PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, New Jersey 08038-0236

PSEG
Nuclear LLC

10 CFR 50.90

SEP 27 2017

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LAR S17-04

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 Relocate the Reactor Coolant System Pressure Isolation Valve Table from the Technical Specifications to the Technical Requirements Manual**

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 Unit 1 TS 3/4.4.6.3, Primary Coolant System Pressure Isolation Valves, and Unit 2 TS 3/4.4.7.2, Operational Leakage. Specifically, this change relocates the reactor coolant system (RCS) pressure isolation valve (PIV) lists, Unit 1 TS Table 4.4-3 and Unit 2 TS Table 3.4-1, from the TS to the Salem Technical Requirements Manual (TRM). In addition, the references to the TS Tables are being removed from the TS Limiting Condition for Operation (LCO) 3.4.6.3 (Unit 1) and 3.4.7.2.f (Unit 2), TS Action 3.4.6.3.a (Unit 1) and Surveillance Requirements (SR) 4.4.6.3 (Unit 1) and 4.4.7.2.2 (Unit 2). The Unit 1 PIV leakage acceptance criteria contained in Unit 1 TS Table 4.4-3 is relocated to Unit 1 SR 4.4.6.3.

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 Mr. Brian Thomas at 856-339-2022.

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

Executed on 9/27/17
(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 Specifications Bases Pages

cc: Mr. D. Dorman, Administrator, Region I, NRC
Mr. R. Ennis, 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

Table of Contents

1.0 DESCRIPTION

The proposed amendment revises Unit 1 TS 3/4.4.6.3, Primary Coolant System Pressure Isolation Valves, and Unit 2 TS 3/4.4.7.2, Operational Leakage. Specifically, this change relocates the reactor coolant system (RCS) pressure isolation valve (PIV) lists, Unit 1 TS Table 4.4-3 and Unit 2 TS Table 3.4-1, from the TS to the Salem Technical Requirements Manual (TRM). In addition, the references to the TS Tables are being removed from the TS Limiting Condition for Operation (LCO) 3.4.6.3 (Unit 1) and 3.4.7.2.f (Unit 2), TS Action 3.4.6.3.a (Unit 1) and Surveillance Requirements (SR) 4.4.6.3 (Unit 1) and 4.4.7.2.2 (Unit 2). The Unit 1 PIV leakage acceptance criteria contained in Unit 1 TS Table 4.4-3 is relocated to Unit 1 SR 4.4.6.3.

The proposed change is consistent with Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," which provides guidance to remove component lists from the TS. This request meets all conditions outlined in the Generic Letter. The proposed change for removal of the PIV Table is also consistent with the NUREG-1431, Revision 4, "Standard Technical Specifications - Westinghouse Plants."

2.0 PROPOSED CHANGE

The proposed changes to the Salem Unit 1 and 2 TS are described below and are indicated on the marked up TS pages provided in Attachment 2 of this submittal.

Salem Unit 1:

- TS LCO 3.4.6.3 will be revised to delete reference to Table 4.4-3.
- TS Action 3.4.6.3.a. will be revised to delete reference to Table 4.4-3
- TS SR 4.4.6.3 will be revised to delete reference to Table 4.4-3 and add the PIV leakage limits contained in Table 4.4-3 and Notes (a) and (b) of Table 4.4-3.
- TS Table 4.4-3 will be deleted. The list of valves will be relocated to the Salem TRM and the PIV leakage limits will be relocated to SR 4.4.6.3.

Salem Unit 2:

- TS LCO 3.4.7.2.f will be revised to delete reference to Table 3.4-1.
- SR 4.4.7.2.2 will be revised to delete reference to Table 3.4-1
- TS Table 3.4-1 will be deleted and will be relocated in its entirety to the Salem TRM.

Proposed changes to the Salem Unit 1 and 2 TS Bases are provided in Attachment 3 for information only; changes to the TS Bases pages will be incorporated in accordance with Unit 1 TS 6.17 and Unit 2 TS 6.16, "Technical Specifications (TS) Bases Control Program."

No changes are being made to the current RCS PIV leakage limits, actions for inoperable PIVs, or surveillance frequencies contained in the Salem Unit 1 and 2 TS.

3.0 BACKGROUND

The function of RCS PIVs is to separate the high pressure RCS from an attached low pressure system. Although PIV TS provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit provides indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components.

