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PSEG
Nuclear L.L.C.

SEP 1 4 2009

10 CFR 50.90

LR-N09-0164
LAR S09-03

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

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

Subject: **License Amendment Request to Correct Technical Specification and Facility Operating License Editorial Items**

In accordance with the provisions of 10CFR50.90, PSEG Nuclear LLC (PSEG) requests an amendment to the facility operating licenses listed above. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed changes correct editorial items in the Technical Specifications (TS) and Facility Operating License (FOL) for Salem Units 1 and 2. These items are either historical in nature, or are errors inadvertently created during the submittal of other license amendment requests and subsequent license amendments. The proposed changes are administrative in nature and fall into one of four categories: (1) correct typographical errors, (2) correct format errors, (3) correct administrative differences between Units, or (4) delete historical requirements that have expired. The TS affected are 3.1.3.3 ACTION b (Unit 1 and 2), 3.2.1 (Unit 2), Surveillance Requirement (SR) 4.2.2.2 (Unit 1 and 2), Table 3.2-1 (Unit 1), Table 3.3-6 (Unit 1 and 2), Table 4.3-3 (Unit 1), 3.3.3.14 (Unit 1 and 2), SR 4.4.11.2 (Unit 2), 6.9.1.5.c (Unit 2), 6.9.1.9 (Unit 2), Amendment 222 (Unit 1) one time changes (various SR), and Amendment 230 (Unit 1) one time changes (3.1.3.1.2, SR 4.1.3.1.1). The FOL sections affected are Unit 1 FOL Attachment 1 and Unit 2 FOL Condition 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25.

Attachment 1 of this submittal provides an evaluation supporting the proposed changes. Attachment 2 provides the marked-up TS and FOL pages, with the proposed changes indicated. No regulatory commitments are contained in this submittal.

The changes in this LAR are not required to address an immediate safety concern; PSEG requests approval of this 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.

A001
NRR

If you have any questions or require additional information, please do not hesitate to contact Mr. Jeff Keenan at (856) 339-5429.

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

Executed on 09/14/09
(Date)

Sincerely,

Robert C. Braun
Site Vice President
Salem Generating Station

Attachments (2)

S. Collins, Regional Administrator - NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Salem
P. Mulligan, Manager IV, NJBNE
Commitment Coordinator - Salem
PSEG Corporate Commitment Manager

License Amendment Request to Correct Technical Specification and Facility Operating License Editorial Items

Table of Contents

1.0 DESCRIPTION

In accordance with the provisions of 10CFR50.90, PSEG Nuclear LLC (PSEG) requests an amendment to the facility operating licenses DPR-70 and DPR-75.

The proposed changes correct editorial items in the Technical Specifications (TS) and Facility Operating License (FOL) for Salem Units 1 and 2. These items are either historical in nature, or are errors inadvertently created during the submittal of other license amendment requests and subsequent license amendments. The proposed changes are administrative in nature and fall into one of three categories: (1) correct typographical errors, (2) correct format errors, (3) correct administrative differences between Units, or (4) delete historical requirements that have expired. The TS/FOL affected sections are 3.1.3.3 ACTION b (Unit 1 and 2), 3.2.1 (Unit 2), Surveillance Requirement (SR) 4.2.2.2 (Unit 1 and 2), Table 3.2-1 (Unit 1), Table 3.3-6 (Unit 1 and 2), Table 4.3-3 (Unit 1), 3.3.3.14 (Unit 1 and 2), SR 4.4.11.2 (Unit 2), 6.9.1.5.c (Unit 2), 6.9.1.9 (Unit 2), Amendment 222 (Unit 1) one time changes (various SR), and Amendment 230 (Unit 1) one time changes (3.1.3.1.2, SR 4.1.3.1.1). The FOL sections affected are Unit 1 FOL Attachment 1 and Unit 2 FOL Condition 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25.

2.0 PROPOSED CHANGES

Item	Description	TS / FOL	Unit	Action
1.	Amendments 201/197 (LCR 94-41) – The Margin Recovery Program removed 3-loop operation references in TS; 3-Loop operation has not been analyzed for Salem. TS 3.1.3.3, ACTION b, was not changed (deleted) due to an oversight. TS 3.1.3.3 ACTION b provides requirements if rod drop time is determined with 3 reactor coolant pumps running in Modes 1 and 2. TS 3.4.1.1 requires that all reactor coolant pumps be in operation in Modes 1 and 2. (Reference 1 and 2)	3.1.3.3 ACTION b	1 and 2	Delete TS 3.1.3.3 ACTION b (Note: Another revision to TS 3.1.3.3 is pending via PSEG LAR S09-01, submitted March 22, 2009)
2.	LCO 3.2.1 contains a typographical error in the acronym for the term "Core Operating Limits Report". The acronym should be "COLR"; instead it is represented as "CORL"	3.2.1	2	Correct typographical error

Item	Description	TS / FOL	Unit	Action
3.	The peaking factor term in SR 4.2.2.2 is incorrectly represented in some cases. Standard convention has the subscript followed by the superscript. The subscript and superscript are reversed in some cases on page 3/4 2-7 of SR 4.2.2.2.	SR 4.2.2.2, page 3/4 2-7	1 and 2	Correct subscript and superscript errors in SR 4.2.2.2, page 3/4 2-7.
4.	SR 4.2.2.2 contains the following note related to Cycle 11: *"*For Cycle 11, when the number of available movable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(1%)] where T is the number of available thimbles."* This note is no longer applicable to subsequent operating cycles; current Salem Unit 2 cycle is Cycle 17.	SR 4.2.2.2	2	Delete historical Cycle 11 footnote
5.	A footnote in Unit 1 TS Table 3.2-1 states that the pressurizer pressure limit is not applicable "during either THERMAL POWER ramp <u>increase</u> in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step <u>increase</u> in excess of 10% RATED THERMAL POWER." The corresponding footnote in Unit 2 TS Table 3.2-1 states that the limit is not applicable "during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER." The word 'increase' in the Unit 1 TS is superfluous and inconsistent with Unit 2 TS, and inconsistent with NUREG 1431 (page 3.4.1-1).	Table 3.2-1	1	Delete the word "increase" (2 instances) from the footnote in Unit 1 TS Table 3.2-1

Item	Description	TS / FOL	Unit	Action
6.	Amendment 280 & 263 (LCR S06-03) – The TS camera ready pages prepared for LCR S06-03 contained format and typographical errors in TS Table 3.3-6; these errors were subsequently included in the TS pages issued with the amendments. Specifically, some symbols were incorrectly reproduced (Unit 1 and 2), and a table heading was misaligned (Unit 2) (Reference 3)	Table 3.3-6	1 and 2	Correct format and typographical errors in Table 3.3-6.
7.	Amendment 225 - a typographical error was noted in the footer of Table 4.3-3, page 3/4 3-38a; the number "1" is missing following the word "Unit" (the "1" has been tabbed-over next to the page number in the center of the footer). (Reference 4)	Table 4.3-3	1	Correct footer alignment
8.	Amendment 282/265 (LCR S05-12) – These amendments approved the relocation of Incore Detection Monitoring TS 3/4.3.3.2 to the UFSAR. During the amendment implementation it was identified that the relocated TS 3.3.3.2 is also referenced in TS 3.3.3.14, Power Distribution Monitoring System (PDMS). The action in TS 3.3.3.14 is to refer to TS 3.3.3.2 if the PDMS was declared inoperable and the incore moveable detector system was to be used. LCR S05-12 was submitted on June 30, 2006 (PSEG letter LR-N06-0079). (Reference 5)	3.3.3.14	1 and 2	Revise TS 3.3.3.14 to remove the reference to the deleted TS 3.3.3.2. The TS will be revised as follows: '...and use the incore movable detector system~~, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2,~~ to obtain any required core power distribution measurements.'

