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PSEG Nuclear LLC

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10 CFR 50.90

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LAR S16-01

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

Salem Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and DPR-75
NRC Docket Nos. 50-272 and 50-311

Subject: **License Amendment Request to Amend the Accident Monitoring Instrumentation Technical Specifications**

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) is submitting a request for an amendment to the Technical Specifications (TS) for Salem Generating Station (Salem) Units 1 and 2.

The proposed amendment revises the TS 3/4.3.3.7, Accident Monitoring Instrumentation. Specifically, this change modifies the list of instruments required to be operable based on implementation of Regulatory Guide (RG) 1.97 for Salem, and revises the allowed outage times and Actions for inoperable channels to be consistent with NUREG-1431, Revision 4, "Standard Technical Specifications - Westinghouse Plants."

Attachment 1 provides an evaluation supporting the proposed changes. Attachment 2 provides the existing TS pages marked up to show the proposed changes. Attachment 3 provides existing TS Bases pages marked up to show the proposed changes and are being provided for information only.

PSEG requests approval of this license amendment request (LAR) in accordance with standard NRC approval process and schedule. Once approved, the amendment will be implemented within 60 days from the date of issuance.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of New Jersey Official.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Ms. Tanya Timberman at 856-339-1426.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 11/17/16
(Date)

Respectfully,

Charles V. McFeaters
Site Vice President
Salem Generating Station

Attachments:
1. Evaluation of Proposed Changes
2. Mark-up of Proposed Technical Specification Pages
3. Mark-up of Proposed Technical Specification Bases Pages

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. C. Parker, Project Manager, NRC
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Chief, NJBNE
PSEG Corporate Commitment Tracking Coordinator
Salem Commitment Tracking Coordinator

Attachment 1

Evaluation of Proposed Changes

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1.0 DESCRIPTION

The proposed amendment revises the TS 3/4.3.3.7, Accident Monitoring Instrumentation. Specifically, this change modifies the list of instruments required to be operable based on implementation of Regulatory Guide (RG) 1.97 for Salem, and revises the allowed outage times and Actions for inoperable channels to be consistent with NUREG-1431, Revision 4, "Standard Technical Specifications - Westinghouse Plants."

Changes are proposed to Technical Specification (TS) Table 3.3-11, to Actions 1, 2, 4 and 6, and to TS Section 6.9.4. In addition, consistent with NUREG-1431 a note is added to Section 3.3.3.7 allowing separate condition entry for each function (i.e., affected instrument).

2.0 PROPOSED CHANGE

This License Amendment Request revises Salem Units 1 and 2 TS Table 3.3-11 and Table 4.3-11, Accident Monitoring Instrumentation as follows:

Instruments removed from TS Table 3.3-11 and TS Table 4.3-11 because they are not regulatory guide (RG) 1.97 Type A, or Category 1:

- Reactor Coolant System Sub-Cooling Margin Monitor*
- PORV Position Indicator (Unit 2 only)*
- PORV Block Valve Position Indicator*
- Pressurizer Safety Valve Position Indicator*
- Containment Pressure (Narrow Range)

* The associated footnotes are also removed.

Instruments added to TS Table 3.3-11 and TS Table 4.3-11 because they are RG 1.97 Type A, or Category 1 instruments that are not currently in the tables:

- Wide Range Neutron Flux Monitors
- Auxiliary Feed Water (AFW) Storage Tank (Condensate Storage Tank) Water Level

The following TS Table 3.3-11 Actions are revised:

<u>Action 1</u>
With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within <u>30</u> days, <u>or submit a special report in accordance with Specification 6.9.4</u>.

<u>Action 2</u>
With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within <u>7 days</u> or be in <u>HOT STANDBY within the next 6 hours and in</u> HOT SHUTDOWN within the <u>following 6</u> hours.

Action 4
With the number of OPERABLE channels one less than the Required Number of channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel; OTHERWISE, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

Action 6
With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

TS 3.3.3.7 is revised to add a note which allows for a Separate Condition entry for each instrument.

The surveillance frequencies for Line Items 22 and 23 that are being added to TS Table 4.3-11 will be contained in the Surveillance Frequency Control Program with the exception of the Channel Functional Test which is not applicable for these instruments. Salem has implemented TSTF Change Traveler TSTF-425, Revision 3 (Reference 11) as approved by the NRC with TS Amendments 299 and 282 (Reference 12) for Units 1 and 2 respectively. The initial surveillance frequency will be 31-days for the CHANNEL CHECK and 18-months for the CHANNEL CALIBRATION consistent with NUREG-1431, Standard Technical Specifications for Westinghouse Plants.

The criteria for relocation of a surveillance frequency to a licensee controlled program in accordance with TSTF-425 were reviewed. These surveillance frequencies are periodic surveillances that: 1) do not reference other approved programs for the specified interval, 2) are not event driven, 3) do not have a time component based on event occurrence, and 4) are not related to a specific condition for performance. Therefore the periodic surveillance frequencies are within the scope of TSTF-425 for location in the licensee controlled Surveillance Frequency Control Program.

Finally, TS 6.9.4, Special Reports, is revised to include Actions 1 and 4 of TS Table 3.3-11.

3.0 BACKGROUND

In August 2015, pursuant to 10 CFR 50.90, PSEG requested an amendment to the Salem facility operating license to remove the Pressurizer Power Operated Relief Valve (PORV) position indication from the Unit 1 Technical Specifications 3/4.3.3.7, Accident Monitoring Instrumentation. The change was requested and approved on an emergency basis as permitted by 10 CFR 50.91(a)(5), and was the result of a 10 CFR Part 21 evaluation associated with the Namco limit switches. Per LAR S15-04,[1] due to the emergency nature of the request, the LAR was only for the specific line item for Salem Unit 1.

[1] PSEG Emergency License Amendment Request to Remove Pressurizer Power Operated Relief Valve (PORV) Position Indication Instrumentation from the Accident Monitoring Instrumentation Technical Specifications, dated August 31, 2015 (accession number ML15243A491).

PSEG is submitting this LAR to revise Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11, Accident Monitoring Instrumentation. The purpose is to align the remainder of the TS tables with the requirements of Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", Revision 2[2] dated December 1980 and the NUREG 1431, Revision 4, Standard Technical Specifications (STS).

NUREG-0737, "Clarification of TMI Action Plan Requirements" (Reference 13), included requirements for accident monitoring instrumentation to be listed in plant Technical Specifications. Subsequently, the NRC Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, which was later codified by changes to 10 CFR 50.36, provided a specific set of objective criteria as guidance for determining which regulatory requirements and operating restrictions should be included In Technical Specifications:

(A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

RG 1.97 describes a method acceptable to the NRC for complying with the regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. Variables and Categories are defined as follows:

Type A Variables are those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Type B Variables are those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control).