On May 6, 1991, Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," was issued to provide guidance to remove component lists from the Technical Specifications. The guidance stipulates that the TS requirements are stated in general terms that describe the types of components to which the requirements apply, and that the removal of component lists does not alter existing TS requirements or those components to which they apply. In addition, the removed lists must be included in a plant procedure that is subject to the change control provisions for plant procedures in the Administrative Controls section of TS.

Generic Letter 91-08 provides guidance for preparing a request for a license amendment to remove component lists from technical specifications (TS). The nuclear industry and the U.S. Nuclear Regulatory Commission (NRC) identified this line-item TS improvement during investigations of TS problems.

The removal of component lists from TS permits administrative control of changes to these lists without processing a license amendment. Any change to component lists contained in plant procedures is subject to the requirements specified in the Administrative Controls section of the TS on changes to plant procedures. Therefore, the change control provisions of the TS provide an adequate means to control changes to these component lists, when they have been incorporated into plant procedures, without including them in TS.

An Enclosure to the Generic Letter provided additional guidance for changing individual TS sections. At the time of issuance in 1991, the Enclosure to GL 91-08 specifically addressed the issue of PIVs stating:

> Guidance on removing from the TS the list of reactor coolant system pressure isolation valves is pending the NRC staff's resolution of generic concerns with existing lists for these valves. In the interim, licensees should not submit proposals to remove this list from the TS.

The NRC has since resolved the Generic Safety Issue referenced in the GL Enclosure. On July 1, 1993, NUREG-1463, "Regulatory Analysis for the Resolution of Generic Safety Issue 105: Interfacing System Loss-of-Coolant Accident in Light-Water Reactors" was issued. The NUREG addressed the outstanding Generic Safety Issue (GSI) 105 regarding Interfacing Systems Loss-of-Coolant Accident (ISLOCA) and PIVs. Additionally, the NRC has since approved NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," which does not include PIV Tables. In addition, the NRC has approved specific LARs for relocation of PIV Tables from TS (see References 7.4, 7.5, 7.6, 7.7).

4.0 TECHNICAL ANALYSIS

PSEG proposes to relocate the RCS PIV component list to the Salem TRM. The TRM is a PSEG controlled document that has been developed to contain requirements relocated from the TS. The TRM is described in Salem Updated Final Safety Analysis Report (UFSAR) section 13.5.4 and is controlled in a manner consistent with procedures fully or partially described in the UFSAR. Revisions to the TRM are reviewed pursuant to 10 CFR 50.59.

Relocating the PIV component list from the TS will eliminate the burden of processing license amendments when changes are made to the PIV Table and will facilitate the more effective utilization of NRC and PSEG resources.

GL 91-08, relating to the issue of removing component lists from the TS, states in part:

> This guidance includes the incorporation of lists into plant procedures that are subject to the change control provisions for plant procedures In the Administrative Controls Section of the TS. The removal of component lists from TS permits administrative control of changes to these lists without processing a license amendment, as is required to update TS component lists. Any change to component lists contained in plant procedures is subject to the requirements specified in the Administrative Controls Section of the TS on changes to plant procedures. Therefore, the change control provisions of the TS provide an adequate means to control changes to these component lists, when they have been incorporated into plant procedures, without including them in TS.

Specific items identified in Enclosure 1 to GL 91-08 to be addressed with a request to remove component lists from the TS include:

1. Each TS should include an appropriate description of the scope of the components to which the TS requirements apply. Components that are defined by regulatory requirements or guidance need not be clarified further. However, the Bases section of the TS should reference the applicable requirements or guidance.

2. If the removal of a component list results in the loss of notes that modify or provide an exception to the TS requirements, the specification should be revised to incorporate that modification or exception. The modification or exception should be stated in terms that identify any group of components by function rather than by plant identification number, if practical.

3. Licensees should confirm that the lists of components removed from the TS are located in appropriately controlled plant procedures. The list of components may be included in the next update of the FSAR. The Bases section of individual specifications also may reference the plant procedures or other documents that identify each component list.

With regard to item (1) above, PIVs are described in NUREG-1431 as any two normally closed valves in series within the reactor coolant pressure boundary which separate the high pressure RCS from an attached low pressure system. The TS requirements for LCO, Actions, and SR relating to PIVs remain applicable. Therefore, removal of the RCS PIV component list does not affect the scope of components to which the TS requirements apply. Per the proposed changes in Attachment 3, the TS Bases now describe PIVs, which is consistent with the NUREG-1431 Bases.

