Southern Company

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Docket Nos.: 52-025
52-026

ND-17-0295
10 CFR 50.90
10 CFR 52.63

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

Southern Nuclear Operating Company
Vogtle Electric Generating Plant Units 3 and 4
Request for License Amendment and Exemption:
Standardization of Instrumentation Setpoint Nomenclature (LAR-17-004)

Ladies and Gentlemen:

Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC), requests an amendment to Combined Licenses (COLs) for Vogtle Electric Generating Plant (VEGP) Units 3 and 4 (License Nos. NPF-91 and NPF-92, respectively). The requested amendment requires changes to the Updated Final Safety Analysis Report (UFSAR) in the form of departures from the Plant-Specific Design Control Document (DCD) Tier 2 information and involves changes to plant-specific Tier 1 information (and corresponding changes to COL Appendix C). Additionally, this request involves changes to the VEGP Units 3 and 4 COL Appendix A, Technical Specifications. Because the proposed changes impact Tier 1 of the Plant-Specific (DCD), Appendix C of the COL, and the Technical Specifications, this activity has been determined to require prior NRC approval. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is also requested for the plant-specific Tier 1 departures.

The proposed changes standardize the Protection and Safety Monitoring System (PMS) setpoint nomenclature.

Enclosure 1 provides the description, technical evaluation, regulatory evaluation (including the Significant Hazards Consideration Determination), and environmental considerations for the proposed changes in the License Amendment Request (LAR).

Enclosure 2 provides the background and supporting basis for the requested exemption.

Enclosure 3 provides the proposed changes to the licensing basis documents.

Enclosure 4 provides conforming Technical Specification Bases changes for information only.

This letter contains no regulatory commitments. This letter has been reviewed and confirmed to not contain security-related information.

In order to support the Vogtle Units 3 and 4 PMS software installation schedule, SNC requests NRC staff review and approval of the license amendment and exemption no later than September 13, 2017. Approval by this date will allow sufficient time to implement licensing basis changes prior to PMS software installation activities. SNC expects to implement the proposed amendment within thirty days of approval. South Carolina Electric & Gas Company (SCE&G) has stated that the current requested approval date for the parallel LAR and Exemption for Virgil C. Summer Nuclear Station (VCSNS) Unit 2 will be November 6, 2017.

In accordance with 10 CFR 50.91, SNC is notifying the State of Georgia of this LAR by transmitting a copy of this letter and enclosures to the designated State Official.

Should you have any questions, please contact Mr. Christopher L. Whitfield at (205) 992-5071.

Mr. Brian H. Whitley states that: he is the Regulatory Affairs Director of Southern Nuclear Operating Company; he is authorized to execute this oath on behalf of Southern Nuclear Operating Company; and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

LISA MYRICK SPEARS
MY COMMISSION EXPIRES June 18, 2019
NOTARY PUBLIC
STATE OF ALABAMA

Brian H. Whitley

Brian H. Whitley

BHW/CLW/ljs

Sworn to and subscribed before me this 24th *day of* February *, 2017*

Notary Public: Lisa Myrick Spears

My commission expires: June 18, 2019

Enclosures:

1) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Request for License Amendment: Standardization of Instrumentation Setpoint Nomenclature (LAR-17-004)
2) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Exemption Request: Standardization of Instrumentation Setpoint Nomenclature (LAR-17-004)
3) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Proposed Changes to the Licensing Basis Documents (LAR-17-004)
4) Vogtle Electric Generating Plant (VEGP) Units 3 and 4 – Conforming Technical Specification Bases Changes (LAR-17-004) – (For Information Only)

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Southern Nuclear Operating Company

ND-17-0295

Enclosure 1

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Request for License Amendment:

Standardization of Instrumentation Setpoint Nomenclature

(LAR-17-004)

(Enclosure 1 consists of 29 pages, including this cover page)

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Pursuant to 10 CFR 52.98(c) and in accordance with 10 CFR 50.90, Southern Nuclear Operating Company (SNC, or the “Licensee”) hereby requests an amendment to Combined License (COL) Nos. NPF-91 and NPF-92 for Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

1. SUMMARY DESCRIPTION

The proposed changes revise the Combined Licenses (COLs) concerning standardizing the Protection and Safety Monitoring System (PMS) setpoint nomenclature. No setpoint values or PMS alarms and actuations are proposed to be changed by this activity.

These proposed changes include the following:

1. Reactor Coolant Pump Bearing Water Temperature “High” changed to “High-2”;
2. Pressurizer Pressure:
    a. “Low” changed to “Low-2” for the Reactor Trip System (RTS) trip; and
    b. “Low” changed to “Low-3” for the Engineered Safety Features Actuation System (ESFAS) trip;
3. Pressurizer Pressure “High” changed to “High-2”;
4. Pressurizer Water Level:
    a. “High” changed to “High-3” for the RTS trip; and
    b. “High” changed to “High-2” or “High-3” for ESFAS trip functions;
5. Reactor Coolant Flow “Low” changed to “Low-2”;
6. Reactor Coolant Pump Speed “Low” changed to “Low-2”;
7. Steam Generator Water Level:
    a. “High” and/or “High-2” changed to “High-3”;
    b. “High” changed to clarify two separate trips -- “High-3” and “Reactor trip plus High Steam Generator Level”; and
    c. One occurrence of “High-2” corrected to “High”;
8. Core Makeup Tank (CMT) Water Level:
    a. “Low-1” (or “Low”) changed to “Low-3” and
    b. “Low-2” (or “Low-Low”) changed to “Low-6”;
9. Steam Line Pressure “Low” changed to “Low-2”;
10. Reactor Coolant System (RCS) Hot Leg Level:
    a. “Low-1” (or “Low”) changed to “Low-2” and
    b. “Low-2” (or “Low”) changed to “Low-4”;
11. RCS Cold Leg Temperature (T_{cold}) “Low” changed to “Low-2”; and
12. Startup Feedwater Flow “Low” changed to “Low-2”;

13. General Consistency changes:
    a. Technical Specifications (TS) – “Low-1” and “High-1” changed to “Low” and “High”
    b. Subsection 15.5.1.2 – “high-3 pressurizer signal” changed to “High-3 pressurizer water level signal”

The requested amendment proposes changes to the Updated Final Safety Analysis Report (UFSAR) in the form of departures from the plant-specific Design Control Document (DCD) Tier 2 information, and involves changes to related plant-specific DCD Tier 1 information, with corresponding changes to the associated COL Appendix C information. In addition, revisions are proposed to COL Appendix A, TS. This enclosure requests approval of the license amendment necessary to implement the Tier 2 and COL Appendix A and Appendix C changes.

Enclosure 2 requests the exemption necessary to implement the involved changes to the plant-specific DCD Tier 1 information.

Enclosure 3 provides the proposed markups to the license basis documents requested for approval.

Enclosure 4 provides conforming Technical Specification Bases changes for information only.

2. DETAILED DESCRIPTION

Background

The PMS provides detection of off-nominal conditions and actuation of appropriate safety-related functions necessary to achieve and maintain the plant in a safe shutdown condition. The PMS initiates reactor trips and ESFAS functions when plant conditions reach specified setpoints. It has four divisions of reactor trip and ESFAS actuation, and two divisions of safety-related post-accident parameter displays. The PMS controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

The AP1000 digital Instrumentation & Control (I&C) systems, such as the PMS and the Plant Control System (PLS), use various parameters and setpoints to control and protect the plant and to provide alarms to plant personnel. Specific values are used for each setpoint, which depend on the specific plant function that uses the given parameter. In addition to specific values, setpoints are also given generic designators or names. The exact setpoint value may change as the result of a design change, but the generic designator or name for the setpoint generally remains unchanged. For example, the reactor coolant pump is tripped on a Low-2 pressurizer level. “Low-2” is the specific designator for the setpoint. The actual pressurizer level value for “Low-2” may change, but the designator name “Low-2” will remain. In some instances, the design includes a higher level setpoint, such as a “Low-1,” to provide a warning alarm to the operator prior to the system reaching the “Low-2” actuation setpoint.

The existing UFSAR and Combined License (COL) Appendix A (Technical Specifications) and Appendix C (and corresponding plant-specific Tier 1 information) are not consistent when referring to PMS setpoint designators. For example, a given setpoint may be referred to with its specific designator, such as Low-3, but in other instances it is referred to with the generic designator "low." Therefore, it is necessary to revise some generic "low" and "high" references which are currently inclusive of multiple setpoint values to avoid any confusion as to which specific setpoint designator is used for a given PMS function. In addition, the format used for the setpoint designator is inconsistent between these documents.

In some instances a given setpoint designator is used for more than one setpoint value. For example, the High-2 steam generator water level designator is used for both an alarm and the PMS reactor trip function, even though they are actuated at different setpoint values. To avoid confusion and address the human factor issues related to labeling different setpoint values with the same setpoint designator, a given setpoint is referred to with its own unique setpoint designator value.

Description of the Activity

The PMS setpoint nomenclature for the AP1000 is proposed to be standardized. The proposed standardization follows the scheme shown in Figure 1. In this setpoint nomenclature scheme, lower and higher setpoint values are designated with a higher degree of "lowness" and "highness," and only one setpoint designator is used per setpoint.

Figure 1: Setpoint Nomenclature Scheme

Low Setpoints	High Setpoints
Low / Low-1*	High / High-1*
Low-2	High-2
Low-3	High-3
[etc.]	[etc.]

* Low and Low-1 are considered to be equivalent terms when referring to the first low setpoint designator. High and High-1 are considered to be equivalent terms when referring to the first high setpoint designator.

The PMS RTS and ESFAS setpoint designators referenced within the UFSAR, COL Appendix A, and COL Appendix C, and the corresponding plant-specific Tier 1 are proposed to be updated, as necessary, to reflect this nomenclature scheme. The proposed updates include:

- Changes in nomenclature format. For example, "low-low" to "Low-2," "Hi-2" to "High-2," and "underspeed" to "Low-2" speed.

- Changes in high and low setpoint designator levels. For example, “Low” to “Low-3” and “High-2” to “High-3.” In some instances these changes are made for internal consistency within the licensing basis. In other instances, the setpoint designator is renamed so that any single setpoint designator is only used for a single setpoint value.
- A new note for UFSAR Table 7.3-1 to explicitly state that “Low” and “Low-1,” and “High” and “High-1,” are equivalent designators. For consistency, the Technical Specifications are proposed to change each “Low 1” and “High 1” reference to “Low” and “High,” respectively.

Proposed Licensing Basis Changes

Table 1 that follows provides the proposed licensing basis changes for COL Appendix A, COL Appendix C (and corresponding plant-specific Tier 1 information), and the involved Tier 2 changes. There are no Tier 2* proposed changes.

Note that Enclosure 4 provides draft TS Bases conforming revisions for information only. TS Bases changes are made after the approval of the amendment request, in accordance with the Bases Control Program.

Table 1: Proposed Licensing Basis Changes

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Reactor Coolant Pump Bearing Water Temperature: “High” changed to “High-2”	Table 2.5.2-2	Table 3.3.1-1, Function 8 Table 3.3.8-1, Function 19	Chapter 1 Section 1.2.1.2.3 Chapter 5 Section 5.4.1.3.4 Section 5.4.1.3.5 Chapter 6 Table 6.2.3-1 Chapter 7 Section 7.2.1.1.3 Table 7.2-2 Figure 7.2-1 (Sheets 2,5) Section 7.3.1.2.5 Section 7.3.1.2.25 Table 7.3-1 WCAP-16675, Section 1.1 Chapter 9 Section 9.2.2.2 Section 9.2.2.3.4 Section 9.2.2.4.5.2 Section 9.2.2.7 Chapter 14 Table 14.3-2

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Pressurizer Pressure: a. “Low” changed to “Low-2” for the RTS trip b. “Low” changed to “Low-3” for the ESFAS trip	Table 2.5.2-2 Table 2.5.2-6	Table 3.3.1-1, Function 5a Table 3.3.8-1, Function 5	<u>Chapter 3</u> Section 3.9.1.1.2.5 Section 3.9.1.1.2.9 Section 3.9.1.1.2.10 Section 3.9.1.1.3.5 <u>Chapter 5</u> Section 5.4.5.1 Section 5.4.5.2.3 Section 5.4.5.3.3 <u>Chapter 6</u> Section 6.2.3.3 Section 6.3.2.5 Section 6.3.3 <u>Chapter 7</u> Section 7.2.1.1.3 Section 7.2.1.1.4 Table 7.2-2 Table 7.2-3 Figure 7.2-1 (sheets 2,5) Section 7.3.1.1 Section 7.3.1.5.5 Table 7.3-1 Table 7.3-2 Section 7.4.1.1 WCAP-16675, Section 1.1 ...

PMS Parameter	COL Appendix C (and corresponding Tier 1 information)	COL Appendix A (TS)	Tier 2 Locations
... (continued) Pressurizer Pressure: a. “Low” changed to “Low-2” for the RTS trip b. “Low” changed to “Low-3” for the ESFAS trip			<u>Chapter 14</u> Table 14.3-2 <u>Chapter 15</u> Table 15.0-4a Table 15.0-6 Section 15.1.4.1 Section 15.1.5.1 Section 15.1.5.5.1 Section 15.1.6.1 Section 15.4.2.1 Section 15.6.1.1 Section 15.6.1.2.2 Section 15.6.1.3 Section 15.6.3.1.2 Section 15.6.3.2.1.2 Section 15.6.5.2 Section 15.6.5.2.2 Section 15.6.5.4B Section 15.6.5.4B.1 Table 15.6.1-1 <u>Chapter 19</u> Section 19E.4.7.3 Section 19E.4.8

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Pressurizer Pressure: “High” changed to “High-2”	Table 2.5.2-2	Table 3.3.1-1, Function 5b	<u>Chapter 5</u> Section 5.4.5.2.3 <u>Chapter 7</u> Section 7.2.1.1.4 Section 7.2.2.2.2 Table 7.2-2 Figure 7.2-1 (sheets 2,6) Section 7.7.1.6 WCAP-16675, Section 1.1 <u>Chapter 15</u> Table 15.0-4a Table 15.0-6 Section 15.2.2.1 Section 15.2.3.2.1 Section 15.2.3.2.2 Section 15.2.8.1 Section 15.2.8.2.1 Table 15.2-1 Section 15.4.2.1 Section 15.5.2.1 Section 15.5.2.2 Section 15.5.2.4

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Pressurizer Water Level: a. "High" changed to "High-3" for RTS b. "High" changed to "High-2" or "High-3" for ESFAS	Table 2.5.2-2 Table 2.5.2-6	N/A	Chapter 7 Section 7.2.2.2.2 Table 7.2-3 Table 7.3-2 WCAP-16675, Section 1.1 Chapter 9 Section 9.3.6.1.1 Section 9.3.6.4.5 Section 9.3.6.5 Chapter 15 Table 15.0-6 Section 15.2.2.1 Section 15.2.3.2.1 Section 15.2.8.2.1 Section 15.4.2.1 Section 15.5.1.2 Section 15.5.1.3 Section 15.5.2.3 Section 15.5.2.4

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Reactor Coolant Flow: “Low” changed to “Low-2”	Table 2.5.2-2 Table 2.5.2-6	Table 3.3.1-1, Function 7	<u>Chapter 3</u> Section 3.9.1.1.2.8 Section 3.9.1.1.4.4 <u>Chapter 7</u> Section 7.2.1.1.3 Section 7.2.1.2.5 Section 7.2.2.2.2 Table 7.2-2 Table 7.2-3 Figure 7.2-1 (sheets 2, 5) WCAP-16675, Section 1.1 <u>Chapter 15</u> Table 15.0-4a Table 15.0-6 Section 15.3.1.1 Section 15.3.1.2.5 Section 15.3.2.1 Section 15.3.2.2.1 Section 15.3.3.1 Section 15.3.4.1 Table 15.3-1 <u>Chapter 19</u> Section 19E.4.3.2 Section 19E.4.4.1

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Reactor Coolant Pump Speed: "Low" changed to "Low-2"	Table 2.5.2-2 Table 2.5.2-6	Table 3.3.1-1, Function 9	<u>Chapter 3</u> Section 3.9.1.1.3.3 <u>Chapter 7</u> Section 7.2.1.1.3 Section 7.2.1.2.5 Section 7.2.2.2.2 Table 7.2-2 Table 7.2-3 Figure 7.2-1 (sheets 2, 5) WCAP-16675, Section 1.1 <u>Chapter 15</u> Table 15.0-4a Table 15.0-6 Section 15.2.3.2.1 Section 15.2.3.2.2 Section 15.2.6.2.1 Table 15.2-1 Section 15.3.2.1 Section 15.3.2.2.1 Section 15.3.2.2.2 Table 15.3-1 <u>Chapter 19</u> Section 19E.4.3.2 Section 19E.4.4.1

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Steam Generator Water Level: a. "High" and/or "High-2" changed to "High-3" b. "High" changed to clarify two separate trips: (i) "High-3" and (ii) "Reactor Trip plus High Steam Generator Level" c. One occurrence of "High-2" corrected to "High"	Table 2.5.2-2 Table 2.5.2-6	Table 3.3.1-1, Function 11 Table 3.3.8-1, Function 23	<u>Chapter 6</u> Table 6.2.3-1 <u>Chapter 7</u> Section 7.2.1.1.6 Section 7.2.1.1.12 Table 7.2-2 Table 7.2-3 Figure 7.2-1 (sheets 2,6,10,14) Section 7.3.1.2.6 Section 7.3.1.2.8 Section 7.3.1.2.13 Section 7.3.1.2.15 Table 7.3-1 WCAP-16675, Section 1.1 <u>Chapter 9</u> Section 9.3.6.3.7 Section 9.3.6.4.5 Section 9.3.6.5 Section 9.3.6.7 <u>Chapter 10</u> Table 10.3.3-1 Section 10.4.9.3 ...

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
... (continued Steam Generator Water Level: a. "High" and/or "High-2" changed to "High-3" b. "High" changed to clarify two separate trips: (i) "High-3" and (ii) "Reactor Trip plus High Steam Generator Level" c. One occurrence of "High-2" corrected to "High"			<u>Chapter 14</u> Table 14.3-2 <u>Chapter 15</u> Table 15.0-4a Table 15.0-6 Section 15.1.2.1 Section 15.1.2.2.1 Section 15.1.2.2.2 Table 15.1.2-1 Section 15.6.3.1.1 Section 15.6.3.1.3 Table 15.6.3-1 <u>Chapter 19</u> Section 19E.4.2.1

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
CMT Water Level: a. “Low-1” (or “Low”) changed to “Low-3” b. “Low-2” (or “Low-Low”) changed to “Low-6”	N/A	Table 3.3.8-1, Functions 15 and 16	<u>Chapter 1</u> Section 1.9.3 item (1)(vii) <u>Chapter 7</u> Figure 7.2-1 (sheet 15) Section 7.3.1.2.4 Table 7.3-1 <u>Chapter 14</u> Table 14.3-2 <u>Chapter 15</u> Table 15.0-4a Table 15.6.5-10 <u>Chapter 19</u> Section 19E.2.3.2.3

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Steam Line Pressure: “Low” changed to “Low-2”	N/A	Table 3.3.8-1, Function 24	<u>Chapter 3</u> Section 3.9.1.1.4.3 <u>Chapter 6</u> Section 6.2.3.3 Table 6.2.3-1 Section 6.3.3 <u>Chapter 7</u> Figure 7.2-1 (sheet 9, 11) Section 7.3.1.1 Section 7.3.1.2.10 Section 7.3.1.2.24 Section 7.3.1.5.5 Table 7.3-1 Table 7.3-2 <u>Chapter 10</u> Section 10.3.2.2.3 Section 10.3.2.2.4 Table 10.3.3-1 <u>Chapter 14</u> Table 14.3-2 ...

PMS Parameter	COL Appendix C (and corresponding Tier 1 information)	COL Appendix A (TS)	Tier 2 Locations
... (continued) Steam Line Pressure: "Low" changed to "Low-2"			Chapter 15 Table 15.0-4a Table 15.0-6 Section 15.1.4.1 Section 15.1.5.1 Section 15.1.5.2.3 Section 15.1.5.5.1 Section 15.1.5.5.2.1 Section 15.1.5.5.3 Table 15.1.2.1 Section 15.2.8.1 Section 15.2.8.2.1 Section 15.2.8.2.2 Table 15.2-1 Section 15.5.2.1 Section 15.5.2.2 Section 15.5.2.4 Section 15.6.3.2.1.2 Section 15.6.3.2.1.3 Table 15.6.3-1 Chapter 19 Section 19E.2.2.2.2 Section 19E.2.3.2.1

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
RCS Hot Leg Water Level: a. "Low-1" (or "Low") changed to "Low-2" b. "Low-2" (or "Low" or "Low-Low") changed to "Low-4"	N/A	Table 3.3.10-1, Functions 1 and 2	<u>Chapter 6</u> Section 6.3.3.4.3 <u>Chapter 7</u> Figure 7.2-1 (sheet 15,16) Section 7.3.1.2.2 Section 7.3.1.2.4 Section 7.3.1.2.22 Table 7.3-1 Table 7.3-2 WCAP-15775, Sections 5.4.1, 5.4.9 and 5.5.6 <u>Chapter 9</u> Section 9.3.6.7 <u>Chapter 19</u> Table 19.59-18 Section 19E.2.1.2.2 Section 19E.2.1.2.4 Section 19E.2.3.2.3 Section 19E.2.3.3.1

PMS Parameter	COL Appendix C (and corresponding Tier 1 information)	COL Appendix A (TS)	Tier 2 Locations
RCS Cold Leg Temperature (T_{cold} and Reactor Coolant Inlet Temperature): "Low" changed to "Low-2"	N/A	Table 3.3.8-1, Function 11	Chapter 6 Section 6.2.1.4.1.3 Section 6.2.3.3 Table 6.2.3-1 Chapter 7 Section 7.3.1.1 Section 7.3.1.2.10 Section 7.3.1.2.13 Table 7.3-1 Table 7.3-2 Chapter 10 Section 10.3.2.2.4 Chapter 14 Table 14.3-2 Chapter 15 Table 15.0-4a Table 15.0-6 Section 15.1.4.1 Section 15.1.4.2.2 Section 15.1.5.1 Section 15.1.5.5.1 Table 15.1.2-1 Section 15.2.7.1 Section 15.2.7.2.2 ...