Item	Description	TS / FOL	Unit	Action
9.	SR 4.4.11.2 - the SR is historical and applicable only to the first 3 cycles of operation: *"Augmented Inservice Inspection Program for Steam Generator Channel Heads - The No. 21 Steam Generator channel head shall be ultrasonically inspected in a selected area during each of the <u>first three refueling outages</u> using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material."* The SR is no longer applicable to cycles subsequent to Cycle 3 (current Salem Unit 2 cycle is Cycle 17.)	SR 4.4.11.2	2	Delete SR 4.4.11.2
10.	Incorrect TS Reference: TS 6.9.1.5.c references TS 3.4.8, however it should reference TS 3.4.9 (Reactor Coolant System Specific Activity) for Unit 2. Section 6.9.1.5 is an administrative section of TS that addresses periodic reports.	6.9.1.5.c	2	Correct reference to TS 3.4.9
11.	Incorrect TS Reference: TS 6.9.1.9.a.1 references TS 3/4.1.1.4; however it should reference 3/4.1.1.3 (Moderator Temperature Coefficient) for Unit 2. Section 6.9.1.9 is an administrative section of TS that addresses periodic reports.	6.9.1.9	2	Correct reference to TS 3/4.1.1.3
12.	Amendment 222 allowed a one-time extension of the TS surveillance interval to the end of fuel Cycle 13 for certain TS surveillance requirements (SRs). Specifically, the amendment extended the surveillance interval in (a) SR 4.3.2.1.3; (b) SRs 4.8.2.3.2.f and 4.8.2.5.2.d; (c) SR 4.8.2.5.2.c.2; (d) SR 4.8.3.1.a.1.; (e) SR 4.1.2.2.c; (f) SRs 4.3.1.1.1, Table 4.3-1, 4.3.2.1.1, Table 4.3-2, and 4.3.3.7, Table 4.3-11; (g) SR 4.5.1.d; (h) SR 4.5.2.e.1; (i) SR 4.7.6.1.d.2; (j) SR 4.7.10.b; and (k) SR	See description	1	Delete historical Cycle 13 notes

Item	Description	TS / FOL	Unit	Action
	4.8.1.1.2.d.7. Because of the length of outage 1R12 and delays in restart, the SRs would have been overdue prior to reaching the next refueling outage (1R13). These Cycle 13 notes are no longer applicable to subsequent operating cycles; current Salem Unit 1 cycle is Cycle 20. Note that the Cycle 13 notes for item (a), SR 4.3.2.1.3, (b), SR 4.8.2.5.2.d (only), (c), SR 4.8.2.5.2.c.2, and (k), SR 4.8.1.1.2.d.7, have already been removed from TS. (Reference 6)			
13.	Amendment 230 modified TS 3.1.3.2.1, Action a.1, to determine the position of Rod 1 SB2 from once every 8 hours to within 8 hours following any movement of the rod until repair of the rod indication system was completed. This change was applicable for the remainder of Unit 1 Cycle 14, or until an outage of sufficient duration occurred to repair the system. Surveillance Requirements (SR) of TSs 4.1.3.1.1 and 4.1.3.4 were also modified to require that the position of Rod 1 SB2 be determined (by the incore system) only following movement of the rod until repair of the indication system was completed. These requirements are historical, they were only applicable for Cycle 14 and are no longer applicable to subsequent operating cycles; current Salem Unit 1 cycle is Cycle 20. (Reference 7)	3.1.3.2.1 4.1.3.1.1 4.1.3.4	1	Delete historical Cycle 14 notes
14.	Unit 1 FOL Attachment 1, "Incomplete preoperational Tests, Startup Tests, and Other Items Which Must be Completed" is a listing of required actions related to Unit 1 initial plant start-up. These items are all historical and can be removed from the FOL.	FOL Attachment 1	1	Delete FOL Attachment 1 and the reference to Attachment 1 on FOL page 5c.

Item	Description	TS / FOL	Unit	Action
15.	Unit 2 FOL Condition 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25 (Unit 2). These conditions comprise actions related to (a) Unit 2 initial plant start-up and operation through the first cycle and refueling outage, (b) the second refueling outage, (c) issuance of the license, or (d) actions required to be completed prior to June 1, 1983. These items are all historical and can be removed from the FOL.	FOL 2.C	2	Delete FOL items 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25.

The marked up TS and FOL pages are provided in Attachment 2.

3.0 BACKGROUND

Background information is provided in the Table in Section 2.0

4.0 TECHNICAL ANALYSIS

The proposed changes are administrative in nature that correct (1) typographical errors, (2) correct format errors, (3) correct administrative differences between Units, or (4) delete historical requirements that have expired, as discussed below.

The proposed deletion of Salem Unit 1 and 2 TS 3.1.3.3, ACTION b is editorial in nature, correcting an oversight in LCR 94-21. LCR 94-41, The Margin Recovery Program, and the subsequent Amendments 201 and 197 removed 3-loop operation references in TS; 3-Loop operation has not been analyzed for Salem. Since 3-loop operation is not approved for Salem, there should be no references for its use in TS. TS 3.1.3.3 ACTION b provides requirements if rod drop time is determined with 3 reactor coolant pumps running in Modes 1 and 2. TS 3.4.1.1 requires that all reactor coolant pumps be in operation in Modes 1 and 2. TS 3.1.3.3, ACTION b, was not changed (deleted) due to an oversight in the preparation of the TS markups in LCR 94-21.

The peaking factor term in SR 4.2.2.2, is incorrectly represented in some cases. Standard convention has the subscript followed by the superscript. The subscript and superscript are reversed in some cases on page 3/4 2-7 of SR 4.2.2.2; this submittal will correct those errors.

The proposed change to LCO 3.2.1 is editorial in nature, correcting a typographical error. The acronym for the term “Core Operating Limits Report” is represented as “CORL”; instead it should be “COLR”.

The proposed change to Salem Unit 2 TS 4.2.2.2 is editorial in nature, eliminating a condition approved only for Cycle 11, which has expired.

The proposed change to Salem Unit 1 TS Table 3.2-1 is editorial in nature, correcting a typographical error; the word ‘increase’ is inserted superfluously in two places. A footnote in Unit 1 TS Table 3.2-1 states that the pressurizer pressure limit is not applicable "during either THERMAL POWER ramp <u>increase</u> in excess of 5% RATED THERMAL POWER per minute or a

THERMAL POWER step <u>increase</u> in excess of 10% RATED THERMAL POWER." The corresponding footnote in Unit 2 TS Table 3.2-1 states that the limit is not applicable "during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER." The word 'increase' in the Unit 1 TS is superfluous and inconsistent with Unit 2 TS, and inconsistent with NUREG 1431 (page 3.4.1-1).

The proposed change to Salem Unit 1 and 2 TS Table 3.3-6 is editorial in nature, correcting typographical and formatting errors. The TS camera ready pages prepared for LCR S06-03 (Amendments 280 and 263) contained format and typographical errors in TS Table 3.3-6; these errors were subsequently included in the TS pages issued with the amendments. Specifically, two symbols ($\leq$ and μ) were incorrectly reproduced (Unit 1 and 2), and a table heading was misaligned (Unit 2).

The proposed change to Salem Unit 1 Table 4.3-3 is editorial in nature, correcting a typographical error. Following the implementation of Amendment 225 a typographical error was noted in the footer of Table 4.3-3, page 3/4 3-38a; specifically the number "1" is missing following the word "Unit" (the "1" has been tabbed-over next to the page number in the center of the footer). This submittal will correct the error.

The proposed change to Salem Unit 1 and 2 TS 3.3.3.14, is editorial in nature, correcting an oversight in LCR S05-12. LCR S05-12 and subsequent Amendments 282 and 265 relocated the Incore Detection Monitoring TS 3/4.3.3.2 to the UFSAR (PSEG intends to ultimately place the requirements in the Technical Requirements Manual (TRM)). During the amendment implementation it was identified that the relocated TS 3.3.3.2 is also referenced in TS 3.3.3.14, Power Distribution Monitoring System (PDMS). The action in TS 3.3.3.14 is to refer to TS 3.3.3.2 if the PDMS was declared inoperable and the incore moveable detector system was to be used. This submittal will revise TS 3.3.3.14 to delete the reference to TS 3.3.3.2 and state that if the PDMS is declared inoperable then the incore movable detector system should be used to obtain any required core power distribution measurements. The requirements for the incore movable detector system have been relocated to a document controlled by 10CFR 50.59 (e.g., the UFSAR or TRM).

The proposed change to Salem Unit 2 SR 4.4.11.2 is editorial in nature, eliminating a condition, approved only for the first 3 refueling outages, which has expired. The SR is historical and no longer applicable to cycles subsequent to Cycle 3 (current Salem Unit 2 cycle is Cycle 17).

The proposed change to Salem Unit 2 TS 6.9.1.5.c is editorial in nature, correcting a typographical error. TS 6.9.1.5.c incorrectly references TS 3.4.8; it should reference TS 3.4.9 (Reactor Coolant System Specific Activity) for Unit 2. Section 6.9.1.5 is an administrative section of TS that addresses periodic reports.

The proposed change to Salem Unit 2 TS 6.9.1.9.a.1 is editorial in nature, correcting a typographical error. TS 6.9.1.5.c incorrectly references TS 3 TS 3/4.1.1.4; it should reference TS 3/4.1.1.3 (Moderator Temperature Coefficient) for Unit 2. Section 6.9.1.9 is an administrative section of TS that addresses periodic reports.

The proposed change to the SRs changed by Amendment 222 is editorial in nature, eliminating conditions approved only for Cycle 13, which has expired.

The proposed change to the TS and SR changed by Amendment 230 is editorial in nature, eliminating conditions approved only for Cycle 14, which has expired.