With regard to item (2) above, the Salem Unit 1 RCS PIV leakage acceptance criteria and associated notes are being relocated from Table 4.4-3 to SR 4.4.6.3. For Salem Unit 2 there are no notes, exceptions, or modifications listed directly in Table 3.4-1.

With regard to item (3), PSEG will relocate the list of PIVs to the TRM, which is an appropriately controlled plant procedure, during the implementation of this LAR .

GL 91-08 provided the guidance for changing individual TS sections. The guidance written in the Generic Letter was written prior to the resolution of GSI 105, which discusses Interfacing Systems Loss of Coolant Accidents. The enclosure to GL 91-08 specifically addresses the issue of PIVs and this GSI stating:

> Guidance on removing from the TS the list of reactor coolant system pressure isolation valves is pending the NRC staff's resolution of generic concerns with existing lists for these valves. In the interim, licensees should not submit proposals to remove this list from the TS.

Explicit guidance on removal of lists of PIVs from the TS has not been issued by the NRC. However, in September 1992, the NRC issued NUREG-1431, Rev 0, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431 TS Section 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," does not contain a list of PIVs.

PSEG concludes that the proposed change to relocate the list of PIVs from the TS to the TRM is administrative in that it merely relocates the component list. No changes are being made to the current RCS PIV leakage limits, actions for inoperable PIVs, or surveillance frequencies contained in the Salem Unit 1 and 2 TS.

PSEG determined that the relocation of PIV component list does not eliminate the requirements for the licensee to ensure that the RCS pressure isolation valves are capable of performing their safety function. Although the PIV component list is relocated from the TSs to the TRM, the information being relocated will be controlled and further revisions to the TRM Table will be subject to 10 CFR 50.59.

5.0 REGULATORY ANALYSIS

5.1 <u>No Significant Hazards Consideration</u>

PSEG requests an amendment to the Salem Unit 1 and 2 Operating Licenses. The proposed amendment revises Unit 1 TS 3/4.4.6.3, Primary Coolant System Pressure Isolation Valves, and Unit 2 TS 3/4.4.7.2, Operational Leakage. Specifically, this change relocates the reactor coolant system (RCS) pressure isolation valve (PIV) lists, Unit 1 TS Table 4.4-3 and Unit 2 TS Table 3.4-1, from the TS to the Salem Technical Requirements Manual (TRM). In addition, the references to the TS Tables are being removed from the TS Limiting Condition for Operation (LCO) 3.4.6.3 (Unit 1) and 3.4.7.2.f (Unit 2), TS Action 3.4.6.3.a (Unit 1) and Surveillance

Requirements (SR) 4.4.6.3 (Unit 1) and 4.4.7.2.2 (Unit 2). The Unit 1 PIV leakage acceptance criteria contained in Unit 1 TS Table 4.4-3 is relocated to Unit 1 SR 4.4.6.3.

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 will not alter the way any structure, system, or component (SSC) functions, and will not alter the manner in which the plant is operated. The proposed changes do not alter the design of any SSC. The relocation of the RCS PIV valve lists from the TS to the TRM is an administrative change. Future revisions to the TRM are subject to 10 CFR 50.59. Therefore the probability of an accident previously evaluated is not significantly increased.

   The proposed changes do not alter the RCS PIV leakage limits contained in the TS nor do they alter the frequency for testing of the RCS PIV. Therefore, the consequences of an accident previously evaluated are not 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 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. The proposed changes are administrative in nature.

   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 RCS PIV TS are administrative in nature. The proposed changes do not alter the RCS PIV leakage limits contained in the TS nor do they alter the frequency for testing of the RCS PIV. The proposed changes will not result in changes to system design or setpoints that are intended to ensure timely identification of plant conditions that could be precursors to accidents or potential degradation of accident mitigation systems.