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
... (continued) T_{cold}: “Low” changed to “Low-2”			Table 15.2-1 Section 15.5.1.1 Section 15.5.1.3 Section 15.5.2.1 Section 15.5.2.2 Section 15.5.2.3 Section 15.5.2.4 Table 15.5-1 <u>Chapter 19</u> Section 19E.4.2.1 Section 19E.4.2.4 Section 19E.4.10.2 Table 19E.4.10-1

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
Low Startup Feedwater Flow: "Low" changed to "Low-2"	N/A	LCO 3.3.11	<u>Chapter 6</u> Section 6.3.3 <u>Chapter 7</u> Section 7.3.1.2.7 Table 7.3-1 <u>Chapter 10</u> Section 10.4.9.5 <u>Chapter 15</u> Table 15.0-6 Section 15.2.6.2.1 Section 15.2.7.1 Section 15.2.7.2.1 Table 15.2-1 <u>Chapter 19</u> Section 19E.4.10.2
TS change only: "Low 1" and "High 1" Function names changed to "Low" and "High" respectively	N/A	Table 3.3.8-1, Function 3 Table 3.3.8-1, Function 6 Table 3.3.8-1, Function 8 Table 3.3.8-1, Function 12	<u>N/A</u>

PMS Parameter	**COL Appendix C (and corresponding Tier 1 information)**	**COL Appendix A (TS)**	**Tier 2 Locations**
UFSAR change only: "Low" and "Low-1" setpoint designators changed to "Low[10]"; and "High" and "High-1" setpoint designators changed to "High[10]" Footnote 10 added stating: "Low and Low-1 are considered to be equivalent terms when referring to the first low setpoint designator. High and High-1 are considered to be equivalent terms when referring to the first high setpoint designator."	N/A	N/A	<u>Chapter 7</u> Table 7.3-1

3. TECHNICAL EVALUATION

No setpoint values or PMS alarms and actuations are proposed to be changed by this activity.

Some proposed changes are made to consistently refer to a given setpoint with the same designator throughout the licensing basis. For example, in some instances the licensing basis may reference the setpoint with its specific setpoint designator (e.g., Low-3 or High-2), but in other instances may simply use the more general reference "low" or "high." In other instances, changes are proposed to standardize the setpoint designator format. These changes do not impact any setpoint values, but provide clarity to the licensing basis so that it is clear which setpoint designator is used for a given PMS engineered safety feature actuation or reactor trip function.

Some proposed changes are made to avoid using a single setpoint designator for more than one setpoint value. These changes are considered to reduce human related errors. Even when a change to the setpoint designator is proposed in this activity (e.g., High-2 to High-3), the actual PMS setpoint is not changing. Nor does this activity change the setpoint values assumed in the safety analysis.

The proposed changes do not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. Beyond the PMS changes proposed, no safety-related structure, system, component (SSC) or function is involved. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated in the plant-specific DCD or UFSAR. The proposed changes do not affect the radiological source terms (i.e., amounts and types of radioactive materials released, their release rates and release durations) used in the accident analyses. No system or design function or equipment qualification is adversely affected by the proposed changes. The changes do not result in a new failure mode, malfunction or sequence of events that could adversely affect a radioactive material barrier or safety-related equipment. The proposed changes do not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in significant fuel cladding failures. The proposed changes do not adversely affect any design code limit allowable value, design analysis, nor do they adversely affect any safety analysis input or result, or design/safety margin.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from the terms and conditions of a Combined License (COL). This activity involves a change to COL Appendix A, Technical Specifications as well as a departure from COL Appendix C (and corresponding plant-specific Tier 1) Inspections, Tests, Analyses and Acceptance Criteria (ITAAC) information; therefore, this activity requires a proposed amendment to the COL. Accordingly, NRC approval is required prior to making the plant-specific changes in this license amendment request.

10 CFR 52, Appendix D, Section VIII.B.5.a allows an applicant or licensee who references this appendix to depart from Tier 2 information, without prior NRC approval,

unless the proposed departure involves a change to or departure from Tier 1 information, Tier 2* information, or the Technical Specifications, or requires a license amendment under paragraphs B.5.b or B.5.c of the section. The proposed changes request a revision to COL Appendix C (and plant-specific DCD Tier 1) and COL Appendix A, Technical Specifications, and thus require prior NRC approval.

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. 10 CFR 52, Appendix D, VIII.A.4 requires a Tier 1 change shall not result in a significant decrease in the level of safety otherwise provided by the design. The proposed changes and their associated Tier 1 information change do not adversely affect any safety-related SSC, function, design analysis or safety analysis. Therefore, the requested changes will not result in a decrease in the level of safety otherwise provided by the design.

10 CFR 52, Appendix D, Section VIII.C.6 states that after issuance of a license, “Changes to the plant-specific TS (Technical Specifications) will be treated as license amendments under 10 CFR 50.90.” 10 CFR 50.90 addresses the applications for amendments of licenses, construction permits, and early site permits. As discussed above, a change to COL Appendix A is requested, and thus a LAR (as supplied herein) is required.

The proposed changes have been evaluated to determine whether applicable regulations continue to be met. It was determined that the proposed changes do not affect conformance with the General Design Criteria (GDC) differently than described in the plant-specific DCD or UFSAR.

4.2 Precedent

None.

4.3 Significant Hazards Consideration Determination

The proposed changes would standardize setpoint nomenclature for Protection and Safety Monitoring System (PMS) functions. The Updated Final Safety Analysis Report (UFSAR) and Combined License (COL) Appendix A (Technical Specifications) and Appendix C (and corresponding plant-specific Tier 1 information) are not consistent when referring to PMS setpoint designators. For example, a given setpoint may be referred to with its specific designator, such as Low-3, but in other instances it is referred to with the generic “low.” Therefore, some generic low and high setpoint designator references were revised to the specific designator to avoid any confusion as to which specific setpoint designator is used for a given PMS function. In addition, the format used for the setpoint designator is revised for consistency (i.e., "Low-Low" vs. "Low-2").

In some instances a given setpoint designator is used for more than one setpoint. For example, the High-2 Steam Generator water level is used to reference both an alarm and the PMS reactor trip function even though they are actuated as different setpoint values. Therefore, the proposed changes provide each setpoint its own setpoint designator to avoid confusion and to address the human factors issues related to labeling different setpoints with the same setpoint designator (i.e., "Low-Low" vs. "Low-2").

The requested amendment proposes changes to UFSAR Tier 2 information. The changes involve Technical Specifications (TS), COL Appendix A information, and Appendix C (and corresponding plant-specific Tier 1) information.

An evaluation to determine whether a significant hazards consideration is involved with the requested amendment was completed by focusing on the three standards set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below:

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

Response: No

No setpoint values or PMS actuations are proposed to be changed by this activity. Nor are any values assumed in the safety analysis changed. This is an administrative change to standardize the PMS setpoint designators. The proposed amendment does not affect the prevention and mitigation of abnormal events, e.g., accidents, anticipated operation occurrences, earthquakes, floods, turbine missiles, and fires or their safety or design analyses. This change does not involve containment of radioactive isotopes or any adverse effect on a fission product barrier. There is no impact on previously evaluated accidents.

These proposed changes have no adverse impact on the support, design, or operation of mechanical and fluid systems. The response of systems to postulated accident conditions is not adversely affected and remains within response time assumed in the accident analysis. There is no change to the predicted radioactive releases due to normal operation or postulated accident conditions. Consequently, the plant response to previously evaluated accidents or external events is not adversely affected, nor does the proposed change create any new accident precursors.

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

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

Response: No

The proposed changes do not involve a new failure mechanism or malfunction, which affects an SSC accident initiator, or interface with any SSC accident initiator or initiating sequence of events considered in the design and licensing bases. There is no adverse effect on radioisotope barriers or the release of radioactive materials. The proposed amendment does not adversely affect any accident, including the possibility of creating a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed changes do not create the possibility of a new or different type of accident from any accident previously evaluated.

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

Response: No

No setpoint values or PMS actuations are proposed to be changed by this activity. This is an administrative change to standardize the PMS setpoint designators. The proposed changes would not affect any safety-related design code, function, design analysis, safety analysis input or result, or existing design/safety margin. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested changes.

Therefore the proposed amendment does not involve a significant reduction in a margin of safety

Based on the above, it is concluded that the proposed amendment does not involve a 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.

4.4 Conclusions

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. Therefore, it is concluded that the requested amendment does not involve a 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.

5. ENVIRONMENTAL CONSIDERATIONS

Details of the proposed changes are provided in Sections 2 and 3 of this license amendment request.

The proposed amendment would standardize setpoint nomenclature to match the PMS functional logic drawings for RTS and ESFAS functions.

A review has determined that facility construction and operation following implementation of the requested 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 requested amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9), in that:

(i) *There is no significant hazards consideration.*

As documented in Section 4.3, Significant Hazards Consideration Determination, of this license amendment request, an evaluation was completed to determine whether or not a significant hazards consideration is involved by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment." The Significant Hazards Consideration determined that (1) the requested amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) the requested amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) the requested amendment does not involve a significant reduction in a margin of safety. Therefore, it is concluded that the requested amendment does not involve a 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.

(ii) *There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.*

The proposed amendment standardizes the PMS setpoint designators. The changes are unrelated to any aspects of plant construction or operation that would introduce any changes to effluent types (e.g., effluents containing chemicals or biocides, sanitary system effluents, and other effluents) or affect any plant radiological or non-radiological effluent release quantities. Furthermore, the proposed change does not diminish the functionality of any design or operational features that are credited with controlling the release of effluents during plant operation. Therefore, it is concluded that the proposed amendment does not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite.

(iii) *There is no significant increase in individual or cumulative occupational radiation exposure.*

The proposed amendment standardizes setpoint nomenclature. The proposed change does not affect walls, floors, or other structures that provide shielding. Plant radiation zones are not affected, and there are no changes to the controls required under 10 CFR Part 20 that preclude a significant increase in occupational radiation exposure. Therefore, the requested amendment does not involve a significant increase in individual or cumulative occupational radiation exposure.

Based on the above review of the requested amendment, it has been determined that anticipated construction and operational impacts of the requested amendment do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the requested amendment meets the eligibility criteria 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 requested amendment and proposed exemption.

6. REFERENCES

None.

Southern Nuclear Operating Company

ND-17-0295

Enclosure 2

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Exemption Request:

Standardization of Instrumentation Setpoint Nomenclature

(LAR-17-004)

(Enclosure 2 consists of 7 pages, including this cover page)

1.0 Purpose

Southern Nuclear Operating Company (SNC, or the “Licensee”) requests a permanent exemption from the provisions of 10 CFR 52, Appendix D, Section III.B, *Design Certification Rule for the AP1000 Design, Scope and Contents*, to allow a departure from elements of the certification information in Tier 1 of the generic AP1000 Design Control Document (DCD). The regulation, 10 CFR 52, Appendix D, Section III.B, requires an applicant or licensee referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in DCD Tier 1. The Tier 1 information for which a plant-specific departure and exemption is being requested includes standardizing setpoint nomenclature for Protection and Safety Monitoring System (PMS) functions.

This request for exemption applies to the requirements of 10 CFR 52, Appendix D, Section VIII.A.4 to allow departures from plant-specific DCD Tier 1 information due to the following proposed changes to ITAAC Tables:

- Tier 1 Table 2.5.2-2, PMS Automatic Reactor Trips:
    - Reactor Coolant Pump High Bearing Water Temperature Trip: Change “High” to “High-2”
    - Pressurizer Low Pressure Trip: Change “Low” to “Low-2”
    - Pressurizer High Pressure Trip: Change “High” to “High-2”
    - Pressurizer High Water Level Trip: Change “High” to “High-3”
    - Low Reactor Coolant Flow Trip: Change “Low” to “Low-2”
    - Low Reactor Coolant Pump Speed Trip: Change “Low” to “Low-2”
    - High-2 Steam Generator Water Level Trip: Change “High-2” to “High-3”
- Tier 1 Table 2.5.2-6, PMS Blocks, Reactor Trip Functions:
    - Pressurizer Low Pressure Trip: Change “Low” to “Low-2”
    - Pressurizer High Water Level Trip: Change “High” to “High-3”
    - Low Reactor Coolant Flow Trip: Change “Low” to “Low-2”
    - Low Reactor Coolant Pump Speed Trip: Change “Low” to “Low-2”
    - High Steam Generator Water Level Trip: Change “High” to “High-3”

This request will apply the requirements for granting exemptions from design certification information, as specified in 10 CFR 52, Appendix D, Section VIII.A.4, 10 CFR 52.63, 10 CFR 52.7, and 10 CFR 50.12.

2.0 Background

The Licensee is the holder of Combined License Nos. NPF-91 and NPF-92, which authorize construction and operation of two Westinghouse Electric Company AP1000 nuclear plants, named Vogtle Electric Generating Plant (VEGP) Units 3 and 4, respectively.

The PMS provides detection of off-nominal conditions and actuation of appropriate safety-related functions necessary to achieve and maintain the plant in a safe shutdown condition. The PMS initiates reactor trips and ESFAS functions when plant conditions reach specified setpoints. It has four divisions of reactor trip and ESFAS actuation, and two divisions of safety-related post-accident parameter displays. The PMS controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

The AP1000 digital Instrumentation & Control (I&C) systems, such as the PMS and the Plant Control System (PLS), use various parameters and setpoints to control and protect the plant and to provide alarms to plant personnel. Specific values are used for each setpoint, which depend on the specific plant function that uses the given parameter. In addition to specific values, the setpoints are given a generic designator or name. The exact setpoint value may change, but the generic designator for the setpoint generally remains unchanged. For example, the reactor coolant pump is tripped on a Low-2 pressurizer level. “Low-2” is the generic designator for the setpoint. The actual pressurizer level value for “Low-2” may change, but would remain named “Low-2.” In some instances, the design includes a higher level setpoint, such as a “Low-1,” to provide a warning alarm to the operator prior to the system reaching the “Low-2” actuation setpoint.

The existing UFSAR and Combined License (COL) Appendix A (Technical Specifications) and Appendix C (and corresponding plant-specific Tier 1 information) are not consistent when referring to PMS setpoint designators. For example, a given setpoint may be referred to with its specific designator, such as Low-3, but in other instances it is referred to with the generic “low.” Therefore, it is necessary to revise some generic “low” and “high” references which are currently inclusive of multiple generic designators to avoid any confusion as to which specific setpoint designator is used for a given PMS function. In addition, the format used for the setpoint designator is inconsistent (i.e., "Low-Low" vs. "Low-2").

In some instances a given setpoint designator is used for more than one setpoint. For example, the High-2 steam generator water level designator is used for both an alarm and the PMS reactor trip function, even though they are actuated at different setpoint values. To avoid confusion and address the human factor issues related to labeling different setpoints with the same setpoint designator, a given setpoint is referred to with its own unique setpoint designator.

3.0 Technical Justification of Acceptability

No setpoint values or PMS actuations are proposed to be changed by this activity.

Some proposed changes are made to consistently refer to a given setpoint with the same designator throughout the licensing basis. For example, in some instances the licensing basis may reference the setpoint with its specific setpoint designator (e.g., Low-3 or High-2), but in other instances may simply use the more general reference “low” or “high.” In other instances, changes are proposed to standardize the setpoint designator format. These changes do not impact any setpoint, but provide clarity to the licensing basis so that it is clear which setpoint designator is used for a given PMS engineered safety feature actuation or reactor trip function.

Some proposed changes are made to avoid using a single setpoint designator for more than one setpoint. These changes are considered to be human factor safety improvements. Even though a change to the setpoint designator is proposed in this activity (e.g., High-2 to High-3), the actual PMS setpoint is not changing. Nor does this activity change the setpoint values assumed in the safety analysis.

Additional detail for supporting the Technical Justification of this exemption is provided in Enclosure 1 of the accompanying license amendment request.

4.0 Justification of Exemption

10 CFR Part 52, Appendix D, Section VIII.A.4 and 10 CFR 52.63(b)(1) govern the issuance of exemptions from elements of the certified design information for AP1000 nuclear power plants. The Licensee has identified changes to plant-specific Tier 1 information (as described in Section 1.0), and as a result, an exemption from the certified design information in Tier 1 is requested.

10 CFR Part 52, Appendix D, and 10 CFR 50.12, §52.7, and §52.63 state that the NRC may grant exemptions from the requirements of the regulations provided six conditions are met: 1) the exemption is authorized by law [§50.12(a)(1)]; 2) the exemption will not present an undue risk to the health and safety of the public [§50.12(a)(1)]; 3) the exemption is consistent with the common defense and security [§50.12(a)(1)]; 4) special circumstances are present [§50.12(a)(2)]; 5) the special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption [§52.63(b)(1)]; and 6) the design change will not result in a significant decrease in the level of safety [Part 52, App. D, VIII.A.4].

The requested exemption satisfies the criteria for granting specific exemptions, as described below.

4.1 This exemption is authorized by law

The NRC has authority under 10 CFR 52.63, §52.7, and §50.12 to grant exemptions from the requirements of NRC regulations. Specifically, 10 CFR 50.12 and §52.7 state that the NRC may grant exemptions from the requirements of 10 CFR Part 52 upon a proper showing. No law exists that would preclude the changes covered by this exemption request. Additionally, granting of the proposed exemption does not result in a violation of the Atomic Energy Act of 1954, as amended, or the Commission’s regulations.

Accordingly, this requested exemption is “authorized by law,” as required by 10 CFR 50.12(a)(1).

4.2 This exemption will not present an undue risk to the health and safety of the public

The proposed exemption from the requirements of 10 CFR 52, Appendix D, Section III.B, would allow changes to elements of the plant-specific Tier 1 DCD to depart from the AP1000 certified (Tier 1) design information. The plant-specific DCD Tier 1 will continue to reflect the approved licensing basis for VEGP Units 3 and 4, and will maintain a consistent level of detail with that which is currently provided elsewhere in Tier 1 of the DCD. Therefore, the affected plant-specific DCD Tier 1 ITAAC will continue to serve its required purpose.

The request to standardize setpoint nomenclature for PMS functions does not represent any adverse impact to the design function of PMS and will continue to protect the health and safety of the public in the same manner. The administrative nomenclature standardization does not introduce any new industrial, chemical, or radiological hazards that would represent a public health or safety risk, nor do they modify or remove any design or operational controls or safeguards intended to mitigate any existing on-site hazards. Furthermore, the proposed change would not allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that would result in fuel cladding failures. Accordingly, this change does not present an undue risk from any existing or proposed equipment or systems.

Therefore, the requested exemption from 10 CFR 52, Appendix D, Section III.B would not present an undue risk to the health and safety of the public.

4.3 The exemption is consistent with the common defense and security

The requested exemption from the requirements of 10 CFR 52, Appendix D, Section III.B, would allow the licensee to depart from elements of the plant specific DCD Tier 1 design information. The proposed exemption does not alter the design, function, or operation of any structures or plant equipment that is necessary to maintain a safe and secure status of the plant. The proposed exemption has no impact on plant security or safeguards procedures.

Therefore, the requested exemption is consistent with the common defense and security.

4.4 Special circumstances are present

10 CFR 50.12(a)(2) lists six “special circumstances” for which an exemption may be granted. Pursuant to the regulation, it is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request. The requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstances as when “[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

The rule under consideration in this request for exemption is 10 CFR 52, Appendix D, Section III.B, which requires that a licensee referencing the AP1000 Design Certification Rule (10 CFR Part 52, Appendix D) shall incorporate by reference and comply with the requirements of Appendix D, including Tier 1 information. The VEGP Units 3 and 4 COLs reference the AP1000 Design Certification Rule and incorporate by reference the requirements of 10 CFR Part 52, Appendix D, including Tier 1 information. The underlying purpose of Appendix D, Section III.B is to describe and define the scope and contents of the AP1000 design certification, and to require compliance with the design certification information in Appendix D.

The proposed exemption would revise Tier 1 tables related to standardizing setpoint nomenclature for PMS functions, consistent with the changes proposed in the associated license amendment request.

The proposed administrative revision for the PMS setpoint nomenclature maintains the required design functions. The proposed changes do not affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. The proposed changes do not involve nor interface with any SSC accident initiator or initiating sequence of events related to the accidents evaluated and therefore do not have an adverse effect on any SSC's design function. Accordingly, this exemption from the certification information will enable the Licensee to safely construct and operate the AP1000 facility consistent with the design certified by the NRC in 10 CFR 52, Appendix D.

Therefore, special circumstances are present, because application of the current generic certified design information in Tier 1 as required by 10 CFR Part 52, Appendix D, Section III.B, in the particular circumstances discussed in this request is not necessary to achieve the underlying purpose of the rule.

4.5 The special circumstances outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

Based on the administrative nature of the changes to the plant-specific Tier 1 information and the understanding that these changes support the design function of PMS, it is expected that this exemption may be requested by other AP1000 licensees and applicants. However, a review of the reduction in standardization resulting from the departure from the standard DCD determined that even if other AP1000 licensees and applicants do not request this same departure, the special circumstances will continue to outweigh any decrease in safety from the reduction in standardization because the key design functions of the components associated with this request will continue to be maintained. Furthermore, the justification provided in the license amendment request and this exemption request and the associated mark-ups demonstrate that there is a limited change from the standard information provided in the generic AP1000 DCD, which is offset by the special circumstances identified above.

Therefore, the special circumstances associated with the requested exemption outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption.

4.6 The design change will not result in a significant decrease in the level of safety.

The exemption revises the plant-specific DCD Tier 1 information by revising the PMS setpoint nomenclature as discussed in Section 1.0. The revision of PMS setpoint nomenclature does not change the design requirements. Because these functions continue to be met, there is no reduction in the level of safety.

5.0 Risk Assessment

A risk assessment was not determined to be applicable to address the acceptability of this proposal.

6.0 Precedent Exemptions

None

7.0 Environmental Consideration

The Licensee requests a departure from elements of the certified information in Tier 1 of the generic AP1000 DCD. The Licensee has determined that the proposed departure would require a permanent exemption from the requirements of 10 CFR 52, Appendix D, Section III.B, Design Certification Rule for the AP1000 Design, Scope and Contents, with respect to installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, or which changes an inspection or a surveillance requirement; however, the Licensee evaluation of the proposed exemption has determined that the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Based on the above review of the proposed exemption, the Licensee has determined that the proposed activity does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment of the proposed exemption is not required.

