December 13, 2019

TSTF-19-09
PROJ0753

Attn: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Transmittal of TSTF-576, Revision 0, "Revise Safety/Relief Valve Requirements"

Enclosed for NRC review is TSTF-576, Revision 0, "Revise Safety/Relief Valve Requirements."

The following information is provided to assist the NRC staff in prioritizing their review of TSTF-576:

- Applicability: TSTF-576 is applicable to Boiling Water Reactor (BWR) plants.
- Classification: TSTF-576 proposes improvements to the Safety/Relief Valve (S/RV) specification to align the requirements with the accident analysis assumptions.
- Specialized Resource Availability: TSTF-576 is a high priority change. The TSTF requests that TSTF-576 be reviewed within 18 months and be made available for adoption under the Consolidated Line Item Improvement Process (CLIIP).

The Technical Specifications Task Force should be billed for the review of the traveler.

Should you have any questions, please do not hesitate to contact us.

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PWROG PWR Owners Group BWR OWNERS' GROUP APOG

Enclosure

cc: Michelle Honcharik, Technical Specifications Branch, NRC
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Technical Specifications Task Force
Improved Standard Technical Specifications Change Traveler

Revise Safety/Relief Valve Requirements

NUREGs Affected: ☐ 1430 ☐ 1431 ☐ 1432 ☑ 1433 ☑ 1434 ☐ 2194

Classification: 1) Technical Change

Recommended for CLIIP?: Yes

Correction or Improvement: Improvement

NRC Fee Status: Not Exempt

Benefit: Increases Equipment Operability

Changes Marked on ISTS Rev 4.0

PWROG RISD & PA (if applicable): N/A,N/A

See attached.

Revision History

OG Revision 0 **Revision Status: Active**

Revision Proposed by: BWROG LC

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 12-Jul-19

Owners Group Comments
Presubmittal meeting held September 12, 2019. Revised traveler distributed to BWROG on October 7.

Owners Group Resolution: Approved Date: 02-Aug-19

TSTF Review Information

TSTF Received Date: 03-Dec-19 Date Distributed for Review 03-Dec-19

TSTF Comments:

A presubmittal meeting was held with the NRC on September 12, 2019. A revised draft was developed and submitted to the NRC on October 21. A presubmittal teleconference was held on December 2. The traveler was finalized addressing the NRC comments.

TSTF Resolution: Approved Date: 12-Dec-19

NRC Review Information

NRC Received Date: 13-Dec-19

NRC Comments:

A presubmittal meeting was held with the NRC on September 12, 2019. A revised draft was developed and submitted to the NRC on October 21. A presubmittal teleconference was held on December 2. The traveler was finalized addressing the NRC comments and submitted December 13, 2019.

Affected Technical Specifications

SR 3.3.6.3.7 Bases	LLS Instrumentation	NUREG(s)- 1433 Only
3.4.3	S/RVs	NUREG(s)- 1433 Only
	Change Description: Specification renamed Overpressure Protection System	
3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: Specification renamed Overpressure Protection System	
Bkgnd 3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
S/A 3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
LCO 3.4.3	S/RVs	NUREG(s)- 1433 Only
LCO 3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
Appl. 3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
Action 3.4.3.A	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
Action 3.4.3.A Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
Action 3.4.3.B	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
Action 3.4.3.B Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
Action 3.4.3.C	S/RVs	NUREG(s)- 1433 Only
	Change Description: Revised and renamed "A"	
Action 3.4.3.C Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: Revised and renamed "A"	
SR 3.4.3.1	S/RVs	NUREG(s)- 1433 Only
SR 3.4.3.1 Bases	S/RVs	NUREG(s)- 1433 Only
SR 3.4.3.2	S/RVs	NUREG(s)- 1433 Only
	Change Description: New	
SR 3.4.3.2	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
SR 3.4.3.2 Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: Deleted	
SR 3.4.3.2 Bases	S/RVs	NUREG(s)- 1433 Only
	Change Description: New	

Ref. 3.4.3 Bases	S/RVs	NUREG(s)- 1433 Only
Bkgnd 3.3.6.5 Bases	Relief and LLS Instrumentation	NUREG(s)- 1434 Only
3.4.4	S/RVs	NUREG(s)- 1434 Only
	Change Description: Specification renamed Overpressure Protection System	
3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only
	Change Description: Specification renamed Overpressure Protection System	
Bkgnd 3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only
S/A 3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only
LCO 3.4.4	S/RVs	NUREG(s)- 1434 Only
LCO 3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only
Appl. 3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only
Action 3.4.4.A	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
Action 3.4.4.A Bases	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
Action 3.4.4.B	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
Action 3.4.4.B Bases	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
Action 3.4.4.C	S/RVs	NUREG(s)- 1434 Only
	Change Description: Renamed A	
Action 3.4.4.C Bases	S/RVs	NUREG(s)- 1434 Only
	Change Description: Renamed A	
SR 3.4.4.1	S/RVs	NUREG(s)- 1434 Only
SR 3.4.4.1 Bases	S/RVs	NUREG(s)- 1434 Only
SR 3.4.4.2	S/RVs	NUREG(s)- 1434 Only
SR 3.4.4.2 Bases	S/RVs	NUREG(s)- 1434 Only
SR 3.4.4.3	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
SR 3.4.4.3 Bases	S/RVs	NUREG(s)- 1434 Only
	Change Description: Deleted	
Ref. 3.4.4 Bases	S/RVs	NUREG(s)- 1434 Only

1. SUMMARY DESCRIPTION

The proposed change revises the Safety/Relief Valve (S/RV) Technical Specifications (TS) to align the requirements with the safety limits and the regulations. The proposed change modifies NUREG-1433, "Standard Technical Specifications, General Electric BWR/4 Plants," and NUREG-1434, "Standard Technical Specifications, General Electric BWR/6 Plants."[1]

2. DETAILED DESCRIPTION

2.1. System Design and Operation

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel to be protected from overpressure during upset conditions by self-actuated safety valves. The overpressure protection system requirements dictate the size and number of S/RVs that are needed such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB) under the most severe transients. Section 5.2.2, "Overpressure Protection," of NUREG-0800, "Standard Review Plan," describes the typical requirements for the overpressure protection system for boiling water reactor (BWR) plants.

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, "General Design Criteria," (GDC), criterion 15 "Reactor coolant system design," states, "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." While many of the operating plants are not committed to the Appendix A GDC, most plants are committed to a similar design requirement as described in their Updated Final Safety Analysis Report (UFSAR).

The overpressure protection system for a BWR utilizes the safety mode, and in some cases the relief mode, of the S/RVs. The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded disk or pilot valve opens when steam pressure overcomes the spring force holding the valve or pilot valve closed. For S/RVs with pilot valves, opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. In the relief mode of operation, pneumatic pressure is used to open the valve, initiated by switches located in the control room or by pressure-sensing instrumentation. Some plants credit a percentage of the total installed S/RV capacity operating via the relief mode for overpressure protection, as permitted by the ASME Code.

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[1] NUREG 1433 is based on the BWR/4 plant design, but is also representative of the BWR/2, BWR/3, and, in this case, BWR/5 designs. NUREG 1434 is based on the BWR/6 plant design.

2.1.1. S/RV Inservice Testing

The S/RVs are tested in accordance with the Inservice Testing (IST) Program, as required by 10 CFR 50.55a(f). Periodic testing is described in Appendix I of the ASME Operations and Maintenance (OM) Code, "Class 1 Main Steam Pressure Relief Valves with Auxiliary Actuating Devices," Section I-3300, "Periodic Testing." This testing is performed during a plant shutdown as a bench test performed at nominal operating temperatures and pressures. The inservice test verifies each S/RV opens within the required "as-found" tolerance around the setpoint.

Safety/Relief Valve nominal setpoints, as-left tolerance limits, and as-found tolerance limits are also established and controlled by the ASME OM Code. ASME OM Code Appendix I Section I-1310(e), states, "The Owner, based upon system and valve design basics or technical specification, shall establish and document acceptance criteria for tests required by this Mandatory Appendix."