[2] Salem was designed prior to the issuance of RG 1.97. The UFSAR was subsequently updated to demonstrate compliance with the intent of RG 1.97, Revision 2. In various site-specific evaluations of PAM instruments, Revision 3 was used by Salem as more clearly presenting guidance for PWR variables while at the same time being essentially equivalent to Revision 2 guidance. The current RG 1.97 (Revision 4) states that it is primarily intended for new reactors and that previous versions of the RG remain in effect for licensees of current operating reactors. Revision 2 remains the Salem licensing basis for RG 1.97 compliance.

Type C Variables are those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Type D Variables are those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Type E Variables are those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

Category 1, in general, provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power.

Category 2, in general, provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power).

Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only offsite power.

The current TS contains instrumentation that does not meet any of the 10 CFR 50.36(c)(2)(ii) screening criteria for inclusion in TS. It also contains TS allowed outage times (7 days for less than required number of OPERABLE channels, 48 hours for less than minimum number of OPERABLE channels) that are overly restrictive and are not commensurate with the safety significance of the instruments.

4.0 TECHNICAL ANALYSIS

4.1 Methodology

The accident monitoring instrumentation for the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11 was assessed as follows:

- The following documents were reviewed and compared with each other, specifically to identify post-accident monitoring (PAM) instruments as Type A or Category 1:

Document	Reference
Regulatory Guide 1.97	1
Salem UFSAR	2
Salem TS Tables 3.3-11/4.3-11	3
Standard Westinghouse TS	4
Engineering Evaluation S-C-GSC-CEE-0470	5

- Based on the comparisons, instruments that should be removed because they are not Type A variable or not Category 1 were identified.
- Based on the comparisons, instruments that should be added to the TS because they are Type A variables, or non-Type A, but Category 1 were identified.

Type A Variables are defined as those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Category 1, in general, provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power.

The intent is to align the instruments in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11 to be consistent with the scope of the NUREG-1431 reviewer's note which states that the TS should include all RG 1.97 Type A instruments and all RG 1.97, Category 1, non-Type A instruments in accordance with the Unit's RG 1.97 Safety Evaluation Report. The instruments proposed for removal from the TS tables are not being removed from the plant and will continue to satisfy their RG 1.97 requirements.

4.2 Technical Evaluation

Each of the instruments listed in Table 1 below, were evaluated and determined to be removed or added to the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Table 1

Instrument	**Salem Unit**	**Action**
Reactor Coolant System Sub-Cooling Margin Monitor	U1 and U2	Remove
PORV Position Indicator	Unit 2 only[3]	Remove
PORV Block Valve Position Indicator	U1 and U2	Remove
Pressurizer Safety Valve Position Indicator	U1 and U2	Remove
Containment Pressure (Narrow Range)	U1 and U2	Remove
Wide Range Neutron Flux Monitors	U1 and U2	Add
AFW (Condensate) Storage Tank Water Level	U1 and U2	Add

4.3 Individual Instrument Evaluations

Reactor Coolant System Sub-Cooling Margin Monitor

The reactor coolant system (RCS) sub-cooling margin monitor (SMM) is not listed in the STS. It should be noted, that this instrument is categorized in Table 3 of RG 1.97 as a Category 2 variable.

For the Salem Generating Station, Engineering Evaluation S-C-GCS-CEE-0470 states that the sub-cooling margin monitor is a Type B Category 2 variable in accordance with the requirements of RG 1.97. It is currently listed in UFSAR Table 7.5-3, as a Type A variable, Category 1. As discussed below, degrees of subcooling should be removed from UFSAR Table 7.5-3 because it does not meet the Type A or Category 1 criteria and should be classified as Type B, Category 2.

[3] The PORV position indicator was removed from Salem Unit 1 by TS Amendment 310

Type B Variables are defined as those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control).

Category 2, in general, provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power).

The RCS sub-cooling margin indication provides information to the operators related to satisfying one of the Safety Injection (SI) termination criteria following a steam line break or Steam Generator Tube Rupture (SGTR) accident. The inputs to the RCS sub-cooling margin monitor are the core exit thermocouples for RCS temperature and the wide range RCS pressure indication for RCS pressure.

The RCS SMM does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."

RCS sub-cooling instrumentation is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

The RCS SMM is not part of a primary success path as indicated in Criterion 3.

The loss of the RCS SMM instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

The RCS sub-cooling margin indication provides information to indicate whether the core cooling safety function is being accomplished, and is therefore considered a Type B variable. The RCS sub-cooling indication is a backup to the core exit thermocouples and RCS pressure, and is therefore considered Category 2. In support of this LAR, the UFSAR (Tables 7.5-1, 7.5-3, and 7.5-4) and associated design documentation, as appropriate, are being revised to reflect that the subcooling margin monitor is not a Type A variable. PSEG has determined these documentation changes can be implemented in accordance with 10 CFR 50.59. The Salem UFSAR will be revised prior to implementation of this proposed TS change.

The RCS sub-cooling margin indication does not meet any of the four screening criteria of the NRC Final Policy Statement in 58 Federal Register 39132 (58FR39132), which provides criteria to determine which items are required to be included in the TS as LCOs. This conclusion is supported by the absence of operability and surveillance requirements for the sub-cooling margin indication in the improved standard TS presented in NUREG-1431. Accordingly, removal of the RCS sub-cooling margin indication from the TS conforms to the STS.

Therefore, the reactor coolant system sub-cooling margin monitor is removed from the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

PORV Position Indication
The reactor coolant system is protected against over-pressurization by control and protective circuits such as the pressurizer pressure high reactor trip and by the PORVs connected to the top of the pressurizer. The PORVs provide a means for pressure relief. Each PORV is equipped with two limit switches to provide Open and Closed indication (i.e. lights) in the control room.

The PORV limit switch position indicators provide information to the control room operators related to the position of the pressurizer PORV's. The Design Basis Accident (DBA) analysis of an inadvertent opening of the PORV does not rely on operator diagnosis and closure of the PORV or block valve, the DBA analysis assumes that automatic safety injection actuation will provide adequate protection. At Salem, this event is bounded by the more limiting inadvertent opening of a code safety valve as described in Salem UFSAR Section 15.2.12.1, Accidental Depressurization of the Reactor Coolant System. At Salem, no credit for diagnosis and re-closure of a PORV, block valve, or code valve is assumed in the DBA analysis.