Specific details of the environmental considerations supporting this request for exemption are provided in Section 5 of the associated License Amendment Request provided in Enclosure 1 of this letter.

8.0 Conclusion

The proposed changes to Tier 1 are necessary to revise the PMS setpoint nomenclature as discussed in Section 1.0. The exemption request meets the requirements of 10 CFR 52.63, *Finality of design certifications*, 10 CFR 52.7, *Specific exemptions*, 10 CFR 50.12, *Specific exemptions*, and 10 CFR 52 Appendix D, *Design Certification Rule for the AP1000*. Specifically, the exemption request meets the criteria of 10 CFR 50.12(a)(1) in that the request is authorized by law, presents no undue risk to public health and safety, and is consistent with the common defense and security as well as providing the special circumstances criteria of 10 CFR 50.12(a)(2)(ii). Furthermore, approval of this request does not result in a significant decrease in the level of safety, satisfies the underlying purpose of the AP1000 Design Certification Rule, and does not present a significant decrease in safety as a result of a reduction in standardization.

9.0 References

None

Southern Nuclear Operating Company

ND-17-0295

Enclosure 3

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Proposed Changes to the Licensing Basis Documents

(LAR-17-004)

Note:

Added text is shown as Blue Underline

Deleted text is shown as ~~Red Strikethrough~~

Omitted text is shown as three asterisks (* * *)

(Enclosure 3 consists of 91 pages, including this cover page)

Proposed Tier 1 and COL Appendix C Changes

Table 2.5.2-2 PMS Automatic Reactor Trips
* * * Reactor Coolant Pump ~~High~~ High-2 Bearing Water Temperature Trip * * * Pressurizer ~~Low~~ Low-2 Pressure Trip Pressurizer ~~High~~ High-2 Pressure Trip Pressurizer ~~High~~ High-3 Water Level Trip ~~Low~~ Low-2 Reactor Coolant Flow Trip ~~Low~~ Low-2 Reactor Coolant Pump Speed Trip * * * ~~High-2~~High-3 Steam Generator Water Level Trip * * *

Table 2.5.2-6 PMS Blocks
Reactor Trip Functions: * * * Pressurizer ~~Low~~ Low-2 Pressure Trip Pressurizer ~~High~~ High-3 Water Level Trip ~~Low~~ Low-2 Reactor Coolant Flow Trip ~~Low~~ Low-2 Reactor Coolant Pump Speed Trip ~~High~~ High-3 Steam Generator Water Level Trip * * *

Proposed Technical Specification Changes

Table 3.3.1-1 (page 1 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
* * *				
5. Pressurizer Pressure				
a. ~~Low~~ Low 2 Setpoint	* * *	* * *	* * *	* * *
b. ~~High~~High 2 Setpoint	* * *	* * *	* * *	* * *
* * *				

Table 3.3.1-1 (page 2 of 2)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS
7. Reactor Coolant Flow – ~~Low~~Low 2	* * *	* * *	* * *	* * *
8. Reactor Coolant Pump (RCP) Bearing Water Temperature – ~~High~~High 2	* * *	* * *	* * *	* * *
9. RCP Speed – ~~Low~~Low 2	* * *	* * *	* * *	* * *
* * *				
11. Steam Generator (SG) Narrow Range Water Level – ~~High 2~~High 3	* * *	* * *	* * *	* * *
* * *				

Proposed Technical Specification Changes (continued)

Table 3.3.8-1 (page 1 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
* * *			
3. Containment Radioactivity – ~~High 1~~ High	* * *	* * *	* * *
* * *			
5. Pressurizer Pressure – ~~Low~~ Low 3	* * *	* * *	* * *
6. Pressurizer Water Level – ~~Low 1~~ Low	* * *	* * *	* * *
* * *			
8. Pressurizer Water Level – ~~High 1~~ High	* * *	* * *	* * *
* * *			
11. RCS Cold Leg Temperature (T_{cold}) – ~~Low~~ Low 2	* * *	* * *	* * *
12. Reactor Coolant Average Temperature (T_{avg}) – ~~Low 1~~ Low	* * *	* * *	* * *
* * *			

Table 3.3.8-1 (page 2 of 2)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
15. Core Makeup Tank (CMT) Level – ~~Low 1~~ Low 3	* * *	* * *	* * *
* * *	* * *	* * *	* * *
16. CMT Level – ~~Low 2~~ Low 6	* * *	* * *	* * *
* * *	* * *	* * *	* * *
19. Reactor Coolant Pump Bearing Water Temperature – ~~High~~ High 2	* * *	* * *	* * *
* * *			
23. SG Narrow Range Water Level – ~~High 2~~ High 3	* * *	* * *	* * *
	* * *	* * *	* * *
24. Steam Line Pressure – ~~Low~~ Low 2	* * *	* * *	* * *
* * *			

Proposed Technical Specification Changes (continued)

Table 3.3.10-1 (page 1 of 1)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS
1. Hot Leg Level – ~~Low 2~~Low 4	4[(a)],5	1 per loop	C
	6[(b)]	1 per loop	D
2. Hot Leg Level – ~~Low 1~~Low 2	4[(a)(c)],5[(c)]	1 per loop	E
	6[(c)(d)]	1 per loop	F

~~~~~~~~~~~~~~~~~~~~

LCO 3.3.11 Two channels of ESFAS Startup Feedwater Flow – Low 2 instrumentation for each startup feedwater line shall be OPERABLE.

~~~~~~~~~~~~~~~~~~~~

Proposed Tier 2 and UFSAR Departures

1.2.1.2.3 Reactor Coolant Pump Design

* * *

The reactor coolant pumps are designed such that they are not damaged due to a loss of all cooling water until a safety-related pump trip occurs on ~~high~~High-2 bearing water temperature. This automatic protection is provided to protect the reactor coolant pumps from an extended loss of coolant water.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

1.9.3 Three Mile Island Issues

* * *

(1)(vii) Automatic Depressurization System Activation (NUREG-0737 Item II.K.3.18)

* * *

AP1000 Response:

* * *

The automatic depressurization system actuates on ~~Low-1~~Low-3 core makeup tank level, coincident with a core makeup tank actuation signal. Therefore * * *

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.2.5 Control Rod Drop

* * *

Control Rod Drop - Case A

* * *

The steam load-reactor power mismatch causes the plant to cool down, eventually leading to a reactor trip on ~~low~~Low-2 pressurizer pressure. Following the reactor trip, * * *

* * *

Control Rod Drop - Case B

* * *

The steam load-reactor power mismatch causes the plant to cool down. With a zero moderator temperature coefficient of reactivity, no reactor power recovery occurs. Plant cooldown continues, causing a reactor trip due to ~~low~~Low-2 pressurizer pressure, which is then followed by turbine trip. * * *

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.2.8 Partial Loss of Reactor Coolant Flow

This transient applies to a partial loss of flow from full power in which a reactor coolant pump is tripped out of service as a result of loss of power to that pump. The consequences of such an accident are a reactor trip on ~~low~~Low-2 reactor coolant flow, followed by a turbine trip; actuation of startup * * *

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.2.9 Inadvertent Reactor Coolant System Depressurization - Umbrella Case

* * *

An inadvertent auxiliary spray occurs if the auxiliary spray valve is opened during normal plant operation because of either failure of a control component or operator error. The opening of the auxiliary spray valve causes an inadvertent spray transient only during the limited time that the makeup pump in the chemical volume and control system is operating. The inadvertent auxiliary spray introduces cold water into the pressurizer, which results in a sharp pressure decrease and, eventually, in a ~~low~~ Low-2 pressure reactor trip.

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.2.10 Excessive Feedwater Flow

* * *

The passive safety injection system is actuated on a ~~low~~ Low-3 pressurizer pressure signal. Main feedwater flow is effectively isolated on the safety injection signal.

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.3.3 Complete Loss of Flow

* * *

This accident involves a complete loss of flow from full power resulting from the simultaneous loss of power to all reactor coolant pumps. The consequences are a reactor trip on ~~low~~ Low-2 pump speed, followed by an automatic turbine trip.

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.3.5 Steam Generator Tube Rupture

This transient is postulated as the double-ended rupture of a single steam generator tube, which results in decreases in pressurizer level and reactor coolant pressure. Assuming no operator action, the reactor eventually trips on overtemperature ΔT or ~~low~~ Low-2 pressurizer pressure.
* * *

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.4.3 Large Feedwater Line Break

* * *

In the analysis, no credit is taken for operation of pressure control systems, steam dump, or steam generator power-operated relief valves. The intact steam generator feeds the break through the main steam header after the faulted steam generator discharges its liquid inventory. Steam flow continues until the main steam lines are isolated on ~~low~~ Low-2 steam line pressure.

~~~~~~~~~~~~~~~~~~~~

3.9.1.1.4.4 Reactor Coolant Pump Locked Rotor

This accident is based on the seizure of the rotating assembly of a reactor coolant pump rotor, with the plant operating at full power. Reactor trip occurs rapidly, as the result of ~~low~~ Low-2 coolant flow in the affected cold leg. * * *

* * *

~~~~~~~~~~~~~~~~~~~~

5.4.1.3.4 Coastdown Capability

* * *

A loss of component cooling water has no impact on coastdown capability. The reactor coolant pump can operate without cooling water until a safety-related pump trip occurs on ~~high~~ High-2 bearing water temperature. This prevents damage that could potentially affect coastdown.

~~~~~~~~~~~~~~~~~~~~

5.4.1.3.5 Bearing Integrity

* * *

The bearing cooling provisions include a temperature monitoring system. The system operates continuously and has at least four redundant indicators per pump. Upon initiation of failure, the system indicates and alarms in the control room as a high bearing water temperature. All of the pumps trip when the ~~high~~ High-2 temperature setpoint is reached.

~~~~~~~~~~~~~~~~~~~~

5.4.5.1 Design Bases

* * *

- A ~~low-~~Low-3 pressurizer pressure engineered safety features actuation signal will not be activated because of a reactor trip and turbine trip, assuming normal operation of control and makeup systems and no failures of the nuclear steam supply systems

~~~~~~~~~~~~~~~~~~~~

5.4.5.2.3 Operation

* * *

During an outsurge of water from the pressurizer, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the ~~low-~~Low-3-pressure engineered safety features actuation setpoint. During an in-surge from the reactor coolant system, the spray system (which is fed from two cold legs) condenses steam in the pressurizer. This prevents the pressurizer pressure from reaching the ~~high~~ High-2-pressure reactor trip setpoint. * * *

~~~~~~~~~~~~~~~~~~~~

5.4.5.3.3 Pressure Setpoints

* * *

The ~~low-~~Low-3 pressurizer pressure engineered safety features actuation signal does not require a coincident low pressurizer water level signal.

~~~~~~~~~~~~~~~~~~~~

6.2.1.4.1.3 Startup Feedwater System Design

The effects on the steam generator mass are maximized in the calculation described in Subsection 6.2.1.4.3.2 by assuming full startup feedwater flow to the faulted steam generator starting at time zero from the safeguard system(s) signal and continuing until automatically terminated on a ~~low~~ Low-2 RCS Tcold signal.

~~~~~~

6.2.3.3 Design Evaluation

A. Engineered safeguards and containment isolation signals automatically isolate * * *

* * *

- ~~Low~~ Low-3 pressurizer pressure
- ~~Low~~ Low-2 steam-line pressure
- ~~Low~~ Low-2 T_{cold}

* * *

The component cooling water lines penetrating * * * The safeguards actuation signal is generated by any of the following conditions.

- ~~Low~~ Low-3 pressurizer pressure
- ~~Low~~ Low-2 steam line pressure
- ~~Low~~ Low-2 reactor coolant inlet temperature

* * *

B. Upon failure of a main steam line, the steam generators * * *

The two redundant train-oriented steam-line isolation signals are initiated upon receipt of any of the following signals:

- ~~Low~~ Low-2 steam-line pressure

* * *

- ~~Low~~ Low-2 T_{cold}

~~~~~~

Table 6.2.3-1 (Sheet 1 of 4)
Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration			Isolation Device						Test		
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
CAS		* * *		CAS-PL-V204 CAS-PL-V205		* * *		None None			* * *	
		* * *		CAS-PL-V014 CAS-PL-V015		* * *		T None			* * *	
CCS		* * *		CCS-PL-V200 CCS-PL-V201		* * *		S, ~~HRCP~~ HRCP2 None			* * *	
		* * *		CCS-PL-V208 CCS-PL-V207		* * *		S, ~~HRCP~~ HRCP2 S, ~~HRCP~~ HRCP2			* * *	
		* * *		CCS-PL-V220		* * *		None			* * *	
CVS		* * *		CVS-PL-V041 CVS-PL-V040 CVS-PL-V042		* * *		None None None			* * *	
		* * *				* * *					* * *	
		* * *		CVS-PL-V047 CVS-PL-V045 CVS-PL-V058		* * *		T T None			* * *	
		* * *		CVS-PL-V090 CVS-PL-V091 CVS-PL-V100		* * *		HR, PL2, S+PL1, ~~SGL~~ SGL3, R+SGL HR, PL2, S+PL1, ~~SGL~~ SGL3, R+SGL None			* * *	

* * *

Table 6.2.3-1 (Sheet 2 of 4)
Containment Mechanical Penetrations and Isolation Valves

System	Containment Penetration			Isolation Device						Test		
	Line	Flow	Closed Sys IRC	Valve/Hatch Identification	Pipe Length	DCD Subsection	Position N-S-A	Signal	Closure Times	Type[1] & Note	Medium	Direction
* * *												
SGS	* * *			SGS-PL-V040A SGS-PL-V027A(7) SGS-PL-V030A,31A,32A,33A,34A,35A SGS-PL-V036A SGS-PL-V240A	* * *			MS ~~LSL~~ None MS MS	* * *	* * *		
				SGS-PL-V040B SGS-PL-V027B(7) SGS-PL-V030B,31B,32B,33B,34B,35B SGS-PL-V036B SGS-PL-V240B				MS ~~LSL~~ None MS MS		* * *		
	* * *											
	* * *							~~LTC, SGL~~ ~~LTC, SGL~~		* * *		

LSL2

LTC2

SGL3, R+SGL

Table 6.2.3-1 (Sheet 4 of 4)
Containment Mechanical Penetrations and Isolation Valves

Explanation of Heading and Acronyms for Table 6.2.3-1

* * *

Signal: Device closure signal * * *

* * *

LSL2: Low-2 ~~LSL:~~ ~~Low~~ steam line pressure

MF: Main feedwater isolation

LTC2: Low-2 ~~LTC:~~ ~~Low~~ T_{cold}

* * *: * * *

R: Reactor Trip signal

HR: High containment radiation

* * * * * *

R+SGL: Reactor trip plus High steam generator level

SGL3: High-3 ~~SGL:~~ ~~High~~ steam generator level

HRCP2: High-2 ~~HRCP:~~ ~~High~~ reactor coolant pump bearing water temperature trip

* * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

6.3.2.5 System Reliability

* * *

The initiating signals for the passive core cooling system are derived from independent sources as measured from process parameters (pressurizer ~~low~~ Low-3 pressure) or environmental (containment high pressure) variables. Redundant, as well as functionally independent variables, are measured to initiate passive core cooling system operation.

~~~~~~~~~~~~~~~~~~~~

6.3.3 Performance Evaluation

* * *

D. Shutdown Events (Chapter 19)

* * *

The events listed in groups A and B are non-LOCA events where the primary protection is provided by the passive core cooling system passive residual heat removal heat exchanger. For these events, the passive residual heat removal heat exchanger is actuated by the protection and monitoring system for the following conditions:

- Steam generator low narrow range level, coincident with startup feedwater ~~low~~ Low-2 flow

* * *

The events listed in group C above are events involving the loss of reactor coolant where the primary protection is by the core makeup tanks and accumulators. For these events the core makeup tanks are actuated by the protection and monitoring system for the following conditions:

- Pressurizer ~~low~~ Low-3 pressure
- * * *
- Steam line ~~low~~ Low-2 pressure

~~~~~~~~~~~~~~~~~~~~

6.3.3.4.3 Loss of Normal Residual Heat Removal Cooling During Reduced Inventory

* * *

* * * four paths are required to be operable in these conditions. The stage four valves are automatically opened by a signal from the protection and monitoring system on a ~~low-~~Low-4 hot leg level signal following a time delay.

The in-containment refueling water storage tank injection squib valves automatically open via the same ~~low-~~Low-4 hot leg level signal that opens the automatic depressurization stage four valves. The operators can also

~~~~~~~~~~~~~~~~~~~~

7.2.1.1.3 Core Heat Removal Trips

* * *

Reactor Trip on ~~Low~~Low-2 Pressurizer Pressure

* * *

Reactor Trip on ~~Low~~Low-2 Reactor Coolant Flow

This trip protects against departure from nucleate boiling in the event of ~~low~~Low-2 reactor coolant flow. Flow in each hot leg is measured at the hot leg elbow. The trip on ~~low~~Low-2 flow in the hot legs is automatically blocked when reactor power is below the P-10 permissive setpoint. * * *

* * *

Reactor Trip on Low-2 Reactor Coolant Pump ~~Unders~~Speed

* * *

The Low-2 reactor coolant pump ~~under~~speed trip provides a direct measurement of the parameter of interest. It permits the plant to ride through many postulated voltage or frequency dip transients without reactor trip if safety limits are not violated. Selection of the Low-2 reactor coolant pump ~~under~~speed trip setpoint and time response provide for the timely initiation of reactor trip

* * *
~~~~~~~~~~~~~~~~~~~~

Reactor Trip on ~~High~~ High-2 Reactor Coolant Pump Bearing Water Temperature

This trip is an anticipatory trip based on the expectation of a complete loss of reactor coolant flow if cooling water is lost to any of the reactor coolant pumps. This trip occurs before the reactor coolant pumps are tripped on the same measurement.

* * *

~~~~~~~~~~~~~~~~~~~~

7.2.1.1.4 Primary Overpressure Trips

Pressurizer ~~High~~High-2 Pressure Reactor Trip

This trip protects the reactor coolant system against system overpressure. The same sensors used for the pressurizer ~~low~~Low-2 pressure reactor trip are used for the ~~high~~ High-2 pressure trip except that separate setpoints are used. The ~~high~~ High-2 pressurizer pressure protection trips the reactor when two out of the four pressurizer pressure channels exceed the trip setpoint. There are no interlocks or permissives associated with this trip function.

* * *

High-3 Pressurizer Water Level Reactor Trip

This trip is provided as backup to the ~~high~~ High-2 pressurizer pressure reactor trip and serves to prevent water relief through the pressurizer safety valves. The high-3 pressurizer water level protection trips the reactor when two out of the four pressurizer water level channels exceed the trip setpoint.

~~~~~~~~~~~~~~~~~~~~

7.2.1.2.5 Design Basis: Reactor Trips for Malfunctions, Accidents, Natural Phenomena, or Credible Events (Paragraph 4.7 and 4.8 of IEEE-603-1991)

* * *

For example, protection is provided for the complete loss of coolant flow event by ~~low~~Low-2 reactor coolant pump speed and by ~~low~~Low-2 coolant flow reactor trips. Complete reliance is not made on a single reactor trip terminating a given event.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

7.2.1.1.6 Feedwater Isolation Trip

~~High-2~~High-3 Steam Generator Water Level in Any Steam Generator

This function is an anticipatory trip based on the expectation that a reactor trip would occur after steam generator feedwater is isolated. The plant control system uses a lower steam generator water level setpoint, High-1, to close the feedwater control valves. This provides an interval for operator action to prevent total isolation of the steam generator and a reactor trip before the ~~High-2~~High-3 setpoint is exceeded. The trip on ~~High-2~~High-3 steam generator water level may be manually blocked

~~~~~~~~~~~~~~~~~~~~

7.2.1.1.12 Reactor Trip System Interlocks

* * *

Steam Generator ~~High-2~~High-3 Water Level Block (One Control for each Division)