The proposed amendment will not result in a design basis or safety limit being exceeded or altered. 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.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criteria (GDC)

Salem was designed and constructed in accordance with Atomic Energy Commission (AEC) proposed General Design Criteria published in July 1967. The applicable AEC proposed criteria, as document in Salem UFSAR Section 3.1, were compared to 10 CFR 50 Appendix A General Design Criteria (GDC) as discussed below. The applicable GDC criteria are GDC 14, 54, and 55.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC Criterion 14 is similar to AEC Criterion 9.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC Criterion 54 is similar to AEC Criterion 51 and 57.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or
(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Salem performed a comparison to GDC Criterion 55 and stated in UFSAR Section 3.1.3 that valve arrangements that do not comply are discussion in UFSAR Section 6.2.4. USFAR section 6.2.4.1 states in part:

- ...the two barriers may consist of: (a) two closed piping systems or vessels, one inside and one outside the containment, (b) two automatic isolation valves, one inside and one outside containment, (c) an automatic isolation valve inside the containment and a closed system outside the containment, (d) an automatic isolation valve outside the containment and a closed system inside the containment, or (e) an automatic isolation valve outside containment and a closed system outside the containment.
- A check valve on an incoming line or a normally closed valve is considered an automatic valve.

10 CFR 50.55a

(c) *Reactor coolant pressure boundary*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code as specified in this paragraph.

(f) *Inservice testing requirements*. Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code and ASME Code for Operation and Maintenance of Nuclear Power Plants as specified in this paragraph.

The administrative change to relocate the RCS PIV component list to the TRM was generically approved by the NRC in NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," which is consistent with the NRC Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, and 10 CFR 50.36.

Generic Letter 91-08, "Removal of Component Lists from Technical Specifications," provides guidance to remove component lists from the Technical Specifications.

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. 10 CFR 50.36, "Technical Specifications"
2. NUREG-1431, Revision 4.0, Standard Technical Specifications Westinghouse Plants, April 2012
3. Generic Letter 91-08, " Removal of Component Lists from Technical Specifications"
4. NRC Safety Evaluation Related to Amendment No. 44 , Seabrook Station, dated November 28, 1995
5. NRC Safety Evaluation Related to Amendment No. 76, River Bend Station, dated March 8, 1995
6. NRC Safety Evaluation Related to Amendment Nos 182 and 144, Limerick Generating Station Unit Nos. 1 and 2, dated February 17, 2006.
7. NRC Safety Evaluation Related to Amendment No. 206, Nine Mile Point Unit 1, dated July 26, 2010.

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
3/4.4.6.3, Primary Coolant System Pressure Isolation Valves	3/4 4-16a, 16b, 16c

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

Technical Specification	Page
3/4.4.7.2, Operational Leakage	3/4 4-17, 18, 19

REACTOR COOLANT SYSTEM

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.6.3 Reactor Coolant System Pressure Isolation Valves ~~specified in table 4.4-3~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the specified limit ~~in Table 4.4-3;~~ isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.3 Each Reactor Coolant System Pressure Isolation Valve ~~specified in Table 4.4-3~~ shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be ~~within its limit:~~ ≤ 5.0 gpm for each valve (a)(b):

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves ~~listed in Table 4.4-3~~ the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