Unit 1 FOL Attachment 1, "Incomplete preoperational Tests, Startup Tests, and Other Items Which Must be Completed" is a listing of required actions solely related to Unit 1 initial plant start-up. These items are all historical and can be removed from the FOL.

Unit 2 FOL Condition 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25 comprise actions solely related to (a) Unit 2 initial plant start-up and operation through the first cycle and refueling outage, (b) the second refueling outage, (c) issuance of the license, or (d) actions required to be completed prior to June 1, 1983. These items are all historical and can be removed from the FOL.

5.0 REGULATORY ANALYSIS

10 CFR 50.36 (a)(1) requires that each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed TS in accordance with the requirements of section 50.36. The TS are part of the FOL and any changes to the FOL and TS must be in accordance with 10 CFR 50.90. The corrections proposed by this license amendment request conform to these regulations.

5.1 No Significant Hazards Consideration

PSEG requests an amendment to the Salem Unit 1 and 2 Operating Licenses. The proposed changes correct editorial items in the Technical Specifications (TS) and Facility Operating License (FOL) for Salem Units 1 and 2. These items are either historical in nature, or are errors inadvertently created during the submittal of other license amendment requests and subsequent license amendments. The proposed changes are administrative in nature and fall into one of four categories: (1) correct typographical errors, (2) correct format errors, (3) correct administrative differences between Units, or (4) delete historical requirements that have expired. The TS involved are 3.1.3.3 ACTION b (Unit 1 and 2), 3.2.1 (Unit 2), Surveillance Requirement (SR) 4.2.2.2 (Unit 1 and 2), Table 3.2-1 (Unit 1), Table 3.3-6 (Unit 1 and 2), Table 4.3-3 (Unit 1), 3.3.3.14 (Unit 1 and 2), SR 4.4.11.2 (Unit 2), 6.9.1.5.c (Unit 2), 6.9.1.9 (Unit 2), Amendment 222 (Unit 1) one time changes (various SR), and Amendment 230 (Unit 1) one time changes (3.1.3.1.2, SR 4.1.3.1.1). The FOL sections affected are Unit 1 FOL Attachment 1 and Unit 2 FOL Condition 2.C.3 through 2.C.9 and 2.C.11 through 2.C.25.

PSEG has evaluated the proposed changes to the TS and FOL for the stations listed above, using the criteria in 10CFR50.92, 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 consideration.

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

   Response: No

   The proposed changes to TS and the FOL are administrative in nature that correct typographical errors, correct format errors, correct inconsistencies between Units, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

The proposed change does not have any impact on structures, systems and components (SSCs) of the plant, and no affect on plant operations. The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility. Therefore, this proposed change does not represent a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No

   The proposed changes to TS and the FOL are administrative in nature that correct typographical errors, correct format errors, correct inconsistencies between Units, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

   The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements 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 TS and the FOL are administrative in nature that correct typographical errors, correct format errors, correct inconsistencies between Units, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

   The proposed change incorporates corrections to the TS and FOL and result in improved accuracy of these licensing documents. There is no change to any design basis, licensing basis or safety limit, no change to any parameters; consequently no safety margins are affected. Therefore, 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 10CFR50.92 (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.

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 a 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) Letter from NRC to PSEG: "Salem Nuclear Generating Station, Unit 1 (TAC NO M95383)", Amendment 201, dated November 26, 1997. (ML011720441)

(2) Letter from NRC to PSEG: "Salem Nuclear Generating Station, Unit 2 (TAC NO M95384)", Amendment 197, dated January 8, 1999. (ML0117230279)

(3) Letter from NRC to PSEG: "SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: ACCIDENT MONITORING INSTRUMENTATION AND SOURCE CHECK DEFINITION (TAC NOS. MD1654, MD1655, MD1656 AND MD1657)", Amendment 280/263, dated April 19, 2007. (ML070920317)

(4) Letter from NRC to PSEG: "SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 AND 2, ISSUANCE OF AMENDMENT RE: ADMINISTRATIVE AND EDITORIAL CHANGES (TAC NOS. MA0180 AND MA0181)", Amendment 225/206, dated November 2, 1999. (ML993240246)

(5) Letter from NRC to PSEG: "SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS RE: RELOCATION OF TECHNICAL SPECIFICATION REQUIREMENTS FOR THE MOVABLE INCORE DETECTORS AND RADIOACTIVE GASEOUS EFFLUENT OXYGEN MONITORING INSTRUMENTATION (TAC NOS. MD2505 AND MD2506)", Amendment 282/265, dated June 6, 2007. (ML071200393)

(6) Letter from NRC to PSEG: "SALEM NUCLEAR GENERATING STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: ONE-TIME EXTENSION OF SURVEILLANCE INTERVAL (TAC NO. MA4554)", dated May 4, 1999 (ML011730078)

(7) Letter from NRC to PSEG: "SALEM NUCLEAR GENERATING STATION, UNIT NO. 1, ISSUANCE OF AMENDMENT RE: EXIGENT REQUEST FOR CHANGES TO TECHNICAL SPECIFICATIONS -POSITION INDICATION SYSTEM (TAC NO. MA8840)", dated May 26, 2000 (ML003719424)

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

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

Technical Specification	Page
4.1.2.2.c	3/4 1-9
4.1.3.1.1	3/4 1-18a
3.1.3.2.1	3/4 1-19
3.1.3.3	3/4 1-21
4.1.3.4	3/4 1-22
4.2.2.2	3/4 2-7
Table 3.2-1	3/4 2-14
Table 4.3-1	3/4 3-13
Table 4.3-2	3/4 3-31a
Table 3.3-6	3/4 3-36a
Table 4.3-3	3/4 3-38a
Table 4.3-11	3/4 3-57a
3.3.3.14	3/4 3-71
4.5.1.d	3/4 5-2
4.5.2.e.1	3/4 5-5
4.7.6.1	3/4 7-21
4.7.10.b	3/4 7-34
4.8.2.3.2.f	3/4 8-9a
4.8.3.1.a.1	3/4 8-14

Facility Operating License	Page
Attachments	5c
Attachment 1	1 through 4

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

Technical Specification	Page
3.1.3.3	3/4 1-18
3.2.1	3/4 2-1
4.2.2.2	3/4 2-7
Table 3.3-6	3/4 3-39a
3.3.3.14	3/4 3-66
4.4.11.2	3/4 4-33
6.9.1.5.c	3/4 6-21
6.9.1.9	3/4 6-24

Facility Operating License	Page
Section 2.C	4 through 21

UNIT 1 TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

Facility Operating License DPR-70

SURVEILLANCE REQUIREMENTS (Continued)

c. * At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

d. At least once per 18 months by verifying that the flow path required by specification 3.1.2.2.a delivers at least 33 gpm to the Reactor Coolant System.

~~* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

a) A reevaluation of each accident analysis of table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

c) A core power distribution measurement is obtained and $F_Q(Z)$ $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER. THERMAL POWER shall be maintained less than or equal to 75% of RATED THERMAL POWER until compliance with ACTIONS 3.1.3.1.c.3.a and 3.1.3.1.c.3.c above are demonstrated.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the limits established in the limiting condition for operation at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.~~*~~

4.1.3.1.2 Each full length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

~~* During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position of Rod 1SB2 will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.~~

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2.1 The shutdown and control rod position indication systems shall be OPERABLE and capable of determining the actual and demanded rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

All Shutdnwn Banks: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

Control Rank A: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 200-228 steps.

Control Bank B: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-30 steps and 160-228 steps.

Control Bank C and D: ±18 steps at ≤ 85% reactor power or if reactor power is > 85% RATED THERMAL POWER ± 12 steps of the group demand counters for withdrawal ranges of 0-228 steps.

b. Group demand counters; ± 2 steps of the pulsed output of the Slave Cycler Circuit over the withdrawal range of 0-228 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one analog rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly using the power distribution monitoring system (if power is above 25% RTP) or using the movable incore detectors (if power is less than 25% RTP or the power distribution monitoring system is inoperable) at least once per 8 hours~~*~~ and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

~~* During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position of Rod 1SB2 will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.~~

b. With two or more analog rod position indicators per bank inoperable, within one hour restore the inoperable rod position indicator(s) to OPERABLE status or be in HOT STANDBY within the next 6 hours. A maximum of one rod position indicator per bank may remain inoperable following the hour, with Action (a) above being applicable from the original entry time into the LCO.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn shall be ≤ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

Note: A proposed change to 3.1.3.3 is pending via PSEG LAR S09-01, submitted March 22, 2009.

a. $T_{avg} \geq 541°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

~~b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to ≤71% of RATED THERMAL POWER.~~

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be FULLY WITHDRAWN.