The ASME Code permits testing 20% of the S/RVs each cycle, with the tested population expanded if failures are found. Alternatively, all of the S/RVs or pilot valves may be removed and replaced, and the as-found testing is performed after removal. The ASME Code requires the as-found testing to be performed within one year. Following testing, the S/RVs or pilot valves are refurbished, tested, and certified for use. The valves are set to a narrower "as-left" tolerance to allow for drift during the period of operation.

If an S/RV fails to open within the established as-found tolerance during testing, the failure is entered into the Corrective Action Process and, according to licensee procedures, evaluated, corrected, and tracked. The extent of condition is also evaluated. Depending on the nature and extent of the failure, the extent of condition could include an evaluation of the ability of the S/RVs to perform their function in the current cycle. For example, in 2016 Southern Company discovered unexpected damage during testing of the S/RVs for Plant Hatch Unit 1. After examination, it was determined that the damage was similar to damage reported in a previous 10 CFR Part 21 report. Extensive extent of condition evaluations were performed on Unit 1 and Unit 2 (see NRC Reactive Inspection Report 05000321/2016009 dated June 10, 2016), which determined the Hatch S/RVs were suspectable to fretting as described in the 10 CFR Part 21 report. As a result, in May of 2016 Southern Company performed a mid-cycle outage on Plant Hatch Unit 2 to replace all eleven S/RVs and to inspect the main valve internals.

The current Technical Specifications (TS) also impose additional actions if an S/RV fails the inservice test.

2.2. <u>Current Technical Specifications Requirements</u>

In addition to the ASME Code requirements, the current TS contain multiple specifications that govern the S/RVs depending on the function they are fulfilling.

- Safety Limit 2.1.2, "Reactor Coolant System Pressure SL," states, "Reactor steam dome pressure shall be $\leq$ 1325 psig." The pressure limit is plant specific. The S/RVs are credited for meeting this safety limit. Safety Limit 2.1.2 limits the reactor steam dome pressure to the lowest transient overpressure allowed in order to ensure the maximum transient pressure allowable in the RCS pressure vessel is less than the ASME Code, Section III, limit of 110% of design pressure.

- BWR/4 and BWR/6 TS 3.5.1, "ECCS - Operating," requires the Automatic Depressurization System (ADS), which uses the S/RVs in the relief mode. The ADS is designed to provide depressurization of the reactor pressure vessel (RPV) during a small break Loss of Coolant Accident (LOCA) if high pressure core injection (BWR/4) or high pressure core spray (BWR/6) fails or is unable to maintain the required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure Emergency Core Cooling System (ECCS) subsystems, so that these subsystems can provide coolant inventory makeup.

- BWR/4 and BWR/6 TS 3.6.1.6, "Low-Low Set (LLS) Valves," requires the S/RVs operating in relief mode. In the LLS mode, a subset of the S/RVs are signaled to open at a lower pressure than the relief or safety mode pressure setpoints and to stay open longer, so that reopening more than one S/RV is prevented on subsequent actuations. The LLS function prevents excessive short duration S/RV cycles with valve actuation at the relief setpoint.

- BWR/4 TS 3.3.6.3, "Low-Low Set (LLS) Instrumentation," and BWR/6 TS 3.3.6.5, "Relief and Low-low Set (LLS) Instrumentation," provide instrumentation requirements that support the S/RVs in the LLS mode of operation. For plants that credit S/RVs in relief mode to prevent overpressurization, the LLS Instrumentation TS also provide the instrumentation requirements to support that function.

- BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4, both titled, "Safety/Relief Valves," require the S/RVs to prevent RCPB overpressurization. For most plants, the most severe pressurization transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position). For most BWR/2, BWR/3, BWR/4, and BWR/5 plants, the S/RVs in the safety mode ensure the Safety Limit is not exceeded during normal operation and Anticipated Operational Occurrences (AOOs). For BWR/6 plants and two non-BWR/6 plants (Dresden 2 and 3 and Quad Cities 1 and 2), some S/RVs in relief mode in addition to the S/RVs in safety mode are credited to ensure the ASME Code overprotection limit is protected.

  The BWR/4 TS 3.4.3 S/RV LCO typically states, "The safety function of *XX* S/RVs shall be operable," with the required number of S/RVs (*XX*) corresponding to the minimum number needed to accommodate the limiting pressure transient without exceeding the Safety Limit using only the safety mode of operation.

  The BWR/6 TS 3.4.4 LCO (which is also applicable to some non-BWR/6 plants) requires the safety function of *XX* S/RVs and the relief function of *YY* additional S/RVs to be operable. The required number of S/RVs in the safety mode (*XX*) and relief mode (*YY*) varies by plant.

  BWR/4 Surveillance Requirement (SR) 3.4.3.1 and BWR/6 SR 3.4.4.1 require verification of the safety function lift setpoints of the required S/RVs. These SRs reflect the performance of the ASME Code inservice testing and also state the number of valves required to open within a specified tolerance (typically 3%) of the given setpoint. The SRs also specify the as-left tolerance (typically 1%) after testing. A review of Licensee Event Reports over the last ten years found over forty events in which S/RVs failed to lift within the SR lift pressure tolerance when bench tested. In all cases in which the SR was not met due to setpoint drift, the Licensee Event Reports concluded that the S/RVs as a group would have retained the capability to protect Safety Limit 2.1.2.

BWR/6 SR 3.4.4.2 requires verification that each relief function S/RV actuates on an actual or simulated automatic initiation signal. Non-BWR/6 plants that require the S/RV relief mode have a similar SR.

BWR/4 SR 3.4.3.2 and BWR/6 SR 3.4.4.3 verify that each S/RV opens when manually actuated.

2.3. Reason for the Proposed Change

The S/RV LCO is written in terms of individual valves. However, the specified safety function is based on the combined pressure relieving capacity of a group of the S/RVs. The failure of some valves to open within the SR tolerance typically would not result in the inability of the S/RVs as a group to perform the specified safety function. Therefore, the LCO should be revised to align with the specified safety function.

Testing of the safety mode of each S/RV is required by the IST Program, which is required by 10 CFR 50.55a(f). It is unnecessary to duplicate this regulatory requirement in the TS when the result of any individual valve test is not required to meet the specified safety function of the system.

2.4. Description of the Proposed Change

The proposed change renames BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4 from "Safety/Relief Valves (S/RVs)" to "Overpressure Protection System (OPS)." This title change requires revision to the Table of Contents and a reference in the Bases of BWR/4 TS 3.3.6.3, "Low-Low Set (LLS) Instrumentation," and BWR/6 TS 3.3.6.5, "Relief and Low-low Set (LLS) Instrumentation."

The proposed change revises the S/RV LCO to require the Overpressure Protection System (OPS) to be operable. The LCO Bases describes an operable OPS as being capable of preventing reactor steam dome pressure from exceeding Safety Limit 2.1.2.

BWR/4 LCO 3.4.3 is revised to state (deletions are struck through; insertions are in italics):

> The *OPS* ~~safety function of the [11] S/RVs~~ shall be OPERABLE.

BWR/6 LCO 3.4.4 is revised to state:

The *OPS* ~~safety function of the [seven] S/RVs~~ shall be OPERABLE,

~~AND~~

~~The relief function of [seven] additional S/RVs shall be OPERABLE.~~

BWR/4 SR 3.4.3.1 is revised to state:

Verify the OPS has the capability to prevent reactor steam dome pressure from exceeding Safety Limit 2.1.2.

~~-----NOTE-----~~
~~≤ [2] [required] S/RVs may be changed to a lower setpoint group.~~
~~----------~~

~~Verify the safety function lift setpoints of the [required] S/RVs are as follows:~~

~~Number of S/RVs~~	~~Setpoint (psig)~~
~~[4]~~	~~[1090 ± 32.7]~~
~~[4]~~	~~[1100 ± 33.0]~~
~~[3]~~	~~[1110 ± 33.3]~~

~~Following testing, lift settings shall be within ± 1%.~~

BWR/6 SR 3.4.4.1 is revised to state:

Verify the OPS has the capability to prevent reactor steam dome pressure from exceeding Safety Limit 2.1.2.

