safety valves during normal operational transients when assuming the following conditions at the plant:

i. The reactor is operating at the licensed core thermal power level.

ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure.

iii. All components, instrumentation, and controls function normally.

B. The design of safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000. Sufficient available margin should account for uncertainties in the design and operation of the plant, assuming the following:

i. The reactor is operating at a power level that will produce the most severe overpressurization transient.

ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

iii. The second safety-grade signal from the reactor protection system initiates the reactor scram.

iv. The discharge flow is based on the rated capacities specified in ASME III for each type of valve.

v. The design of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000.

C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of ASME Code Article NB-7511.1.

3. <u>Design Requirements for PWRs Operating at Power</u>

A. For overpressure protection during power operation of the PWR reactor, the design of the PORVs or the pressurizer should have sufficient capacity to preclude actuation of safety valves during normal operational transients, when assuming the following conditions at the plant:

i. The reactor is operating at the licensed core thermal power level.

ii. All system and core parameters have values within normal operating range that produce the highest anticipated pressure.

iii. All components, instrumentation, and controls function normally.

B. The designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000. Also, sufficient available margin should account for uncertainties in the design and operation of the plant assuming:

I. The reactor is operating at a power level that will produce the most severe overpressurization transient.

ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.

iii. The second safety-grade signal from the reactor protection system initiates the reactor scram.

iv. The discharge flow is based on the rated capacities specified in ASME Code Article NB-7000 for each type of valve.

In addition, the designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe infrequent event, as specified by ASME Code Article NB-7000.

C. A single malfunction or failure of an active component should not preclude safety-related portions of the system from functioning as required during normal operations, adverse environmental occurrences, and accident conditions, including loss of offsite power.

Full credit is allowed for spring-loaded safety valves designed in accordance with the requirements of ASME Code Article NB-7511.1.

4. <u>Design Requirements for PWRs Operating at Low Temperature (Startup, Shutdown)</u>

The design of the low-temperature overpressure protection (LTOP) system or the cold overpressure mitigation system (COMS) should be in accordance with the requirements of Branch Technical Position (BTP) RSB 5-2, attached to this SRP section. The LTOP system or COMS should be operable during startup and shutdown conditions below the enable temperature defined in paragraph II.2 of BTP RSB 5-2.

5. Testing and Inspections

The performance of tests and inspections should occur before operation and during startup to functionally demonstrate that the overpressure protection system, as installed, meets all design requirements.

6. Technical Specifications

The technical specifications should specify appropriate limiting conditions of operation and inservice surveillance to ensure continued system reliability, including, for PWRs, specific limiting conditions of operation and testing of the LTOP system as specified in NUREG-1430 through NUREG-1434, Generic Letters No. 82-16, 83-02, and 90-06.

7. TMI Action Plan Requirements

Section II.D.1 of the TMI Action Plan requires an applicant submit a plant specific report regarding relief valve (RV) and safety valve (SV) testing. Section II.D.3 of the TMI Action Plan requires that RVs and SVs be provided with direct valve position indication. Generic Letters No. 82-16 and 83-02 requires sections II.D.1 and II.D.3 be covered by technical specifications while NUREG -0737 section II.K.3.3 specifies reporting for section II.D.1 and II.D.3.

Technical Rationale

The technical rationale for application of these requirements and/or SRP acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 15 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 condition. The overpressure protection system maintains RCS pressure within acceptable design limits during certain analyzed transients. Application of GDC 15 to the overpressure protection system provides assurance that the RCPB will have an extremely low probability of failure during transients.

2. GDC 30 requires that the reactor coolant pressure boundary be designed, fabricated, erected and tested to the highest quality standards practical. Application of GDC 30 to overpressure protection system provides assurance that the reactor coolant pressure boundary will have an extremely low probability of failure because of manufacturing or design defects.

3. GDC 31 requires that the RCPB is designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. During certain conditions in which the RCPB might behave in a brittle manner, the overpressure protection system maintains the RCS pressure below brittle fracture limits. Application of GDC 31 to the overpressure protection system

provides assurance that the RCPB will have an extremely low probability of failure because of brittle fracture.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

For each area of review specified in Subsection I of this SRP section, the review procedure is identified below. These review procedures are based on the identified SRP acceptance criteria. For deviations from these specific acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. System Design

The staff uses the procedures below during the review to assure that the design criteria and bases and the preliminary design as set forth in the preliminary safety analysis report (SAR) meet the acceptance criteria in Subsection II of this SRP section.