The Salem UFSAR identifies "Primary System Safety Relief Valve Positions (including PORV and code valves)" as Type D, Category 2 Variables, consistent with RG 1.97 (UFSAR Section 7.5). Type D variables are those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. Category 2 instruments are designed to less stringent qualifications that do not require seismic qualification, redundancy, or continuous display, and require only a high reliability power source, not necessarily standby power. Removing the PORV position indication from the TS conforms with the NRC position on application of the screening criteria to post-accident monitoring instrumentation.

The PORVs themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a DBA and therefore meet Criterion 3 of the NRC Final Policy Statement. For example, they are credited in Salem UFSAR Section 15.2.14, Spurious Operation of the Safety Injection System at Power. The operability of the PORVs is therefore required by TS 3.4.3, "Relief Valves." However, PORV position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. PORV position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the PORVs themselves is part of the primary success path in the UFSAR, PORV position indication is not part of the primary success path. UFSAR accident analysis assumes that the PORVs open as designed to reduce reactor pressure and no operator action based on PORV position indication is required. Therefore, PORV position indication is not part of the primary success path as indicated in Criterion 3.

The loss of PORV position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

PORV position indication instrumentation does not meet any of the four screening criteria of the NRC Final Policy Statement in 58 Federal Register 39132 (58FR39132), which provides criteria to determine which items are required to be included in the TS as LCOs. This conclusion is supported by the absence of operability and surveillance requirements for the PORV position indication instrumentation in the improved standard TS presented in NUREG-1431.

Accordingly, removal of PORVs from TS conforms to the STS. The PORV position indicator was previously removed from Salem Unit 1 through TS Amendment 310.

Therefore, the PORV position indication is removed from the Salem Unit 2 TS Tables 3.3-11 and 4.3-11.

PORV Block Valve Position Indication
The PORVs can be isolated by the PORV block valves which are connected in line with the PORVs. The PORV block valves provide closure redundancy to the PORVs, to isolate PORVs with excessive seat leakage or that stick open. Use of the PORV block valves in the event of a PORV malfunction can prevent a severe depressurization of the RCS with potential for uncovering of the reactor core.

The PORV block valve limit switch position indication provides information to the control room operators on the position of the pressurizer PORV block valves. It could be used to diagnose the availability of the pressurizer PORV's for use in depressurizing the RCS or to indicate the isolation of a stuck open PORV (LOCA) at lower RCS temperatures. The PORV block valve limit switch position indication does not provide an indication for operator actions for which no automatic control is provided or impact the response of the PORVs to a design basis accident. Furthermore, this instrumentation is not needed for manual operator action necessary for safety systems to accomplish their safety function for the design-basis events.

The PORVs themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. In order to provide this function, the PORV block valves must be open or capable of being manually opened. The operability of the PORV block valves is therefore required by Unit 1 TS 3.4.3 and Unit 2 TS 3.4.5, "Relief Valves." However, PORV block valve position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation installed to identify the source of actual leakage, for example valve position indicators. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."

PORV block valve position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the PORVs themselves is part of the primary success path in the UFSAR, the PORV block valve position indication is not part of the primary success path. UFSAR accident analysis assumes that the PORVs open as designed to reduce reactor pressure. In the event a PORV block valve is closed to isolate PORV valve seat leakage as allowed by the technical specifications, the emergency operating procedures direct the operators to manually open the PORV and associated block valve. This action does not rely on the PORV block valve position indication. Therefore, PORV block valve position indication is not part of the primary success path as indicated in Criterion 3.

The loss of PORV block valve position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

PORV block valve position indication does not meet any of the four screening criteria of the NRC Final Policy Statement in 58 Federal Register 39132 (58FR39132), which provides criteria to determine which items are required to be included in the TS as LCOs. This conclusion is supported by the absence of operability and surveillance requirements for the PORV block valve position indication in the improved standard TS presented in NUREG-1431. Accordingly, removal of the PORV block valve position indication from the TS conforms to the STS.

Thus, since the PORV block valve limit switch positions provide information to indicate the status of the pressurizer PORV block valves which are used to isolate the PORVs in the event of excessive PORV leakage, they are a Type D[4] variable. For the same reasons as the PORV position indications, the block valves position indicators do not meet Criterion 1, 2, 3 or 4 of the NRC Final Policy Statement and an LCO is not required concerning the PORV block valve position indicators. Because the PORV block valve position indicator is not classified as a Type A variable, or a non-Type A but Category 1 variable, the PORV block valve position indication is not required to be in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Therefore, the PORV block valve position indication is removed from the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Pressurizer Safety Valve Position Indication
The Pressurizer Safety Valve Position Indication provides information to the control room operators on the position of the pressurizer safety valves. It could be used to diagnose high RCS pressures or a stuck open safety valve (i.e. LOCA) at lower RCS pressures. They are both considered Type D variable, Category 2.

The Pressurizer safety relief valves themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. The operability of the Pressurizer safety valves is therefore required by Unit 1 TS 3.4.2 and Unit 2 TS 3.4.2 & 3.4.3, "Safety Valves." However, the Pressurizer safety valve position indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation installed to identify the source of actual leakage, for example valve position indicators. The instrumentation that satisfies Criterion 1 is contained in Salem Unit 1 TS 3.4.6.1 and Unit 2 TS 3.4.7.1 "Reactor Coolant System Leakage, Leakage Detection Systems."

Pressurizer safety relief valve position indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

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[4] UFSAR Section 7.5 describes implementation of RG 1.97 at Salem. Type D variables are defined in RG 1.97, as variables that provide information to indicate the operation of individual safety systems and other systems important to safety.

While the function of the Pressurizer safety relief valves themselves is part of the primary success path in the UFSAR, the valve position indication is not part of the primary success path. UFSAR accident analysis assumes that the Pressurizer safety relief valves open as designed to reduce reactor pressure and no operator action based on valve position indication is required. Therefore, the Pressurizer safety valve position indication is not part of the primary success path as indicated in Criterion 3.

The loss of Pressurizer safety valve position indication instrumentation has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

Pressurizer Safety Valve position indication does not meet any of the four screening criteria of the NRC Final Policy Statement in 58 Federal Register 39132 (58FR39132), which provides criteria to determine which items are required to be included in the TS as LCOs. This conclusion is supported by the absence of operability and surveillance requirements for the Pressurizer Safety Valve position indication in the improved standard TS presented in NUREG-1431. Accordingly, removal of the Pressurizer Safety Valve position indication from the TS conforms to the STS.

Since the position indication for these valves does not provide an indication for operator actions for which no automatic control is provided, it does not satisfy Criterion 1, 2, 3 or 4 of the NRC Final Policy Statement for the same reasons as discussed for the PORVs above.