APPLICABILITY: MODES 1*, and 2*#@

ACTION:

With a maximum of one shutdown rod not FULLY WITHDRAWN, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. FULLY WITHDRAW the rod, or,

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be FULLY WITHDRAWN by use of the group demand counters, and verified by the analog rod position indicators**,~~***~~:

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

* See Special Test Exceptions 3.10.2 and 3.10.3

**For power levels below 50% one hour thermal "soak time" is permitted. During this soak time, the absolute value of rod motion is limited to six steps.

~~*** During Cycle 14, the position of Rod 1SB2 will be determined indirectly by the movable incore detectors within 8 hours following its movement until the repair of the indication system for this rod. During reactor startup, the fully withdrawn position of Rod 1SB2 will be determined by current traces and subsequently verified by the movable incore detectors prior to entry into Mode 1.~~

\# With Keff greater than or equal to 1.0

@ Surveillance 4.1.3.4.a is applicable prior to withdrawing control banks in preparation for startup (Mode 2).

SURVEILLANCE REQUIREMENTS (Continued)

b) At least once per 31 EFPD, whichever occurs first.

2. When the $F_{xy}{}^{C}{}_{\cancel{xy}}$ is less than or equal to the $F_{xy}{}^{RTP}{}_{\cancel{xy}}$ limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and $F_{xy}{}^{C}{}_{\cancel{xy}}$ compared to $F_{xy}{}^{RTP}{}_{\cancel{xy}}$ and $F_{xy}{}^{L}{}_{\cancel{xy}}$ at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power ($F_{xy}{}^{RTP}{}_{\cancel{xy}}$) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15% inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ±2%, 46.4 ±2%, 60.6 ±2%, and 74.9 ±2% inclusive.
4. Core plane regions within ±2% of core height (±2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever $F_{xy}{}^{C}{}_{\cancel{xy}}$ exceeds $F_{xy}{}^{L}{}_{\cancel{xy}}$.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS 4 Loops In Operation
Reactor Coolant System T_{avg}	$\leq$ 582.9°F
Pressurizer Pressure	$\geq$ 2200 psia*
Reactor Coolant System Flow	$\geq$ 341,000 gpm#

* Limit not applicable during either THERMAL POWER ramp ~~increase~~ in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step ~~increase~~ in excess of 10% RATED THERMAL POWER.

\# Includes a 2.4% flow measurement uncertainty plus a 0.1% measurement uncertainty due to feedwater venturi fouling.

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual SSPS functional input check every 18 months. ~~**~~

(5) - Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

~~** A one-time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	R~~*~~	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure-High	S	R	Q(3)	1,2,3
d. Pressurizer Pressure--Low	S	R	Q	1,2,3
e. Differential Pressure Between Steam Lines--High	S	R	Q	1,2,3
f. Steam Flow in Two Steam Lines--High coincident with Tavg--Low-Low or Steam Line Pressure-Low	S	R	Q	1,2,3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	R	1,2,3,4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1,2,3,4
c. Containment Pressure--High-High	S	R	Q(3)	1,2,3

~~*A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

TABLE 3.3-6 (Continued)
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 3.0 \times 10^{-2} \mu Ci/cm^3$ (Alarm only)	10^{-3}-$10^{1} \mu Ci/cm^3$	23
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 1.0 \times 10^{2} \mu Ci/cm^3$	10^{-1}-$10^{5} \mu Ci/cm^3$ (Alárm only)	23
3) Condenser Exhaust System	1	1,2,3&4	$\leq 1.27 \times 10^{4}$ cpm (Alarm only)	1-10^{6} cpm	23
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	$\leq 2.48 \times 10^{3}$ cpm	10^{1}-10^{7} cpm	24, 25

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNELS CHECKS	SOURCE CHECKS	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
2) High Range Auxiliary Building Exhaust System (Plant Vent)	S	M	R	Q	1, 2, 3 & 4
3) Condenser Exh. Sys.	S	M	R	Q	1, 2, 3 & 4
3. CONTROL ROOM					
a. Air Intake - Radiation Level	S	M	R	Q	**

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

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

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
12. PORV Position Indicator	M	N.A.	R
13. PORV Block Valve Position Indicator	M	N.A.	Q*
14. Pressurizer Safety Valve Position Indicator	M	N.A.	R
15. Containment Pressure - Narrow Range	M	N.A.	N.A.
16. Containment Pressure - Wide Range	M	R	N.A.
17. Containment Water Level - Wide Range	M	R~~**~~	N.A.
18. Core Exit Thermocouples	M	R	N.A.
19. Reactor Vessel Level Instrumentation System (RVLIS)	M	R	N.A.
20. Containment High Range Accident Radiation Monitor	S	R	Q

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

~~** A one-time extension to this surveillance requirement is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system~~,~~ ~~satisfying the OPERABILITY requirements listed in Specification 3.3.3.2,~~ to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required:

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

b. At least once per 31 days and within 6 hours after each solution volume increase of $\geq 1\%$ of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is greater than 1000 psig by verifying that the power lockout switch is in lockout.

d.~~*~~ At least once per 18 months by verifying that each accumulator isolation valve opens automatically upon receipt of a safety injection test signal.

~~*A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

   1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

   2. At least once daily (24 hour consecutive period) the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

   1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:

   1.~~*~~ Verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal.

   2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

      a) Centrifugal charging pump

      b) Safety injection pump

      c) Residual heat removal pump

~~* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

2.* Verifying that on a safety injection test signal or control room intake high radiation test signal, the system automatically actuates in the pressurization mode by opening the outside air supply and diverting air flow through the HEPA filter and charcoal adsorber bank.

3. Deleted

4. Verifying that on a manual actuation signal, the system will actuate to the required pressurization or recirculation operating mode.

5. Verify each CREACS train has the capability to remove the assumed heat load.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal absorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place while operating the filter system at a flow rate of 8000 cfm ± 10%.

4.7.6.2 Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program (Refer to T.S. 6.18).

~~* A one time extension to this surveillance requirement which is satisfied by performance of the Manual SI test is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

LIMITING CONDITION FOR OPERATION

ACTION: MODES 5 and 6 or during movement of irradiated fuel assemblies. *

a. With one chiller inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Restore the chiller to OPERABLE status within 14 days or;

3. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

b. With two chillers inoperable:

1. Remove the appropriate non-essential heat loads from the chilled water system within 4 hours and;

2. Align the control room emergency air conditioning system (CREACs) for single filtration operation using the Salem Unit 2 train within 4 hours and;

3. Restore at least one chiller to OPERABLE status within 72 hours or;

4. Suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

c. With one chilled water pump inoperable, restore the chilled water pump to OPERABLE status within 7 days or suspend CORE ALTERATIONS and movement of irradiated fuel assemblies.

SURVEILLANCE REQUIREMENTS

4.7.10 The chilled water loop which services the safety-related loads in the Auxiliary Building shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each manual valve in the chilled water system flow path servicing safety related components that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. ~~**~~At least once per 18 months, by verifying that each automatic valve actuates to its correct position on a Safeguards Initiation signal.

c. At least once per 92 days by verifying that each chiller starts and runs.

* During Modes 5 and 6 and during movement of irradiated fuel assemblies, chilled water components are not considered to be inoperable solely on the basis that the backup emergency power source, diesel generator, is inoperable.

~~**A one time extension to this surveillance requirement for performance of relay time response and sequence testing of the safeguard equipment control (SEC) system, which partially satisfies the surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

3. The connection resistance is:
$\leq$ 150 micro ohms for inter-cell connections,
$\leq$ 350 micro ohms for inter-rack connections,
$\leq$ 350 micro ohms for inter-tier connections,
$\leq$ 70 micro ohms for field cable terminal connections, and
$\leq$2500 micro ohms for the total battery connection resistance which includes all inter-cell connections (including bus bars), all inter-rack connections (including cable resistance), all inter-tier connections (including cable resistance), and all field terminal connections at the battery.

e. At least once per 18 months by verifying that the battery charger will supply at least 170 amperes at 125 volts for at least 4 hours.

f.* At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

g. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Satisfactory completion of this performance discharge test shall also satisfy the requirements of Specification 4.8.2.3.2.f if the performance discharge test is conducted during a shutdown where that test and the battery service test would both be required.

h. At least once per 12 months, during shutdown, if the battery shows signs of degradation OR has reached 85% of the service life with a capacity less than 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its capacity on the previous performance test, or is below 90% of the manufacturer's rating.

i. At least once per 24 months, during shutdown, if the battery has reached 85% of the service life with capacity greater than or equal to 100% of manufacturers rating, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test.