~~-----NOTE-----~~
~~≤ [2] [required] S/RVs may be changed to a lower setpoint group.~~
~~----------~~

~~Verify the safety function lift setpoints of the [required] S/RVs are as follows:~~

~~Number of S/RVs~~	~~Setpoint (psig)~~
~~[8]~~	~~[1165 ± 34.9]~~
~~[6]~~	~~[1180 ± 35.4]~~
~~[6]~~	~~[1190 ± 35.7]~~

> ~~Following testing, lift settings shall be within ± 1%.~~

The current frequency has three options: In accordance with the Inservice Testing Program, [18] months, or in accordance with the Surveillance Frequency Control Program. The [18] month and Surveillance Frequency Control Program options are deleted.

BWR/6 SR 3.4.4.2 is revised to state:

> Verify each ~~[required]~~ ~~relief function~~ *safety/relief valve acting in the relief mode* ~~S/RV~~ actuates on an actual or simulated automatic initiation signal.

The revised BWR/6 SR 3.4.4.2 is added to the BWR/4 STS as optional surveillance SR 3.4.3.2.

BWR/4 SR 3.4.3.2 and BWR/6 SR 3.4.4.3, which state, "Verify each [required] S/RV opens when manually actuated," are deleted.

The changes to the LCO and SRs result in changes to the TS Actions.

BWR/4 and BWR/6 Condition A, "One [or two] [required] S/RV[s] inoperable," and "One [required] S/RV inoperable," respectively, are deleted as the LCO and SRs no longer contain requirements on individual S/RVs.

Condition B, the default action when Condition A's Required Action and associated Completion Time is not met, is no longer required after deletion of Condition A.

BWR/4 and BWR/6 Condition C, "[Three] or more [required] S/RVs inoperable," and "[Two] or more [required] S/RVs inoperable," respectively, are replaced with a new Condition, "OPS inoperable." The new Condition retains the existing Required Actions to be in Mode 3 in 12 hours and in Mode 4 in 36 hours.

The TS Bases are revised to reflect the changes to the TS. The regulation at Title 10 of the Code of Federal Regulations (10 CFR), Part 50.36, states, "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." A licensee may make changes to the TS Bases without prior NRC review and approval in accordance with the Technical Specifications Bases Control Program. The proposed TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132). Therefore, the Bases changes are provided for information and approval of the Bases is not requested.

A model application is attached. The model may be used by licensees desiring to adopt the traveler following NRC approval.

3. TECHNICAL EVALUATION

Specification Name Change.

As discussed in Section 2.2, there are several specifications which provide requirements on the S/RVs. It is confusing to title the specification "Safety/Relief Valves," because that name implies it is the only specification that governs the equipment. Just as TS 3.5.1 refers to the "Automatic Depressurization System (ADS)" function of the S/RVs, it is more appropriate to title BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4 "Overpressure Protection System (OPS)," to represent the functional capability required by the specification. Renaming the specifications is consistent with the STS convention that an LCO requires a system to be operable and the LCO Bases describe what is required for the system to capable of performing its specified safety function. The term "overpressure protection system," is not new. The NRC Standard Review Plan (NUREG-0800), Section 5.2.2, is titled, "Overpressure Protection," and many BWR plants have a similar Updated Final Safety Analysis Report (UFSAR) section. In addition, the existing BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4 "Applicable Safety Analysis" section of the Bases begins, "The overpressure protection system must accommodate the most severe pressurization transient." As a result, referring to the S/RV overpressure protection function as the "Overpressure Protection System (OPS)," is a clearer representation of the requirement.

LCO Changes

The current S/RV Limiting Condition for Operation (LCO) is overly restrictive based on the relevant regulations. Title 10 of the CFR, Paragraph 50.36(c)(2)(i) states, "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility." The existing S/RV LCO is written in terms of individual valves, but the specified safety function is based on the combined pressure relieving capacity of the credited S/RVs (which may be less than the installed complement of valves). The failure of a particular valve or valves to open within the SR tolerance may not (and based on historical performance, is very unlikely to) result in the inability of the credited S/RVs as a group to perform the specified safety function. Given that the LCO is overly restrictive with respect to the regulatory requirements, it is revised to represent the lowest functional capability or performance level of equipment required for safe operation of the facility.

TSTF-GG-05-02, "Writer's Guide for Plant-Specific Improved Technical Specifications," (ADAMS Accession No. ML070660229) Section 4.1.4, "Chapter 3 LCO Content," states, "The LCO describes as simply as possible the lowest functional capability or performance levels of equipment required for safe operation of the facility. ... It is acceptable to generically refer to the system, subsystem, component or parameter which is the subject of the LCO and provide the specific scope/boundaries in the Bases." Following this guidance, the BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4 LCOs are revised to require the OPS to be operable. The LCO Bases are revised to state, "The OPS is OPERABLE when it can ensure that the ASME Code limit on peak reactor pressure, as stated in Safety Limit 2.1.2, will be protected using the safety mode of the S/RVs [and the relief mode of additional S/RVs]. OPERABILITY of the OPS is only dependent on the capability of the S/RVs to open to relieve excess pressure, and may credit less than the full complement of installed S/RVs." The phrase "and the relief mode of additional S/RVs" is bracketed (i.e., plant-specific) in the BWR/4 TS Bases since it is applicable to only a few plants. The phrase is not bracketed in the BWR/6 TS since it is applicable to all BWR/6 plants.

The Bases and the revised BWR/4 SR 3.4.3.2 and BWR/6 SR 3.4.4.2 use the terms "safety mode" and "relief mode." The terms "safety function" and "relief function" are used in BWR/4 TS 3.4.3 and BWR/6 TS 3.4.4. However, the TS Bases, "Background" section uses the terms "safety mode" and "relief mode." For example, the Bases state, "The S/RVs can actuate by either of two modes: the safety mode or the relief mode," and "The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves." The term "safety function" could be easily confused with the term "specified safety function" used in the definition of operability. For clarity and for consistency with the existing TS Bases, the terms are replaced with "safety mode" and "relief mode." This is an administrative change with no change in intent.

The LCO is revised to no longer specify the number of credited operable S/RVs. As stated previously, the overpressure protection function is provided by the collective action of the credited S/RVs, not individual S/RVs. This change is consistent with the required function and the 10 CFR 50.36 requirement that the LCO represent the lowest functional capability required for safe operation of the facility.

The BWR/6 LCO requires relief mode of operation for a subset of S/RVs. These plant designs permit crediting some of the pressure relieving capability of electrically operated pressure relief valves in the overpressure analysis. Two non-BWR/6 plants, Dresden 2 and 3 and Quad Cities 1 and 2, also credit electrically operated pressure relief valves and changes are proposed to address that design. The revision to the LCO to require the OPS to be operable includes a change to the LCO Bases to describe the role of the S/RVs in relief mode. The LCO Bases are revised to specify that the S/RVs credited for the relief mode cannot also be credited for meeting the safety mode portion of LCO, consistent with the term "additional" which appears in the current LCO.

BWR/4 SR 3.4.3.1 and BWR/6 SR 3.4.4.1 Changes

Paragraph 10 CFR 50.36(c)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met." While the existing test of each S/RV assures the necessary quality of the components is maintained, the testing is duplicative of the IST Program, which is required by regulation (10 CFR 50.55a(f)). The existing SR does not meet the regulatory guidance that the SR will assure that facility operation will be within safety limits, as the SR requirements are overly conservative for that purpose. The existing SR verifies that the existing LCO is met, but as discussed above, the LCO is also overly restrictive with respect to the regulatory requirements.

The existing SR verifies each S/RV lifts within the tolerance around the specified setpoint. As previously discussed, the specified safety function is based on the collective capability of the credited S/RVs to relieve pressure, not the ability of each S/RV to lift within a specified tolerance. As discussed above, the LCO requirement that the S/RVs be operable is replaced with a requirement that the Overpressure Protection System be operable. Therefore, the SR is replaced with a requirement to verify the OPS has the capability to prevent reactor steam dome pressure from exceeding Safety Limit 2.1.2.