The Pressurizer Safety Valve indication is not included in the STS. They are not Category 1 in RG 1.97 nor are they Type A or Category 1 within the Salem UFSAR or Engineering Evaluation S-C-GCS-CEE-0470, thus, the Pressurizer Safety Valve Position Indication is not required to be in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Therefore, the Pressurizer Safety Valve Position Indication is proposed to be removed from the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

<u>Containment Pressure (Narrow Range)</u>
The containment pressure indication provides information for assessing an inadequate containment cooling condition and for determining the potential challenge to the containment pressure retaining integrity. The wide range containment pressure instrumentation, which is also listed in Salem Unit 1 and Unit 2 TS Table 3.3-11 (Line Item #16), provides an adequate range and sensitivity for this purpose. Only the wide range instrumentation is used in the emergency operating procedures (EOPs) to define the potential for a challenge to containment integrity due to over-pressurization.

If containment heat removal systems are functioning properly, no challenge to containment integrity should occur due to containment pressure. Containment pressure instrumentation is used in the severe accident management guidelines (SAMG) to indicate a possible containment integrity challenge and to initiate the assessment of containment venting strategies. It is also used as an indicator of the potential loss of a fission product barrier in the emergency plan. Containment pressure is a key indicator in the declaration of a General Emergency level and the potential need for offsite radiological protection actions. Therefore, the containment wide range instrumentation (by itself) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

The STS requires only the Containment Pressure (Wide Range) instrumentation be included in the PAM TS. Per Salem UFSAR 7.5-3, the measurement of Containment Pressure is a Type A, Category 1 variable; however, no distinction is made between the use of wide range or narrow range instruments.

Since the redundancy requirements and other requirements of RG 1.97 are met with the Containment Pressure (Wide Range) channels, the Containment Pressure (Narrow Range) channels are not required to be in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

In support of this LAR, the UFSAR (Tables 7.5-1, 7.5-3, and 7.5-4, as appropriate) and Item 11 of S-C-GCS-CEE-0470 will be revised to reflect that the narrow range containment pressure instrumentation is not a Type A variable. PSEG has determined these documentation changes can be implemented in accordance with 10 CFR 50.59. The Salem UFSAR will be revised prior to implementation of this proposed TS change.

The containment pressure narrow instrument channels themselves are part of the primary success path in the UFSAR accident analysis because they are assumed to actuate to mitigate a Design Basis Accident (DBA) and therefore meet Criterion 3 of the NRC Final Policy Statement. The operability of the containment pressure narrow range channels is therefore required by Unit 1 and 2 TS 3.3.2.1, "Engineered Safety Feature Actuation System Instrumentation." However, containment pressure narrow range indication does not detect or indicate a significant abnormal degradation of the reactor coolant pressure boundary, as required by Criterion 1. This is consistent with the NRC Final Policy Statement, which provided that Criterion 1 is intended to ensure that those instruments specifically installed to detect excessive reactor coolant system leakage be included in the TS. Criterion 1 is not to be interpreted to include instrumentation installed to identify the source of actual leakage, for example valve position indicators.

Containment pressure narrow range indication is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis considered in Criterion 2.

While the function of the containment pressure narrow range channels themselves is part of the primary success path in the UFSAR, the pressure indication is not part of the primary success path. UFSAR accident analyses assume automatic actuation of the safety injection and containment spray system based on containment narrow range pressures and no operator actions based on containment pressure narrow range indication are required. Therefore, the containment pressure narrow range indication is not part of the primary success path as indicated in Criterion 3.

The Containment Pressure narrow range indication has no effect on the probabilistic safety assessment, and has not been shown to be significant to health and safety as considered in Criterion 4.

Containment Pressure (Narrow Range) indication does not meet any of the four screening criteria of the NRC Final Policy Statement in 58 Federal Register 39132 (58FR39132), which provides criteria to determine which items are required to be included in the TS as LCOs. This conclusion is supported by the absence of operability and surveillance requirements for the Containment Pressure (Narrow Range) indication in the improved standard TS presented in

NUREG-1431. Accordingly, removal of Containment Pressure (Narrow Range) indication from the TS conforms to the STS.

Therefore, the Containment Pressure (Narrow Range) is removed from the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Wide Range Neutron Flux Monitors (Gamma-Metrics Post-Accident Neutron Monitoring)
The Neutron Flux is listed as an instrument within the STS, and is listed as Category 1 in RG 1.97, and is Category 1 in Engineering Evaluation S-C-GSC-CEE-0470.

The bases in Westinghouse STS (Reference 4) state that power range and source range neutron flux indication is provided to verify reactor shutdown. The two ranges are necessary to cover the full range of flux that may occur post-accident. It also states that neutron flux is used for accident diagnosis, verification of sub-criticality, and diagnosis of positive reactivity insertion.

Gamma-Metrics Post-Accident Neutron Monitoring (PANM) system was installed at Salem to provide reliable neutron flux monitoring in a harsh environment from plant shutdown to full power, and thereby specifically satisfies RG 1.97 requirements. In the wide-range mode, the PANM output range meets the requirement of RG 1.97 (Reference 1).

Therefore, the Wide Range Neutron Flux Monitors are added to the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Auxiliary Feed Water (AFW) Storage Tank (Condensate Storage Tank[5]) Water Level
The AFW storage tank water level is listed as a Type A variable per the Salem UFSAR Table 7.5-3 and Category 1 by Table 3 in RG 1.97. It is also listed as Type A and Category 1 in Engineering Evaluation S-C-GCS-CEE-0470. Because of this site-specific determination the instrumentation should be added to TS Tables 3.3-11 and 4.3-11 for Unit 1 and Unit 2 in accordance with the guidance of the reviewer's notes in NUREG-1431.

Therefore, the Auxiliary Feed Water (AFW) Storage Tank (Condensate Storage Tank) Water Level is added to the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

Exceptions:
All instruments that are Type A variables, or non-Type A, but Category 1 were evaluated and are proposed to be included in the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11 with the following exceptions:

- Containment Isolation Valve (CIV) Position Indicators
  PSEG is not proposing to add CIV position indication to TS Tables 3.3-11 and 4.3-11.