The procedures below are used to verify that the initial design criteria and bases have been appropriately implemented in the final design in the final SAR (FSAR). The ASME Code requires the latter report, which is the basis during the review for many of the individual review steps outlined below.

The reviewer can use previously reviewed designs as a guide, but must verify that any changes are justified.

A. BWRs Operating at Power

i. The reviewer examines the piping and instrumentation diagrams to determine the number, type, and location of the SRVs on the BWR RCS main steamlines and on the primary side of any auxiliary or emergency system that interfaces with the RCS. The reviewer also analyzes the functions of other pressure dissipation systems, such as isolation condensers.

ii. The reviewer identifies all other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems. This includes any blowdown or heat dissipation systems connected to the discharge side of any pressure-relieving devices such as the suppression pool. The reviewer determines the effects of these other functions or systems on the operation of the overpressure protection system.

iii. The reviewer identifies the capacities, setpoints, and setpoint tolerances for all SRVs or other overpressure protection system devices. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of pressure and temperature based on RCPB material parameters. The reviewer identifies allowable power levels with one or more inoperable SRVs to ensure that they are suitably conservative, as specified in RS-001, and confirms that the plant technical specifications limit power operation as appropriate.

iv. The reviewer identifies all of the reactor trip signals that occur during overpressure transients, including their setpoints and setpoint tolerances. The reviewer verifies that the second reactor trip signal, under worst-case conditions during an overpressure transient, is adequate to provide overpressure protection to the RCPB in conjunction with the installed overpressure protection systems or devices.

v. The reviewer examines all transients analyzed in the accident analysis section of the SAR that result in an increase in the pressure experienced by the RCPB. The reviewer identifies the predicted peak pressures and assesses the operating conditions and setpoints used in the analysis to ensure that they are suitably conservative.

vi. Relevant industry codes and classifications applied to the system analysis for power operations should be clearly identified, as specified in 10 CFR 50, Appendix G and Regulatory Issue Summary (RIS) 2004-04. Assumptions used in the analysis, including the initial plant conditions and system parameters, should also be identified and justified. The reviewer should identify studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. The reviewer also should consult Section 5.2.2 of the FSAR for new plant designs to obtain insight on the overpressure protection methodology.

B. <u>PWRs Operating at Power</u>

i. The reviewer examines the piping and instrumentation diagrams to determine the number, type, and location of the SRVs in both the primary and secondary systems and in discharge lines, instrumentation, and other components. A large-capacity pressurizer may eliminate the need for a PORV in the PWR primary system.

ii. The reviewer identifies all other functions of the components, instruments, or controls used for overpressure protection and the interfaces with all other systems. This includes any blowdown or heat dissipation system connected to the discharge side of any pressure-relieving devices such as a quench tank or containment atmosphere. The reviewer determines the effects of these other functions or systems on operation of the overpressure protection system.

iii. The reviewer identifies the capacities, setpoints, and setpoint tolerances for all primary and secondary SRVs or other overpressure protection system devices. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of pressure and temperature based on RCPB material parameters.

iv. The reviewer identifies allowable power levels with one or more inoperable main steamline safety valves to ensure that they are suitably conservative. The reviewer also confirms that plant technical specifications limit power operation as appropriate per RS-001.

v. The reviewer identifies all of the reactor trip signals that occur during overpressure transients, including their setpoints and setpoint tolerances. The reviewer verifies that the second reactor trip signal, under worst-case conditions during an overpressure transient, is adequate to provide overpressure protection to the RCPB in conjunction with the installed overpressure protection systems or devices.

vi. The reviewer examines all transients analyzed in the accident analysis section of the SAR that result in an increase in the pressure experienced by the RCPB. The reviewer identifies predicated peak pressures and assesses the operating conditions and setpoints used in the analysis to ensure that they are suitably conservative.

vii. The reviewer ensures that nonsafety-grade, pressure-operated relief valves are not credited for the mitigation of events that result in an increase in reactor coolant inventory. Furthermore, the reviewer ensures that the pressurizer does not reach a water-solid condition in response to events that lead to an increase in reactor coolant inventory per RS-001.

viii. The reviewer ensures that relevant industry codes and classifications applied to the system analysis for power operations should be clearly identified as specified in RIS 2004-04. Assumptions used in the analysis, including the initial plant conditions and system parameters, should also be identified and justified. The reviewer should identify studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. The reviewer should consult Section 5.2.2 of the FSAR for new plant designs to obtain insight into the overpressure protection methodology.