The SR Bases are revised to discuss the relationship between the SR and the inservice testing of the S/RVs required by the ASME Code. The proposed Bases state:

> This Surveillance verifies that the OPS has the capability to prevent the reactor steam dome pressure from exceeding Safety Limit 2.1.2. The testing of the S/RV safety mode lift settings is performed on valves removed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The SR consists of a combination of testing and calculation. The measured S/RV mechanical lift values tested in accordance with the Inservice Testing Program are reviewed and aggregated to verify that the collective performance of the credited S/RVs will ensure Safety Limit 2.1.2 is protected. Should one or more of the credited S/RVs not actuate within the assumed tolerance, the actual lift values will be used to evaluate the affected overpressure analyses to determine whether the Safety Limit would have been protected.

The description of the OPS in the LCO Bases includes the safety mode of the S/RVs and, when necessary, the relief mode of additional S/RVs. Therefore, when the SR requires verification that the OPS will protect the Safety Limit, it includes the relief mode when credited in the overpressure protection analysis. Therefore, it is not necessary to call out S/RVs operating in the relief mode in the SR.

The proposed change to the SR no longer specifies the number of S/RVs set at each lift setpoint and the as-found tolerance around the setpoint. This information is controlled by the ASME OM Code, which is required to be followed by 10 CFR 50.55a. Appendix I of the ASME OM Code provides S/RV testing requirements, including establishment of setpoints, as-found tolerances, and as-left tolerances. ASME OM Code requirement I-1310(e), states, "The Owner, based upon system and valve design basics or technical specification, shall establish and document acceptance criteria for tests required by this Mandatory Appendix." Therefore, the acceptance criteria will be specified in the licensee-controlled documents. The number of S/RVs required at each setpoint and the as-found and as-left setpoints are not specified in the LCO Bases of the proposed change, as inclusion of that information would inappropriately tie individual valve performance to operability of the OPS.

Periodic testing of S/RVs will still be performed as required by Appendix I of the ASME OM Code, "Class 1 Main Steam Pressure Relief Valves with Auxiliary Actuating Devices," Section I-3300, "Periodic Testing." Title 10 of the CFR, Part 50, paragraph 55a, "Codes and standards," requires licensees to follow the ASME OM Code. The results of the OM Code-required testing will be used to evaluate S/RV performance under the proposed BWR/4 SR 3.4.3.1 and BWR/6 SR 3.4.4.1.

The S/RV safety mode lift setpoints and tolerances are also inputs to accident analyses. The licensee may set the number of S/RVs at each lift setpoint, and the as-found and as-left tolerances, as specified by the OM Code, and verify that the overpressure and accident analyses provide acceptable results using NRC-approved methods. The S/RV lift setpoints and tolerances used in accident analyses will be maintained in licensee-controlled documents subject to the 10 CFR 50.59 change controls, similar to other analysis assumptions.

The proposed approach is similar to other SRs that require analysis to evaluate whether the SR is met. For example, BWR/4 SR 3.7.5.1 states, "Verify each [control room AC] subsystem has the capability to remove the assumed heat load." As discussed in the associated Bases, performance of the SR consists of a combination of testing and calculation, just as the proposed S/RV SR will require a combination of testing and calculation to verify the SR is met.

The S/RV relief mode setpoints will continue to be specified in BWR/6 TS 3.3.6.5, "Relief and Low-low Set (LLS) Instrumentation," and in some plant-specific TS not based on the STS.

Under the proposed SR, the results of the Inservice Testing Program individual valve testing will be reviewed and aggregated to verify that the collective performance of the S/RVs will ensure Safety Limit 2.1.2 is protected. If all of the required S/RVs actuate within the assumed tolerance, the SR is met. Should an S/RV not actuate within the assumed tolerance, the actual lift settings will be used to evaluate the affected overpressure analyses to determine whether the Safety Limit would have been protected. The evaluation will use NRC-approved methods with assumptions similar to the overpressure analysis of record, except that the measured S/RV lift pressures will be used. If the Safety Limit would have continued to be protected, the SR is met. If not, the SR is not met. Failure to meet the proposed SR could occur due to testing of S/RVs that were removed from the plant and replaced with spares. Failure of the SR will require consideration of whether the condition is reportable in accordance with 10 CFR 50.72 and 50.73. If other analyses (safety analyses, structural analyses, etc.) would have been potentially affected by an S/RV not actuating with the assumed tolerance, the condition will be evaluated under the licensee's corrective action program, as is the current requirement, and reported as required by the regulations.

The Boiling Water Reactor Owners' Group (BWROG) has been working to improve S/RV performance for many years and has trended the performance of problematic two-stage S/RVs. This work is funded for 2020 and there are plans to request funding for the activity in the future in order to continue to monitor and improve S/RV performance across the BWR fleet.

Safety/Relief Valves that are removed from the plant for testing are refurbished, certified, and reset to within the as-left tolerance prior to reinstallation. Future operation with the reinstalled S/RVs is expected to meet all design and licensing basis requirements.

In summary, the proposed BWR/4 SR 3.4.3.1 and BWR/6 SR 3.4.4.1 will verify that the Overpressure Protection System, which represents the collective function of the S/RVs, will perform the specified safety function and will confirm that facility operation will be within safety limits. The requirements on individual S/RVs will be adequately controlled by 10 CFR 50.55a, the ASME OM Code, and 10 CFR 50.59, and do not need to appear in the TS.

<u>BWR/6 SR 3.4.4.2 Changes and BWR/4 SR 3.4.3.2 Addition</u>

BWR/6 SR 3.4.4.2 states, "Verify each [required] relief function S/RV actuates on an actual or simulated automatic initiation signal." The SR is revised to refer to the "safety/relief valve acting in the relief mode" instead of the "relief function" as previously discussed. The brackets around the word "required" are removed. Brackets indicate a plant-specific option. The equivalent SR in all four BWR/6 plants contains the word "required." Therefore, the brackets are removed to make the STS consistent with the plant TS.

At least two BWR/4 plants' overpressure protection analyses assume some S/RVs actuating in the relief mode. In order to accommodate those designs, an SR equivalent to BWR/6 SR 3.4.4.2 is added to the BWR/4 TS as new SR 3.4.3.2. The SR is in brackets, indicating that it is plant-specific.

BWR/4 SR 3.4.3.2 and BWR/6 SR 3.4.4.3 Changes

The existing SRs verify that each S/RV opens when manually actuated. The Bases for the SRs state that a manual actuation is performed to verify that, mechanically, the valve is functioning properly, and no blockage exists in the valve discharge line. This SR is relocated to the licensee's control as part of post-maintenance testing.

There is no analysis that credits manual opening of the S/RVs for overpressure protection. The inability to open an S/RV manually would not render the S/RV incapable of performing in the safety mode. The existing SR is a post-maintenance test to assure the S/RVs are operable after completion of the Inservice Testing. However, one goal of the Standard Technical Specifications was removal of post-maintenance SRs. As discussed in the SR 3.0.1 Bases, "Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2." Therefore, the existing SR is unnecessary and inconsistent with the Standard Technical Specifications and is removed.

Manual operation of the S/RV valves in the relief mode will continue to be tested by BWR/4 SR 3.5.1.12 and BWR/6 SR 3.5.1.7, "Verify each ADS valve opens when manually actuated," and BWR/4 and BWR/6 SR 3.6.1.6.1, "Verify each LLS valve opens when manually actuated."

Action Changes

Existing BWR/4 TS 3.4.4, Condition A, applies when one or two [required] S/RVs are inoperable. This action is no longer needed as the revised LCO and SR no longer contain requirements on each S/RV, but on the Overpressure Protection System. An inoperable S/RV would not render the OPS inoperable unless Safety Limit 2.1.2 would not be protected, in which case a plant shutdown is warranted.

Existing BWR/4 TS 3.4.4, Condition B, applies when the Required Action and associated Completion Time of Condition A is not met. As Condition A is deleted, Condition B is no longer needed and is also deleted.

Existing BWR/4 TS 3.4.4, Condition C, applies when [three] or more [required] S/RVs are inoperable. The Condition is renumbered Condition A and revised to state, "OPS inoperable." The existing Required Actions to be in Mode 3 in 12 hours and Mode 4 in 36 hours are retained.

The existing BWR/6 TS 3.4.4 (i.e., NUREG-1434) Actions are not consistent with the plant TS of the four BWR/6 plants. TS 3.4.4 for the BWR/6 plants (River Bend, Grand Gulf, Perry, and Clinton) contains a single action for one or more required S/RVs inoperable, that requires being in Mode 3 in 12 hours and in Mode 4 in 36 hours. Therefore, consistent with the existing plant TS and the proposed BWR/4 TS, existing Condition A is revised to apply when the OPS is inoperable and retains the existing Required Actions to be in Mode 3 in 12 hours and Mode 4 in 36 hours.