The steam generator ~~High-2~~High-3 reactor trip may be manually blocked upon the occurrence of the P-11 permissive. This trip function is automatically reset when * * *

~~~~~~~~~~~~~~~~~~~~

7.2.2.2.2 Conformance to the Single Failure Criterion for Reactor Trip (Paragraph 5.1 of IEEE 603-1991, IEEE 379-2000)

A single failure in the protection and safety monitoring system or the reactor trip actuation divisions does not prevent a reactor trip, even when a reactor trip channel is bypassed for test or maintenance. Conformance of the equipment to this requirement is discussed in WCAP-15776 (Reference 2). In addition to the redundancy of equipment, diversity of reactor trip functions is incorporated. Most Condition II, III, or IV events requiring a reactor trip are protected by trips from diverse parameters. For example, reactor trip, because of an uncontrolled rod cluster control assembly bank withdrawal at power, may occur on power range high neutron flux, overtemperature, overpower, pressurizer ~~high~~ High-2 pressure or pressurizer ~~high~~ High-3 water level. Reactor trip on complete loss of reactor coolant flow may occur on ~~low~~Low-2 flow or from the diverse parameter of ~~low~~Low-2 reactor coolant pump speed.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Table 7.2-2 (Sheet 1 of 2)

Reactor Trip[1]	No. of Channels	Division Trip Logic
Source Range High Neutron Flux Reactor Trip	* * *	
Intermediate Range High Neutron Flux Reactor Trip		
Power Range High Neutron Flux (Low Setpoint) Trip	* * *	
Power Range High Neutron Flux (High Setpoint) Trip		
High Positive Flux Rate Trip		
High-2 Reactor Coolant Pump Bearing Water Temperature	* * *	
Overtemperature ΔT		
Overpower ΔT		
Pressurizer ~~Low~~ Low-2 Pressure Trip		
Pressurizer ~~High~~ High-2 Pressure Trip		
High-3 Pressurizer Water Level Trip		
Low-2 ~~Low~~ Reactor Coolant Flow	* * *	
Low-2 Reactor Coolant Pump ~~Underspeed~~		
Low Steam Generator Water Level		
High-3 ~~High-2~~ Steam Generator Water Level	* * *	

Table 7.2-3 (Sheet 1 of 2)
Reactor Trip Permissives and Interlocks

* * *

P-10	Power range nuclear power above setpoint	* * * (d) Allows reactor trip on ~~low~~ Low-2 coolant flow (e) Allows reactor trip on ~~low~~ Low-2 reactor coolant pump speed (f) Allows reactor trip on ~~high~~ High-3 pressurizer water level (g) Allows reactor trip on ~~low~~ Low-2 pressurizer pressure

Table 7.2-3 (Sheet 2 of 2)
Reactor Trip Permissives and Interlocks

$\overline{\text{P-10}}$	Power range nuclear power below setpoint	* * * * * * (d) Blocks reactor trip on ~~low~~ Low-2 coolant flow (e) Blocks reactor trip on ~~low~~ Low-2 reactor coolant pump speed (f) Blocks reactor trip on ~~high~~ High-3 pressurizer water level (g) Blocks reactor trip on ~~low~~ Low-2 pressurizer pressure
P-11	Pressurizer pressure below setpoint	Allows manual block of ~~High-2~~ High-3 steam generator water level reactor trip
$\overline{\text{P-11}}$	Pressurizer pressure above setpoint	Automatically resets ~~High-2~~ High-3 steam generator water level reactor trip

Figure 7.2-1 (Sheet 2 of 21)

* * *

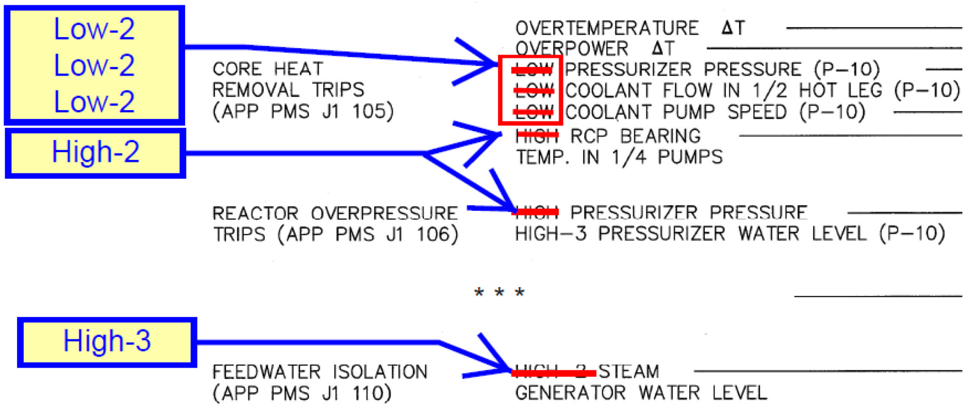

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Figure 7.2-1 (Sheet 5 of 21)

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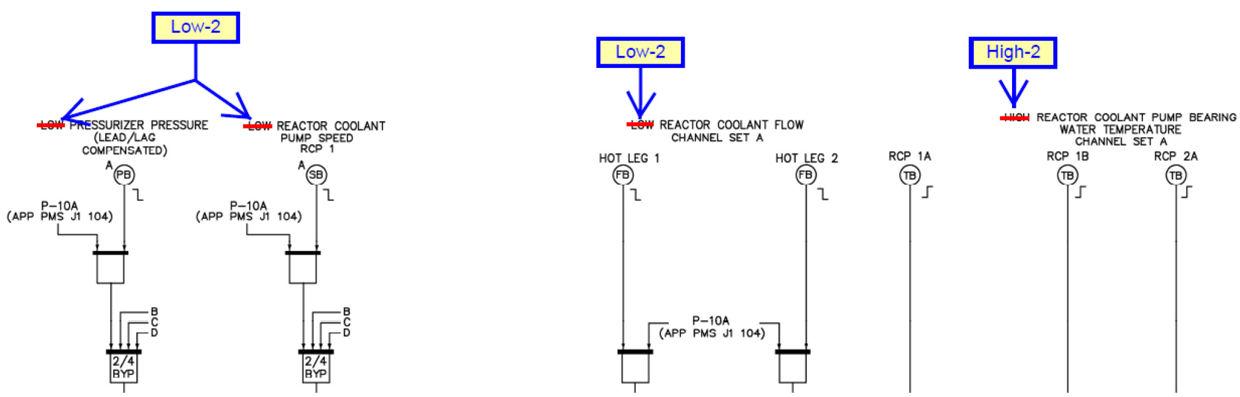

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Figure 7.2-1 (Sheet 6 of 21)

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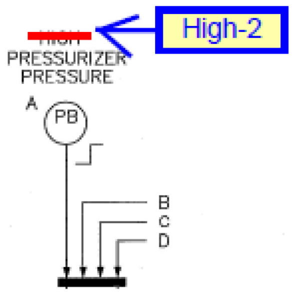

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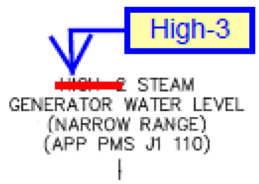

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Figure 7.2-1 (Sheet 9 of 21)

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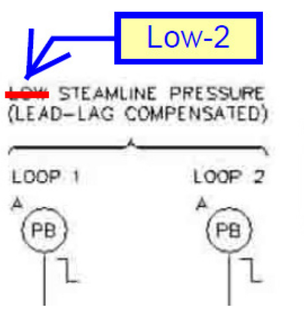

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Figure 7.2-1 (Sheet 10 of 21)

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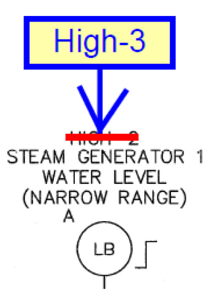

Figure 7.2-1 (Sheet 11 of 21)

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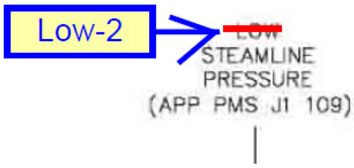

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Figure 7.2-1 (Sheet 14 of 21)

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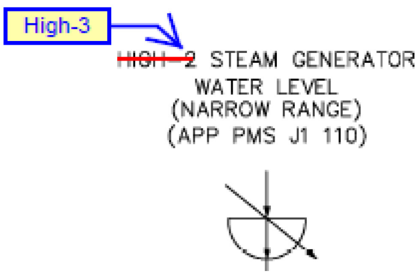

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Figure 7.2-1 (Sheet 15 of 21)

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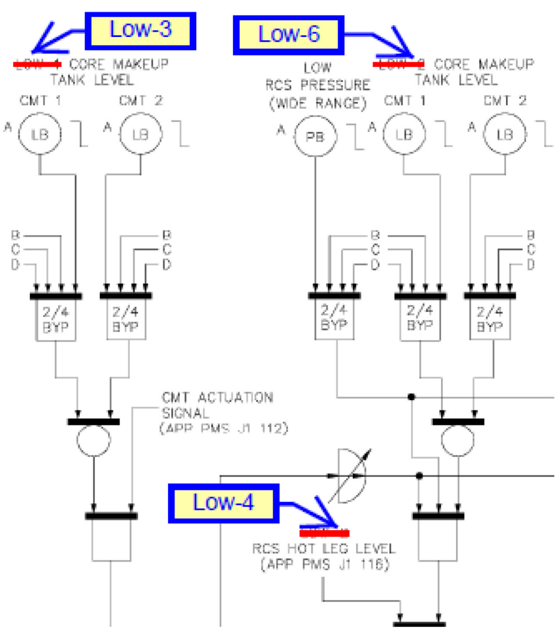

Figure 7.2-1 (Sheet 16 of 21)

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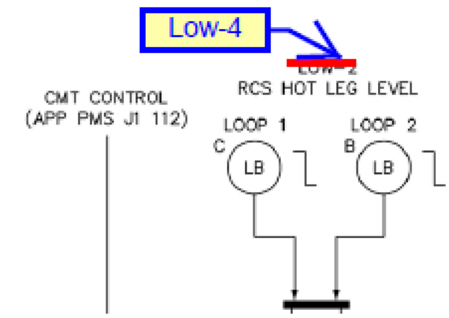

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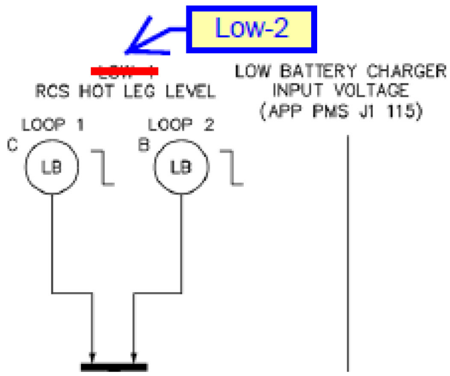

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7.3.1.1 Safeguards Actuation (S) Signal

* * *

The safeguards actuation signal is generated from any of the following initiating conditions:

1. ~~Low~~Low-3 pressurizer pressure

2. ~~Low~~Low-2 lead-lag compensated steam line pressure

3. ~~Low~~Low-2 cold leg temperature

4. * * *

5. * * *

Condition 1 results from the coincidence of pressurizer pressure below the ~~Low~~Low-3 setpoint in any two of the four divisions.
~~~~~~~~~~~~~~~~~~~~~

Condition 2 results from the coincidence of two of the four divisions of compensated steam line pressure below the ~~Low~~Low-2 setpoint in either of the two steam lines. The steam line pressure signal is lead-lag compensated to improve system response.

Condition 3 results from the coincidence of two of the four divisions of reactor coolant system cold leg temperature below the ~~Low~~Low-2 setpoint in any loop.

* * *

To permit startup and cooldown, the safeguards actuation signals generated from ~~Low~~Low-3 pressurizer pressure, ~~low~~Low-2 steam line pressure, or ~~low~~Low-2 reactor coolant inlet temperature can be manually blocked when pressurizer pressure is below the P-11 setpoint. The signal is automatically unblocked when the pressurizer pressure is above the P-11 setpoint.

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.2 In-Containment Refueling Water Storage Tank Injection

Signals to align the in-containment refueling water storage tank for injection are generated from the following conditions:

1. * * *

2. Coincidence loop 1 and loop 2 hot leg levels below ~~Low-2~~Low-4 setpoint for a duration exceeding an adjustable time delay

* * *

In-containment refueling water storage tank injection on ~~Low-2~~Low-4 hot leg level is automatically blocked when the pressurizer water level is above the P-12 setpoint. This reduces the probability of a spurious injection. * * *

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.4 Automatic Depressurization System Actuation

A signal to actuate the first stage of the automatic depressurization system is generated from any of the following conditions:

1. Core makeup tank injection alignment signal (Subsection 7.3.1.2.3) coincident with core makeup tank level less than the ~~Low-1~~Low-3 setpoint in either core makeup tank in two of the four divisions

* * *

The fourth stage is actuated upon the coincidence of a ~~Low-2~~Low-6 core makeup tank level and Low reactor coolant system pressure following a preset time delay after the third stage depressurization valves are sent a signal to open. The ~~Low-2~~Low-6 core makeup tank level input is based on the core makeup tank level being less than the ~~Low-2~~Low-6 setpoint in two of the four divisions in either core makeup tank. Upon a fourth stage actuation signal, * * *

* * *

A signal to initiate the opening sequence of the fourth stage is also generated upon the occurrence of coincidence loop 1 and loop 2 hot leg levels below the ~~Low-2~~Low-4 setpoint for a duration exceeding an adjustable time delay. This signal also initiates in-containment refueling water storage tank injection. As discussed * * *.

7.3.1.2.5 Reactor Coolant Pump Trip

* * *

6. ~~High~~ High-2 reactor coolant pump bearing water temperature

* * *

Condition 6 is derived from a coincidence of two of the four divisions of ~~high~~ High-2 reactor coolant pump bearing water temperature for any reactor coolant pump. All of the reactor coolant pumps are tripped simultaneously if Condition 6 is met for the bearing water temperature of any reactor coolant pump. This function is included for equipment protection. The high temperature setpoint and dynamic compensation are the same as used in the ~~high~~ High-2 reactor coolant pump bearing water temperature reactor trip (Subsection 7.2.1.1.3) but with the inclusion of preset time delay.

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.6 Main Feedwater Isolation

Signals to isolate the main feedwater supply to the steam generators are generated from any of the following conditions:

* * *

3. ~~High-2~~High-3 steam generator narrow range water level

* * *

Condition 3 is derived from a coincidence of two of the four divisions of narrow range steam generator water level above the ~~High-2~~High-3 setpoint for either steam generator. In addition to tripping the turbine and isolating the main feedwater supply, condition 3 also * * *

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.7 Passive Residual Heat Removal Heat Exchanger Alignment

A signal to align the passive heat removal heat exchanger to passively remove core heat is generated from any of the following conditions:

* * *

4. Low narrow range steam generator level coincident with ~~Low~~Low-2 startup feedwater flow

* * *

Condition 4 results from the coincidence of two of the four divisions of narrow range steam generator level below the Low setpoint, after a preset time delay, coincident with a ~~Low~~Low-2 startup feedwater flow in a particular steam generator. This function is provided for each

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.8 Turbine Trip

A signal to initiate turbine trip is generated from any of the following conditions:

1. Reactor trip (Table 7.3-2, interlock P-4)

2. ~~High-2~~High-3 steam generator narrow-range water level

* * *

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the ~~High-2~~High-3 setpoint for either steam generator.

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.10 Steam Line Isolation

A signal to isolate the steam line is generated from any one of the following conditions:

1. * * *

2. * * *

3. ~~Low~~Low-2 lead-lag compensated steam line pressure

4. * * *

5. ~~Low~~Low-2 cold leg temperature

* * *

Condition 3 results from the coincidence of two of the four divisions of compensated steam line pressure below the ~~Low~~Low-2 setpoint. The steam line pressure signal is lead-lag compensated to improve system response. If the pressure is below this setpoint, in either steam line, both main steam lines are isolated.

* * *

Condition 5 results from the coincidence of reactor coolant system cold leg temperature below the ~~Low~~Low-2 T_{cold} setpoint in any loop.

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.13 Startup Feedwater Isolation

Signals to isolate the startup feedwater supply to the steam generators are generated from either of the following conditions:

1. ~~Low~~Low-2 cold leg temperature
2. ~~High-2~~High-3 steam generator narrow range water level

* * *

Condition 1 results from the coincidence of reactor coolant system cold leg temperature below the ~~Low~~Low-2 T_{cold} setpoint in any loop. Startup feedwater isolation on this condition may be manually blocked when the pressurizer pressure is below the P-11 setpoint. This function is automatically unblocked when the pressurizer pressure is above the P-11 setpoint.

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the ~~High-2~~High-3 setpoint for either steam generator.

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.15 Chemical and Volume Control System Isolation

A signal to close the isolation valves of the chemical and volume control system is generated from any of the following conditions:

1. High-2 pressurizer level
2. ~~High-2~~High-3 steam generator narrow range water level

* * *

Condition 2 results from a coincidence of two of the four divisions of narrow range steam generator water level above the ~~High-2~~High-3 setpoint for either steam generator

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

7.3.1.2.22 Chemical and Volume Control System Letdown Isolation

A signal to isolate the letdown valves of the chemical and volume control system is generated upon the occurrence of a ~~Low-1~~Low-2 hot leg level in either of the two hot leg loops. This helps to maintain reactor coolant system * * *

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.24 Steam Generator Relief Isolation

A signal for closing the steam generator power operated relief valves and their block valves is generated from any of the following conditions:

1. Manual initiation

2. ~~Low~~Low-2 lead-lag compensated steam line pressure

Condition 2 results from the coincidence of two of the four divisions of compensated steam line pressure below the ~~Low~~Low-2 setpoint. The steam line pressure signal is lead-lag compensated to improve system response. * * *

~~~~~~~~~~~~~~~~~~~~

7.3.1.2.25 Component Cooling System Containment Isolation Valve Closure

* * *

A signal to close the component cooling system containment isolation valves is derived from a coincidence of two of the four divisions of ~~high~~ High-2 reactor coolant pump bearing water temperature for any reactor coolant pump. The high temperature setpoint and dynamic compensation are the same as used in the ~~high~~ High-2 reactor coolant pump bearing water temperature reactor coolant pump trip (Subsection 7.3.1.2.5, condition 6), but with the inclusion of preset time delay.

* * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

7.3.1.5.5 Design Basis: Engineered Safety Features for Malfunctions, Accidents, Natural Phenomena, or Credible Events (Paragraph 4.7 and 4.8 of IEEE 603-1991)

* * *

Functional diversity is used in determining the actuation signals for engineered safety features. For example, a safeguards actuation signal is generated from high containment pressure, ~~low~~Low-3 pressurizer pressure, and ~~low~~Low-2 compensated steam line pressure.

~~~~~~~~~~~~~~~~~~~~

Table 7.3-1 (Sheet 1 of 9)

* * *

	Actuation Signal
	1. Safeguards Actuation Signal (
Low-3	a. ~~Low~~ pressurizer pressure
Low-2	b. ~~Low~~ lead-lag compensated steam line pressure
Low-2	c. ~~Low~~ cold leg temperature (~~Low~~ T_{cold})

* * *
~~~~~~~~~~~~~~~~~~~~

Table 7.3-1 (Sheet 2 of 9)

* * *

	Actuation Signal
3.	**Automatic Depressurization System**
a.	Core makeup tank injection coincident with
	Core makeup tank level less than ~~Low-1~~ Low-3 setpoint
	* * *
	* * *
	* * *
	~~Low~~ Low[10] reactor coolant system pressure or
	3rd stage depressurization
e.	Core makeup tank level less than ~~Low-2~~ Low-6 setpoint coincident with
	~~Low~~ Low[10] reactor coolant system pressure and coincident with
	3rd stage depressurization
f.	Coincident loop 1 and loop 2 ~~Low-2~~ Low-4 hot leg level (after delay)

* * *

Table 7.3-1 (Sheet 3 of 9)

* * *

	Actuation Signal
	4. Main Feedwater Isolation (Figure
	* * * . * * *
High-3	c. ~~High-2~~ steam generator narrow range level
Low[10]	d. ~~Low~~ reactor coolant temperature (Low-~~1~~ T_{avg}) coincident with
	Reactor trip (P-4)
	(Trip of Main Feedwate
	* * * . * * *
High-3	c. ~~High-2~~ steam generator narrow range level

* * *

Table 7.3-1 (Sheet 4 of 9)

* * *

Actuation Signal
d. * * *
* * *
e.
f. ~~High~~ High-2 reactor coolant pump bearing water temperature
6. Core Makeup Tank Injection (Figure
a. * * *

* * *

7. Turbine Trip (Figure 7.2-1, Sheet 14)
a. * * *
b.
c. ~~High-2~~ High-3 steam generator narrow range level

Table 7.3-1 (Sheet 5 of 9)

* * *

	Actuation Signal
	b. * * *
Low-2	c. ~~Low~~ lead-lag compensated steam line pressure[4]
	d. High steam line negative pressure rate
Low-2	e. ~~Low~~ cold leg temperature (~~Low~~ T_{cold})
	9. Steam Generator Blowdown Syst

* * *

	11. Startup Feedwater Isolation (Figure
Low-2	a. ~~Low~~ cold leg temperature (~~Low~~ T_{cold})
High-3	b. ~~High-2~~ steam generator narrow range water level
	* * *
High[10]	d. ~~High~~ steam generator narrow range level coincident with

* * *

Table 7.3-1 (Sheet 6 of 9)

* * *

	Actuation Signal
	12. Passive Residual Heat Removal
	a. * * *
	b. * * * coincident with
Low-2	~~Low~~ startup feedwater flow

* * *

	14. Chemical Volume Control System
	a. * * *
High-3	b. ~~High-2~~ steam generator narrow range level

Table 7.3-1 (Sheet 7 of 9)

* * *

Actuation Signal	
	High[10] ~~High-1~~ pressurizer water level
d.	* * *
e.	* * *
f.	
g.	High[10] ~~High~~ steam generator narrow range level coincident with
	Reactor trip (P-4)
15. Steam Dump Block (Figure 7.2-1	

* * *

17. Auxiliary Spray and Purification	
a.	Low[10] ~~Low-1~~ pressurizer level
b.	* * *

Table 7.3-1 (Sheet 8 of 9)

* * *

Actuation Signal	
c.	* * *
18. Containment Air Filtration System	
a.	Containment isolation
b.	High[10] ~~High-1~~ containment radioactivity

* * *

24. Chemical and Volume Control System Letdown Iso		
a.	Low-2 ~~Low-1~~ hot leg level	* * *

Table 7.3-1 (Sheet 9 of 9)

* * *

Actuation Signal
a. * * *
b. ~~Low~~ Low-2 lead-lag compensated steam line pressure[4]
27. Close Component Cooling System
a. ~~High~~ High-2 reactor coolant pump bearing water temperature

* * *

Notes:

1. * * *
2. .
3. * * *
4.
5. .
   * * *
6. .
7. * * *
8.

10. Low and Low-1 are considered to be equivalent terms when referring to the first low setpoint designator. High and High-1 are considered to be equivalent terms when referring to the first high setpoint designator.

Reviewer's Note:
Table 7.3-1 Note 9 is proposed to be added by separate LAR.
Note 10 is proposed as-is, irrespective of the review of the separate LAR

Table 7.3-2 (Sheet 1 of 4)

Designation	Derivation	Function
* * *	* * *	* * *
P-11	Pressurizer pressure below setpoint	(a) Permits manual block of safeguards actuation on ~~low~~ Low-3 pressurizer pressure, ~~low~~ Low-2 compensated steam line pressure, or ~~low~~ Low-2 reactor coolant inlet temperature (b) Permits manual block of steam line isolation on ~~low~~ Low-2 reactor coolant inlet temperature (c) Permits manual block of steam line isolation and steam generator power-operated relief valve block valve closure on ~~low~~ Low-2 compensated steam line pressure (d) Coincident with manual actions of (b) or (c), automatically unblocks steam line isolation on high negative steam line pressure rate (e) Permits manual block of main feedwater isolation on low reactor coolant temperature

Table 7.3-2 (Sheet 2 of 4)

Designation	Derivation	Function
P-11 (continued)	Pressurizer pressure below setpoint	(f) Permits manual block of startup feedwater isolation on ~~low~~ Low-2 reactor coolant inlet temperature (g) Permits manual block of steam dump block on low reactor coolant temperature (h) Permits manual block of normal residual heat removal system isolation on high containment radioactivity.
$\overline{\text{P-11}}$	Pressurizer pressure above setpoint	(a) Prevents manual block of safeguards actuation on ~~low~~ Low-3 pressurizer pressure, ~~low~~ Low-2 compensated steam line pressure, or ~~low~~ Low-2 reactor coolant inlet temperature (b) Prevents manual block of steam line isolation on ~~low~~ Low-2 reactor coolant inlet temperature (c) Prevents manual block of steam line isolation and steam generator power-operated relief valve block valve closure on ~~low~~ Low-2 compensated steam line pressure (d) Automatic block of steam line isolation on high negative steam line pressure rate (e) Prevents manual block of feedwater isolation on low reactor coolant temperature (f) Prevents manual block of startup feedwater isolation on ~~low~~ Low-2 reactor coolant inlet temperature (g) Prevents manual block of normal residual heat removal system isolation on high containment radioactivity

Table 7.3-2 (Sheet 3 of 4)

Designation	Derivation	Function
P-12	Pressurizer level below setpoint	* * * . * * * . * * * . * * * . * * * . * * * (d) Coincident with manual action of (a), automatically unblocks fourth stage automatic depressurization system initiation on ~~low~~ Low-4 hot leg level to provide protection during mid-loop operation. (e) Automatically unblocks chemical and volume control system letdown isolation on ~~Low 1~~ Low-2 hot leg level
$\overline{\text{P-12}}$	Pressurizer level above setpoint	* * * . * * * . * * * . * * * . * * * . * * * (e) Automatically blocks fourth stage automatic depressurization system initiation on ~~low~~ Low-4 hot leg level to reduce the probability of spurious actuation. (f) Permits manual block of chemical and volume control system letdown isolation on ~~Low 1~~ Low-2 hot leg level

Table 7.3-2 (Sheet 4 of 4)

Designation	Derivation	Function
P-19	Reactor coolant system pressure below setpoint	(a) Permits manual block of chemical and volume control system isolation on ~~high~~ High-2 pressurizer water level (b) Permits manual block of passive residual heat removal heat exchanger alignment on ~~high~~ High-3 pressurizer water level (c) Permits manual block of the pressurizer heater trip on ~~high~~ High-3 pressurizer water level
$\overline{\text{P-19}}$	Reactor coolant system pressure above setpoint	(a) Prevents manual block of chemical and volume control system isolation on ~~high~~ High-2 pressurizer water level (b) Prevents manual block of passive residual heat removal heat exchanger alignment on ~~high~~ High-3 pressurizer water level (c) Prevents manual block of the pressurizer heater trip on ~~high~~ High-3 pressurizer water level