C. <u>PWRs Operating at Low Temperature (Startup, Shutdown)</u>

i. The reviewer examines the piping and instrumentation diagrams to determine the number, type, and location of the SRVs in the primary system and of discharge lines, instrumentation, and other components in interfacing systems.

ii. The reviewer examines failures of the makeup and letdown system or the RHR system to ensure overpressure protection during low-temperature operation of the plant.

iii. The reviewer identifies the capacities, setpoints, and setpoint tolerances for all SRVs designated for low-temperature overpressure protection. The reviewer verifies that these constraints are adequate to provide overpressure protection to the RCPB at critical values of temperature based on RCPB material parameters.

iv. Relevant industry codes and classifications applied to the system analysis for low-temperature operation should be clearly identified as specified in RIS 2004-04. Assumptions used in the analysis, including the initial plant conditions and system parameters, should also be identified and justified. The reviewer should identify studies that show the sensitivity of the system's performance to variations in these conditions, parameters, and characteristics. The reviewer should consult Section 5.2.2 of the FSAR for new plant designs to obtain insight on the overpressure protection methodology.

2. Testing and Inspections

To ensure operational readiness, the overpressure protection system should be testable. The reviewer should verify the following:

A. Tests for SRV operability are scheduled to be conducted as specified in Section III of the ASME Code Article NB-7000.

B. For PWRs, PORV and PORV block-valves are included in the inservice inspection program as specified in Generic Letter No. 90-06.

C. For PWRs, PORV and PORV block-valves are included in the QA and preventive maintenance program as specified in Generic Letter No. 90-06.

3. Technical Specifications

The review includes the technical specifications to ensure that they are adequate in regard to limiting conditions of operation and periodic surveillance testing. The review should include the following:

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 are adequate.

C. Verify compliance with the technical specification guidance of Generic Letter No. 90-06 for PORVs, PORV block-valves, and the LTOP system (PWRs only).

D. Verify compliance with TMI Action Plan Item II.K.3.3 of NUREG-0737 regarding reporting of SRV challenges and failures. Generic Letters No. 82-16 and 83-02 provide descriptions of this NUREG-0737 item, include guidance regarding appropriate technical specifications to address the reporting requirements of II.K.3.3 of Section 5.6.4 of Standard Technical Specifications NUREG-1430 through NUREG-1434 regarding monthly operating reports, and offer related guidance on an appropriate technical specification to address this issue for those applicants implementing improved technical specifications.

E. Verify that appropriate references to the pressure-temperature limit report (PTLR) exist and that PTLR technical specification administrative controls (Standard

Technical Specifications 5.6.6) are established as required by Generic Letter No. 96-03.

4. For reviews of DC and COL applications under 10 CFR Part 52, the reviewer should follow the above procedures to verify that the design set forth in the safety analysis report, and if applicable, site interface requirements meet the acceptance criteria. For DC applications, the reviewer should identify necessary COL action items. With respect to COL applications, the scope of the review is dependent on whether the COL applicant references a DC, an ESP or other NRC-approved material, applications, and/or reports.

   After this review, SRP Section 14.3 should be followed for the review of Tier I information for the design, including the postulated site parameters, interface criteria, and ITAAC.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) 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. BWRs

   The overpressure protection system prevents overpressurization of the RCPB under the most severe transients and limits the reactor pressure during normal operational transients. The isolation condenser system provides overpressure protection.

   The SRVs located on the ____ main steamlines between the reactor vessel and the first isolation valve inside the drywell provide defense-in-depth. The SRVs are distributed among the ____ main steamlines such that a single accident cannot disable the automatic overpressure protection function. The valves discharge through piping to the suppression pool. The valves have setpoints that range from ______ to ______ kilopascal gauge (kPag) (______ to ______ pounds per square inch gauge (psig)). The total capacity at their setpoints is ______ percent of rated steam flow.

   To determine the ability of the overpressure protection system to prevent overpressurization, the applicant has analyzed the most severe anticipated overpressure transients. The analysis assumed that (1) the plant is in operation at design conditions of ____ percent of rated steam flow and a reactor vessel dome pressure of ____ kPag (__ psig) and (2) the reactor is shut down by ______. The calculated peak pressure at the bottom of the vessel is ______ kPag (______ psig), a value within the code allowable of ______ kPag (______ psig) (110 percent of vessel design pressure).