Insert Notes (a) and (b) from Table 4.4-3

Relocate to the Technical Requirements Manual

Leakage Criteria moved to SR 4.4.6.3

TABLE 4.4-3

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum[(a) (b)] Allowable Leakage
Low Pressure Safety Injection		≤ 5.0 GPM each valve
Loop 11, cold leg	11SJ56	≤ 5.0 GPM each valve
	11SJ43	≤ 5.0 GPM each valve
Loop 12, cold leg	12SJ56	≤ 5.0 GPM each valve
	12SJ43	≤ 5.0 GPM each valve
Loop 13, cold leg	13SJ56	≤ 5.0 GPM each valve
	13SJ43	≤ 5.0 GPM each valve
Loop 13, hot leg	13SJ156	≤ 5.0 GPM each valve
	13RH27	≤ 5.0 GPM each valve
Loop 14, cold leg	14SJ56	≤ 5.0 GPM each valve
	14SJ43	≤ 5.0 GPM each valve
Loop 14, hot leg	14SJ156	≤ 5.0 GPM each valve
	14RH27	≤ 5.0 GPM each valve
Intermediate Pressure Safety Injection		≤ 5.0 GPM each valve
Loop 11, cold leg	11SJ144	≤ 5.0 GPM each valve
Loop 11, hot leg	11SJ156	≤ 5.0 GPM each valve
	11SJ139	≤ 5.0 GPM each valve
Loop 12, cold leg	12SJ144	≤ 5.0 GPM each valve
Loop 12, hot leg	12SJ156	≤ 5.0 GPM each valve
	12SJ139	≤ 5.0 GPM each valve
Loop 13, cold leg	13SJ144	≤ 5.0 GPM each valve
Loop 13, hot leg	13SJ156	≤ 5.0 GPM each valve
	13SJ139	≤ 5.0 GPM each valve
Loop 14, cold leg	14SJ144	≤ 5.0 GPM each valve
Loop 14, hot leg	14SJ156	≤ 5.0 GPM each valve
	14SJ139	≤ 5.0 GPM each valve
Safety Injection Accumulators to cold leg		
loop 11, cold leg	11SJ55	≤ 5.0 GPM each valve
loop 12, cold leg	12SJ55	≤ 5.0 GPM each valve
loop 13, cold leg	13SJ55	≤ 5.0 GPM each valve
loop 14, cold leg	14SJ55	≤ 5.0 GPM each valve
Safety Injection Boron Injection to cold legs		
loop 11, cold leg	11SJ17	≤ 5.0 GPM each valve
loop 12, cold leg	12SJ17	≤ 5.0 GPM each valve
loop 13, cold leg	13SJ17	≤ 5.0 GPM each valve
loop 14, cold leg	14SJ17	≤ 5.0 GPM each valve
	1SJ150	≤ 5.0 GPM each valve
RHR Suction		
loop 11	1RH1	≤ 5.0 GPM each valve
loop 11	1RH2	≤ 5.0 GPM each valve

~~TABLE 4.4-3 (CONT'D)~~

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable. However, for initial tests, or tests following valve repair or replacement, leakage rates less than or equal to 5.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

(b) Minimum differential test pressure shall not be less than 150 psid.

Notes (a) and (b) are relocated to SR 4.4.6.3

Insert

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. NOT USED

f. 1 GPM leakage at a Reactor Coolant System pressure of 2230 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-1~~.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or primary-to-secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, and primary-to-secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor in accordance with the Surveillance Frequency Control Program.

b. Monitoring the containment sump inventory in accordance with the Surveillance Frequency Control Program.

c*. Verifying primary-to-secondary leakage is ≤ 150 gallons per day through any one steam generator in accordance with the Surveillance Frequency Control Program during steady state operation,

d*. Performance of a Reactor Coolant System water inventory balance** in accordance with the Surveillance Frequency Control Program. The water inventory balance shall be performed with the plant at steady state conditions. The provisions of specification 4.0.4 are not applicable for entry into Mode 4, and

e. Monitoring the reactor head flange leakoff system in accordance with the Surveillance Frequency Control Program.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve ~~specified in Table 3.4-1~~ shall be demonstrated OPERABLE pursuant to the INSERVICE TESTING PROGRAM, except that in lieu of any leakage testing required by the INSERVICE TESTING PROGRAM, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. In accordance with the Surveillance Frequency Control Program.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

d. For the Residual Heat Removal and Safety Injection Systems hot and cold leg injection valves and accumulator valves ~~listed in Table 3.4-1~~ the testing will be done within 24 hours following valve actuation due to automatic or manual action or flow through the valve. For all other systems testing will be done once per refueling.

The provisions of specification 4.0.4 are not applicable for entry into MODE 3 or 4.

* Not required to be completed until 12 hours after establishment of steady state operation.