~~~~~~~~~~~~~~~~~~~~

7.4.1.1 Safe Shutdown Using Safety-Related Systems

* * *

The engineered safety system actuation signal generated on ~~low~~Low-3 pressurizer pressure also actuates containment isolation. This prevents loss of water inventory from containment and permits indefinite operation of the passive residual heat removal heat exchanger and the in-containment refueling water storage tank.

~~~~~~~~~~~~~~~~~~~~

7.7.1.6 Pressurizer Pressure Control System

* * *

Pressure increases too fast to be handled by reducing the variable heater output result in spray actuation. Spray continues until pressure decreases to the point that the variable heaters alone can regulate pressure. For normal transients including a full-load rejection, the pressurizer pressure control system acts promptly to prevent reaching the ~~high~~High-2 pressurizer pressure reactor trip setpoint.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

9.2.2.2 System Description

* * *

Component cooling water is distributed to the components by this single supply/return header. Components are grouped in branch lines according to plant arrangement, with one branch line cooling the components inside containment. Loads inside containment are automatically isolated in response to a safety injection signal, which also trips the reactor coolant pumps, and in response to a ~~high~~High-2 bearing water temperature trip signal from one of the reactor coolant pumps. Individual components, except the reactor coolant pumps, can be isolated locally to permit maintenance while supplying the remaining components with cooling water.

~~~~~~~~~~~~~~~~~~~~

9.2.2.3.4 Component Cooling Water System Valves

* * *

Three motor-operated isolation valves and a check valve provide containment isolation for the supply and return component cooling water system lines that penetrate the containment barrier. The motor-operated valves are normally open; however, they are closed upon receipt of a safety injection signal, or a ~~high~~High-2 bearing water temperature reactor trip signal. They are controlled from the main control room and fail as-is.

* * *

* * * One relief valve is also provided in each Safety Class C section of piping, just inside the innermost containment isolation valves on the CCS 10-inch supply and return lines penetrating the containment barrier, to ensure protection of the containment isolation valves from excess pressure while closing on a reactor coolant pump ~~high~~High-2 bearing water temperature trip signal intended to isolate a potential external heat exchanger tube leak. In addition, these valves provide protection of containment isolation valves in the event of a letdown heat exchanger tube rupture

~~~~~~~~~~~~~~~~~~~~

9.2.2.4.5.2 Leakage into the Component Cooling Water System from a High Pressure Source

* * *

* * * These containment isolation valves close automatically if the leak rate is sufficiently large to cause a ~~high~~High-2 bearing water temperature reactor and pump trip signal to be generated by the protection and safety monitoring system (PMS). The containment isolation valves can also be closed manually by the operator after being alerted to a reactor coolant pump leak by alarms from component cooling water system instrumentation (surge tank level and/or radiation

level in the CCS pump suction header) or from the flow instruments located on the inlet and outlet lines from the leaking reactor coolant pump external heat exchanger. Manual closure of one CCS outlet isolation valve will result in a ~~high~~High-2 bearing water temperature trip of the plant if the affected reactor coolant pump continues to operate.

~~~~~~~~~~~~~~~~~~~~

9.2.2.7 Instrumentation Requirements

* * *

Flow alarms in the main control room, produced by the two flow channels located on the CCS reactor coolant pump cooling water inlet and outlet lines, will alert the operator to a leak from the reactor coolant pump external heat exchanger into the component cooling water system. Signals generated by the PMS, in the event of a ~~high~~High-2 bearing water temperature trip of the reactor, also close the CCS containment isolation valves to eliminate the possibility of reactor coolant from a faulted external heat exchanger tube discharging to portions of the CCS outside the containment.

~~~~~~~~~~~~~~~~~~~~

9.3.6.1.1 Safety Design Basis

The safety functions provided by the chemical and volume control system are limited to containment isolation of chemical and volume control system lines penetrating containment, termination of inadvertent reactor coolant system boron dilution, isolation of makeup on a steam generator High-3 or pressurizer ~~high~~High-2 level signal, and preservation of the reactor coolant system pressure boundary, including isolation of normal chemical and volume control system letdown from the reactor coolant system.

~~~~~~~~~~~~~~~~~~~~

9.3.6.3.7 Chemical and Volume Control System Valves

* * *

Makeup Line Containment Isolation Valves

These normally open, motor-operated globe valves provide containment isolation of the chemical and volume control system makeup line and automatically close on a high-2 pressurizer level, ~~high~~High-3 steam generator level, or high-2 containment radiation signal from the protection and safety monitoring system

~~~~~~~~~~~~~~~~~~~~

9.3.6.4.5 Accident Operation

* * *

To protect against steam generator overfill, the makeup function is isolated by closing the makeup line containment isolation valves, if a ~~high~~High-3 steam generator level signal is generated. These valves also close and isolate the system on a ~~high~~High-2 pressurizer level signal.

~~~~~~~~~~~~~~~~~~~~

9.3.6.5 Design Evaluation

* * *

The chemical and volume control system also incorporates a safety-related method of isolating the makeup to the reactor coolant system upon receipt of a ~~high~~High-3 steam generator level signal or a ~~high~~High-2 pressurizer level signal, as described in Subsection 9.3.6.4.5. Other chemical and volume control system components are not safety-related.

~~~~~~~~~~~~~~~~~~~~

9.3.6.7 Instrumentation Requirements

* * *

- **Letdown isolation valves –** * * * The letdown isolation valves also receive a signal from the protection and safety monitoring system to automatically close and isolate letdown during midloop operations based on a ~~low~~Low-2 hot leg level. Manual control * * *

* * *

- **Makeup isolation valves –** To isolate the makeup flow to the reactor coolant system, two valves are provided in the chemical and volume control system makeup line. These valves automatically close on a signal from the protection and safety monitoring system derived from source range flux doubling, high-2 pressurizer level, ~~high~~High-3 steam generator level, a safeguards signal coincident with high-1 pressurizer level, or high steam generator water level coincident with P-4 permissive (reactor trip) * * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

10.3.2.2.3 Power-Operated Atmospheric Relief Valves

* * *

An isolation valve with remote controls is provided upstream of each power operated relief valve providing isolation of a leaking or stuck-open valve. The upstream location allows for maintenance on the power-operated relief valve operator at power. The motor-operated isolation valve employs a safety-related operator and closes automatically on ~~low~~Low-2 steam line pressure to terminate steam line depressurization transients.

~~~~~~~~~~~~~~~~~~~~

10.3.2.2.4 Main Steam Isolation Valves

* * *

Closure of the main steam isolation valves and main steam isolation bypass valves is initiated by the following:

* * *

- ~~Low~~Low-2 steam line pressure in one of two loops
- * * *
- * * *
- ~~Low~~Low-2 T_{cold} in either reactor coolant loop

~~~~~~~~~~~~~~~~~~~~

Table 10.3.3-1 (Sheet 3 of 10)

* * *	Method of Failure Detection	Failure Effect on System Safety Function Capability
	* * *	
* * *	c. Position indication on main control room & remote shutdown workstation	c. None, automatic redundant isolation of the PORV ~~on low~~ Low-2 steam line pressure

Table 10.3.3-1 (Sheet 8 of 10)

* * *	Method of Failure Detection	Failure Effect on System Safety Function Capability
* * * * * * * * *	a. Higher pressure and/or water level in SG A(B)	a. None, 5 out of 6 safety valves for SG A(B) still available with PORV available to supplement relief capacity; also, plant trip occurs on ~~high~~ High-3 steam generator level

~~~~~~~~~~~~~~~~~~~~

10.4.9.3 Safety Evaluation

* * *

- The startup feedwater isolation valves and the startup feedwater control valves automatically close upon receipt of a feedwater isolation signal, which occurs on a steam generator ~~high~~High-3 water level and other appropriate engineered safeguards * * *

~~~~~~~~~~~~~~~~~~~~

10.4.9.5 Instrumentation Applications

* * *

* * * The startup feedwater flow transmitters also provide redundant indication of startup feedwater and automatic safeguards actuation input on ~~low~~Low-2 flow coincident with low, narrow range steam generator level. See Section 7.3.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Table 14.3-2 (Sheet 2 of 17)

Reference	Design Feature	Value
* * *	* * *	
.	.	
.	.	
* * *	* * *	
.	.	
.	.	
* * *	* * *	
Section 5.4.1.3.4	A safety-related pump trip occurs on ~~high~~ High-2 bearing water temperature.	

* * *

Table 14.3-2 (Sheet 8 of 17)

Reference	Design Feature	Value
* * *	* * *	
Section 7.3.1.2.4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a ~~low 2~~ Low-6 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	

* * *

Table 14.3-2 (Sheet 9 of 17)

Reference	Design Feature	Value
Section 7.3.1.2.4 Low-3	The first stage valves of the automatic depressurization system open upon receipt of a signal generated from a core makeup tank injection alignment signal coincident with core makeup tank water level less than the ~~Low-1~~ setpoint in either core makeup tank via the protection and safety monitoring system.	

* * *

Reference	Design Feature	Value
Section 7.3.1.2.15 High-3	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and monitoring system derived from a source range flux doubling, high-2 pressurizer level, ~~high-2~~ steam generator level signal, a safeguards signal coincident with high-1 pressurizer level, high steam generator water level coincident with P-4 permissive (reactor trip) or high-2 containment radioactivity.	

* * *

Table 14.3-2 (Sheet 12 of 17)

Reference	Design Feature	Value
* * *	* * *	
Section 9.3.6.7 High-3	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and safety monitoring system derived from a source range flux doubling, high-2 pressurizer level, ~~high~~ steam generator level signal, high steam generator water level coincident with P-4 permissive (reactor trip), high-2 containment radioactivity, or a safeguards signal coincident with high-1 pressurizer level.	

* * *

Table 14.3-2 (Sheet 15 of 17)

Reference	Design Feature
* * *	* * *
Section 15.1.2.1	Continuous addition of excessive feedwater is prevented by the steam generator ~~high-2~~ High-3 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps and reactor.
Section 15.1.4.1	For an inadvertent opening of a steam generator relief or safety valve, core makeup tank actuation occurs from one of four sources: – Two out of four ~~low~~ Low-3 pressurizer pressure signals – Two out of four low-2 pressurizer level signals – Two out of four ~~low~~ Low-2 T_{cold} signals in any one loop – Two out of four ~~low~~ Low-2 steam line pressure signals in any one loop

* * *

Table 14.3-2 (Sheet 16 of 17)

Reference	Design Feature	Value
Section 15.1.5.1	Following a steam line rupture, core makeup tank actuation occurs from one of five sources: – Two out of four ~~low~~ Low-3 pressurizer pressure signals – Two out of four high-2 containment pressure signals – Two out of four ~~low~~ Low-2 steam line pressure signals in any loop – Two out of four ~~low~~ Low-2 T_{cold} signals in any one loop – Two out of four low-2 pressurizer level signals	
* * *		
Section 15.2.8.2.1	Receipt of a ~~low~~ Low-2 steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives a safeguards signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.	

~~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~~

Table 15.0-4a (Sheet 1 of 2)

Function
* * *
Reactor trip on ~~high~~ High-2 pressurizer pressure
Reactor trip on ~~low~~ Low-2 pressurizer pressure
Reactor trip on ~~low~~ Low-2 reactor coolant flow in either hot leg
Reactor trip on Low-2 reactor coolant pump ~~under~~ speed
* * *   .   .   * * *   .   .   * * *
~~High-2~~ High-3 steam generator level
* * *   .   * * *
“S” signal and steam line isolation on ~~low~~ Low-2 T_{cold}

Table 15.0-4a (Sheet 2 of 2)

Function
"S" signal and steam line isolation on ~~low~~ steam line pressure Low-2
Low-3 "S" signal on ~~low~~ pressurizer pressure

* * *

ADS Stage 1 actuation on core makeup tank ~~low~~ Low-3 level signal(1)
ADS Stage 4 actuation on core makeup tank ~~low-low~~ Low-6 level signal(1)

* * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Table 15.0-6 (Sheet 1 of 5)

Incident	Reactor Trip Functions	ESF Actuation Functions
* * *		
Inadvertent opening of a steam generator safety valve	Power range high flux, overtemperature ΔT, overpower ΔT, ~~Low~~ Low-2 pressurizer pressure, "S", manual	~~Low~~ Low-3 pressurizer pressure, ~~low~~ Low-2 compensated steam line pressure, ~~low~~ Low-2 T_{cold}, low-2 pressurizer level
Steam system piping failure	Power range high flux, overtemperature ΔT, overpower ΔT, ~~Low~~ Low-2 pressurizer pressure, "S", manual	~~Low~~ Low-3 pressurizer pressure, ~~low~~ Low-2 compensated steam line pressure, high-2 containment pressure, ~~low~~ Low-2 T_{cold}, manual
Inadvertent operation of the PRHR	PRHR discharge valve position	~~Low~~ Low-3 pressurizer pressure, ~~low~~ Low-2 T_{cold}, low-2 pressurizer level

Table 15.0-6 (Sheet 2 of 5)

Incident	Reactor Trip Functions	ESF Actuation Functions
Section 15.2		
Decrease in heat removal by the secondary system		
Loss of external load/turbine trip	~~High~~ High-2 pressurizer pressure, ~~high~~ High-3 pressurizer water level, overtemperature ΔT, overpower ΔT, Steam generator low narrow range level, ~~low~~ Low-2 RCP speed, manual	–
Loss of nonemergency ac power to the station auxiliaries	Steam generator low narrow range level, ~~high~~ High-2 pressurizer pressure, ~~high~~ High-3 pressurizer level, ~~low~~ Low-2 RCP speed, manual	Steam generator low narrow range level coincident with ~~low~~ Low-2 startup water flow, steam generator low wide range level
Loss of normal feedwater flow	Steam generator low narrow range level, ~~high~~ High-2 pressurizer pressure, ~~high~~ High-3 pressurizer level, manual	Steam generator low narrow range level coincident with ~~low~~ Low-2 startup water flow, steam generator low wide range level
Feedwater system pipe break	Steam generator low narrow range level, ~~high~~ High-2 pressurizer pressure, ~~high~~ High-3 pressurizer level, overtemperature ΔT, manual	Steam generator low narrow range level coincident with ~~low~~ Low-2 startup feedwater flow, Steam generator low wide range level, ~~low~~ Low-2 steam line pressure, high-2 containment pressure

Table 15.0-6 (Sheet 3 of 5)

Incident	Reactor Trip Functions	ESF Actuation Functions
Section 15.3		
Decrease in reactor coolant system flow rate	Low-2 RCP	
Partial and complete loss of forced reactor coolant flow	~~Low~~ Low-2 flow, ~~under~~speed, manual	–
Reactor coolant pump shaft seizure (locked rotor)	~~Low~~ Low-2 flow, ~~high~~ High-2 pressurizer pressure, manual	–

* * *

Incident	Reactor Trip Functions	ESF Actuation Functions
Uncontrolled RCCA bank withdrawal at power	Power range high neutron flux, high power range positive neutron flux rate, overtemperature ΔT, over-power ΔT, ~~high~~ High-2 pressurizer pressure, ~~high~~ High-3 pressurizer water level, manual	–
RCCA misalignment	Overtemperature ΔT, ~~low~~ Low-2 pressurizer pressure, manual	–

Table 15.0-6 (Sheet 4 of 5)

Incident	Reactor Trip Functions	ESF Actuation Functions
Section 15.5		
Increase in reactor coolant inventory		
Inadvertent operation of the CMT during power operation High-2	~~High~~ pressurizer pressure, manual, "safeguards" trip, ~~high~~ pressurizer level High-3	~~High~~ pressurizer level, ~~low~~ T_{cold} Low-2
Chemical and volume control system malfunction that increases reactor coolant inventory	~~High~~ pressurizer pressure, "safeguards" trip, ~~high~~ pressurizer level, manual	~~High~~ pressurizer level, ~~low~~ T_{cold}, ~~low~~ steam line pressure Low-2

* * *

Section 15.6		
Decrease in reactor coolant inventory	Low-2	Low-3
Inadvertent opening of a pressurizer safety valve or ADS path	~~Low~~ pressurizer pressure, overtemperature ΔT, manual	~~Low~~ pressurizer pressure

Table 15.0-6 (Sheet 5 of 5)

Incident	Reactor Trip Functions	ESF Actuation Functions
Section 15.6 (Cont.)		
Steam generator tube rupture	~~Low~~ Low-2 pressurizer pressure, overtemperature ΔT, safeguards (“S”), manual	~~Low~~ Low-3 pressurizer pressure, ~~high-2~~ High-3 steam generator water level, high steam generator level coincident with reactor trip (P-4), ~~low~~ Low-2 steam line pressure, low pressurizer level
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	~~Low~~ Low-2 pressurizer pressure, safeguards (“S”), manual	High-2 containment pressure, ~~low~~ Low-3 pressurizer pressure

~~~~~~~~~~~~~~~~~~~~~

15.1.2.1 Identification of Causes and Accident Description

* * *

Continuous addition of excessive feedwater is prevented by the steam generator ~~high-2~~High-3 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

~~~~~~~~~~~~~~~~~~~~~

15.1.2.2.1 Method of Analysis

* * *

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

* * *

- The feedwater flow resulting from a fully open control valve is terminated by a steam generator ~~high-2~~High-3 level trip signal, which closes feedwater control and isolation valves and trips the main feedwater pumps, the turbine, and the reactor.

~~~~~~~~~~~~~~~~~~~~

15.1.2.2.2 Results

* * *

When the steam generator water level in the faulted loop reaches the ~~high-2~~High-3 level setpoint, the feedwater control valves and feedwater isolation valves are automatically closed and the main feedwater pumps are tripped. This prevents * * *

~~~~~~~~~~~~~~~~~~~~

15.1.4.1 Identification of Causes and Accident Description

* * *

- Core makeup tank actuation from one of the following signals:
  - Safeguards (“S”) signal
    - Two out of four ~~low~~Low-3 pressurizer pressure signals
    - * * *
    - Two out of four ~~low~~Low-2 T_{cold} signals in any one loop or
    - Two out of four ~~low~~Low-2 steam line pressure signals in any one loop

    * * *
- Redundant isolation of the startup feedwater system
  - Sustained high startup feedwater flow causes additional cooldown. Therefore, the ~~low~~Low-2 T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

  – Two out of four ~~low~~Low-2 steam line pressure signals in any one loop (above permissive P-11)

  – * * *

  – Two out of four ~~low~~Low-2 T_{cold} signals in any one loop, or

  * * *

~~~~~~~~~~~~~~~~~~~~

15.1.4.2.2 Results

* * *

Core makeup tank injection and the associated tripping of the reactor coolant pumps are initiated automatically by the ~~low~~Low-2 T_{cold} "S" signal. Boron solution at * * *

~~~~~~~~~~~~~~~~~~~~

15.1.5.1 Identification of Causes and Accident Description

* * *

- Core makeup tank actuation from any of the following:

  – Safeguards ("S") signal from:

  – Two out of four ~~low~~Low-3 pressurizer pressure signals

  – * * *

  – Two out of four ~~low~~Low-2 T_{cold} signals in one loop, or

  – Two out of four ~~low~~Low-2 steam line pressure signals one loop

  * * *

- Redundant isolation of the startup feedwater system

  Sustained high startup feedwater flow causes additional cooldown. Therefore, the ~~low~~Low-2 T_{cold} signal closes the startup feedwater control and isolation valves.

- Trip of the fast-acting main steam line isolation valves (assumed to close in less than 10 seconds) on one of the following signals:

  – Two out of four ~~low~~Low-2 steam line pressure signals in any one loop (above permissive P- 11)

  – * * *

  – Two out of four ~~low~~Low-2 T_{cold} signals in any one loop, or

  * * *

~~~~~~~~~~~~~~~~~~~~

15.1.5.2.3 Core Power and Reactor Coolant System Transient

* * *

* * * The transient shown assumes an uncontrolled steam release from only one steam generator. Steam release from more than one steam generator is prevented by automatic trip of the main steam isolation valves in the steam lines by ~~low~~Low-2 steam line pressure signals. Even with the failure of one valve

~~~~~~~~~~~~~~~~~~~~

15.1.5.5.1 Identification of Causes and Accident Description

* * *

Depending upon the break size, the reactor may be tripped on any of the following trip signals to provide the necessary protection against the rupture of a main steam line.

- Overpower ΔT
- ~~Low~~Low-2 pressurizer pressure
- Safeguards ("S') actuation signal

  - ~~low~~Low-2 steam line pressure
  - ~~low~~Low-2 cold leg temperature

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

15.1.5.5.2.1 Method of Analysis

* * *

The following assumptions are made in the transient analysis:

* * *

2. Break Size – A spectrum of break sizes was analyzed. Small breaks do not result in a reactor trip. Intermediate breaks result in a reactor trip on overpower ΔT. Larger break sizes result in a reactor trip on ~~low~~Low-2 steam line pressure safeguards actuation.

* * *

5. Protection System – The protection system features that mitigate the effects of a steam line beak are described in subsection 15.1.5. This analysis only considers the initial phase of the transient initiated from an at-power condition. Protection in this phase of the transient is provided by reactor trip, if necessary (specifically overpower ΔT, and ~~low~~Low-2 steam line pressure safeguards actuation).