2. PWRs

   The overpressure protection system prevents overpressurization of the RCPB under the most severe transients and limits the reactor pressure during normal operational transients. The safety valves provide overpressurization protection. These valves discharge to containment atmosphere or a pressurizer quench tank through a common header from the pressurizer. The SRVs or the safety valves and a large capacity pressurizer in the primary, in conjunction with the steam generator SRVs in the secondary, and the reactor protection and safeguards systems will protect the primary system against overpressure in the event of a complete loss of heat sink.

The peak primary system pressure following the most severe anticipated transient is limited to the ASME Code allowable (110 percent of the design pressure) with no credit taken for nonsafety-grade relief systems. The assumption was that the ______ plant was operating at design conditions (percent of rated power) and the reactor is shut down by a scram. The calculated pressure at the bottom of the vessel is kPag (___ psig), a value within the code allowable of_kPag (___ psig) (110 percent of vessel design pressure).

Overpressure protection during low-temperature operation (defined in BTP RSB 5-2) of the plant is provided by________ .

In addition, the applicant has incorporated into the design the recommendations of Task Action Plan Items II.D.1 and II.D.3, as described in NUREG-0737, and has met the related requirements of 10 CFR 50.34(f)(2)(x), 10 CFR 50.34(f)(2)(xi), or 10 CFR 52.47(a)(8), as applicable.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

2. 10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor Coolant System Design."

3. 10 CFR Part 50, Appendix A, General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary."

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

5. 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”

6. 10 CFR 50.34(f), “Additional TMI-related Requirements.”

7. 10 CFR 52.47, “Contents of Applications; Technical Information.”

8. 10 CFR 52.79, “Contents of Applications; Technical Information in Final Safety Analysis Report.”

9. 10 CFR 52.97, “Issuance of Combined Licenses.”

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

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

12. ASME Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components.

13. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, “Overpressure Protection,” American Society of Mechanical Engineers.

14. ASME Boiler and Pressure Vessel Code, Section III, Article NB-7511.1, “Spring-Loaded Valves.”

15. ANSI/ANS 51.1, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants-Replaces ANSI N18.2."

16. ANSI/ANS 52.1, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants.”

17. Branch Technical Position RSB 5-2, “Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures,” attached to this SRP section.

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

19. NUREG-1430, “Standard Technical Specifications, Babcock and Wilcox Plants.”

20. NUREG-1431, “Standard Technical Specifications, Westinghouse Plants.”

21. NUREG-1432, “Standard Technical Specifications, Combustion Engineering Plants.”

22. NUREG-1433, “Standard Technical Specifications, General Electric Plants, BWR/4.”

23. NUREG-1434, “Standard Technical Specifications, General Electric Plants, BWR/6.”

24. NUREG-1511, Supplement 2, “Reactor Pressure Vessel Status Report.”

25. NRC Letter to All Pressurized Power Reactor Licensees, “NUREG-0737 Technical Specifications,” Generic Letter 82-16, September 20, 1982.

26. NRC Letter to All Boiling Water Reactor Licensees, “NUREG-0737 Technical Specifications,” Generic Letter 83-02, January 10, 1983.

27. NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, "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,'" Generic Letter No. 90-06, June 25, 1990.
28. RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003.
29. NRC Regulatory Issue Summary 2004-04, "Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits," RIS 2004-04, April 5, 2004.
30. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, "Relocation of the Pressure Temperature Limit Curves and Low-Temperature Overpressure Protection System Limits," Generic Letter 96-03, January 31, 1996.
31. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."
32. Regulatory Guide 1.29, "Seismic Design Classification."
33. Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III."

BRANCH TECHNICAL POSITION RSB 5-2

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

I. 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 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.

II. Branch 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 branch 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 ICSB 1 in Appendix 7-A to SRP Chapter 7).

III. 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 ICSB 1, (BTP-7A-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."

SRP Section 5.2.2
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 3, dated June 1996 of this SRP. See ADAMS accession number ML061930482.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, “Standard Review Plan (SRP) Process.” The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 2, dated 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

1. AREAS OF REVIEW

Delineated the four major topical areas covered within this section
a. BWR power operation
b. BWR low temperature operation
c. PWR power operation
d. PWR low temperature operation

Added general plant design requirement excerpts from ANSI/ANS 51.1 and 51.2 to the Areas of Review. There are no specific overpressure protection system requirements in these standards that are not already part of ASME III; therefore, the standards were not included in the section list of references. Also, at the current time, the standards are withdrawn. However, the standards are in the process of being reinstituted by the ANS Standards Committee and the 1988 reaffirmation of the standards remain in common use.