Testing to verify that the OPS is operable is typically performed during a shutdown when the LCO is not applicable. Therefore, it is worthwhile to consider how the Actions would be applied at power. As an example, consider a unit operating at 100% power. A vendor bulletin is received that identifies several installed S/RVs have faulty parts that could reduce the S/RV's relief capacity by 10%. Under the Corrective Action process, the licensee must evaluate whether the Overpressure Protection System is operable. An evaluation is performed to determine if the reduction in relief capacity in the affected S/RVs would render the OPS incapable of protecting Safety Limit 2.1.2 in the limiting event. If the remaining relief capacity is capable of protecting the Safety Limit, the OPS is operable. If not, the OPS is inoperable, and Action A would require a plant shutdown. Even though the proposed LCO and SRs do not specify the number of required S/RVs or their setpoints, there is no change in the need to ensure the ability to protect Safety Limit 2.1.2 when a condition is identified that could affect the S/RVs.

4. REGULATORY EVALUATION

4.1. <u>Applicable Regulatory Requirements/Criteria</u>

Section IV, "The Commission Policy," of the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 Federal Register 39132), dated July 22, 1993, states in part:

> The purpose of Technical Specifications is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by identifying those features that are of controlling importance to safety and establishing on them certain conditions of operation which cannot be changed without prior Commission approval.
>
> …[T]he Commission will also entertain requests to adopt portions of the improved STS, even if the licensee does not adopt all STS improvements.
>
> …The Commission encourages all licensees who submit Technical Specification related submittals based on this Policy Statement to emphasize human factors principles.
>
> …In accordance with this Policy Statement, improved STS have been developed and will be maintained for [BWR designs]. The Commission encourages licensees to use the improved STS as the basis for plant-specific Technical Specifications.
>
> …[I]t is the Commission intent that the wording and Bases of the improved STS be used … to the extent practicable.

As described in the Commission's "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," recommendations were made by NRC and industry task groups for new STS that include greater emphasis on human factors principles in order to add clarity and understanding to the text of the STS, and provide improvements to the Bases of STS, which provides the purpose for each requirement in the specification. Improved vendor-specific STS were developed and issued by the NRC in September 1992.

The regulation at Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(a)(1) requires an applicant for an operating license to include in the application proposed TS in accordance with the requirements of 10 CFR 50.36. The applicant must include in the application a "summary statement of the bases or reasons for such specifications, other than those covering administrative controls…." However, per 10 CFR 50.36(a)(1), these technical specification bases "shall not become part of the technical specifications." The Final Policy Statement provides the following description of the scope and the purpose of the Technical Specification Bases:

> Appropriate Surveillance Requirements and Actions should be retained for each LCO [limiting condition for operation] which remains or is included in the Technical Specifications. Each LCO, Action, and Surveillance Requirement should have supporting Bases. The Bases should at a minimum address the following questions and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases.
>
> 1. What is the justification for the Technical Specification, i.e., which Policy Statement criterion requires it to be in the Technical Specifications?
>
> 2. What are the Bases for each LCO, i.e., why was it determined to be the lowest functional capability or performance level for the system or component in question necessary for safe operation of the facility and, what are the reasons for the Applicability of the LCO?
>
> 3. What are the Bases for each Action, i.e., why should this remedial action be taken if the associated LCO cannot be met; how does this Action relate to other Actions associated with the LCO; and what justifies continued operation of the system or component at the reduced state from the state specified in the LCO for the allowed time period?
>
> 4. What are the Bases for each Safety Limit?
>
> 5. What are the Bases for each Surveillance Requirement and Surveillance Frequency; i.e., what specific functional requirement is the surveillance designed to verify? Why is this surveillance necessary at the specified frequency to assure that the system or component function is maintained, that facility operation will be within the Safety Limits, and that the LCO will be met?
>
> Note: In answering these questions the Bases for each number (e.g., Allowable Value, Response Time, Completion Time, Surveillance Frequency), state, condition, and definition (e.g., operability) should be clearly specified. As an example, a number might be based on engineering judgment, past experience, or PSA [probabilistic safety assessment] insights; but this should be clearly stated.

Additionally, 10 CFR 50.36(b) requires:

> Each license authorizing operation of a … utilization facility … will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 ["Contents of applications; technical information"]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The categories of items required to be in the TS are provided in 10 CFR 50.36(c). As required by 10 CFR 50.36(c)(2)(i), the TS will include LCOs, which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Per 10 CFR 50.36(c)(2)(i), when an LCO of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the TS until the condition can be met.

The regulation at 10 CFR 50.36(c)(3) requires TS to include items in the category of SRs, which are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

Per 10 CFR 50.90, whenever a holder of a license desires to amend the license, application for an amendment must be filed with the Commission, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

Per 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate.

The NRC staff's guidance for the review of TS is in Chapter 16, "Technical Specifications," of NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared Standard Technical Specifications for each of the light-water reactor nuclear designs.

4.2. <u>Conclusions</u>

In conclusion, based on the considerations discussed above, the proposed revision does not alter the current manner of operation and (1) there is reasonable assurance that the health and safety of the public will not be endangered by continued operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

5. REFERENCES

None.

Model Application

[DATE] 10 CFR 50.90

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

PLANT NAME
DOCKET NO. 50-[xxx]
SUBJECT: Application to Revise Technical Specifications to Adopt TSTF-576, "Revise Safety/Relief Valve Requirements"

Pursuant to 10 CFR 50.90, [LICENSEE] is submitting a request for an amendment to the Technical Specifications (TS) for [PLANT NAME, UNIT NOS.].

[LICENSEE] requests adoption of TSTF-576, "Revise Safety/Relief Valve Requirements." The proposed change revises the Safety/Relief Valve (S/RV) TS to align the requirements with the safety limits and the regulations.

The enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides the existing TS Bases pages marked to show revised text associated with the proposed TS changes and is provided for information only.

[[LICENSEE] requests that the amendment be reviewed under the Consolidated Line Item Improvement Process (CLIIP).] Approval of the proposed amendment is requested by [date]. Once approved, the amendment shall be implemented within [ ] days.

There are no regulatory commitments made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated [STATE] Official.

[In accordance with 10 CFR 50.30(b), a license amendment request must be executed in a signed original under oath or affirmation. This can be accomplished by attaching a notarized affidavit confirming the signature authority of the signatory, or by including the following statement in the cover letter: "I declare under penalty of perjury that the foregoing is true and correct. Executed on (date)." The alternative statement is pursuant to 28 USC 1746. It does not require notarization.]

If you should have any questions regarding this submittal, please contact [NAME, TELEPHONE NUMBER].

Sincerely,

[Name, Title]

Enclosure: Description and Assessment

Attachments:
1. Proposed Technical Specification Changes (Mark-Up)
2. Revised Technical Specification Pages
3. Proposed Technical Specification Bases Changes (Mark-Up) – For Information Only

[The attachments are to be provided by the licensee and are not included in the model application.]

cc: NRC Project Manager
NRC Regional Office
NRC Resident Inspector
State Contact

ENCLOSURE

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

[LICENSEE] requests adoption of TSTF-576, "Revise Safety/Relief Valve Requirements." The proposed change revises the Safety/Relief Valve (S/RV) Technical Specifications (TS) to align the requirements with the safety limits and the regulations.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

[LICENSEE] has reviewed the safety evaluation for TSTF-576 provided to the Technical Specifications Task Force in a letter dated [DATE]. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-576. [As described herein,] [LICENSEE] has concluded that the justifications presented in TSTF-576 and the safety evaluation prepared by the NRC staff are applicable to [PLANT, UNIT NOS.] and justify this amendment for the incorporation of the changes to the [PLANT] TS.