~~* A one time extension to this surveillance requirement is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

LIMITING CONDITION FOR OPERATION

3.8.3.1 All containment penetration conductor overcurrent protective devices required to provide thermal protection of penetrations shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping either the primary or backup protective device, or racking out or removing the primary or backup device within 72 hours, declare the affected system or component inoperable, and verify the primary or backup protective device to be tripped, or the primary or backup device racked out or removed at least once per 7 days thereafter; or

b. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 All required containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. ~~*,**~~For at least one 4.16 KV reactor coolant pump circuit, such that all reactor coolant pump circuits are demonstrated OPERABLE at least once per 72 months, by performance of:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

~~* A one time extension to this surveillance requirement for inspection calibration and meggaring of the 1F 4KV Bus overload relays, which partially satisfies this surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

~~** A one time extension to this surveillance requirement for inspection calibration and meggaring of the 1A, 1B, and 1C 460 transformer relays and CT's, which partially satisfy this surveillance requirement, is granted during fuel cycle thirteen allowing Unit 1 operations to continue to the thirteenth refueling outage (1R13). The surveillance testing is to be completed at the appropriate time during the 1R13 outage, prior to the unit returning to Mode 4 upon outage completion.~~

UNIT 1 FACILITY OPERATING LICENSE PAGES WITH PROPOSED CHANGES

Facility Operating License DPR-70

10. TERMINATION

Pursuant to the provisions of 10 CFR 75.41, the Commission will inform the licensee, in writing, when its installation is no longer subject to Article 39(b) of the principal text of the US/IAEA Safeguards Agreement. The IAEA Safeguards License Conditions incorporating Code 7. of the Facility Attachment as part of NRC License DPR-70 will be terminated as of the date of such notice from the Commission. However, since the IAEA may elect to maintain the licensee's installation under Article 2(a) of the Protocol, provisions equivalent to Codes 1. through 6. of the Facility Attachment (with possible appropriate modifications) may still apply, and accordingly all other IAEA Safeguards License Conditions to NRC License No. DPR-70 will remain in effect until the Commission notifies the licensee otherwise. If this option is not selected by the IAEA, the Commission will then notify the licensee that all License Conditions pertaining to the US/IAEA Safeguards Agreement are terminated.

J. RELOCATED TECHNICAL SPECIFICATIONS

PSEG Nuclear LLC shall relocate certain technical specification requirements to licensee-controlled documents as described below. The location of these requirements shall be retained by the licensee.

a. This license condition approves the relocation of certain technical specification requirements to licensee-controlled documents (UFSAR), as described in the licensee's applications with the staff's safety evaluation approval and Amendment No. as noted below:

Licensee's Applications	Safety Evaluations	Amendment Nos.
September 25, 1996	January 30, 1997	189

Implementation shall include the relocation of technical specifications requirements to the appropriate licensee-controlled document as identified in the licensee's application.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
1. ~~Incomplete Preoperational Tests, Startup Tests, and Other Items Which Must Be Completed~~ DELETED
2. Page Changes to Technical Specifications, Appendix A

Date of Issuance: December 1, 1976

~~ATTACHMENT 1 LICENSE DPR-70~~

~~Incomplete preoperational Tests, Startup Tests, and Other Items Which Must be Completed~~

~~This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to certain specified Operational Modes. Public Service Electric and Gas Company shall not proceed beyond the authorized Operational Modes without prior written authorization from the Commission.~~

~~A. Public Service Electric and Gas Company may at the license issue date proceed directly to Operational Mode 6 (initial fuel loading), and may subsequently proceed to Operational Mode 5 (cold shutdown).~~

~~B. Prior to proceeding to Operational Mode 4 (hot shutdown); Public service Electric and Gas Company shall test the response times of primary sensors in the reactor coolant system per SUP 20.1. Subsequent to the verification by the Office of Inspection and Enforcement of the acceptable completion of this item, and upon written authorization by the Commission, Public service Electric and Gas company may proceed to Operational Mode 4 (hot shutdown).~~

~~C. Prior to proceeding to Operational Mode 3 (hot standby), Public Service Electric and Gas Company shall complete the following items:~~

~~1. Testing operation of RHR pump recirculation valves llRH29 and 12RH29 per SUP. 50.0.~~

~~2. Testing motor winding temperatures of RHR pump motors Nos. 11 and 12 per SUP 12.~~

~~3. Testing the following snubbers per SUP 50.4:~~

~~RHRH 11-29A~~
~~RHRH ll-29B~~
~~RHRH 12-348~~
~~RHRH 12-34C~~

~~4. Testing the boron recycle system per SUP 10.5.~~

~~5. Demonstrate beta dosimetry capability.~~

~~6. Testing process radiation monitors, excluding those required for fuel loading, per SUP 21.~~

~~7. Testing service water system per SUP 28.~~

~~(Revised September 10, 1976)~~

8. ~~Testing chilled water portion of the control room air conditioning system per SUP 19.7.~~

9. ~~Prepare the following radiochemistry procedures:~~

    ~~(a) PD 3.3.010 - procedure to determine the average energy of gamma emitting isotopes;~~

    ~~(b) PD 3.3.011 - procedure for detecting fission gases by gamma spectroscopy in the presence of other gases;~~

    ~~(c) PD 3.3.003 - procedure to determine the dose equivalent Iodine 131 in the primary coolant.~~

10. ~~Replace the existing standby charcoal filters in the auxiliary building ventilation system with charcoal filters capable of renovinq 90 percent of the organic iodines.~~

~~Subsequent to verification by the Office of Inspection and Enforcement of the acceptable completion of the above listed items, and upon written authorization from the commission, the Public Service Electric and Gas Company may proceed to Operational Mode 3 (hot standby).~~

~~D. Prior to proceeding to Operational Mode 2 (initial criticality), Public Service Electric and Gas company shall complete the following items:~~

1. ~~Testing high temperature alarm TE463A on pressurizer relief line per SUP 50.6.~~

2. ~~Testing control of steam generator blowdown flow by valves GB8 and GBl0 per SUP 50.13.~~

3. ~~Testing upper motor bearing of reactor coolant pump No. 14 per SUP 50.0.~~

4. ~~Testing pump seal of reactor coolant pump No. 11 per SUP 50.0.~~

5. ~~Testing RID's Nos. 423B, 431A, 433B and 440B in the reactor coolant system per SUP 50.7.~~

6. ~~Testing the following snubbers per SUP 50.4:~~

~~1-PRA-146~~	~~1-PRSN-7~~	~~1-PRSN-28~~	~~1-PRSN-400~~
~~1-PRA-150~~	~~1-PRSN-9~~	~~1-PRSN-29~~	~~1-PRSN-401~~
~~1-PRA-154~~	~~1-PRSN-10~~	~~1-PRSN-30~~	~~1-PRSN-402~~
~~1-PRA-158~~	~~1-PRSN-11~~	~~1-PRSN-32A~~	~~1-PRSN-405~~
~~1-PRA-162~~	~~1-PRSN-12~~	~~1-PRSN-32B~~	~~1-PRSN-405A~~
	~~1-PRSN-13~~	~~1-PRSN-33~~	~~1-PRSN-406~~
~~1-PRSN-1~~	~~1-PRSN-16~~	~~1-PRSN-34~~	~~1-PRSN-406A~~
~~1-PRSN-2~~	~~1-PRSN-17~~	~~1-PRSN-36~~	
~~1-PRSN-3~~	~~1-PRSN-19~~	~~1-PRSN-37~~	
~~1-PRSN-3A~~	~~1-PRSN-20~~	~~1-PRSN-38A~~	
~~1-PRSN-4~~	~~1-PRSN-23~~	~~1-PRSN-38B~~	
~~1-PRSN-5~~	~~1-PRSN-25~~	~~1-PRSN-39~~	
~~1-PRSN-5A~~	~~1-PRSN-27~~	~~1-PRSN-42~~	

~~Subsequent to verification by the Office of Inspection and Enforcement of the acceptable completion of the above items, and upon written authorization from the Commission, public Service Electric and Gas Company may proceed to Operational Mode 2 (initial criticality).~~

~~E. Prior to proceeding to Operation Mode 1 (power operation), the following items shall be completed:~~

~~1. Reactor Vessel Overpressure Alarm - A reactor vessel overpressure alarm shall be installed in the control room. This alarm shall be operable whenever the system is in cold shutdown or hot shutdown, shall be actuated whenever the system pressure exceeds the technical specification limits, and shall not compromize safety related equipment.~~

~~2. Maintenance procedures - The maintenance procedures delineated in Inspection and Enforcement Report 50-272/76-38 shall be completed.~~

~~Subsequent to verification by the Office of Inspection and Enforcement of the acceptable completion of the *above* items, and upon written authorization by the Commission, Public service Electric and Gas Company may proceed in its power ascension program to Operational Mode 1, with the power level limited to twenty percent of rated core power.~~

~~F. Prior to exceeding the forty percent power limit, the snubber tests delineated in Item F above shall be repeated at a power level between thirty and forty percent of rated core power. Upon written acceptance by the Commission of the above items, Public Service Electric and Gas Company may proceed in its power ascension program to a power level not exceeding ninety percent of rated core power.~~

~~G. Prior to exceeding the ninety percent power limit, the snubber tests delineated in Item F above shall be repeated at a power level between eighty and ninety percent of rated. Upon written acceptance by the Commission of these tests, Public Service Electric and Gas Company may proceed in its power ascension program to full power.~~

~~Upon attaining full power, or as soon as possible thereafter, Public Service Electric and Gas Company shall perform a final verification test of these snubbers. The Office of Inspection and Enforcement will review the results of these verification tests, and absent any notification to the contrary, Public Service Electric and Gas Company may sustain full power operation.~~

~~(Revised December 1, 1976)~~

UNIT 2 TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

Facility Operating License DPR-75

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from 228 steps withdrawn shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

Note: A proposed change to 3.1.3.3 is pending via PSEG LAR S09-01, submitted March 22, 2009.