Added review interfaces for SRP sections 5.4.11, 6.2.1.1.C, 3.9.6, 5.2.4, 5.4.14, 6.3, and 14.3.

Updated references to the review of COL applications under 10 CFR Part 52.

2. ACCEPTANCE CRITERIA

Included overpressure mitigation features of advanced light water reactors (ALWR), such as the use of isolation condensers in BWRs and large capacity pressurizers in PWRs.

Added appropriate references to inspection, test, analysis, and acceptance criteria (ITAAC).

Added acceptance criteria related to Three Mile Island requirements and ITAAC.

Updated references to the review of COL applications under 10 CFR Part 52.

3. REVIEW PROCEDURES

Included overpressure mitigation features of advanced light water reactors (ALWR), such as the use of isolation condensers in BWRs and large capacity pressurizers in PWRs.

Added a reference to evaluating license amendment requests and the Review Standard for Extended Power Uprates (RS-001).

Added appropriate references to inspection, test, analysis, and acceptance criteria (ITAAC).

Added a reference to single failure criteria in the explanation of the BWR and PWR design requirements related to the acceptance criteria.

Added a reference to the review of section 5.2.2 of the applicable ALWR Final Safety Analysis Report in the review procedures.

Added steps to the review procedure to verify PORV and PORV block-valve inclusion in a QA and preventive maintenance program and an Inservice Inspection program per Generic Letter No. 90-06.

Updated references to the review of COL applications under 10 CFR Part 52.

4. EVALUATION FINDINGS

Revised parameters units in the conclusion statements.

Added ITAAC conclusion summary statement for DC and COL applications under 10 CFR Part 52.

5. IMPLEMENTATION

Added use of SRP for DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52.

6. REFERENCES

Added or revised the listing of the following references:

10 CFR Part 50, Appendix A, General Design Criterion 1
10 CFR Part 50, Appendix A, General Design Criterion 30
10 CFR 50.34(f), "Additional TMI-Related Requirements."
10 CFR 52.47, "Contents of Applications."
10 CFR 52.79, "Contents of Applications; Technical Information."
10 CFR 52.97, "Issuance of Combined Licenses."
ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Overpressure Protection."
ASME Boiler and Pressure Vessel Code, Section III, Article NB-7511.1, "Spring-Loaded Valves."
NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants."
NUREG-1431, "Standard Technical Specifications, Westinghouse Plants."

NUREG-1432, “Standard Technical Specifications, Combustion Engineering Plants.”
NUREG-1433, “Standard Technical Specifications, General Electric Plants, BWR/4.”
NUREG-1434, “Standard Technical Specifications, General Electric Plants, BWR/6.”
NUREG-1511, Supplement 2, “Reactor Pressure Vessel Status Report.”

NRC Letter to All Pressurized Power Reactor Licensees, “NUREG-0737 Technical Specifications,” Generic Letter 82-16, September 20, 1982.
NRC Letter to All Boiling Water Reactor Licensees, “NUREG-0737 Technical Specifications,” Generic Letter 83-02, January 10, 1983.
NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, “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,’” Generic Letter No. 90-06, June 25, 1990.

RS-001, Revision 0, “Review Standard for Extended Power Uprates,” December 2003.

NRC Regulatory Issue Summary 2004-04, “Use of Code Cases N-588, N-640, and N-641 in Developing Pressure-Temperature Operating Limits,” RIS 2004-04, April 5, 2004.
NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, “Relocation of the Pressure Temperature Limit Curves and Low-Temperature Overpressure Protection System Limits,” Generic Letter 96-03, January 31, 1996.

Regulatory Guides 1.26, 1.29 and 1.84

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

7. <u>BRANCH TECHNICAL POSITION RSB 5-2</u>

A. <u>BACKGROUND</u>

None.

B. <u>BRANCH POSITION</u>

Updated the IEEE 603 reference.

C. <u>REFERENCES</u>

Added or revised the listing of the following references:

10 CFR Part 50, Appendix A, General Design Criterion 15, “Reactor Coolant System Design.”
10 CFR Part 50, Appendix G, “Fracture Toughness Requirements.”
NUREG-0138, “Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR, to NRR Staff.”
IEEE Standard 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (as endorsed by Regulatory Guide 1.153).
ASME Boiler and Pressure Vessel Code, Section II, “Materials Specifications.”

ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components.”
Branch Technical Position ICSB 3 BTP-7A-3, “Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service”
Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.”
Regulatory Guide 1.29, “Seismic Design Classification.”