2.2 Optional Changes and Variations

[LICENSEE is not proposing any variations from the TS changes described in TSTF-576 or the applicable parts of the NRC staff's safety evaluation dated [DATE].] [LICENSEE is proposing the following variations from the TS changes described in TSTF-576 or the applicable parts of the NRC staff's safety evaluation: describe the variations]

[The [PLANT] TS utilize different [numbering][and][titles] than the Standard Technical Specifications on which TSTF-576 was based. Specifically, [describe differences between the plant-specific TS numbering and/or titles and the TSTF-576 numbering and titles.] These differences are administrative and do not affect the applicability of TSTF-576 to the [PLANT] TS.]

[The [PLANT] TS contain requirements that differ from the Standard Technical Specifications on which TSTF-576 was based but are encompassed in the TSTF-576 justification. [Describe differences and why TSTF-576 is still applicable.]]

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

[LICENSEE] requests adoption of TSTF-576, "Revise Safety/Relief Valve Requirements." The proposed change revises the Safety/Relief Valve (S/RV) Technical Specifications (TS) to align the requirements with the safety limits and the regulations. The Limiting Condition for Operation (LCO) and Surveillance Requirements (SRs) are revised to replace requirements on each credited S/RV with a requirement that the Overpressure Protection System (OPS) be operable. Operability of the OPS is defined as the capability to prevent an overpressure event from exceeding Safety Limit 2.1.2, "Reactor Coolant System Pressure." An SR that tests the ability of the S/RVs to be capable of manual operation is removed as that capability is not credited in any safety analysis. [An SR that verifies the ability of credited S/RVs acting in the relief mode is [revised/added] to be consistent with the revised LCO.] The TS Actions are revised to be consistent with the changes to the LCO and SRs. Administrative changes are made to the TS for clarity and consistency.

[LICENSEE] has evaluated if a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change revises the S/RV TS to align the requirements with the safety limits and the regulations. The proposed change does not change the design or operation of the S/RVs. Therefore, the probability of an accident previously evaluated is not affected. Each S/RV will continue to be tested under the auspices of the Inservice Testing Program required by 10 CFR 50.55a(f) and the aggregate performance of the S/RVs will be verified to ensure the overpressure Safety Limit will still be met. The accident analyses consider the aggregate operation of the credited S/RVs, not the performance of individual valves. The proposed change moves the S/RV setpoints and tolerances to licensee control, to be governed by the Inservice Testing Program, which is required by 10 CFR 50.55a. Altering the control process for these values has no effect on previously performed accident evaluations. Therefore, the ability of the S/RVs to mitigate any accident previously evaluated is not significantly decreased.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change revises the S/RV TS to align the requirements with the safety limits and the regulations. The proposed change does not alter the design function or operation of the S/RVs. The proposed change does not create any new credible failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing basis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The proposed change revises the S/RV TS to align the requirements with the safety limits and the regulations. The proposed change ensures that the S/RVs can protect Safety Limit 2.1.2. Although the setpoints and tolerances of specific S/RVs are moved to licensee control, the safety margin provided by the aggregate S/RV capability, which ensures the Safety Limit is protected, is not changed. The proposed change does not alter a design basis limit or a safety limit, and, therefore, does not reduce the margin of safety.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, [LICENSEE] concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL EVALUATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Technical Specifications and Bases Changes

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Overpressure Protection System (OPS) ~~Safety/Relief Valves (S/RVs)~~

LCO 3.4.3 The OPS ~~safety function of [11] S/RVs~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. [One [or two] [required] S/RV[s] inoperable.~~	~~A.1 Restore the [required] S/RV[s] to OPERABLE status.~~	~~14 days]~~
~~B. [Required Action and associated Completion Time of Condition A not met.]~~	~~B.1 -------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------------~~ ~~Be in MODE 3.~~	~~12 hours~~
A~~C~~. OPS inoperable. ~~[Three] or more [required] S/RVs inoperable.~~	A~~C~~.1 Be in MODE 3. AND A~~C~~.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the OPS has the capability to prevent reactor steam dome pressure from exceeding Safety Limit 2.1.2. ~~-----NOTE-----~~ ~~≤ [2] [required] S/RVs may be changed to a lower setpoint group.~~ ~~----------~~ ~~Verify the safety function lift setpoints of the [required] S/RVs are as follows:~~ ~~Number of S/RVs — Setpoint (psig)~~ ~~[4] — [1090 ± 32.7]~~ ~~[4] — [1100 ± 33.0]~~ ~~[3] — [1110 ± 33.3]~~ ~~Following testing, lift settings shall be within ± 1%.~~	~~[~~ In accordance with the Inservice Testing Program ~~OR~~ ~~[[18] months]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
[SR 3.4.3.2	-----NOTE----- Valve actuation may be excluded. ---------- Verify each required safety/relief valve acting in the relief mode actuates on an actual or simulated automatic initiation signal.	[[18] months OR In accordance with the Surveillance Frequency Control Program]]
~~SR 3.4.3.2~~	~~-----NOTE-----~~ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ ~~----------~~ ~~Verify each [required] S/RV opens when manually actuated.~~	~~[[18] months [on a STAGGERED~~

SURVEILLANCE	FREQUENCY
	~~TEST BASIS for each valve solenoid~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]]~~

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BASES

SURVEILLANCE REQUIREMENTS (continued)

----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.3.6.3.7

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specified channel. The system functional testing performed in LCO 3.4.3, "Overpressure Protection System (OPS) ~~Safety/Relief Valves(S/RVs)~~" and LCO 3.6.1.8, "Low-Low Set (LLS) Safety/Relief Valves (S/RVs)," for S/RVs overlaps this test to provide complete testing of the assumed safety function.

[The Frequency of once every 18 months for SR 3.3.6.3.7 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

REFERENCES

1. FSAR, Figure [ ] .

2. FSAR, Section [5.5.17].

3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.3.7

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specified channel. The system functional testing performed in LCO 3.4.3, "Overpressure Protection System (OPS) ~~Safety/Relief Valves(S/RVs)~~" and LCO 3.6.1.8, "Low-Low Set (LLS) Safety/Relief Valves (S/RVs)," for S/RVs overlaps this test to provide complete testing of the assumed safety function.

[The Frequency of once every 18 months for SR 3.3.6.3.7 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

REFERENCES

1. FSAR, Figure [ ] .
2. FSAR, Section [5.5.17].
3. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 ~~Safety/Relief Valves (S/RVs)~~Overpressure Protection System (OPS)

BASES

BACKGROUND

The Overpressure Protection System (OPS) prevents overpressurization of the nuclear system by discharging reactor steam to the suppression pool. This action protects the reactor coolant pressure boundary (RCPB) from failure which could result in the release of fission products (Ref. 1).

The American Society of Mechanical Engineers (ASME) ~~-~~Boiler and Pressure Vessel Code (Ref. 2) requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. ~~. As part of the nuclear pressure relief system, t~~The size and number of safety/relief valves (S/RVs) are selected such that peak pressure ~~in the nuclear system~~ will not exceed the ASME Code limits for the ~~reactor coolant pressure boundary (~~RCPB~~)~~.

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. The safety mode is credited for overpressure protection. ~~This satisfies the Code requirement.~~

[In the relief mode (or power actuated mode of operation), a pneumatic piston or cylinder and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. Some S/RVs operating in the relief mode are also credited for overpressure protection.]

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."

APPLICABLE SAFETY ANALYSES

The OPS~~overpressure protection system~~ must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). The S/RV discharge piping is designed to accommodate forces resulting from relief action including interactions with the suppression pool and is supported for reactions due to flow at maximum S/RV discharge capacity so that system integrity is maintained. For the purpose of the overpressure protection analyses (Ref. 1), [six~~11~~] S/RVs are assumed to operate in the safety mode of operation [and [seven] S/RVs are assumed to operate in the relief mode]. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the ~~Design~~ design ~~Basis~~ basis ~~Event~~event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference ~~2~~ 3 discusses additional events that are expected to actuate the S/RVs.