~~~~~~~~~~~~~~~~~~~~

15.1.5.5.3 Results

A spectrum of steam line break sizes was analyzed * * * the reactor trips on the ~~low~~Low-2 steam line pressure safeguards actuation signal.

~~~~~~~~~~~~~~~~~~~~

15.1.6.1 Identification of Causes and Accident Description

* * *

The inadvertent actuation of the PRHR heat exchanger event is a Condition II event, a fault of moderate frequency. Plant systems and equipment available to mitigate the effects of the accident are discussed in subsection 15.0.8 and listed in Table 15.0.6. The following reactor protection system functions are available to provide protection in the event of an inadvertent PRHR heat exchanger actuation:

- PRHR discharge valve not closed
- Overpower/overtemperature reactor trips (neutron flux and ΔT)
- Two out of four ~~low~~Low-3 pressurizer pressure signals

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Table 15.1.2-1 (Sheet 1 of 2)

* * *

Accident	Event
* * *	
Feedwater system malfunctions that result in an increase in feedwater flow	Both main feedwater control valves fail fully open Minimum DNBR occurs Turbine trip/feedwater isolation and reactor trip on ~~high~~ High-3 steam generator level Rod motion begins

~~~~~~~~~~~~~~~~~~~~

Table 15.1.2-1 (Sheet 2 of 2)

Accident	Event
Inadvertent opening of a steam generator relief or safety valve	Inadvertent opening of one main steam safety or relief valve "S" actuation signal on safeguards ~~low~~ Low-2 T_{cold} Core makeup tank actuation Boron reaches core
Steam system piping failure	Steam line ruptures "S" actuation signal on safeguards ~~low~~ Low-2 steam line pressure Criticality attained Boron reaches core Pressurizer and surgeline empty

~~~~~~~~~~~~~~~~~~~~

15.2.2.1 Identification of Causes and Accident Description

* * *

If a safety limit is approached, protection is provided by ~~high~~High-2 pressurizer pressure, ~~high~~High-3 pressurizer water level, and overtemperature ΔT trips. * * *

* * *

If the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the ~~high~~High-2 pressurizer pressure signal, the ~~high~~High-3 pressurizer water level signal, or the overtemperature ΔT signal. * * *

~~~~~~~~~~~~~~~~~~~~

15.2.3.2.1 Method of Analysis

* * *

Reactor Trip

Reactor trip is actuated by the first reactor trip setpoint reached, with no credit taken for the rapid power reduction on the turbine trip. Trip signals are expected due to ~~high~~High-2 pressurizer pressure, overtemperature ΔT, ~~low~~Low-2 RCP speed, ~~high~~High-3 pressurizer water level, or low steam generator water level.

~~~~~~~~~~~~~~~~~~~~

15.2.3.2.2 Results

* * *

Minimum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

The results for these cases analyzed to address RCS pressure concerns are shown in Figure 15.2.3-15 through 15.2.3-20. In the case with offsite power available, the reactor is tripped by the ~~high~~High-2 pressurizer pressure trip function. The pressurizer safety valves are actuated in this case and maintain the reactor coolant system pressure below 110 percent of the design value.

If offsite power is lost, the reactor is tripped by the ~~low~~Low-2 reactor coolant pump speed reactor trip function. Offsite power is assumed to be lost 3 seconds after turbine trip. This causes a reduction in the reactor coolant system flow, which is illustrated in Figure 15.2.3-20.

* * *

Minimum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

* * *

The case without offsite power is tripped by the ~~low~~Low-2 reactor coolant pump speed trip function. The minimum DNBR remains above the safety analysis DNBR limit value at all times, as shown in Figure 15.2.3-6; * * *

* * *

Maximum Reactivity Feedback, With Pressurizer Spray, With and Without Offsite Power Available

* * *

The case without offsite power is tripped by the ~~low~~Low-2 reactor coolant pump speed trip function. The DNBR transient is similar to, and bounded by, the minimum feedback case with pressurizer spray and without offsite power discussed above.

* * *

Maximum Reactivity Feedback, Without Pressurizer Spray, With and Without Offsite Power Available

Figures 15.2.3-21 through 15.2.3-26 show the transient responses for the two other cases analyzed to address RCS pressure concerns, with and without offsite power available. In the case with offsite power available, the reactor is tripped by the ~~high~~High-2 pressurizer pressure function.

The case without offsite power is tripped by the ~~low~~Low-2 reactor coolant pump speed trip function. RCS pressure for both cases is shown in Figure 15.2.3-22,

* * *

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15.2.6.2.1 Method of Analysis

* * *

The assumptions used in the analysis are as follows:

* * *

- Reactor trip occurs on RCP Low-2 speed ~~low~~

* * *

- The PRHR heat exchanger is actuated by the low steam generator water level (narrow range coincident with ~~low~~Low-2 start up feed water flow).

* * *

~~~~~~~~~~~~~~~~~~~~

15.2.7.1 Identification of Causes and Accident Description

* * *

The following occurs upon loss of normal feedwater (assuming main feedwater pump fails or valve malfunctions):

* * *

- If startup feedwater is not available, the PRHR heat exchanger is actuated on either a low steam generator water level (narrow range), coincident with a ~~low~~Low-2 startup feedwater flow rate signal or a low steam generator water level (wide range) signal.
- The PRHR heat exchanger extracts heat from the reactor coolant system leading to an "S" signal on a ~~Low~~Low-2 T_{cold} signal. This actuates the core makeup * * *

~~~~~~~~~~~~~~~~~~~~

15.2.7.2.1 Method of Analysis

* * *

The assumptions used in the analysis are as follows:

* * *

- The principle safety function required after reactor trip is ...* * *

The PRHR heat exchanger is actuated by the low steam generator water level narrow range signal, coincident with ~~low~~Low-2 start up feedwater flow or by the low steam generator water level wide range signal.

~~~~~~~~~~~~~~~~~~~~

15.2.7.2.2 Results

* * *

The capacity of the PRHR heat exchanger, when the reactor coolant pumps are operating, is much larger than the decay heat, and in the first part of the transient, the reactor coolant system is cooled down and the pressurizer pressure and water volume decrease. The cool down continues until the reactor coolant temperature reaches the ~~low~~Low-2 T_{cold} setpoint. When the lowLow-2 T_{cold} setpoint is reached, the reactor coolant pumps are tripped and the core makeup tanks are actuated.

~~~~~~~~~~~~~~~~~~~~

15.2.8.1 Identification of Causes and Accident Description

* * *

* * * The case with offsite power available is not explicitly examined because, due to the fast generation of an "S" signal (generated by the ~~low~~Low-2 steam line pressure), the reactor coolant pumps would be tripped by the protection and safety monitoring system shortly after the reactor trip. The only difference between the cases with and without offsite power available would be a small difference in when the reactor coolant pumps are tripped.

The following provides the protection for a main feedwater line rupture:

- A reactor trip on any of the following four conditions:

  – HighHigh-2 pressurizer pressure

* * *

  - "S" signals from either of the following:

    - Two out of four ~~low~~Low-2 steam line pressure in either steam generator

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

15.2.8.2.1 Method of Analysis

* * *

- No credit is taken for the following four protection and safety monitoring system reactor trip signals to mitigate the consequences of the accident:

  – ~~High~~High-2 pressurizer pressure

  – Overtemperature ΔT

  – ~~High~~High-3 pressurizer water level

  – High containment pressure

The PRHR heat exchanger is initiated once the steam generator water level drops to the low steam generator level (wide range). Similarly, receipt of a ~~low~~Low-2 steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives an "S" signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.

~~~~~~~~~~~~~~~~~~~~

15.2.8.2.2 Results

* * *

After the trip, the core makeup tanks are actuated on ~~low~~Low-2 steam line pressure in the ruptured loop while the PRHR heat exchanger is actuated on a low steam generator water level (wide range).

~~~~~~~~~~~~~~~~~~~~

Table 15.2-1 (Sheet 1 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
I. Turbine trip		
A.1. With pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater	0.0
	Minimum DNBR (2.336) occurs	10.7
	Initiation of steam release from steam generator safety valves	11.5
	OTDT reactor trip setpoint reached	19.1
	Rods begin to drop	21.1
A.2. With pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
Low-2	~~Low~~ reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Minimum DNBR (1.575/1.554, typical/thimble) occurs	6.2
	Initiation of steam release from steam generator safety valves	16.6

Table 15.2-1 (Sheet 2 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
B.1. With pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	Minimum DNBR (2.393) occurs	0.0[1]
	Initiation of steam release from steam generator safety valves	11.7
	OTDT reactor trip setpoint reached	21.0
	Rod motion begins	23.0
B.2. With pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
Low-2	~~Low~~ reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Minimum DNBR (2.168/2.117 typical/thimble) occurs	5.2
	Initiation of steam release from steam generator safety valves	18.8

Table 15.2-1 (Sheet 3 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
C.1. Without pressurizer control, minimum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
High-2	~~High~~ pressurizer pressure reactor trip point reached	5.1
	Rods begin to drop	7.1
	Initiation of steam release from steam generator safety valves	8.9
	Peak RCS pressure (2728psia) occurs	8.9
C.2. Without pressurizer control, minimum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
Low-2	~~Low~~ reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Peak RCS pressure (2708 psia)occurs	6.4
	Initiation of steam release from steam generator safety valves	10.7

Table 15.2-1 (Sheet 4 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
D.1. Without pressurizer control, maximum reactivity feedback, with offsite power available	Turbine trip; loss of main feedwater flow	0.0
	~~High~~ High-2 pressurizer pressure reactor trip	5.1
	Rods begin to drop	7.1
	Peak RCS pressure (2710 psia) occurs	8.2
	Initiation of steam release from steam generator safety valves	8.8
D.2. Without pressurizer control, maximum reactivity feedback, without offsite power available	Turbine trip; loss of main feedwater	0.0
	Offsite power lost, reactor coolant pumps begin coasting down	3.0
	~~Low~~ Low-2 reactor coolant pump speed reactor trip setpoint reached	3.55
	Rods begin to drop	4.35
	Peak RCS pressure (2668 psia) occurs	6.1
	Initiation of steam release from steam generator safety valves	10.9

Table 15.2-1 (Sheet 5 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
II.A. Loss of ac power to the plant auxiliaries	Offsite ac power is lost, feedwater is lost, RCPs begin to coast down, turbine trip	0.0
	RCP ~~speed low~~ Low-2 speed reactor trip set point is reached	0.5
	Rods begin to drop	1.3
	Pressurizer safety valves open	~3.0
	Maximum pressurizer pressure reached	3.0
	Pressurizer safety valves close	~7.5
	Pressurizer safety valves open	47.0 [1]
	Steam generator 1 safety valves open	89.0[1]
	Steam generator 2 safety valves open	91.0[1]
	Maximum pressurizer water volume reached	401.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with ~~low~~ Low-2 start up flow rate)	401.0
	PRHR heat exchanger extracted heat matches decay heat	~ 18,500

* * *

Table 15.2-1 (Sheet 6 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
IIIA. Loss of normal feedwater flow	Feedwater is lost	0.0
	Low steam generator water level (narrow range) reactor trip reached	48.2
	Rods begin to drop	50.2
	Minimum DNBR is reached	51.0
	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with ~~low~~ Low-2 start up feeedwater flow rate)	110.2
	Cold leg temperature reaches ~~low~~ Low-2 T_{cold} setpoint	1,915.7
	Reactor coolant pump trip on ~~low~~ Low-2 T_{cold} "S" signal	1,922.4
	Steam line isolation on ~~low~~ Low-2 T_{cold} "S" signal	1,927.7
	Core makeup tank actuation on ~~low~~ Low-2 T_{cold} "S" signal	1,932.7

* * *

Table 15.2-1 (Sheet 7 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
III.B Loss of normal feedwater flow with a consequential loss of ac power	Feedwater is lost	10.0
	Low steam generator water level setpoint is reached	58.2
	Rods begin to drop	60.2
	Minimum DNBR is reached	61.0
	RCP trip due to loss of ac power	67.6
	Steam generator safety valves open	98.6
	Pressurizer safety valves open	~104.5
Low-2	PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with ~~low~~ start up flow rate)	120.2
	* * *	

Table 15.2-1 (Sheet 8 of 8)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (seconds)
IV. Feedwater system pipe break	Main feedwater flow to both steam generators stops due to interaction between the break and the main feedwater control system	10.0
	Low steam generator water level (narrow range) setpoint reached	60.3
	Rods begin to drop	62.3
	Reverse flow from the faulted steam generator through a full double-ended rupture starts	62.3
	Loss of offsite power	70.3
Low-2	~~Low~~ steam line pressure setpoint is reached	76.7
	* * *	

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

15.3.1.1 Identification of Causes and Accident Description

* * *

Protection against this event is provided by the ~~low~~Low-2 primary coolant flow reactor trip signal, which is actuated by two-out-of-four ~~low-~~Low-2 flow signals. Above permissive P10, ~~low~~Low-2 flow in either hot leg actuates a reactor trip (see Section 7.2).

~~~~~~~~~~~~~~~~~~~~

15.3.1.2.6 Results

* * *

The affected reactor coolant pumps coast down and the core flow reaches a new equilibrium value. The plant is tripped by the ~~low-~~Low-2 flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values.

~~~~~~~~~~~~~~~~~~~~

15.3.2.1 Identification of Causes and Accident Description

* * *

The following signals provide protection against this event:

- Low-2 Reactor coolant pump ~~under~~speed

* * *

The reactor trip on Low-2 reactor coolant pump ~~under~~speed protects against conditions that can cause a loss of voltage to two-out-of-four reactor coolant pumps. This function is blocked below approximately 10-percent power (permissive P10). The reactor trip on Low-2 reactor coolant pump ~~under~~speed also protects against an underfrequency condition resulting from frequency disturbances on the power grid, * * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

15.3.2.2.1 Method of Analysis

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps.

For the scenario of a complete loss of voltage, which results in all the reactor coolant pumps coasting down, the method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in subsection 15.3.1, with two exceptions. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the Low-2 reactor coolant pump ~~under~~speed trip instead of the ~~low~~Low-2 primary coolant flow trip. * * *

~~~~~~~~~~~~~~~~~~~~

15.3.2.2.2 Results

Figures 15.3.2-1 through 15.3.2-6 show the transient response for the complete loss of voltage to all four reactor coolant pumps. The reactor is tripped on the Low-2 reactor coolant pump ~~under~~speed signal. Figure 15.3.2-6 demonstrates that the DNBR is always greater than the safety analysis limit value * * *

~~~~~~~~~~~~~~~~~~~~

15.3.3.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor, as discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a ~~low-~~Low-2 flow signal. * * *

~~~~~~~~~~~~~~~~~~~~

15.3.4.1 Identification of Causes and Accident Description

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a ~~low-~~Low-2 flow signal in the affected loop.

* * *

~~~~~~~~~~~~~~~~~~~~

Table 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

Accident	Event	Time (seconds)
Partial loss of forced reactor coolant flow		
– Loss of two pumps with four pumps running	Two pumps lose power and begin coasting down ~~Low~~ Low-2 flow reactor trip setpoint reached Rods begin to drop Minimum DNBR occurs	0.00 1.45 3.42 5.50
Complete loss of forced reactor coolant		
– Loss of four pumps with four pumps running	All pumps lose power and begin coasting down Low-2 Reactor coolant pump ~~under~~speed trip setpoint reached Rods begin to drop Minimum DNBR occurs	0.00 0.55 1.35 3.20
Reactor coolant pump shaft seizure (locked rotor)		
– One locked rotor with four pumps running without offsite power available	Rotor on one pump locks ~~Low~~ Low-2 flow trip point reached Rods begin to drop Maximum reactor coolant system pressure occurs Maximum cladding temperature occurs	0.00 0.10 1.55 3.40 4.10

~~~~~~~~~~~~~~~~~~~~

15.4.2.1 Identification of Causes and Accident Description

* * *

The automatic features of the PMS that prevent core damage following the postulated accident include the following:

* * *

- A ~~high~~High-2 pressurizer pressure reactor trip is actuated from any two out of four pressure divisions when a set pressure is exceeded. This set pressure is less than the set pressure for the pressurizer safety valves.

- A ~~high~~High-3 pressurizer water level reactor trip is actuated from any two out of four level divisions that exceed the setpoint when the reactor power is above approximately 10 percent (permissive-P10).
~~~~~~~~~~~~~~~~~~~~

* * *

The area of permissible operation (power, pressure, and temperature) is bounded by the combination of reactor trips:

- High neutron flux (fixed setpoint)
- ~~High~~High-2 pressurizer pressure (fixed setpoint)
- ~~Low~~Low-2 pressurizer pressure (fixed setpoint)
- Overpower and overtemperature ΔT (variable setpoints)

~~~~~~~~~~~~~~~~~~~~~~~~~

15.5.1.1 Identification of the Causes and Accident Description

* * *

The PRHR heat exchanger extracts heat from the reactor coolant system leading to an "S" signal on a ~~Low~~Low-2 T_{cold} signal. The PRHR heat exchanger may inject asymmetrically into the steam generator outlet plenum such that a higher percentage of the PRHR flow is in one of two cold legs coming from the steam generator on the PRHR loop. To account for this, the analysis assumes that the ~~Low~~Low-2 T_{cold} setpoint is reached coincident with * * *

~~~~~~~~~~~~~~~~~~~~~~~~~

15.5.1.2 Analysis of Effects and Consequences

* * *

- Moderator and Doppler coefficients of reactivity

  A least-negative moderator temperature coefficient, a Low (absolute value) Doppler power coefficient, and a minimum boron worth are assumed. With these minimum feedback parameters and the operability of the pressurizer spray system and automatic rod control system assumed, the reactivity effects of the boron injection from the core makeup tanks is counteracted. As a result, the ~~high-3~~High-3 pressurizer water level signal is the first reactor trip signal generated during the transient.

~~~~~~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~~~~~~

15.5.1.3 Results

* * *

If the PRHR heat exchanger coolant asymmetrically injects into the steam generator outlet plenum, then one cold leg could reach the ~~Low~~Low-2 T_{cold} “S” setpoint more quickly than if the flow were split evenly. To conservatively account for this effect, the ~~Low~~Low-2 T_{cold} “S” signal is modeled to actuate simultaneously with the actuation of the PRHR heat exchanger (2,768.6 seconds). The ~~Low~~Low-2 T_{cold} “S” signal activates the second core makeup tank, which then begins injecting additional

* * *

As noted above, the limiting case presented here models explicit operator action 45 minutes after receipt of the high-2 pressurizer level signal. For pressurizer level increase events, the operator would take action to reduce the increase in coolant inventory. As the pressurizer water level would increase above the ~~high~~High-2 pressurizer water level that normally isolates chemical and volume control system makeup ~~(high-2)~~, the normal letdown line could be placed into service to reduce the increase in coolant inventory. * * *

~~~~~~~~~~~~~~~~~~~~

15.5.2.1 Identification of Causes and Accident Description

* * *

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an “S” signal will be generated by either the ~~low~~Low-2 T_{cold} or ~~low~~Low-2 steam line pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similarly to, and is only slightly more limiting than, a spurious “S” signal event. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a ~~high~~High-2 pressure reactor trip signal, no “S” signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

Under typical operating conditions for the AP1000, the boron concentration of the injected chemical and volume control system water is equal to that of the reactor coolant system. If the chemical and volume control system is functioning in this manner and the pressurizer spray system functions properly to prevent a ~~high~~High-2 pressure reactor trip signal, no “S” signals are generated and this specific event is also terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

* * *

This scenario is as follows:

* * *
~~~~~~~~~~~~~~~~~~~~

- The non-safety-related pressurizer spray is assumed to be available, so that a ~~high~~High-2 pressurizer pressure reactor trip is prevented.

Due to the boron addition in the core, the plant cools down until an “S” signal is generated on ~~low~~Low-2 cold leg temperature. On the “S” signal * * *

~~~~~~~~~~~~~~~~~~~~

15.5.2.2 Analysis of Effects and Consequences

* * *

- Pressurizer spray

The spray system controls the pressurizer pressure so that a ~~high~~High-2 pressurizer pressure reactor trip is prevented.

- Boron injection

After 10 seconds at steady state, the chemical and volume control system pumps start injecting borated water, which is slightly above the reactor coolant system boron concentration. Upon receipt of an “S” signal, the core makeup tanks begin injecting 3400 ppm borated water. The chemical and volume control system pumps are isolated on high-2 pressurizer level. In this analysis the boron concentration of the chemical and volume control system is iterated upon until the high-2 pressurizer level and the ~~low~~Low-2 T_{cold} “S” setpoint are reached at the same time. This begins core makeup tank injection when the chemical and volume control system pumps are isolated, which is conservative with respect to filling the pressurizer

* * *

- Protection and safety monitoring system actuations

If the automatic rod control system is modeled and the pressurizer spray system functions properly, no reactor trip signal is expected to occur. Instead, the event is terminated by automatic isolation of the chemical and volume control system on the safety grade high-2 pressurizer level setpoint. If the automatic rod control system is not active and the pressurizer spray system is assumed to be available, reactor trip may be initiated on either ~~low~~Low-2 T_{cold} “S” or a ~~low~~Low-2 steam line pressure “S” signal.