APPLICABILITY: MODES 1 & 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

~~b. With the rod drop times within limits but determined with 3 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 76% of RATED THERMAL POWER.~~

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE shall be maintained within the target band about the target flux difference as specified in the CORE OPERATING LIMITS REPORT (COLR ~~CORL~~).

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference as specified in the COLR and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the target band as specified in the COLR for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits as specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

   b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits as specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

2. When the $F_{xy}^{C}{}_{\cancel{xy}}$ is less than or equal to the $F_{xy}^{RTP}{}_{\cancel{xy}}$ limit for the appropriate measured core plane, additional core power distribution measurements shall be taken and $F_{xy}^{C}{}_{\cancel{xy}}$ compared to $F_{xy}^{RTP}{}_{\cancel{xy}}$ and $F_{xy}^{L}{}_{\cancel{xy}}$ at least once per 31 EFPD.

e. The F_{xy} limit for Rated Thermal Power ($F_{xy}^{RTP}{}_{\cancel{xy}}$) shall be provided for all core planes containing bank "D" control rods and all unrodded core planes in the COLR per specification 6.9.1.9.

f. The F_{xy} limits of e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0% to 15%, inclusive.
2. Upper core region from 85% to 100%, inclusive.
3. Grid plane regions at 17.8% ± 2%, 32.1% ± 2%, 46.4% ± 2%, 60.6% ± 2% and 74.9% ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. Evaluating the effects of F_{xy} on $F_Q(Z)$ to determine if $F_Q(Z)$ is within its limit whenever $F_{xy}^{C}{}_{\cancel{xy}}$ exceeds $F_{xy}^{L}{}_{\cancel{xy}}$.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a core power distribution measurement and increased by the applicable manufacturing and measurement uncertainties~~*~~ as specified in the COLR.

~~* For Cycle 11, when the number of available movable detector thimbles is greater than or equal to 50% and less than 75% of the total, the 5% measurement uncertainty shall be increased to [5% + (3-T/14.5)(1%)] where T is the number of available thimbles.~~

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2. PROCESS MONITORS					
b. Noble Gas Effluent Monitors					
1) Medium Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 3.0 \times 10^{-2}$ μCi/cm^3 (Alarm only)	10^{-3}-10^{1} μCi/cm^3	26
2) High Range Auxiliary Building Exhaust System (Plant Vent)	1	1,2,3&4	$\leq 1.0 \times 10^{2}$ μCi/cm^3 (Alarm only)	10^{-1}-10^{5} μCi/cm^3	26
3) Condenser Exhaust System	1	1,2,3&4	$\leq 7.12 \times 10^{4}$ cpm (Alarm only)	1-10^{6} cpm	26
3. CONTROL ROOM					
a. Air Intake - Radiation Level	2/Intake##	**	$\leq 2.48 \times 10^{3}$ cpm	10^{1}-10^{7} cpm	27,28

Control Room air intakes shared between Unit 1 and 2.

** ALL MODES and during movement of irradiated fuel assemblies and during CORE ALTERATIONS.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILTY.- MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, or 3.3.3.14.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system~~, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2,~~ to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.14.1 The operability criteria listed in 3.3.3.14.a, 3.3.3.14.b, and 3.3.3.14.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.14.2 Calibration of the PDMS is required:

a. At least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.14.b.1 and 3.3.3.14.b.2 are satisfied, or

b. At least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.14.b.3 is satisfied.

3.4.11 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.11.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.11.1.

APPLICABILITY: ALL MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.11.1 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

~~4.4.11.2 Augmented Inservice Inspection Program for Steam Generator Channel Heads -- The No. 21 Steam Generator channel head shall be ultrasonically inspected in a selected area during each of the first three refueling outages using the same ultrasonic inspection procedures and equipment used to generate the baseline data. These inservice ultrasonic inspections shall verify that the cracks observed in the stainless steel cladding prior to operation have not propagated into the base material.~~

6.9.1.5 Reports required on an annual basis shall include:

a. DELETED

b. DELETED

c. The results of any specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.~~8~~9. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit; results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 DELETED

6.9.1.9 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Moderator Temperature Coefficient Beginning of Life (BOL) and End of Life (EOL) limits and 300 ppm surveillance limit for Specification 3/4.1.1.3 ~~4~~,

2. Control Bank Insertion Limits for Specification 3/4.1.3.5,

3. Axial Flux Difference Limits and target band for Specification 3/4.2.1,

4. Heat Flux Hot Channel Factor, F_Q, its variation with core height, K(z), and Power Factor Multiplier PF_{xy}, Specification 3/4.2.2, and

5. Nuclear Enthalpy Hot Channel Factor, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

6. Refueling boron concentration per Specification 3.9.1

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, (W Proprietary), Methodology for Specifications listed in 6.9.1.9.a.

UNIT 2 FACILITY OPERATING LICENSE PAGES WITH PROPOSED CHANGES

Facility Operating License DPR-75

(2) Technical Specifications and Environmental Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 275, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

~~(3) Special Low Power Test Program~~

~~PSE&G shall complete the training portion of the Special Low Power Test Program in accordance with PSE&G's letter dated September 5, 1980 and in accordance with the Commission's Safety Evaluation Report "Special Low Power Test Program", dated August 22, 1980 (See Amendment No. 2 to DPR-75 for the Salem Nuclear Generating Station, Unit No. 2) prior to operating the facility at a power level above five percent.~~

~~Within 31 days following completion of the power ascension testing program outlined in Chapter 13 of the Final Safety Analysis Report, PSE&G shall perform a boron mixing and cooldown test using decay heat and Natural Circulation. PSE&G shall submit the test procedure to the NRC for review and approval prior to performance of the test. The results of this test shall be submitted to the NRC prior to starting up following the first refueling outage.~~

~~(4) Initial Test Program~~

~~PSE&G shall conduct the post-fuel-loading initial test program (set forth in Chapter 13 of the Final Safety Analysis Report, as amended) without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:~~

~~(a) Elimination of any test identified in Chapter 13 of the Final Safety Analysis Report, as amended, as essential;~~

~~(b) Modification of test objectives, methods or acceptance criteria for any test identified in Chapter 13 of the Final Safety Analysis Report, as amended, as essential;~~

~~(c) Performance of any test at a power level different by more than five percent of rated power from there described; and~~

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ITEMS 3 THROUGH 9 DELETED

~~(d) Failure to complete all tests included in the described program (planned or scheduled for power levels up to the authorized power level) prior to exceeding a core burnup of 120 effective full power days.~~

~~(5) Instrument Trip Setpoints~~

~~PSE&G shall submit for NRC review within six months of the date of issuance of this operating license the following values for each Reactor Protection System and Engineered Safety Features instrumentation channel:~~

~~(a) the Technical Specification allowable value (the Technical Specification trip setpoint plus the instrument drift assumed in the accident analysis);~~

~~(b) the instrument drift assumed to occur during the interval between Technical Specification surveillance tests;~~

~~(c) the components of the cumulative instrument bias; and~~

~~(d) the maximum margin between the Technical Specification trip setpoint and the new trip value assumed in the accident analysis.~~

~~(6) SMII-6 Open Items List~~

~~Prior to exceeding five percent rated thermal power, PSE&G will resolve to the satisfaction of the NRC's Office of Inspection and Enforcement all remaining construction and testing deficiencies on the SMII-6 Open Items List designated for completion prior to the commencement of power range testing. All listed items deferred beyond the commencement of power range testing will be subject to review by NRC Region I inspectors.~~

~~(7) Compliance With Regulatory Guide 1.97~~

~~By June 1, 1983, PSE&G shall implement to the satisfaction of the NRC the provisions of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," as modified by PSE&G's commitments to NUREG-0588 and NUREG-0737.~~

~~(8) Snubbers~~

~~(a) Within 4 months after issuance of the license, PSE&G shall provide a Technical Specification listing of mechanical snubbers. In the interim, PSE&G will conduct a comprehensive mechanical snubber inspection program implemented by plant instructions.~~