  For valves which receive an automatic isolation signal to close in order to accomplish containment isolation, valve position is a Type B, Category 1 variable. Type B variables are those which provide information to indicate whether plant safety functions are being accomplished. In the Reference 8 letter to vendor owners

[5] In various PAM instrument documents, Condensate Storage Tank water-level is listed because the CST is frequently the source of auxiliary feedwater. At Salem, the AFW storage tank is the source of auxiliary feedwater. Hence AFW storage level and CST water level are used interchangeably in this document.

groups on application of the policy statement criteria to standard technical specifications, the NRC staff concurred that that only Type A variables meet Criterion 3 as a structure. system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the Integrity of a fission product barrier. However, the staff was unable to confirm the owners groups' conclusion that non-Type A Category 1 instrumentation can be relocated from Technical Specifications

Inclusion of CIV position indication in Salem TS Tables 3.3-10 and 4.3-10 is not required to ensure adequate information is available to operators to verify containment operability and the accomplishment of containment isolation when required. The operability requirements for the containment are controlled by TS 3.6.1. Limiting Condition for Operation (LCO) 3.6.1.1 requires primary CONTAINMENT INTEGRITY to be maintained in Modes 1, 2, 3 and 4. CONTAINMENT INTEGRITY requires that all penetrations required to be closed during accident conditions are either capable of being closed by an OPERABLE containment automatic isolation valve system, or closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open under administrative control as permitted by TS 3.6.3.1.

The operability requirements for automatic containment isolation valves are controlled in TS 3.6.3.1 (unit 1) and TS 3.6.3 (Unit 2). LCO 3.6.3.1 (3.6.3 Unit 2) requires each containment isolation valve to be OPERABLE in Modes 1, 2, 3 and 4. Surveillance Requirement (SR) 4.6.3.1.2 (SR 4.6.3.2 Unit 2) requires each containment isolation valve to be demonstrated OPERABLE by periodically verifying the valves actuate to their isolation positions in response to the required containment isolation signals. SR 4.6.3.1.3 (SR 4.6.3.3 Unit 2) requires periodic verification that on a main steam Isolation test signal, each main steam isolation valve actuates to its isolation position. SR 4.6.3.1.4 (SR 4.6.3.4 Unit 2) requires the isolation time of each automatic containment isolation valve to be determined to be within its limit when tested pursuant to the Inservice Testing Program. Proper position indication is verified during conduct of surveillance testing of the associated valves.

CIV position indication was not included in the Amendment which added accident monitoring instrumentation to the Unit 1 Technical Specifications (Reference 9) or in the original Unit 2 Technical Specifications (Reference 10). The CIV position indication meets the associated Regulatory Guide 1.97 requirements and the current TS requirements are sufficient to ensure adequate information is available to operators to verify containment operability and the accomplishment of containment isolation when required.

Therefore, the Containment Isolation Valve Position Indicators is not added to the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

- Containment Hydrogen Concentration
  Containment Hydrogen Concentration is listed as a Type A, Category I variable in Engineering Evaluation S-C-GSC-CEE-0470, not listed as Type A in UFSAR Table 7.5-3, but also listed as Category I in RG 1.97. However, it is not listed in the STS.

This instrument was included in the rulemaking for 10 CFR 50.44. The statement of considerations for the 50.44 rulemaking states that this instrument can be relocated from the Technical Specifications and can be re-classified as Type C, Category 3 per the RG 1.97 definitions. Not including these instruments is consistent with Salem LCR S06-05 and Salem TS Amendments 281 and 264 (Reference 7).

Therefore, the Containment Hydrogen Monitors are not added to the Salem Unit 1 and Unit 2 TS Tables 3.3-11 and 4.3-11.

4.4 LCO Times and Separate Condition Entry

The allowed outage times (AOT) in current Technical Specification 3.3.3.7 have their origin in NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors". The STS (NUREG-1431) extended the 7-day completion time for one inoperable instrument channel to 30 days and the 48-hour completion time for two inoperable channels to 7 days. Additionally, the STS removed the shutdown requirement for a single inoperable instrument channel.

With one channel inoperable beyond 30 days, a special report outlining the preplanned method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels must be submitted to the NRC within the next 14 days. With two channels inoperable for more than 7 days, the STS requires either a plant shutdown or submittal of a special report, as discussed above, depending on the particular channel that is out of service. The STS also contain provisions that permit a separate condition entry for each inoperable instrument function.

The STS 30 day completion time is based on operating experience and takes into account the remaining OPERABLE channel (or in the case of a function that has only one required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

When the required actions and 30 day completion time is not met, a written report is required to be submitted. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of unit conditions that would require information provided by this instrumentation.

The STS completion time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information.

Since the TS is being revised to better align to the STS, this amendment revises the following instruments within the Salem Unit 1 and Unit 2 TS Table 3.3-11 to adopt the allowable outage times in STS of 30 days for one inoperable channel less than the required number of channels and 7 days for two or more channels less than the required number of channels (versus the current 7 days and 48 hours, respectively). Other minor changes to the listed actions are discussed below.

Instrument		Action
1.	Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	1, 2
2.	Reactor Coolant Outlet Temperature - T_{Cold} (Wide Range)	1, 2
3.	Reactor Coolant Pressure (Wide Range)	1, 2
4.	Pressurizer Water Level	1, 2
5.	Steam Line Pressure	1, 2
6.	Steam Generator Water Level (Narrow Range)	1, 2
7.	Steam Generator Water Level (Wide Range)	1, 2
8.	Refueling Water Storage Tank Water Level	1, 2
10.	Auxiliary Feedwater Flow Rate	4,6
16.	Containment Pressure – Wide Range	7,2
17.	Containment Water Level – Wide Range	7,2
18.	Core Exit Thermocouples	1, 2
22.	Wide Range Neutron Flux Monitors	1, 2
23.	AFW (Condensate) Storage Tank Water Level	1, 2

The following TS Table 3.3-11 Actions are revised:

Action 1
With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

Action 2
With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Action 4
With the number of OPERABLE channels one less than the Required Number of channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate channel; OTHERWISE, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

Action 6
With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

This amendment also revises the Salem TS 3.3.3.7 to include the STS note which allows for a separate condition entry for each instrument. This will allow each instrument the full associated LCO time, if multiple PAM instruments become inoperable at different times.

The surveillance frequencies for Line Items 22 and 23 that are being added to TS Table 4.3-11 will be contained in the Surveillance Frequency Control Program with the exception of the

Channel Functional Test which is not applicable for these instruments. Salem has implemented TSTF Change Traveler TSTF-425, Revision 3 (Reference 11) as approved by the NRC with TS Amendments 299 and 282 (Reference 12) for Units 1 and 2 respectively. The initial surveillance frequency will be 31-days for the CHANNEL CHECK and 18-months for the CHANNEL CALIBRATION consistent with NUREG-1431, Standard Technical Specifications for Westinghouse Plants.