~~~~~~~~~~~~~~~~~~~~

15.5.2.3 Results

* * *

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At 2,271.3 seconds, the ~~low~~Low-2 T_{cold} setpoint is reached, the reactor trips on the resulting “S” signal, and * * *

* * *

* * * For pressurizer level increase events, the operator could take other actions to reduce the increase in coolant inventory. As the pressurizer water level would increase above the ~~high~~High-2 pressurizer water level that normally isolates chemical and volume control system makeup, the normal letdown line could be placed into service to reduce the increase in coolant inventory.

~~~~~~~~~~~~~~~~~~~~

15.5.2.4 Conclusions

* * *

If the automatic rod control system and the pressurizer spray systems are assumed to function, no reactor trip signal is expected to occur. Instead, the event would be terminated by automatic isolation of the chemical and volume control system on the safety grade ~~high-2~~High-2 pressurizer level setpoint. If manual rod control is assumed and the pressurizer spray system is assumed to be unavailable, reactor trip may be initiated on either a ~~high~~High-2 pressurizer pressure, ~~low~~Low-2 T_{cold} “S”, or a ~~low~~Low-2 steamline pressure “S” signal.

~~~~~~~~~~~~~~~~~~~~

Table 15.5-1 (Sheet 1 of 2)

Accident	Event
Inadvertent operation of the core makeup tanks during power operation	Core makeup tank discharge valves open
	High-2 pressurizer level setpoint reached
	High-3 pressurizer level setpoint reached
	Rod motion begins
	Loss of offsite power
	Reactor coolant pumps trip
	High-3 pressurizer level setpoint reached
	PRHR heat exchanger actuated
Low-2	~~Low~~ T_{cold} "S" setpoint is reached
	Second CMT starts recirculating

Table 15.5-1 (Sheet 2 of 2)

Accident	Event
Chemical and volume control system malfunction that increases reactor coolant inventory	Chemical and volume control system charging pumps start
Low-2	~~Low~~ T_{cold} "S" signal and high-2 pressurizer level signals are reached
	Core makeup tank discharge valves open

~~~~~~~~~~~~~~~~~~~~~~~~

15.6.1.1 Identification of Causes and Accident Description

* * *

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer ~~low~~Low-2 pressure
~~~~~~~~~~~~~~~~~~~~~~~~

~~~~~~~~~~~~~~~~~~~~

15.6.1.2.2 Results

* * *

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the ~~low~~Low-2 pressurizer pressure reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

~~~~~~~~~~~~~~~~~~~~

15.6.1.3 Conclusion

The results of the analysis show that the ~~low~~Low-2 pressurizer pressure reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR * * *

~~~~~~~~~~~~~~~~~~~~

15.6.3.1.1 Introduction

* * *

The AP1000 design provides automatic protective actions to mitigate the consequences of an SGTR. The automatic actions include reactor trip, actuation of the passive residual heat removal (PRHR) heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of chemical and volume control system flow and startup feedwater flow on ~~high-2~~High-3 steam generator level or high steam generator level coincident with reactor trip (P-4).

~~~~~~~~~~~~~~~~~~~~

15.6.3.1.2 Sequence of Events for a Steam Generator Tube Rupture

* * *

- Continued loss of reactor coolant inventory leads to a reactor trip generated by a ~~low~~Low-2 pressurizer pressure or over-temperature ΔT signal. Following reactor trip, the SGTR leads to a decrease in reactor coolant pressure and pressurizer level, counteracted by chemical and volume control system flow and pressurizer heater operation. A safeguards (“S”) signal from ~~low~~Low-3 pressurizer pressure, actuates the core makeup tanks. * * *

~~~~~~~~~~~~~~~~~~~~

15.6.3.1.3 Steam Generator Tube Rupture Automatic Recovery Actions

* * * secondary level increases as break flow accumulates in the steam generator. Eventually, the ruptured steam generator secondary level reaches the high and ~~high-2~~High-3 steam generator narrow range level setpoint, which is near the top of the narrow range level span.

~~~~~~~~~~~~~~~~~~~~

15.6.3.2.1.2 Analysis Assumptions

* * *

* * * The chemical and volume control system and pressurizer heater modeling is conservatively chosen to delay the ~~low~~Low-3 pressurizer pressure "S" and the low-2 pressurizer level signal and associated protection system actions.

* * *

The valve is subsequently isolated when the associated block valve is automatically closed on a ~~low~~Low-2 steam line pressure protection system signal.

~~~~~~~~~~~~~~~~~~~~

15.6.3.2.1.3 Results

* * *

The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure (Figures 15.6.3-1 and 15.6.3-2). Depressurization of the primary and secondary systems continues until the ~~low~~Low-2 steam line pressure setpoint is reached. As a result, the steam line isolation valves and intact and ruptured steam generator power-operated relief block valves are closed.

~~~~~~~~~~~~~~~~~~~~

15.6.5.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer ~~low~~Low-2-pressure trip setpoint is reached. A safeguards actuation ("S") signal is generated when the * * *

~~~~~~~~~~~~~~~~~~~~

15.6.5.2.2 Description of Small-Break LOCA Transient

* * *

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer ~~low~~Low-2-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer ~~low~~Low-3-pressure setpoint is reached. * * *

~~~~~~~~~~~~~~~~~~~~

15.6.5.4B Small-Break LOCA Analyses

Should a small break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer ~~low~~Low-2-pressure trip setpoint is reached. An "S" signal is generated when the appropriate setpoint is reached. * * *

~~~~~~~~~~~~~~~~~~~~

15.6.5.4B.1 Description of Small-Break LOCA Transient

* * *

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer ~~low~~Low-2 pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer ~~low~~Low-3-pressure setpoint is reached. * * *

~~~~~~~~~~~~~~~~~~~~

Table 15.6.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (seconds)
Inadvertent opening of a pressurizer safety valve	Pressurizer safety valve opens fully	0.00
	~~Low~~ Low-2 pressurizer pressure reactor trip setpoint reached	15.50
	Rods begin to drop	17.50
	Minimum DNBR occurs	18.30
Inadvertent opening of two ADS Stage 1 trains	ADS valves begin to open	0.00
	~~Low~~ Low-2 pressurizer pressure reactor trip setpoint reached	17.83
	Rods begin to drop	19.83
	ADS valves fully open	20.00
	Minimum DNBR occurs	20.70

~~~~~~~~~~~~~~~~~~~~~

Table 15.6.3-1

* * *

Events
* * *
Ruptured steam generator power-operated relief valve block valve closes on ~~low~~ Low-2 steam line pressure signal
Chemical and volume control system isolated on ~~high-2~~ High steam generator narrow range level setpoint
Break flow terminated

~~~~~~~~~~~~~~~~~~~~~

Table 15.6.5-10

AP1000 ADS PARAMETERS

Actuation Signal [percentage of core makeup tank level]		Actuation Time (seconds)	Minimum Valve Flow Area (for each path, in^2)	Number of Paths	Valve Opening Time (seconds)
Stage 1 – Control ~~Low 1~~ Low-3	67.5	32 after CMT ~~Low 1~~ Low-3	4.6	2 out of 2	≤ 40
* * *					
* * *					
* * *					

Notes:

1. The interlock requires coincidence of CMT ~~Low-2~~ Low-6 level as well as 128 seconds after the Stage 3 actuation signal is generated.

* * *

~~~~~~~~~~~~~~~~~~~~~

Table 19.59-18 (Sheet 13 of 25)
AP1000 PRA-Based Insights

Insight
9. The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution and to preserve containment integrity by isolation of the CVS lines penetrating the containment. The CVS provides a nonsafety-related means to perform the following functions: – Makeup water to the RCS during normal plant operation. – Boration following a failure of reactor trip – Makeup water to the pressurizer auxiliary spray line. Two makeup pumps are provided. Each pump provides capability for normal makeup. Two safety-related air-operated valves provide isolation of normal CVS letdown during shutdown operation on ~~low~~ Low-2 hot leg level.

~~~~~~~~~~~~~~~~~~~~~

19E.2.1.2.2 RCS Instrumentation

RCS Hot Leg Level

* * *

These channels provide signals for the following protection functions:

- Isolation of letdown on ~~low~~Low-4 level on a one-out-of-two basis.
- Actuation of fourth-stage ADS valves on ~~low~~Low-4 (empty) hot leg level on a two-out-of-two basis. Actuation of fourth-stage ADS causes actuation of IRWST injection.

~~~~~~~~~~~~~~~~~~~~

19E.2.1.2.4 Improved RCS Draindown Method

* * *

* * * the control room if the RCS level falls below the automatic letdown valve closure setpoint so that the operator is alerted to manually isolate the letdown line. Furthermore, an automatic isolation of the letdown line is actuated on ~~low~~Low-2 hot leg level. This draindown method provides a reliable means of attaining mid-loop conditions.

~~~~~~~~~~~~~~~~~~~~

19E.2.2.2.2 Safety-Related Actuation in Shutdown Modes

The AP1000 has safety-related actuations associated with * * * If actuated, this signal causes the MSIVs to close to terminate the blowdown of the SGS following a steam line break. This signal is placed into service below the setpoint that disables the ~~low~~Low-2 steam line pressure signal (P11) that actuates steam line isolation as discussed in Section 7.3. When the operator manually blocks the ~~low~~Low-2 steam line pressure signal, the steam line high pressure-negative rate signal is automatically enabled.

~~~~~~~~~~~~~~~~~~~~

19E.2.3.2.1 Core Makeup Tanks

The CMTs provide RCS makeup. During shutdown, the CMTs are available in Modes 3, 4, and 5, until the RCS pressure boundary is open and the pressurizer water level is reduced. During power operation, the CMTs are automatically actuated on various signals including a safeguards actuation signal (low RCS pressure, low RCS temperature, ~~low~~Low-2 steam line pressure, and high containment pressure) and on low pressurizer water level. See Chapter 7 for a description of the AP1000 PMS actuation logic. In shutdown modes, portions of the safeguards actuation signal are disabled to allow the RCS to be cooled and depressurized for
~~~~~~~~~~~~~~~~~~~~

shutdown. For instance, the low RCS pressure and temperature, and ~~low~~Low-2 steam line pressure signals are blocked in Mode 3 prior to cooling and depressurizing the RCS. Therefore, during shutdown Modes 3, 4, and 5, the primary signal that actuates the CMTs due to a loss of inventory is the pressurizer level signal. In Mode 5, with the RCS open (in preparation for reduced inventory operations), the low pressurizer level signal is blocked prior to draining the pressurizer. Therefore, in Mode 5 with the RCS open, the CMTs are not required to be available and the RCS makeup function is provided by the IRWST.

The CMTs also provide an emergency boration function for accidents such as steam line breaks. However, the signals that provide the primary protection for this function (~~low~~Low-2 steam line pressure, low RCS pressure, and low RCS temperature) are blocked in Mode 3 as discussed above. Prior to blocking these signals in Mode 3,

~~~~~~~~~~~~~~~~~~~~

19E.2.3.2.3 In-containment Refueling Water Storage Tank

* * *

The IRWST injection paths are actuated on a ~~low-2~~Low-6 CMT water level. This signal is available in shutdown Modes 3, 4, and 5, with the RCS intact. When the RCS is open to transition to reduced inventory operations, the CMT actuation logic on low pressurizer level is removed, and the CMTs can be taken out of service. For these modes, automatic actuation of the IRWST can be initiated (on a two-out-of-two basis) on ~~low~~Low-4 hot leg level.

~~~~~~~~~~~~~~~~~~~~

19E.2.3.3.1 Operation During Loss of Normal Residual Heat Removal Cooling During Mid-loop Events

* * *

The IRWST injection squib valves and fourth stage ADS valves are automatically opened if the RCS hot leg level indication decreases below a ~~low~~Low-4 setpoint. A time delay is provided to provide time for the operators

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

19E.4.2.1 Feedwater System Malfunctions Which Increase Heat Removal from the Primary System

* * *

Additional PMS functions are provided to detect and protect against asymmetrical feedwater system malfunctions. Automatic reactor trip, closure of the main feedwater control and isolation valves, closure of the startup feedwater control and isolation valves, tripping of the booster/main feedwater pumps, and tripping of the startup feedwater pumps occur if the level in a single steam generator is above the ~~high-2~~High-3 water level setpoint. Similar actions occur if cold leg temperature in a single RCS loop decreases below the ~~low~~Low-2 T_{cold} setpoint. The ~~high-2~~High-3 steam generator level setpoint is active in Modes 1 through 4 unless the various feedwater valves are closed. This ensures that the steam generators cannot inadvertently be overfilled. The ~~low~~Low-2 T_{cold} signal is available in Modes 1 through 3. In Mode 3 prior to blocking the ~~low~~Low-2 T_{cold} signal, the RCS must be borated to cold shutdown conditions. With the RCS borated, no feedwater malfunction can be postulated to cool the RCS such that a core power excursion would occur.

~~~~~~~~~~~~~~~~~~~~

19E.4.2.4 Inadvertent PRHR HX Operation

* * *

* * * the cold leg temperature dropping below the ~~low~~Low-2 T_{cold} safeguards signal setpoint. This function actuates a reactor trip, initiates boration by the CMTs, and most importantly, trips all the RCPs. When the RCPs trip, natural circulation flow begins in the RCS and the PRHR HX loop. When natural circulation flow is initiated, the heat removal capability of the PRHR HX decreases to approximately 1.5 percent of full power and the severity of the transient is minimized. With the RCS in natural circulation, the cooldown rate of the RCS is also slowed. If criticality is obtained, boration by the CMTs will bring the core subcritical again.

The ~~low~~Low-2 T_{cold} safeguards signal may be blocked by the operator in Mode 3 to allow plant depressurization and cooldown to lower modes. However, prior to blocking the ~~low~~Low-2 T_{cold} safeguards signal, the RCS is borated to the shutdown margin requirements at cold shutdown (200°F). * * *

~~~~~~~~~~~~~~~~~~~~

19E.4.3.2 Loss of ac Power

A discussion and an analysis of a loss of ac power event are provided in Subsection 15.2.6. The loss of ac power results in the loss of forced primary coolant flow and the loss of main feedwater flow. This results in a heatup and pressurization of the RCS. If the reactor is at power, the event is mitigated by tripping the reactor. The reactor may be automatically tripped on ~~low~~Low-2 RCP speed, ~~low~~Low-2 RCS flow, low steam generator level, or several other primary side heatup signals. Also reactor trip may occur due to the loss of power to the control rod drive mechanisms.

~~~~~~~~~~~~~~~~~~~~

19E.4.4.1 Partial and Complete Loss of Forced RCS Flow

* * *

Protection for loss of forced RCS flow events is provided by tripping the reactor. This reduces reactor power and preserves margin-to-DNB limits. The AP1000 PMS includes a reactor trip on ~~low~~Low-2 RCS flow in any cold leg and a reactor trip on ~~low~~Low-2 RCP speed in any two of four RCPs. These two reactor trips are used to detect all possible partial and complete loss of RCS flow transients. * * *

~~~~~~~~~~~~~~~~~~~~

19E.4.7.3 Steam Generator Tube Rupture in Lower Modes

* * *

* * * Therefore, the initial margin to overfill directly impacts the final margin. For the AP1000, the primary cooldown and depressurization occur automatically when the PRHR HX is actuated on a ~~low~~Low-3 pressurizer pressure “S” signal or low pressurizer level * * *

~~~~~~~~~~~~~~~~~~~~

19E.4.8 Loss-of-Coolant Accident Events in Shutdown Modes

* * *

* * * The ~~low~~Low-3 pressurizer pressure safeguards signal is also assumed to be disabled because the initial pressure is below the setpoint.

~~~~~~~~~~~~~~~~~~~~

19E.4.10.2 Shutdown Temperature Evaluation

* * *

Summarizing this transient, the loss of normal ac power occurs, followed by the reactor trip. The PRHR heat exchanger is actuated on the low steam generator narrow range level coincident with ~~low~~Low-2 startup feed water flow rate signal. Eventually a safeguards actuation signal is actuated on ~~Low~~Low-2 cold leg temperature and the CMTs are actuated.

~~~~~~~~~~~~~~~~~~~~

Table 19E.4.10-1

Event
* * *
* * *
* * *
~~Low~~ Low-2 T_{cold} Setpoint Reached
Steam Line Isolation on ~~Low~~ Low-2 T_{cold} Signal
CMTs Actuated on ~~Low~~ Low-2 T_{cold} Signal
IRWST Reaches Saturation Temperature

~~~~~~~~~~~~~~~~~~~~

UFSAR Appendix 7A, "INSTRUMENTATION AND CONTROLS LICENSING BASIS DOCUMENT CHANGES":

[REVIEWER'S NOTE: For clarity, entirely new text that is being added to UFSAR Appendix 7A is depicted in blue font in these markups; new markups to add or delete text to the existing Appendix 7A text are identified with blue underlined font or red strike-out font, respectively; and existing markups in Appendix 7A that are unchanged by this LAR are depicted with black font (underlined or strike-out) as per the current UFSAR Appendix 7A.]

7A.1 WCAP-15775, AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report

* * *

- Revise Section 4.2, Determining Diversity – Guideline 2, under diversity aspect number 4, Human Diversity, as follows:

* * *

- Revise Section 5.4, "Reactor Coolant Inventory Control" section as follows:

5.4.1 During normal plant operation, the pressurizer level control function of the PLS automatically controls the operation of the nonsafety chemical and volume control system (CVS) to maintain RCS inventory. In the event of a small RCS leak, the CVS makeup pumps automatically start on a ~~low~~ <u>Low-2</u> pressurizer level signal."

* * *

5.4.9 During shutdown operations, the IRWST discharge isolation valves are normally closed with actuation power available. The PMS automatically opens these valves to initiate IRWST injection on a ~~low-low~~ <u>Low-4</u> RCS hot leg level.

* * *

- Revise Section 5.5, "Core Decay Heat Removal" section as follows:

* * *

5.5.6 During plant conditions when the RCS is not intact or with reduced RCS inventory (such as midloop operation), the RNS is normally operating and will automatically restart when power is restored following a loss of power to the RNS pumps. Various PXS components including the CMTs, accumulators, and PRHR heat exchangers are not available. The IRWST will automatically actuate on ~~low-low~~ <u>Low-4</u> RCS hot leg level. The IRWST can also be manually actuated.

- Revise Section 6, References, by adding Reference 9, as follows:

* * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

7A.8 WCAP-16675-P and WCAP-16675-NP, AP1000 Protection and Safety Monitoring System Architecture Technical Report

The UFSAR incorporates by reference Tier 2 documents WCAP-16675-P and WCAP-16675-NP, AP1000 Protection and Safety Monitoring System Architecture Technical Report. See Table 1.6-1. WCAP-16675, Revision 5, includes the following revisions and additions as indicated by strikethroughs and underlines.

- Revised Section 1.1, "Reactor Trip Functions" section as follows:

  The PMS generates an automatic reactor trip for the following conditions:

  * * *

  8. Reactor Trip on Pressurizer Low-2 ~~Pressurizer~~ Pressure as described in Reference 9.
  9. Reactor Trip on Low-2 Reactor Coolant Flow as described in Reference 9.
  10. Reactor Trip on Reactor Coolant Pump Low-2 Speed ~~Underspeed~~ as described in Reference 9.
  11. High-2 Reactor Coolant Pump Bearing Water Temperature Trip as described in Reference 9.
  12. Pressurizer High-2 Pressure Reactor Trip as described in Reference 9.
  13. High-3 Pressurizer ~~High~~ Water Level Reactor Trip as described in Reference 9.
  15. ~~High-2~~High-3 Steam Generator Water Level in any Steam Generator as described in Reference 9.

- Revise Section 1.3, "Qualified Data Processing System Functions" as follows:

  * * *

Southern Nuclear Operating Company

ND-17-0295

Enclosure 4

Vogtle Electric Generating Plant (VEGP) Units 3 and 4

Conforming Technical Specification Bases Changes

(LAR-17-004)

(For Information Only)

Note:

Added text is shown as Blue Underline

Deleted text is shown as ~~Red Strikethrough~~

Omitted text is shown as three asterisks (* * *)

(Enclosure 4 consists of 19 pages, including this cover page)

Revise Technical Specifications Bases Section 2.1.2 as shown below:

* * *

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor ~~high~~High 2 pressurizer pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

* * *

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor ~~high~~High 2 pressurizer pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). The safety analyses for both the ~~high~~High 2 pressurizer pressure trip and * * *

~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.1.9 as shown below:

* * *

APPLICABILITY

* * *... In the applicable MODES, the need to isolate the CVS makeup to the RCS is detected by the pressurizer level instruments (~~high 1~~High setpoint coincident with safeguards actuation or high 2 setpoint) or the steam generator narrow range level instruments (high setpoint coincident with reactor trip (P-4) or ~~high 2~~High 3 setpoint).

~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.3.1 as shown below:

* * *

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

* * *

Power Range Neutron Flux, P-10

* * *:

(1) On increasing power, the P-10 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure – ~~Low~~Low 2 Setpoint,
- Pressurizer Water Level – High 3,

- Reactor Coolant Flow – ~~Low~~Low 2, and
- RCP Speed – ~~Low~~Low 2.

* * *

(5) On decreasing power, the P-10 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure – ~~Low~~Low 2 Setpoint,
- Pressurizer Water Level – High 3,
- Reactor Coolant Flow – ~~Low~~Low 2, and
- RCP Speed – ~~Low~~Low 2.