~~(b) The functional testing of hydraulic and mechanical snubbers in accordance with Technical Specification 3.7.9 shall commence with the first refueling outage. The initial functional testing shall be completed prior to resuming power operation following the first refueling outage.~~

~~(9) Environmental Qualification (Section 3.11, Supplement 5)*~~

~~PSE&G shall take the following remedial actions, or alternative actions acceptable to the NRC, with regard to the environmental qualification requirements for Class IE equipment:~~

~~(a) No later than June 30, 1982, the wide-range resistance temperature detectors for the reactor coolant system shall be qualified for radiation exposure for the 40-year plant life and appropriate exposure condition due to design basis accidents. Pending completion of such qualification and acceptance by the NRC, PSE&G shall replace each of these detectors at each refueling outage.~~

~~(b) Prior to completion of the first refueling outage or June 30, 1982, whichever is earliest, PSE&G shall replace the Scotchcast No. 9 resin seals, used at the electrical connection interface on the NAMCO limit switches, with Conax Electric Conduction Seal Assemblies.~~

~~(c) By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588, "Interim Staff Position in Environmental Qualification of Safety-Related Electrical Equipment," December 1979.~~

~~* References are to the appropriate sections of the Safety Evaluation Report (NUREG-0517) and its supplements.~~

~~(d) Complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.~~

~~(e) Within 90 days of receipt of the equipment qualification safety evaluation, the licensee shall either (i) provide missing documentation identified in Sections 3 and 4 of the equipment qualification safety evaluation which will demonstrate compliance of the applicable equipment with NUREG-0588, or (ii) commit to corrective actions which will result in documentation of compliance of applicable equipment with NUREG-0588 not later than June 30, 1982.~~

(10) Fire Protection

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report, and as approved in the NRC Safety Evaluation Report, dated November 20, 1979, and in its supplements, and in the NRC Safety Evaluation dated January 7, 2004 subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

PAGES 8 THROUGH ~~8, 9, AND 10~~ 20 ARE INTENTIONALLY BLANK
ITEMS 11 THROUGH 25 DELETED

~~(11) Containment Isolation (Section 6.2.3, Supplements 4 and 5)~~

~~Within 90 days after issuance of the license, PSE&G shall demonstrated to the satisfaction of the NRC that the present containment isolation provisions for the main feedwater lines comply with the requirements of General Design Criterion 57 under all postulated accident conditions, or propose a design change that will achieve compliance. If necessary, the design change shall be implemented during the first refueling outage.~~

~~(12) Main Condenser (Section 14.0, Supplement 4)~~

~~Prior to exceeding 50 percent power, PSE&G shall complete the preoperational testing of the remaining three of six circulators to be tested in the main condenser for the circulating water system.~~

~~(13) River Traffic Accidents (Section 2.2.1, Supplement 1)~~

~~PSE&G shall also report for the Salem facility any information reported for the Hope Creek facility relating to circumstances which suggest that the risk from flammable gas clouds (resulting from river traffic accidents on the Delaware River) varies significantly from that previously considered.~~

~~(14) Waterhammer Test (Appendix C, A-1, Supplement 4 and Section 22.2, II.E.1.1. Supplement 5)~~

~~Prior to exceeding 90 percent power, PSE&G shall perform a test program to show that unacceptable waterhammer damage will not result from anticipated feedwater transients to the steam generator. Prior to performing the test program, PSE&G shall obtain NRC approval of the test procedures.~~

~~(15) Prior to resuming power operation following the first refueling outage:~~

~~(a) Control Rod Guide Thimble (Section 4.2.2, Supplement 4)~~

~~PSE&G shall submit the details of the inspection program for control rod guide thimble tube wall wear for NRC approval.~~

~~(c) Pressure Isolation Valves (Section 5.3.2, Supplement 5)~~

~~PSE&G shall install leak test connections on the pressure isolation valves; until installation of the leak test connections, PSE&G may substitute multiple valve leak tests for Technical Specification 3.4.7.2.f, such that the cumulative leakage from two valves in parallel lines shall not exceed two gallons per minute, and the cumulative leakage from three valves in parallel lines shall not exceed three gallons per minute.~~

~~(d) Diesel Generator Reliability (Section 8.3.4, Supplement 5)~~

~~PSE&G shall implement the following design and procedural modifications with respect to diesel generator reliability:~~

~~(i) Complete a formal training program for all the mechanical and electrical maintenance and quality control personnel, including supervisors, who are responsible for the maintenance and availability of the diesel generators. The depth and quality of this training program shall be at least equivalent to that of training programs normally conducted by major diesel engine manufacturers.~~

~~(ii) Develop operating procedures that require loading the diesel engine to a minimum of 25 percent of full load for one hour after eight hours of continuous no load operation or as recommended by the engine manufacturer.~~

~~(e) Containment Sump Model Test (Appendix C, A-43, Supplement 4)~~

~~PSE&G shall submit the confirmatory results of the containment sump model test program, along with a description of any sump modifications resulting from the tests.~~

~~(f) Under-Voltage Protection (Section 8.4.1, Supplement 4)~~

~~PSE&G shall install a second level of undervoltage protection for the emergency buses.~~

~~(g) Reactor Containment Electrical Penetrations (Section 8.4.3., Supplement 4)~~

~~PSE&G shall add a fuse in series with the primary device of each one of 12 circuits fed from 230 volt ac motor control centers to provide backup protection for reactor containment electrical penetrations. Each fuse shall be located in a independent compartment in the control center of the present primary device.~~

~~(16) Loss of Non-Class IE Instrumentation and Control Power Bus During Operation (Section 7.9, Supplement 5)~~

~~PSE&G shall implement the design modifications identified in the PSE&G letter dated July 31, 1980 prior to resuming power operation following the first refueling outage.~~

~~(17) Turbine Inspection (Section 3.5.1, Supplement 5)~~

~~Prior to resuming power operation following the second refueling outage, PSE&G shall subject the low pressure turbines to an inservice inspection. The inspection shall consist of visual and volumetric examinations. The visual examination shall be applied to 100 percent of all the accessible surface of the rotors, discs and blading. The volumetric examination shall use an ultrasonic technique to fully examine the bore and keyway region of the discs in each low pressure turbine.~~

~~The inspection results and evaluation of this inservice inspection shall be reported to the NRC and shall be accepted by the NRC prior to startup following the second refueling outage.~~

~~(18) Vibration Dynamics Effects Test (Section 3.9.1, SER)~~

~~PSE&G shall conduct a preoperational vibration dynamic effects test program for all ASME 1, 2 and 3 piping systems and piping restraints during startup test programs and initial operation.~~

~~(19) Differential Pressure Baseline Data (Part II, Section I.G, Supplement 4)~~

~~PSE&G shall obtain baseline data regarding differential pressure across the elbow pressure taps in each reactor coolant loop for various pump combinations during startup and initial operation.~~

~~(20) Engineered Safety Feature Reset Controls (Section 7.10, Supplement 5)~~

~~In conformance with IE Bulletin 80-06, PSE&G shall correct the reset actions for the two sets of valves identified in the PSE&G letter dated June 13, 1980, as corrected by the PSE&G letter dated July 18, 1980, prior to operating the facility at a power level above five percent. PSE&G shall also perform the additional testing required by IE Bulletin 80-06 prior to operation above five percent power.~~

~~(21) Sump Performance (Section 6.3.3, Supplement 5)~~

~~(a) Prior to resuming power operation following the first refueling outage, PSE&G shall provide a detailed survey of insulation materials.~~

~~(b) Prior to operation above five percent power, control room operators shall be trained in the recognition and mitigation of LPI performance degradation.~~

~~(22) Radiation Protection Organization (Section 12.0, Supplement 5)~~

~~PSE&G shall complete the reorganization actions and programs associated with radiation protection no later than November 1, 1981.~~

~~(23) Category I Masonry Walls (Section 3.8.3, Supplement 5)~~

~~(a) Prior to operation above five percent power, PSE&G shall submit the information requested in the NRC letter dated January 8, 1981.~~

~~(b) Prior to startup following the first refueling, PSE&G shall resolve the difference between the staff criteria and the criteria used by PSE&G to the satisfaction of the NRC and implement the required fixes that might result from such as resolution.~~

~~(24) TMI Action Plan Conditions (Section 22.2, Supplement 5)~~

~~Unless otherwise noted, each of the following conditions references the appropriate section of Supplement No. 5 to the Safety Evaluation Report (NUREG-0517) for the Salem Nuclear Generation Station, Unit 2, dated January 1981 and shall be completed to the satisfaction of the NRC by the times indicated.~~

~~(a) DELETED~~

~~(b) Short-Term Accident Analysis and Procedure Revision (Section 22.2, I.C.1. and I.C.8)~~