The OPS satisfies ~~S/RVs satisfy~~ Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES

LCO

The OPS is OPERABLE when it can ensure that the ASME Code limit on peak reactor pressure, as stated in Safety Limit 2.1.2, will be protected using the safety ~~function~~ mode of the ~~[11]~~ S/RVs [and the relief mode of additional S/RVs].~~are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2)~~. OPERABILITY of the OPS is only dependent on ~~The requirements of this LCO are applicable only to~~ the capability of the S/RVs to ~~mechanically~~ open to relieve excess pressure, and may credit less than the full complement of installed S/RVs. ~~when the lift setpoint is exceeded (safety function).~~

~~The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the FSAR are based on these setpoints, but also include the additional uncertainties of ± 1% of the nominal setpoint drift to provide an added degree of conservatism.~~

An inoperable OPS ~~Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits,~~ could result in a more severe reactor response to a transient than predicted, possibly resulting in Safety Limit 2.1.2 ~~the ASME Code limit on reactor pressure~~ being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, the OPS ~~all S/RVs~~ must be OPERABLE~~,~~ since there may be considerable energy ~~may be~~ in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The OPS ~~S/RVs~~ may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The OPS ~~S/RV function~~ is not needed during these conditions.

ACTIONS

~~[A.1~~

~~With the safety function of one [or two] [required] S/RV[s] inoperable, the remaining OPERABLE S/RVs are capable of providing the necessary overpressure protection. Because of additional design margin, the ASME Code limits for the RCPB can also be satisfied with two S/RVs inoperable. However, the overall reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE S/RVs could~~

~~result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.~~

BASES

ACTIONS (continued)

~~The 14 day Completion Time to restore the inoperable required S/RVs to OPERABLE status is based on the relief capability of the remaining S/RVs, the low probability of an event requiring S/RV actuation, and a reasonable time to complete the Required Action.]~~

~~B.1~~

~~-----------------------------------REVIEWER'S NOTE -----------------------------------~~
~~Adoption of a MODE 3 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in TSTF-IG-05-02, Implementation Guidance for TSTF-423, Revision 2, "Technical Specifications End States, NEDC-32988-A," November 2009.~~
~~---~~

~~If the safety function of the inoperable required S/RVs cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to MODE 3 within 12 hours.~~

~~Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 3) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low risk state.~~

~~Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

BASES

ACTIONS (continued)

~~The allowed Completion Time is reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

A~~C~~.1 and A~~C~~.2

If the OPS is inoperable, ~~[three] or more [required] S/RVs are inoperable,~~ a transient may result in the violation of the ASME Code limit on reactor pressure. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance ~~requires~~ verifies that the OPS has the capability to ~~that the [required] S/RVs~~ prevent the reactor steam dome pressure from exceeding Safety Limit 2.1.2~~will open at the pressures assumed in the safety analysis of Reference 1~~. The testing of the ~~demonstration of the~~ S/RV ~~safe~~ safety mode lift settings is ~~must be~~ performed during shutdown, since this is a bench test, ~~[~~to be done in accordance with the Inservice Testing Program~~]~~. The measured S/RV mechanical lift values tested in accordance with the Inservice Testing Program are reviewed and aggregated to verify that the collective performance of the credited S/RVs will ensure Safety Limit 2.1.2 is protected. Should one or more of the credited S/RVs not actuate within the assumed tolerance, the actual lift values will be used to evaluate the affected overpressure analyses to determine whether the Safety Limit would have been protected. In this case, the SR consists of a combination of testing and calculation. ~~The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. [A Note is provided to allow up to [two] of the required [11] S/RVs to be physically replaced with S/RVs with lower setpoints. This provides operational flexibility which maintains the assumptions in the over-pressure analysis.]~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~If the testing is within the scope of the licensee's Inservice Testing Program, the Frequency "In accordance with the Inservice Testing Program" should be used.~~

~~Otherwise, the periodic Frequency of 18 months or the reference to the Surveillance Frequency Control Program should be used.~~

~~--~~

~~[The 18 month Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~---------------------------------REVIEWER'S NOTE---------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

[SR 3.4.3.2

The OPS assumes that the required relief mode S/RVs actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify the mechanical portions of the automatic relief mode operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.7 overlaps this SR to provide complete testing of the relief mode function.

[The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

---------------------------------REVIEWER'S NOTE---------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.

---]

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.]

~~SR 3.4.3.2~~

~~A manual actuation of each [required] S/RV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is [920] psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by [at least 1.25 turbine bypass valves open, or total steam flow ≥10^6 lb/hr]. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.~~

~~[The [18] month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code (Ref. 4). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~OR~~

SURVEILLANCE REQUIREMENTS (continued)

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~

~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~--]~~

REFERENCES

1. FSAR, Section [5.2.2.2.4].

<u>4</u>~~2~~<u>. ASME Code for Operation and Maintenance of Nuclear Power Plants.</u>

<u>3</u>~~2~~. FSAR, Section [15].

~~3. NEDC-32988-A, Revision 2, Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants, December 2002.~~

~~4. ASME Code for Operation and Maintenance of Nuclear Power Plants.~~

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3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Overpressure Protection System (OPS) ~~Safety/Relief Valves (S/RVs)~~

LCO 3.4.4 The OPS shall be OPERABLE.~~The safety function of [seven] S/RVs shall be OPERABLE,~~

~~AND~~

~~The relief function of [seven] additional S/RVs shall be OPERABLE.~~

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. [One [required] S/RV inoperable.~~	~~A.1 Restore [required] S/RV to OPERABLE status.~~	~~14 days]~~
~~B. [Required Action and associated Completion Time of Condition A not met.]~~	~~B.1 -------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 3. ---------------------------------~~ ~~Be in MODE 3.~~	~~12 hours~~
A~~C~~. OPS inoperable.~~[Two] or more [required] S/RVs inoperable.~~	A~~C~~.1 Be in MODE 3. AND A~~C~~.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the OPS has the capability to prevent reactor steam dome pressure from exceeding Safety Limit 2.1.2. ~~-----NOTE-----~~ ~~≤ [2] [required] S/RVs may be changed to a lower setpoint group.~~ ~~----------~~ ~~Verify the safety function lift setpoints of the [required] S/RVs are as follows:~~ ~~Number of S/RVs — Setpoint (psig)~~ ~~[8] — [1165 ± 34.9]~~ ~~[6] — [1180 ± 35.4]~~ ~~[6] — [1190 ± 35.7]~~ ~~Following testing, lift settings shall be within ± 1%.~~	~~[~~ In accordance with the Inservice Testing Program ~~OR~~ ~~[[18] months]~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~
SR 3.4.4.2	-----NOTE----- Valve actuation may be excluded. ---------- Verify each [required] ~~relief function~~ safety/relief valve acting in the relief mode ~~S/RV~~ actuates on an actual or simulated automatic initiation signal.	[[18] months OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.4.4.3~~	~~-----------------------------NOTE-----------------------------~~ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ ~~---~~ ~~Verify each [required] S/RV opens when manually actuated.~~	~~[[18] months on a STAGGERED TEST BASIS for each valve solenoid~~ ~~OR~~ ~~In accordance with the Surveillance Frequency Control Program]~~

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B 3.3 INSTRUMENTATION

B 3.3.6.5A Relief and Low-Low Set (LLS) Instrumentation (Without Setpoint Control Program)

BASES

BACKGROUND

The safety/relief valves (S/RVs) prevent overpressurization of the nuclear steam system. Instrumentation is provided to support two modes of S/RV operation - the relief function (all valves) and the LLS function (selected valves). Refer to LCO 3.4.4, "Overpressure Protection System (OPS)~~Safety/Relief Valves (S/RVs)~~," and LCO 3.6.1.6, "Low-Low Set (LLS) Safety/Relief Valves (S/RVs)," for Applicability Bases for additional information of these modes of S/RV operation. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the Safety/Relief valve instrumentation, as well as LCOs on other reactor system parameters, and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that an SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

---------------------------------REVIEWER'S NOTE-----------------------------------
The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.

"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.

B 3.3 INSTRUMENTATION

B 3.3.6.5B Relief and Low-Low Set (LLS) Instrumentation (With Setpoint Control Program)

BASES

BACKGROUND

The safety/relief valves (S/RVs) prevent overpressurization of the nuclear steam system. Instrumentation is provided to support two modes of S/RV operation - the relief function (all valves) and the LLS function (selected valves). Refer to LCO 3.4.4, " Overpressure Protection System (OPS)~~Safety/Relief Valves (S/RVs)~~," and LCO 3.6.1.6, "Low-Low Set (LLS) Safety/Relief Valves (S/RVs)," for Applicability Bases for additional information of these modes of S/RV operation. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the Safety/Relief valve instrumentation, as well as LCOs on other reactor system parameters, and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that an SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur. The LSSS values are identified and maintained in the Setpoint Control Program (SCP) controlled by 10 CFR 50.59.