* * *

Pressurizer Pressure, P-11

With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Steam Generator Narrow Range Water Level – ~~High 2~~High 3 reactor Trip. This allows rod testing with the steam generators in cold wet layup. With pressurizer pressure channels greater than P-11 setpoint, the Steam Generator Narrow Range Water Level – ~~High 2~~High 3 reactor Trip is automatically enabled. The operator can also enable these actuations by use of the respective manual reset.

* * *

5. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure – ~~High~~High 2 and – ~~Low~~Low 2 trips and the Overtemperature ΔT trip.

a. Pressurizer Pressure – ~~Low~~Low 2

The Pressurizer Pressure – ~~Low~~Low 2 trip Function ensures that * * *

The LCO requires four channels of Pressurizer Pressure – ~~Low~~Low 2 to be OPERABLE in MODE 1 above * * *

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – ~~Low~~Low 2 trip must be OPERABLE. * * *

b. Pressurizer Pressure – ~~High~~High 2

The Pressurizer Pressure – ~~High~~High 2 trip Function ensures that protection is provided against overpressurizing * * *

The LCO requires four channels of the Pressurizer Pressure – ~~High~~High 2 to be OPERABLE in MODES 1 and 2. * * *

In MODE 1 or 2, the Pressurizer Pressure – ~~High~~High 2 trip must be OPERABLE to help prevent RCS overpressurization and LCOs, and minimizes challenges to the safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure – ~~High~~High 2 trip Function does not have to be OPERABLE * * *

6. Pressurizer Water Level – High 3

The Pressurizer Water Level – High 3 trip Function provides a backup signal for the Pressurizer Pressure – ~~High 3~~High 2 trip and also provides protection against water relief through * * *

* * *

7. Reactor Coolant Flow – ~~Low~~Low 2

The Reactor Coolant Flow – ~~Low~~Low 2 trip Function ensures * * *

* * *

The LCO requires four Reactor Coolant Flow – ~~Low~~Low 2 channels per hot leg to be OPERABLE in MODE 1 above P-10. * * *

In MODE 1 above the P-10 setpoint, when a loss of flow in one RCS hot leg could result in DNB conditions in the core, the Reactor Coolant Flow – ~~Low~~Low 2 trip must be OPERABLE.

8. Reactor Coolant Pump (RCP) Bearing Water Temperature – ~~High~~High 2

The RCP Bearing Water Temperature – ~~high~~High 2 reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS cold leg. The Trip Setpoint reflects only steady state instrument uncertainties as the detectors do not provide primary protection for any event that results in a harsh environment.

The LCO requires four RCP Bearing Water Temperature – ~~High~~High 2 channels per RCP to be OPERABLE in MODE 1 or 2. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

In MODE 1 or 2, when a loss of flow in any RCS cold leg could result in DNB conditions in the core, the RCP Bearing Water Temperature – ~~High~~High 2 trip must be OPERABLE.

9. Reactor Coolant Pump Speed – ~~Low~~Low 2

The RCP Speed – ~~Low~~Low 2 trip Function ensures that * * *

The LCO requires four RCP Speed – ~~Low~~Low 2 channels (one per pump) to be OPERABLE in MODE 1 above P-10. * * *

In MODE 1 above the P-10 setpoint, the RCP Speed – ~~Low~~Low 2 trip must be OPERABLE. Below the P-10 setpoint, all * * *

* * *

11. Steam Generator Narrow Range Water Level – ~~High 2~~High 3

The SG Narrow Range Water Level – ~~High 2~~High 3 trip * * *

The LCO requires four channels of SG Narrow Range Water Level – ~~High 2~~High 3 per SG to be OPERABLE. Four channels * * *

In MODES 1 and 2 above the P-11 interlock, the SG Narrow Range Water Level – ~~High 2~~High 3 trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater System (non-safety related). The Main Feedwater System is only in operation in MODES 1 and 2. In MODE 3, 4, 5, or 6, the SG Narrow Range Water Level – ~~High 2~~High 3 Function does not have to be * * *

* * *

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.3.8 as shown below:

* * *

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure – ~~Low~~Low 3 is a primary actuation signal for small loss of coolant accidents (LOCAs) and * * *

* * *

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY (continued)

Pressurizer Pressure, P-11

The P-11 interlock permits a normal unit cooldown and depressurization without Safeguards Actuation or main steam line and feedwater isolation. With pressurizer pressure channels less than the P-11 setpoint, the operator can manually block the Pressurizer pressure – ~~Low~~Low 3, Steam Line Pressure – ~~Low~~Low 2, and T_{cold} – ~~Low~~Low 2 Safeguards Actuation signals and the Steam Line Pressure – ~~Low~~Low 2 and T_{cold} – ~~Low~~Low 2 steam line isolation signals. When the Steam Line Pressure – ~~Low~~Low 2 is manually blocked, a main steam isolation signal on Steam Line Pressure-Negative Rate – High is enabled. This provides protection for an SLB by closure of the main steam isolation valves. Manual block of feedwater isolation on T_{avg} – ~~Low 1~~Low, Low 2, and T_{cold} – ~~Low~~Low 2 is also permitted below P-11. With pressurizer pressure channels ≥ P-11 setpoint, the Pressurizer Pressure – ~~Low~~Low 3, Steam Line Pressure – ~~Low~~Low 2, and T_{cold} – ~~Low~~Low 2 Safeguards Actuation signals and the Steam Line Pressure ~~Low~~Low 2 and T_{cold} – ~~Low~~Low 2 steam line isolation signals are automatically enabled. The feedwater isolation signals on T_{cold} – ~~Low~~Low 2, T_{avg} – ~~Low 1~~Low and Low 2 are also automatically enabled above P-11. The operator can also enable these signals by use of the respective manual reset buttons. When the Steam Line Pressure – ~~Low~~Low 2 and T_{cold} – ~~Low~~Low 2 steam line isolation signals * * *

Pressurizer Level, P-12

* * * P-12 setpoint, the operator can manually block Pressurizer Water Level – ~~Low 1~~Low and Pressurizer Water Level – Low 2 signals used for these actuations. Concurrent with blocking CMT actuation on Pressurizer Water Level – Low 2, ADS 4^{th} Stage actuation on ~~Low 2~~Low 4 RCS hot leg level is enabled. Also CVS letdown isolation on ~~Low 1~~Low 2 RCS hot leg level is enabled. When the pressurizer level is above * * *

* * *

Safeguards Actuation

* * *

These other Functions ensure:

- * * * ; and
- Enabling automatic depressurization of the RCS on CMT Level – ~~Low 1~~Low 3 to ensure continued safeguards actuated injection.

Safeguards Actuation is initiated by the following signals:

- Containment Pressure – High 2;
- Pressurizer Pressure – ~~Low~~Low 3;
- RCS Cold Leg Temperature (T_{cold}) – ~~Low~~Low 2;
- Steam Line Pressure – ~~Low~~Low 2; and
- Safeguards Actuation – Manual Initiation.

* * *

Containment Air Filtration System Isolation

* * *

Containment Air Filtration System Isolation is actuated on the following signals:

- Containment Radioactivity – ~~High 1~~High; and
- Containment Isolation Actuation.

* * *

Steam Line Isolation

* * *

Steam Line Isolation is actuated by the following signals:

- Containment Pressure – High 2;
- RCS Cold Leg Temperature (T_{cold}) – ~~Low~~Low 2;
- Steam Line Pressure – ~~Low~~Low 2;
- Steam Line Pressure – Negative Rate – High; and

- Steam Line Isolation – Manual Initiation.

SG Power Operated Relief Valve and Block Valve Isolation

The Function of the SG Power Operated Relief Valve and Block Valve
* * *
the PORV and/or the block valve which receive a close signal on ~~low~~Low 2 steam line pressure. Additionally, the PORV flow path can be isolated manually.

SG Power Operated Relief Valve and Block Valve Isolation is actuated by the following signals:

- Steam Line Pressure – ~~Low~~Low 2; and
- SG Power Operated Relief Valve and Block Valve Isolation – Manual Initiation.

* * *

Turbine Trip

* * *

Turbine Trip is actuated by the following signals:

- SG Narrow Range Water Level – ~~High 2~~High 3;
- Reactor Trip Signal (P-4); and
- Feedwater Isolation – Manual Initiation.

* * *

Main Feedwater Control Valve Isolation

* * *

Main Feedwater Control Valve Isolation is actuated by the following signals:

- SG Narrow Range Water Level – ~~High 2~~High 3;
- Safeguards Actuation;
- Reactor Coolant Average Temperature (T_{avg}) – ~~Low 1~~Low coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

Main Feedwater Pump Trip and Valve Isolation

* * *

Main Feedwater Pump Trip and Valve Isolation is actuated by the following signals:

- SG Narrow Range Water Level – ~~High 2~~High 3;
- Safeguards Actuation;
- Reactor Coolant Average Temperature (T_{avg}) – Low 2 coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

Startup Feedwater Isolation

* * *

Startup Feedwater Isolation is actuated by the following signals:

- SG Narrow Range Water Level – ~~High 2~~High 3;
- RCS Cold Leg Temperature (T_{cold}) – ~~Low~~Low 2;
- SG Narrow Range Water Level – High coincident with Reactor Trip Signal (P-4); and
- Main Feedwater Control Valve Isolation – Manual Initiation.

* * *

ADS Stages 1, 2, & 3 Actuation

* * *

ADS Stages 1, 2, & 3 is actuated on the following signals:

- CMT Level – ~~Low 1~~Low 3 coincident with CMT Actuation; and
- ADS Stages 1, 2 & 3 Actuation – Manual Initiation.

ADS Stage 4 Actuation

* * *

ADS Stage 4 is actuated on the following signals:

- CMT Level – ~~Low 2~~Low 6 coincident with both ADS Stage 1, 2, & 3

Actuation and RCS Wide Range Pressure – Low;

- Hot Leg Loop 1 Level ~~Low – 2~~Low 4 coincident with Hot Leg Loop 2 Level ~~Low – 2~~Low 4;

- ADS Stage 4 Actuation – Manual Initiation coincident with ADS Stages 1, 2, & 3 Actuation; and

- ADS Stage 4 Actuation – Manual Initiation coincident with RCS Wide Range Pressure – Low.

Reactor Coolant Pump Trip

Reactor Coolant Pump (RCP) Trip allows the passive injection * * * maintenance of SHUTDOWN MARGIN following a steam line break. RCP trip on ~~high~~High 2 bearing water temperature protects the RCP coast down.

RCP trip is actuated on the following signals:

- Safeguards Actuation;

- ADS Stages 1, 2, & 3 Actuation;

- Reactor Coolant Pump Bearing Water Temperature – ~~High~~High 2;

- Pressurizer Water Level – Low 2; and

- CMT Injection Actuation – Manual Initiation.

Component Cooling Water System Containment Isolation Valve Closure

The function of the Component Cooling Water System (CCS) * * * the turbine building. CCS Isolation Valve Closure is actuated by Reactor Coolant Pump Bearing Water Temperature – ~~High~~High 2.

* * *

Passive Residual Heat Removal (PRHR) Heat Exchanger Actuation

* * *

PRHR is actuated on the following signals:

- SG Narrow Range Water Level – Low coincident with Startup Feedwater Flow – ~~low~~Low 2;

- * * *.

* * *

Chemical Volume and Control System Makeup Isolation

* * *

Chemical Volume and Control System Makeup Line Isolation is actuated on the following signals:

- Containment Radioactivity – High 2;
- Pressurizer Water Level – High 2;
- Pressurizer Water Level – High 1 coincident with unlatched Safeguards Actuation;
- Source Range Neutron Flux Doubling;
- SG Narrow Range Water Level – ~~High 2~~High 3;
- * * *

Chemical and Volume Control System Letdown Isolation

The CVS provides letdown to the liquid radwaste system to maintain the pressurizer level. To help maintain RCS inventory in the event of a LOCA, the CVS Letdown Isolation is actuated on Hot Leg Level – ~~Low 1~~Low 2.

Auxiliary Spray and Purification Line Isolation

* * *

Auxiliary Spray and Purification Line Isolation is actuated on the following signals:

- Pressurizer Water Level – ~~Low 1~~Low; and
- Chemical Volume and Control System Makeup Isolation – Manual Initiation.

* * *

3. Containment Radioactivity – ~~High 1~~High

   This signal to isolate Containment Air Filtration System results from the coincidence of containment radioactivity above the ~~High 1~~High setpoint in any two of the four divisions.

   The Containment Air Filtration System Isolation ESFAS protective function is actuated by Containment Radioactivity – ~~High 1~~High.

Four channels of Containment Radioactivity – ~~High 1~~High are required to be OPERABLE in MODES 1, 2, and 3, and * * *

* * *

5. Pressurizer Pressure – ~~Low~~Low 3

* * *

The Safeguards Actuation ESFAS protective function is actuated by Pressurizer Pressure – ~~Low~~Low 3. The transmitters are * * *

The LCO requires four channels of Pressurizer Pressure – ~~Low~~Low 3 to be OPERABLE in MODES 1, 2, and 3 (above * * *

* * *

6. Pressurizer Water Level – ~~Low 1~~Low

A signal to isolate the purification line and the auxiliary spray line is generated upon the coincidence of pressurizer level below the ~~Low 1~~Low setpoint in any two-out-of-four divisions.

The Auxiliary Spray and Purification Line Isolation ESFAS protective function is actuated by Pressurizer Water Level – ~~Low 1~~Low.

Four channels of Pressurizer Water Level - ~~Low 1~~Low are required to be OPERABLE in MODES 1 and 2 to help maintain RCS inventory.
* * *

8. Pressurizer Water Level – ~~High 1~~High

Four channels of pressurizer level are provided on the pressurizer. Two-out-of-four channels on indicating level greater than the ~~High 1~~High setpoint coincident with a Safeguards Actuation * * *

The Chemical Volume and Control System Makeup Isolation ESFAS protective function is actuated by Pressurizer Water Level – ~~High 1~~High.

* * *

11. RCS Cold Leg Temperature (T_{cold}) – ~~Low~~Low 2

* * *

The ESFAS protective functions actuated by RCS Cold Leg Temperature (T_{cold}) – ~~Low~~Low 2 are:

- * * *.

* * *

The LCO requires four channels of T_{cold} – ~~Low~~Low 2 to be OPERABLE in MODES 1 and 2, and in MODE 3 * * *

12. T_{avg} ~~Low 1~~Low

This signal provides protection against excessive feedwater flow by closing the main feedwater control valves. This signal results from a coincidence of two of the four divisions of reactor loop average temperature below the ~~Low 1~~Low setpoint coincident with the P-4 permissive. Four channels are provided to permit one * * *

The Main Feedwater Control Valve Isolation ESFAS protective function is actuated by T_{avg} ~~Low 1~~Low.

Closing the Main Feedwater Control Valves on T_{avg} ~~Low 1~~Low coincident with Reactor Trip (P-4) is required to be * * *

* * *

15. CMT Level – ~~Low 1~~Low 3

This Function ensures continued passive injection or borated water to the RCS following a small break LOCA. ADS Stages 1, 2 and 3 actuation is initiated when the CMT Level reaches its ~~Low 1~~Low 3 setpoint coincident with any CMT Actuation signal. * * *

The ADS Stages 1, 2, & 3 Actuation ESFAS protective function is actuated by CMT Level – ~~Low 1~~Low 3.

* * *

16. CMT Level – ~~Low 2~~Low 6

The fourth stage depressurization valves open on CMT Level – ~~Low 2~~Low 6 in two-out-of-four channels in either CMT. * * *

The ADS Stage 4 Actuation ESFAS protective function is actuated by CMT Level – ~~Low 2~~Low 6.

* * *

19. Reactor Coolant Pump Bearing Water Temperature – ~~High~~High 2

The CCS containment isolation valves are closed and the RCPs are tripped if two-out-of-four sensors on any RCP indicate ~~high~~High 2 bearing water temperature.

The ESFAS protective functions actuated by Reactor Coolant Pump Bearing Water Temperature – ~~High~~High 2 are:

- * * *.

20. SG Narrow Range Water Level – Low

PRHR is actuated when the SG Narrow Range Level reaches its low setpoint coincident with an indication of ~~low~~Low 2 Startup Feed Water Flow. * * *

* * *

23. SG Narrow Range Water Level – ~~High 2~~High 3

* * *

The ESFAS protective functions actuated by SG Narrow Range Water Level – ~~High 2~~High 3 are:

- * * *

The transmitters (d/p cells) are located inside * * * The LCO requires four channels of SG Narrow Range Water Level – ~~High 2~~High 3 instrumentation per steam generator to be OPERABLE in MODES 1, 2, 3, and 4 when there is significant mass * * *

24. Steam Line Pressure – ~~Low~~Low 2

Steam Line Pressure – ~~Low~~Low 2 provides protection against the following accidents:

- * * *

Steam Line Pressure – ~~Low~~Low 2 provides closure of the PORV flow paths in the event of SGTR in which the PORV(s) open, to limit the radiological releases from the ruptured steam generator into the atmosphere. Steam Line Pressure – ~~Low~~Low 2 also provides closure of the MSIVs in the event of an SLB to limit the mass and energy release to containment and limit blowdown to a single SG.

* * *

The ESFAS protective functions actuated by Steam Line Pressure – ~~Low~~Low 2 are:

- * * *

The LCO requires four channels per steam line of Steam Line

Pressure – ~~Low~~Low 2 Function to be OPERABLE in * * *

25. Steam Line Pressure-Negative Rate – High

Steam Line Pressure-Negative Rate – High provides * * * manually blocks the Steam Line Pressure – ~~Low~~Low 2 when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate – High signal is automatically enabled.

* * *

MODES 1 and 2, and in MODE 3 when above the P-11 setpoint with the RCS boron concentration below that necessary to meet the SDM requirements at an RCS temperature of 200°F, this signal is automatically disabled and the Steam Line Pressure – ~~Low~~Low 2 signal is automatically enabled.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.3.10 as shown below:

* * *

* * *

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

* * *

* * *

1. Hot Leg Level – ~~Low 2~~Low 4

A signal to automatically open the ADS Stage 4 is generated when * * *

3. Hot Leg Level – ~~Low 1~~Low 2

A signal to isolate the Chemical and Volume Control System (CVS) letdown valves is generated upon the occurrence of a ~~Low 1~~Low 2 hot leg level in either of the two RCS hot leg loops. * * *

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.3.11 as shown below:

* * *

* * *

APPLICABLE SAFETY ANALYSES, LCOs, and APPLICABILITY

* * *

PRHR is actuated when the Steam Generator Narrow Range Level reaches its low setpoint coincident with an indication of ~~low~~Low 2 Startup Feedwater Flow.

Startup Feedwater Flow – ~~Low~~Low 2 uses a one-out-of-two logic on each of the two startup feedwater lines. This Function is required * * * Therefore, startup feedwater flow is not required, and PRHR actuation on ~~low~~Low 2 startup feedwater flow is not required.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.4.11 as shown below:

BACKGROUND * * *

The automatic depressurization valves are designed to open automatically when actuated, and to remain open for the duration of any automatic depressurization event. The valves are actuated sequentially. The stage 1 valves are actuated on a ~~low~~Low 3 core makeup tank (CMT) level. Stages 2 and 3 are actuated on the stage 1 signal plus time delays. Stage 4 is actuated on a ~~Low 2~~Low 6 CMT level signal with a minimum time delay after stage 3. Stage 4 is blocked from actuating at normal RCS pressure.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.6.7 as shown below:

* * *

BACKGROUND (continued)

* * *

The risk of overdraining the RCS has been significantly reduced due to the automatic protection features associated with the hot leg level instruments which isolate letdown on ~~low~~Low 2 hot leg water level. Overdraining the RCS is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.6.9 as shown below:

BACKGROUND * * *

The containment pressure vessel contains two 100% capacity vacuum * * * isolation signal, as well as on ~~High-1~~High containment radioactivity. Each flow path contains a normally closed, self-actuated check valve inside containment that opens on a negative differential pressure of 0.2 psi. A vacuum relief flow path consists of one MOV and one check valve, and the shared containment penetration.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.7.1 as shown below:

* * *

APPLICABLE SAFETY ANALYSES * * *

* * * This analysis demonstrates that the DNB design basis is met. Another analysis is performed assuming no primary system pressure control, but crediting reactor trip on ~~high~~High 2 pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.7.2 as shown below:

BACKGROUND * * *

The MSIVs, turbine stop and control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves close on a main steam isolation signal generated by either ~~low~~Low 2 steam line pressure, high containment pressure, ~~Low~~Low 2 T_{cold}, or high negative steam pressure rate. The MSIVs close on an actuation signal * * *

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.7.3 as shown below:

BACKGROUND * * *

The MFIVs and MFCVs close on receipt of engineered safeguards
~~~~~~~~~~~~~~~~~~~~

feedwater isolation signal generated from any of the following conditions:

- Automatic or manual safeguards actuation “S” signal
- ~~High~~High 3 steam generator level
- * * *

* * *

Additionally, the MFIVs close automatically on a ~~Low 1~~Low T_{avg} coincident with reactor trip (P-4). Each valve may be actuated manually. * * *

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs and MFCVs is established by the analyses * * * feedwater event upon the receipt of a steam generator water level – ~~High 2~~High 3 signal.

* * *

LCO

* * *

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A main feedwater isolation signal on ~~high~~High 3 steam generator level is relied on to terminate an excess feedwater flow event, and therefore failure to meet the LCO may result in the introduction of water into the main steam lines.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.7.7 as shown below:

* * *

APPLICABLE SAFETY ANALYSES

* * *

~~Low~~Low 2 T_{cold} or high steam generator level signals close the startup feedwater control and isolation valves and trips the startup feedwater pumps.

~~~~~~~~~~~~~~~~~~~~

Revise Technical Specifications Bases Section 3.7.10 as shown below:

BACKGROUND * * *

A PORV is installed in a 6 inch branch line off of the main steam line piping from each steam generator, to provide for controlled * * * Protection and Safety Monitoring System (PMS) isolation signal on ~~low~~Low 2 steam line pressure. The block valve is also a containment isolation valve.

* * *

APPLICABLE SAFETY ANALYSES

The PORV flow paths must be isolated following an SGTR to * * * subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on ~~low~~Low 2 steam line pressure.

~~~~~~~~~~~~~~~~~~~~
~~~~~~~~~~~~~~~~~~~~