~~The operators shall be briefed on the revisions to the emergency operation instruction within 30 effective full power days of operation.~~

~~(c) Auxiliary Feedwater System Reliability Evaluation (Section 22.2, II.E.1.1)~~

~~(i) PSE&G shall install auxiliary feedwater storage tank level indications and alarms in accordance with the PSE&G letter of May 5, 1980 prior to startup after the first refueling.~~

~~(ii) PSE&G shall perform a 48-hour endurance test on all auxiliary feedwater system pumps prior to operation at 100 percent power. PSE&G shall provide a report on the results of these tests to NRC within 60 days of completion of the tests.~~

~~(iii) PSE&G shall resolve to NRC's satisfaction the issue concerning time available for operator action to prevent pump damage prior to operation above five percent power.~~

~~(d) Upgrade Emergency Preparedness (Section 22.2, III.A.1.1 and Section 22.3, III.A.2)~~

~~(i) No later than 90 days from the date of issuance of this license, PSE&G shall report to the NRC the status of any items related to emergency preparedness identified by FEMA or the NRC as requuiring further action.~~

~~(ii) PSE&G shall provide meteorological and dose assessment remote interrogation capability to meet the criteria of Appendix 2, NUREG-0654, Revision 1 as follows: (a) a functional description of upgraded capabilities by January 1, 1982, (b) installation of hardware and software by July 1, 1982 provided that NRC approval is received by four months prior to that time and (c) full operation capability by October 1, 1982.~~

~~(iii) PSE&G shall provide substantiation that the back-up source of meteorological information from the NWS Office, Greater Wilmington Airport adequately characterizes the site conditions with respect to wind direction and wind speed by July 1, 1981.~~

~~(iv) PSE&G shall provide substantiation that uncertainties associated with plume trajectory prediction, associated with the occurrence of sea-land breeze circulations within the plume exposure pathway zone, are compatible with the planned recommendations for protective actions that would be based upon such projections by July 1, 1981.~~

~~(e) Primary Coolant Sources Outside Containment (Section 22.2, III.D.1.1)~~

~~(i) For those systems in which leakage is measured during shutdown, PSE&G shall make and report leak rate measurements prior to initial startup.~~

~~(ii) For those systems in which leakage is measured during operations, PSE&G will make and report leak rate measurements within 60 effective full-power days of plant operation.~~

~~(25) TMI Action Plan Dated Conditions (Section 22.3, Supplement 5)~~

~~Each of the following conditions references the appropriate section of Supplement No. 5 to the Safety Evaluation Report (NUREG-0517) for the Salem Nuclear Generating Station, dated January 1981, and shall be completed to the satisfaction of the NRC by the times indicated.~~

~~(a) Short-Term Accident Analysis and Procedure Revision (Section 22.3, I.C.I)~~

~~PSE&G shall implement the requirement of Item I.C.1 specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," no later than the implementation dates established in NUREG-0737.~~

~~(b) Reactor Coolant System Vents (Section 22.3, II.B.1)~~

~~PSE&G shall submit procedural guidelines for and a description of the reactor coolant system vents by July 1, 1981. The reactor coolant system vents shall be installed no later than July 1, 1982.~~

~~(c) Plant Shielding (Section 22.3, II.B.2)~~

~~PSE&G shall complete modifications to assure adequate access to vital areas and protection of safety equipment following an accident resulting in a degraded core not later than January 1, 1982.~~

~~(d) Deleted~~

~~(e) Relief, Safety and Block Valve Test Requirements (Section 22.3, II.D.1)~~

~~PSE&G shall qualify the reactor coolant system relief, safety and block valves under expected operating conditions for design basis transients and accidents in accordance with the plant-specific requirements and schedules established in NUREG-0737, "Clarification of TMI Action Plan Requirements."~~

~~(f) Auxiliary Feedwater Initiation and Indication (Section 22.3, II.E.1.2)~~

~~PSE&G shall upgrade, as necessary, automatic initiation of the auxiliary feedwater system and indication of auxiliary feedwater flow to each steam generator to safety grade quality no later than July 1, 1981.~~

~~(g) Containment Isolation Dependability (Section 22.3, II.E.4.2)~~

~~(i) PSE&G shall reduce the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum compatible with normal operating conditions no later than July 1, 1981.~~

~~(ii) PSE&G shall install a high radiation isolation signal on the containment purge and vent isolation valves no later than July 1, 1981.~~

~~(h) Additional Accident Monitoring Instrumentation (Section 22.3, II.F.1)~~

~~PSE&G shall install and demonstrate the operability of instruments for continuous indication in the control room of the following variables. Each item shall be completed by the specified date in the condition.~~

~~(i) Containment pressure form minus five psig to three times the design pressure of the containment no later than January 1, 1982;~~

~~(ii) Containment water level from (i) the bottom to the top of the containment sump, and (ii) the bottom of the containment to an elevation equivalent to a 600,000 gallon capacity no later than July 1, 1981;~~

~~(iii) Containment atmosphere hydrogen concentration from 0 to 10 volume percent no later than July 1, 1982;~~

~~2.C(25)(h)(iv)~~

~~Containment gamma radiation up to 10^7 rad/hr. at the first outage of sufficient duration but no later than prior to startup following the first refueling outage; and~~

~~(v) Noble gas effluent from each potential release point from normal concentrations up to 10^5 uCi/cc (Xe-133) no later than prior to startup following the first refueling outage.~~

~~PSE&G shall provide the capability to continuously sample gaseous effluents and analyze these samples no later than prior to startup following the first refueling outage.~~

~~Until the above installation is completed, PSE&G shall use interim monitoring procedures and equipment.~~

~~PSE&G shall provide the capability to continuously sample gaseous effluents and analyze these samples no later than January 1, 1982.~~

~~Until the above installation is completed, PSE&G shall use interim monitoring procedures and equipment.~~

~~(i) Inadequate Core Cooling Instruments (Section 22.3, II.F.2)~~

~~PSE&G shall install and demonstrate the operabiliqty of additional instruments or controls needed to supplement installed equipment in order to provide unambiguous, easy-to-interpret indication of inadequate core cooling at the first outage of sufficient duration but no later than prior to startup following the first refueling outage.~~

~~(j) Thermal Mechanical Report (Section 22.3, II.K.2.13)~~

~~PSE&G shall submit a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater no later than January 1, 1982.~~

~~(k) Analysis of Voiding Potential (Section 22.3, II.K.2.17)~~

~~PSE&G shall analyze the potential for voiding in the reactor coolant system (RCS) during anticipated transients. PSE&G shall submit this analysis no later than January 1, 1982.~~

~~(l) Sequential Auxiliary Feedwater Flow Analysis (Section 22.3, II.K.2.19)~~

~~PSE&G shall provide a benchmark analysis of sequential auxiliary feedwater (AFW) flow to the steam generators following a loss of main feedwater no later than January 1, 1982.~~

~~(m) Effect of Loss of Alternating Current Power on Pump Seals (Section 22.3, II.K.3.25)~~

~~PSE&G shall determine, by analysis or experiment, the consequences of a loss of cooling water to the reactor coolant pump seals. PSE&G shall submit the results of the evaluation and proposed modifications no later than January 1, 1982.~~

~~(n) Revised Small Break Loss-of-Coolant-Accident Methods (Section 22.3, II.K.3.30)~~

~~PSE&G shall comply with the requirements of this position as specified in NUREG-0737, "Clarification of TMI Action Plan Requirements."~~

~~(o) Compliance With 10 CFR Part 50.46 (Section 22.3, II.K.3.31)~~

~~PSE&G shall perform plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) to show compliance with 10 CFR Part 50.46. PSE&G shall submit these calculations by January 1, 1983, or one year after NRC approval of LOCA analysis models, whichever is later, only if model changes have been made.~~

~~(p) Emergency Support Facilities (Section 22.3, III.A.1.2)~~

~~PSE&G shall maintain in effect an interim Technical Support Center and an interim Emergency Operations Facility until such time as the final facilities are complete.~~

(26) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 227 are hereby incorporated into this license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(27) PSE&G TO PSEG Nuclear LLC License Transfer Condtions

a. PSEG Nuclear LLC shall take all necessary steps to ensure that the decommissioning trust is maintained in accordance with the application, the requirements of the Order Approving Transfer of License and Conforming Amendment, dated February 16, 2000, and the related Safety Evaluation dated February 16, 2000.

b. The decommissioning trust agreement shall provide that:

1) The use of assets in both the qualified and non-qualified funds shall be limited to expenses related to decommissioning of the unit as defined by the NRC in its regulations and issuances, and as provided in the unit's license and any amendments thereto. However, upon completion of decommissioning, as defined above, the assets may be used for any purpose authorized by law.