--------------------------------REVIEWER'S NOTE----------------------------------
The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.

"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 ~~Safety/Relief Valves (S/RVs)~~Overpressure Protection System (OPS)

BASES

BACKGROUND

The Overpressure Protection System (OPS) prevents overpressurization of the nuclear system by discharging reactor steam to the suppression pool. This action protects the reactor coolant pressure boundary (RCPB) from failure which could result in the release of fission products (Ref. 1).

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 2~~1~~) requires the r~~R~~eactor p~~P~~ressure v~~V~~essel be protected from overpressure during upset conditions by self~~-~~-actuated safety valves. ~~As part of the nuclear pressure relief system, t~~The size and number of safety/relief valves (S/RVs) are selected such that peak pressure ~~in the nuclear system~~ will not exceed the ASME Code limits for the ~~reactor coolant pressure boundary (~~RCPB~~)~~.

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the direct action of the steam pressure in the main steam lines will act against a spring loaded disk that will pop open when the valve inlet pressure exceeds the spring force. The safety mode is credited for overpressure protection.

In the relief mode (or power actuated mode of operation), a pneumatic piston or cylinder and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. Some S/RVs operating in the relief mode are also credited for overpressure protection.

Some ~~Six~~ of the S/RVs providing the relief function also provide the low-low set relief function specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves." Some ~~Eight~~ of the S/RVs that provide the relief function are part of the Automatic Depressurization System specified in LCO 3.5.1, "ECCS - Operating." The instrumentation associated with the relief valve function and low-low set relief function is discussed in the Bases for LCO 3.3.6.5, "Relief and Low-Low Set (LLS) Instrumentation," and instrumentation for the ADS function is discussed in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation."

APPLICABLE SAFETY ANALYSES

The OPS ~~overpressure protection system~~ must accommodate the most severe pressure transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1~~2~~). The S/RV discharge piping is designed to accommodate forces resulting from relief action including interactions with the suppression pool and is supported for reactions due to flow at maximum S/RV discharge capacity so that system integrity is maintained. For the purpose of the overpressure protection analyses (Ref. 1), [six] S/RVs are assumed to operate in the safety mode of operation and [seven] S/RVs are assumed to operate in the relief mode.~~[six] of the S/RVs are assumed to operate in the relief mode, and seven in the safety mode.~~ The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below

APPLICABLE SAFETY ANALYSES (continued)

the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the design basis event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 3 discusses additional events that are expected to actuate the S/RVs.

The OPS satisfies ~~S/RVs satisfy~~ Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPS is OPERABLE when it can ensure that the ASME Code limit on peak reactor pressure, as stated in Safety Limit 2.1.2, will be protected using the safety ~~function~~ mode of the ~~seven~~ S/RVs and the relief mode of additional S/RVs.~~is required to be OPERABLE in the safety mode, and an additional seven S/RVs (other than the seven S/RVs that satisfy the safety function) must be OPERABLE in the relief mode.~~ OPERABILITY of the OPS is only dependent on ~~The requirements of this LCO are applicable only to~~ the capability of the S/RVs to ~~mechanically~~ open to relieve excess pressure, and may credit less than the full complement of installed S/RVs. ~~In Reference 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of seven S/RVs in the safety mode and six S/RVs in the relief mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded.~~

~~The S/RV setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure~~

~~(1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for conditions. The transient evaluations in Reference 3 are based on these setpoints, but also include the additional uncertainties of ± 1% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.~~

An inoperable OPS ~~Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits,~~ could result in a more severe reactor response to a transient than predicted, possibly resulting in Safety Limit 2.1.2 ~~the ASME Code limit on reactor pressure~~ being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, the OPS ~~specified number of S/RVs~~ must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The OPS ~~S/RVs~~ may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

BASES

APPLICABILITY (continued)

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The OPS ~~S/RV function~~ is not needed during these conditions.

ACTIONS

~~A.1~~

~~With the safety function of one [required] S/RV inoperable, the remaining OPERABLE S/RVs are capable of providing the necessary overpressure protection. Because of additional design margin, the ASME Code limits for the RCPB can also be satisfied with two S/RVs inoperable. However, the overall reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE S/RVs could result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.~~

~~The 14 day Completion Time to restore the inoperable required S/RVs to OPERABLE status is based on the relief capability of the remaining S/RVs, the low probability of an event requiring S/RV actuation, and a reasonable time to complete the Required Action.~~

~~B.1~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Adoption of a MODE 3 end state requires the licensee to make the following commitments:~~

~~1. [LICENSEE] will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 3, July 2000.~~

~~2. [LICENSEE] will follow the guidance established in TSTF-IG-05-02, Implementation Guidance for TSTF-423, Revision 2, "Technical Specifications End States, NEDC-32988-A," November 2009.~~
~~---~~

~~If the inoperable required S/RV cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, the plant must be brought to a MODE in which overall plant risk is minimized. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours.~~

ACTIONS (continued)

~~Remaining in the Applicability of the LCO is acceptable because the plant risk in MODE 3 is similar to or lower than the risk in MODE 4 (Ref. 4) and because the time spent in MODE 3 to perform the necessary repairs to restore the system to OPERABLE status will be short. However, voluntary entry into MODE 4 may be made as it is also an acceptable low-risk state.~~

~~Required Action B.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 3. This Note prohibits the use of LCO 3.0.4.a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.~~

~~The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.~~

A~~C~~.1 and A~~C~~.2

If the OPS is inoperable, ~~[two] or more [required] S/RVs are inoperable,~~ a transient may result in the violation of the ASME Code limit on reactor pressure. The plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance verifies that the OPS has the capability to prevent the ~~demonstrates that the [required] S/RVs~~ reactor steam dome pressure from exceeding Safety Limit 2.1.2~~will open at the pressures assumed in the safety analysis of Reference 2~~. The testing of the ~~demonstration of the~~ S/RV safety mode ~~safety function~~ lift settings is ~~must be~~ performed during shutdown, since this is a bench test~~[~~, to be done ~~and~~ in accordance with the Inservice Testing Program~~]~~. The measured S/RV mechanical lift values tested in accordance with the Inservice Testing Program are reviewed and aggregated to verify that the collective performance of the credited S/RVs will ensure Safety Limit 2.1.2 is protected. Should one or more of the credited S/RVs not actuate within the assumed tolerance, the actual lift values will be used to evaluate the affected overpressure analyses to determine whether the Safety Limit would have been protected. In this case, the SR is met by a combination of testing and calculation. ~~The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. [A Note is provided to allow up to [two] of the required [11] S/RVs to be physically~~

~~replaced with S/RVs with lower setpoints. This provides operational flexibility which maintains the assumptions in the over-pressure analysis.]~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~If the testing is within the scope of the licensee's Inservice Testing Program, the Frequency "In accordance with the Inservice Testing Program" should be used. Otherwise, the periodic Frequency of 18 months or the reference to the Surveillance Frequency Control Program should be used.~~
~~---~~

~~[The [18 month] Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

SR 3.4.4.2

The OPS assumes that t~~T~~he required ~~[required]~~ relief ~~function~~ mode S/RVs ~~are required to~~ actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify the mechanical portions of the automatic relief ~~function~~ mode operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.5.4 overlaps this SR to provide complete testing of the relief mode ~~safety~~ function.

[The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

~~SR 3.4.4.3~~

~~A manual actuation of each [required] S/RV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 950 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by [at least 1.25 turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr]. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.~~

~~[The [18] month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~ASME (Ref. 1). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

~~OR~~

~~The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.~~

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~
~~Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.~~
~~---]~~

REFERENCES

1. FSAR, Section [5.2.5.5.3].

2~~1~~. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~2. FSAR, Section [5.2.5.5.3].~~

3. FSAR, Section [15].

~~4. NEDC-32988-A, Revision 2, Technical Justification to Support Risk-Informed Modification to Selected Required End States for BWR Plants, December 2002.~~