The criteria for relocation of a surveillance frequency to a licensee controlled program in accordance with TSTF-425 were reviewed. These surveillance frequencies are periodic surveillances that: 1) do not reference other approved programs for the specified interval, 2) are not event driven, 3) do not have a time component based on event occurrence, and 4) are not related to a specific condition for performance. Therefore the periodic surveillance frequencies are within the scope of TSTF-425 for location in the licensee controlled Surveillance Frequency Control Program.

Finally, TS 6.9.4, Special Reports, is revised to include Actions 1 and 4 of TS Table 3.3-11.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG requests an amendment to the Salem Unit 1 and Unit 2 Operating Licenses. The proposed changes would modify Accident Monitoring Technical Specifications (TS) Tables 3.3-11 and 4.3-11, titled “Accident Monitoring Instrumentation” and “Surveillance Requirements for Accident Monitoring Instrumentation” respectively. Specifically, these changes would add Regulatory Guide (RG) 1.97 Type A or Category 1 instruments that are currently absent into Tables 3.3-11 and 4.3-11, and would remove instruments that are not Type A, nor Category 1 from Tables 3.3-11 and 4.3-11. The specific instruments to be removed or added are summarized below:

Instruments removed from TS Table 3.3-11 and TS Table 4.3-11 because they are not regulatory guide (RG) 1.97 Type A, or Category 1:

- Reactor Coolant System Sub-Cooling Margin Monitor*
- PORV Position Indicator (Unit 2 only)*
- PORV Block Valve Position Indicator*
- Pressurizer Safety Valve Position Indicator*
- Containment Pressure (Narrow Range)

* The associated footnotes are also removed.

Instruments added to TS Table 3.3-11 and TS Table 4.3-11:

- Wide Range Neutron Flux Monitors
- Auxiliary Feed Water (AFW) Storage Tank (Condensate Storage Tank) Water Level

Additionally, allowed outage times (AOT) and required actions of TS Table 3.3-11 Actions 1, 2, 4 and 6 would be revised to align with the Westinghouse Standard Technical Specifications (STS), NUREG-1431. TS Section 6.9.4 is revised to reflect these changes to Actions.

PSEG has evaluated the proposed changes to the TS using the criteria in 10 CFR 50.92, and determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards;

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

   Response: No

   The proposed changes to the TS modify Accident Monitoring Instrumentation TS Tables 3.3-11 and 4.3-11 of Salem Units 1 and 2 by removing or adding instruments as listed above, and updating the AOT and required actions to better align with the Westinghouse STS, NUREG-1431. The instruments listed above are not assumed to be initiators of any analyzed event of Chapter 15 in the Updated Final Safety Analysis Report (UFSAR). Therefore the probability of an accident previously evaluated is not significantly increased.

   The proposed changes do not alter the design of any system, structure, or component (SSC). The proposed changes conform to NRC regulatory guidance regarding the content of plant TS, as identified in 10 CFR 50.36, NUREG-1431, and the NRC Final Policy Statement in 58 FR 39132.

   TS Operability requirements are retained for Type A and Category 1 variables. Operability of these instruments ensures sufficient information is available to monitor and assess plant status during and following an accident. Alternate means for diagnosing and responding to instrument malfunctions are unaffected by the proposed change. Therefore, the consequences of an accident previously evaluated are not significantly increased.

   Therefore, these proposed changes do not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed changes to the TS would modify the TS Tables 3.3-11 and 4.3-11 of Salem Units 1 and 2, by removing or adding instruments as listed above, and updating the AOT and required actions to better align with the Westinghouse STS. The proposed changes do not involve a modification to the physical configuration of the plant or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirement or introduce a new accident initiator, accident precursor, or malfunction mechanism.

   Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The proposed changes to the TS would modify the TS Tables 3.3-11 and 4.3-11 of Salem Units 1 and 2, by removing or adding instruments as listed above, and updating the AOT and required actions to better align with the Westinghouse STS. The instruments removed from Tables 3.3-11 and 4.3-11 are not needed for manual operator action necessary for safety systems to accomplish their safety function for the design basis events. The instruments listed for removal are indication-only with the exception of containment pressure narrow range instruments; thus, they do not provide an input to any automatic trip functions. In the case where similar or related instruments (e.g., containment pressure-narrow range) are associated with important trips (i.e., RPS or ESF trips), such instruments are governed by separate existing TS sections which are not altered by this request.

   Therefore, since the proposed changes do not impact the response of the plant to a design basis accident, the proposed changes do not involve a significant reduction in a margin of safety.

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

In conclusion, based on the considerations discussed above, (1) there is a 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 NRC'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.

5.2 <u>Applicable Regulatory Requirements/Criteria</u>

10 CFR 50.36(c)(2), *Limiting conditions for operation*, states: (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

Regulatory Guide (RG) 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident". The proposed changes to the Salem TS do not alter the classification of instrumentation required by RG 1.97. The proposed changes only modify the instruments listed in TS 3/4.3.3.7.

Therefore, 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.

6.0 ENVIRONMENTAL CONSIDERATION

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

7.0 REFERENCES

1. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", Revision 2 dated December 1980

2. Salem Generating Station (SGS) UFSAR, Section 7.5, "Safety-Related Display Instrumentation"

3. Salem Unit 1 and Unit 2 Technical Specification Table, 3.3-11, Accident Monitoring Instrumentation, and Table 4.3-11, Surveillance Requirements for Accident Monitoring Instruments

4. NUREG-1431, Volume 1, Specifications, Revision 4.0, "Standard Technical Specifications - Westinghouse Plants," dated April 2012 (ADAMS Accession No. ML12100A222)

5. S-C-GSC-CEE-0470, "Engineering Evaluation of SGS 1 & 2 Regulatory Guide 1.97 Instrumentation Compliance with Physical Separation and Electrical Isolation Criteria," Revision 1, July 1996

6. PSEG Emergency License Amendment Request to Remove Pressurizer Power Operated Relief Valve (PORV) Position Indication Instrumentation from the Accident Monitoring Instrumentation Technical Specifications, dated August 31, 2015 (ADAMS Accession No. ML15243A491)

7. PSEG License Amendment Request regarding Elimination of Requirements for Hydrogen Recombiners and Hydrogen Analyzers Technical Specifications, dated June 7, 2006

8. NRC Letter to Owners Group on the NRC Staff Review of Nuclear Steam Supply System Vendor Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications, dated May 9, 1988

9. NRC approval letter regarding Amendment No. 39 which incorporates the requirements for implementation of the TMI-2 Lessons Learned Category "A" Items for Salem Nuclear Generating Station, Unit No. 1, dated October 08, 1981