** Not applicable to primary-to-secondary leakage.

REACTOR COOLANT SYSTEM

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NO	FUNCTION
21SJ43	Safety Injection (L.P. from RHR Pumps to Cold Legs)
22SJ43	Safety Injection (L.P. from RHR Pumps to Cold Legs)
23SJ43	Safety Injection (L.P. from RHR Pumps to Cold Legs)
24SJ43	Safety Injection (L.P. from RHR Pumps to Cold Legs)
21SJ55	Safety Injection (Accumulator Discharge to Cold Legs)
22SJ55	Safety Injection (Accumulator Discharge to Cold Legs)
23SJ55	Safety Injection (Accumulator Discharge to Cold Legs)
24SJ55	Safety Injection (Accumulator Discharge to Cold Legs)
21SJ56	Safety Injection (Accumulator Discharge to Cold Legs)
22SJ56	Safety Injection (Accumulator Discharge to Cold Legs)
23SJ56	Safety Injection (Accumulator Discharge to Cold Legs)
24SJ56	Safety Injection (Accumulator Discharge to Cold Legs)
21SJ17	Safety Injection (Boron Injection to Cold Legs)
22SJ17	Safety Injection (Boron Injection to Cold Legs)
23SJ17	Safety Injection (Boron Injection to Cold Legs)
24SJ17	Safety Injection (Boron Injection to Cold Legs)
2SJ150	Safety Injection (Boron Injection to Cold Legs)
21SJ139	Safety Injection (H.P. from SI Pumps to Hot Legs)
22SJ139	Safety Injection (H.P. from SI Pumps to Hot Legs)
23SJ139	Safety Injection (H.P. from SI Pumps to Hot Legs)
24SJ139	Safety Injection (H.P. from SI Pumps to Hot Legs)
21SJ156	Safety Injection (H.P. from SI Pumps to Hot Legs)
22SJ156	Safety Injection (H.P. from SI Pumps to Hot Legs)
23SJ156	Safety Injection (H.P. from SI Pumps to Hot Legs)
24SJ156	Safety Injection (H.P. from SI Pumps to Hot Legs)
21SJ144	Safety Injection (H.P. from SI Pumps to Cold Legs)
22SJ144	Safety Injection (H.P. from SI Pumps to Cold Legs)
23SJ144	Safety Injection (H.P. from SI Pumps to Cold Legs)
24SJ144	Safety Injection (H.P. from SI Pumps to Cold Legs)
2RH1	RHR Suction from Hot Leg No. 21
2RH2	RHR Suction from Hot Leg No. 21
23RH27	RHR Discharge to Hot Leg No. 23
24RH27	RHR Discharge to Hot Leg No. 24

Relocate to the Technical Requirements Manual

INSERT

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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:

<u>**Technical Specification**</u>	<u>**Page**</u>
3/4.4.6.3, Primary Coolant System Pressure Isolation Valves (new)	B 3/4 4-4b

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

<u>**Technical Specification Bases**</u>	<u>**Page**</u>
3/4.4.7.2, Operational Leakage	B 3/4 4-4

BASES Insert 1:

3/4.4.6.3 REACTOR COOLANT SYSTEM (RCS) PRESSURE ISOLATION VALVES (PIV)

The function of the RCS PIVs is to separate the high pressure RCS from the attached low pressure systems. The PIV leakage limit applies to each individual valve listed in the Technical Requirements Manual. Leakage through both series PIVs in a line must be included as part of the IDENTIFIED LEAKAGE, governed by LCO 3.4.6.2, "Operational Leakage." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 4.4.6.2.d). A known component of the IDENTIFIED LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational leakage if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

Bases Insert 2:

The function of the RCS PIVs is to separate the high pressure RCS from the attached low pressure systems. The PIV leakage limit applies to each individual valve listed in the Technical Requirements Manual. Leakage through both series PIVs in a line must be included as part of the IDENTIFIED LEAKAGE, governed by LCO 3.4.7.2, "Operational Leakage." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 4.4.7.2.1.d). A known component of the IDENTIFIED LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational leakage if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The action times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Surveillances

Verifying RCS leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance. The RCS water inventory must be met with the reactor at steady state conditions. The surveillance is modified by a Note that the surveillance is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program.

Mode ascension to MODE 1-3 is acceptable without a current RCS Inventory Balance, provided the asterisked note of "Not required to be completed until 12 hours after establishment of steady state operations", is complied with.

Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If SR 4.4.6.2.c is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines). If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one Steam Generator. The Surveillance is modified by a Note which states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-~~secondary leakage determination, steady state is defined as stable RCS~~ pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling (in accordance with EPRI PWR Primary-to-Secondary Leak Guidelines).

Insert 1

3/4.4.7

THIS SECTION DELETED

3/4.4.7 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.7.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.7.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The ~~surveillance~~ requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity ~~thereby~~ reducing the probability of gross valve failure and consequent intersystem LOCA. ~~Leakage~~ from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a ~~portion of the~~ allowed limit.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Primary to Secondary Leakage Through Any One SG

The primary-to-secondary leakage rate limit applies to leakage through any one Steam Generator. The limit of 150 gallons per day per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures. The dosage contribution from the tube leakage will be within 10 CFR 50.67 limits in the event of either a steam generator tube rupture or steam line break. The analyses are based on the total primary to secondary leakage from all SGs of 1 gallon per minute as a result of accident induced conditions.

INSERT 2