10. NUREG-0546, "Technical Specifications for Salem Nuclear Generating Station Unit No. 2," dated May 1981

11. Technical Specification Task Force (TSTF) Change Traveler TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b," NRC approved July 6, 2009

12. Letter Richard B. Ennis (NRC) to Thomas Joyce (PSEG), "Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendments RE: Relocation of Specific Surveillance Frequencies to a Licensee-Controlled Program Based on Technical Specification Task Force (TSTF) Change TSTF-425 (TAC Nos. ME3574 and ME3575)," March 21, 2011

13. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated November 1980

Attachment 2

Mark-up of Proposed Technical Specification Pages

The following Technical Specifications pages for Renewed Facility Operating License DPR-70 are affected by this change request:

Technical Specification	Page
3.3.3.7, Accident Monitoring Instrumentation	3/4 3-53
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-54
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-55
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-56
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-56a
Table 4.3-11, Surveillance Requirements for Accident Monitoring Instrumentation	3/4 3-57
Table 4.3-11, Surveillance Requirements for Accident Monitoring Instrumentation	3/4 3-57a
6.9.4, Special Reports	6-24b

The following Technical Specifications pages for Renewed Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
3.3.3.7, Accident Monitoring Instrumentation	3/4 3-50
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-51
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-51a
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-51b
Table 3.3-11, Accident Monitoring Instrumentation	3/4 3-51c
Table 4.3-11, Surveillance Requirements for Accident Monitoring Instrumentation	3/4 3-52
Table 4.3-11, Surveillance Requirements for Accident Monitoring Instrumentation	3/4 3-52a
6.9.4, Special Reports	6-24b

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-11.

-----NOTE-----

Separate Condition entry is allowed for each Function.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4 (1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. ~~Reactor Coolant System Subcooling Margin Monitor~~ Deleted	~~2~~	~~1~~	~~1, 2~~
12. Deleted			

TABLE 3.3-11 (CONTINUED)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. ~~PORV Block Valve Position Indicator~~ Deleted	~~2/valve**~~	~~1~~	~~1, 2~~
14. ~~Pressurizer Safety Valve Position Indicator~~ Deleted	~~2/valve**~~	~~1~~	~~1, 2~~
15. ~~Containment Pressure - Narrow Range~~ Deleted	~~2~~	~~1~~	~~1, 2~~
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/MS Line	1/MS Line	10
INSERT 22. Wide Range Neutron Flux Monitors	2	1	1, 2
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level	2	1	1, 2

~~(**) Total number of channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV Block, or Safety Valve position: Tailpipe Temperatures for the valves, Pressurizer Relief Tank Temperature Pressurizer Relief Tank Level OPERABLE.~~ Delete

TABLE 3.3-11 (continued)

HOT STANDBY within the next 6 hours and in

TABLE NOTATION

submit a special report in accordance with Specification 6.9.4

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within ~~7~~ 30 days, or ~~be in HOT SHUTDOWN within the next 12 hours~~.

ACTION 2 With the number of OPERABLE accident monitoring channels less than the MINIMUM Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within ~~48 hours~~ 7 days or be in HOT SHUTDOWN within the ~~next 12~~ following 6 hours.

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERATABLE Auxiliary Feedwater Flow Rate channel~~.~~; OTHERWISE, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 5 deleted

following 6

TABLE 3.3-11 (continued)

HOT STANDBY within the next 6 hours and in

TABLE NOTATION

ACTION 6 With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the ~~next~~ ~~12~~ hours.

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

ACTION 10 With the number of OPERABLE Channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)			N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)			N.A.
3. Reactor Coolant Pressure (Wide Range)			N.A.
4. Pressurizer Water Level			N.A.
5. Steam Line Pressure			N.A.
6. Steam Generator Water Level (Narrow Range)			N.A.
7. Steam Generator Water Level (Wide Range)			N.A.
8. Refueling Water Storage Tank Water Level			N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#		N.A.
11. ~~Reactor Coolant System Subcooling Margin Monitor~~ Deleted		~~N.A*~~	~~N.A.~~

\# Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

* ~~The instruments used to develop RCS subcooling margin are calibrated in accordance with the Surveillance Frequency Control Program; the monitor will be compared with calculated subcooling margin for known input values in accordance with the Surveillance Frequency Control Program.~~ Delete

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
12. Deleted Deleted			
13. ~~PORV Block Valve Position Indicator~~		~~N.A.~~	~~*~~
14. ~~Pressurizer Safety Valve Position Indicator~~ Deleted		~~N.A.~~	
15. ~~Containment Pressure - Narrow Range~~ Deleted			~~N.A.~~
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation Monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			

Table Notation

INSERT

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

22. Wide Range Neutron Flux Monitors			N.A.
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level			N.A.

~~*~~ ~~Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.3.~~ Delete

h. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

i. The calculated accident induced leakage rate from the portion of the tubes below 15.21 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.16 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined,

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

Insert: 1, 4,

6.9.4 When a report is required by ACTION 8 or 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be operable.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. As shown in Table 3.3-11.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-11.

-----NOTE-----

Separate Condition entry is allowed for each Function.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1	1, 2
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1	1, 2
3. Reactor Coolant Pressure (Wide Range)	2	1	1, 2
4. Pressurizer Water Level	2	1	1, 2
5. Steam Line Pressure	2/Steam Generator	1/Steam Generator	1, 2
6. Steam Generator Water Level (Narrow Range)	2/Steam Generator	1/Steam Generator	1, 2
7. Steam Generator Water Level (Wide Range)	4(1/Steam Generator)	3 (1/Steam Generator)	1, 2
8. Refueling Water Storage Tank Water Level	2	1	1, 2
9. deleted			
10. Auxiliary Feedwater Flow Rate	4 (1/Steam Generator)	3 (1/Steam Generator)	4, 6
11. ~~Reactor Coolant System Subcooling Margin Monitor~~ Deleted	~~2~~	~~1~~	~~1, 2~~
12. ~~PORV Position Indicator~~ Deleted	~~2/valve**~~	~~1~~	~~1, 2~~

TABLE 3.3-11 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM NO. OF CHANNELS	ACTION
13. ~~PORV Block Valve Position Indicator~~ Deleted	~~2/valve**~~	~~1~~	~~1, 2~~
14. ~~Pressurizer Safety Valve Position Indicator~~ Deleted	~~2/valve**~~	~~1~~	~~1, 2~~
15. ~~Containment Pressure - Narrow Range~~ Deleted	~~2~~	~~1~~	~~1, 2~~
16. Containment Pressure - Wide Range	2	1	7, 2
17. Containment Water Level - Wide Range	2	1	7, 2
18. Core Exit Thermocouples	4/core quadrant	2/core quadrant	1, 2
19. Reactor Vessel Level Instrumentation System (RVLIS)	2	1	8, 9
20. Containment High Range Accident Radiation Monitor	2	2	10
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor	1/ MS Line	1/ MS Line	10
22. Wide Range Neutron Flux Monitors	2	1	1, 2
23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level	2	1	1, 2

INSERT

~~(**) Total number of channels is considered to be two (2) with one (1) of the channels being any one (1) of the following alternate means of determining PORV, PORV Block, or Safety Valve position: Tailpipe Temperatures for the valves, Pressurizer Relief Tank Temperature Pressurizer Relief Tank Level OPERABLE.~~ Delete

HOT STANDBY within the next 6 hours and in

submit a special report in accordance with Specification 6.9.4

TABLE 3.3-11 (continued)

TABLE NOTATION

ACTION 1 With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within ~~7~~ days, or ~~be in HOT SHUTDOWN within the next 12 hours~~.

30

ACTION 2 With the number of OPERABLE accident monitoring channels less than the Minimum Number of Channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within ~~48~~ ~~hours~~ or be in HOT SHUTDOWN within the ~~next~~ ~~12~~ hours.

7 days

following 6

ACTION 3 deleted

ACTION 4 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operations may proceed provided that an OPERABLE Steam Generator Wide Range Level channel is available as an alternate means of indication for the Steam Generator with no OPERABLE Auxiliary Feedwater Flow Rate Channel~~.~~

ACTION 5 deleted

; OTHERWISE, restore the inoperable channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

TABLE 3.3-11 (continued)

HOT STANDBY within the next 6 hours and in

following 6

TABLE NOTATION

ACTION 6 With the number of OPERABLE channels less than the Minimum Number of channels shown in Table 3.3-11, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the ~~next 12~~ hours.

ACTION 7 With the number of OPERABLE channels one less than the Required Number of Channels shown in Table 3.3-11, operation may proceed until the next CHANNEL CALIBRATION (which shall be performed upon the next entry into MODE 5, COLD SHUTDOWN).

ACTION 8 With one RVLIS channel inoperable, restore the RVLIS channel to OPERABLE status within 30 days, or submit a special report in accordance with Specification 6.9.4.

ACTION 9 With both RVLIS channels inoperable, restore one channel to OPERABLE status within 7 days or submit a special report in accordance with Specification 6.9.4.

ACTION 10 With the number of OPERABLE Channels less than required by the minimum channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter within 72 hours, and:

1) either restore the inoperable Channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the actions taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-11
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
1. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)			N.A.
2. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)			N.A.
3. Reactor Coolant Pressure (Wide Range)			N.A.
4. Pressurizer Water Level			N.A.
5. Steam Line Pressure			N.A.
6. Steam Generator Water Level (Narrow Range)			N.A.
7. Steam Generator Water Level (Wide Range)			N.A.
8. Refueling Water Storage Tank Water Level			N.A.
9. deleted			
10. Auxiliary Feedwater Flow Rate	S/U#		N.A.
11. ~~Reactor Coolant System Subcooling Margin Monitor~~ Deleted		~~N.A.*~~	~~N.A.~~

Auxiliary Feedwater System is used on each startup and flow rate indication is verified at that time.

* ~~The instruments used to develop RCS subcooling margin are calibrated in accordance with the Surveillance Frequency Control Program; the monitor will be compared with calculated subcooling margin for known input values in accordance with the Surveillance Frequency Control Program.~~ Delete

TABLE 4.3-11 (Continued)
SURVEILLANCE REQUIREMENTS FOR
ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECKS(1)	CHANNEL CALIBRATION(1)	CHANNEL FUNCTIONAL TEST(1)
12. ~~PORV Position Indicator~~ Deleted		~~N.A.~~	
13. ~~PORV Block Valve Position Indicator~~ Deleted		~~N.A.~~	~~*~~
14. ~~Pressurizer Safety Valve Position Indicator~~ Deleted		~~N.A.~~	
15. ~~Containment Pressure - Narrow Range~~ Deleted			~~N.A.~~
16. Containment Pressure - Wide Range			N.A.
17. Containment Water Level - Wide Range			N.A.
18. Core Exit Thermocouples			N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)			N.A.
20. Containment High Range Accident Radiation monitor			
21. Main Steamline Discharge (Safety Valves and Atmospheric Steam Dumps) Monitor			
INSERT 22. Wide Range Neutron Flux Monitors			N.A.
INSERT 23. Auxiliary Feed Water Storage Tank (Condensate Storage Tank) Water Level			N.A.

Table Notation

(1) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

~~* Unless the block valve is closed in order to meet the requirements of Action b, or c in specification 3.4.5.~~ Delete

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Administrator, USNRC Region I within the time period specified for each report.

6.9.3 DELETED

Insert: 1, 4,

6.9.4 When a report is required by ACTION 8 OR 9 of Table 3.3-11 "Accident Monitoring Instrumentation", a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring for inadequate core cooling, the cause of the inoperability, and the plans and schedule for restoring the instrument channels to OPERABLE status.

Attachment 3

Mark-up of Proposed Technical Specification Bases Pages

The following Technical Specifications pages for Renewed Facility Operating License DPR-70 are affected by this change request:

Technical Specification Bases	Page
3/4.3.3.7, Accident Monitoring Instrumentation	B 3/4 3-3

The following Technical Specifications pages for Renewed Facility Operating License DPR-75 are affected by this change request:

Technical Specification Bases	Page
3/4.3.3.7, Accident Monitoring Instrumentation	B 3/4 3-3

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 THIS SECTION DELETED

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the Recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The purpose of tank level indicating devices is to assure the detection and control of leaks that if not controlled could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS.

The Wide Range Neutron Flux Monitors are the Gamma-Metrics Post-Accident Neutron Monitors.

3/4.3.3.9

THIS SECTION DELETED

3/4.3.3.10

THIS SECTION DELETED

3/4.3.3.11

THIS SECTION DELETED

3/4.3.3.12

THIS SECTION DELETED

3/4.3.3.13

THIS SECTION DELETED

Immediate action(s), in accordance with the LCO Action Statements, means that the required action should be pursued without delay and in a controlled manner.

3/4.3.3.2

THIS SECTION DELETED

3/4.3.3.3

THIS SECTION DELETED

3/4.3.3.4

THIS SECTION DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.3.6

THIS SECTION DELETED

3/4.3.3.7 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the Recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578,"TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The Wide Range Neutron Flux Monitors are the Gamma-Metrics Post-Accident Neutron Monitors.