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

AUG 1 7 2001

LRN-01-0248
LCR S01-05

PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

**REQUEST FOR CHANGE TO
TECHNICAL SPECIFICATIONS 6.8.4 e
POST ACCIDENT SAMPLING
SALEM GENERATING STATION
UNIT NOS. 1 AND 2
DOCKET NOS. 50-272 AND 50-311**

In accordance with the requirements of 10CFR50.90, PSEG Nuclear (PSEG) hereby transmits a request for revision of the Technical Specifications (TS) for Salem Generating Station Unit Nos. 1 and 2 respectively. Pursuant to the requirements of 10CFR50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed TS change contained herein deletes Technical Specification (TS) Administrative Controls 6.8.4.e "Post Accident Sampling," and thereby eliminates the requirements to have and maintain the Post Accident Sampling System (PASS) at Salem Station. The change is consistent with NRC approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-366, 'Elimination of Requirements for a Post Accident Sampling System (PASS)." The availability of this technical specification improvement was announced in the Federal Register on October 31, 2000 as part of the consolidated line item improvement process (CLIIP).

Attachment 1 provides a description of the proposed change and the required confirmation of applicability, plant specific verifications, and commitments. Attachment 2 provides the existing Technical Specifications pages marked up to show the proposed changes.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and it has been determined that this request involves no significant hazards considerations. Copies of the proposed change have been submitted in accordance with the requirements of 10 CFR 50.91.

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PSEG requests that the amendment be approved by end of March 2002, and be made effective upon approval and issuance. However, PSEG will delay full implementation of the amendment until the next scheduled refueling outages for each of the Salem Units. These outages are scheduled for the Spring of 2002 for Salem Unit 2 (12th refueling outage), and the Fall of 2002 for Salem Unit 1 (15th refueling outage). This implementation period will provide sufficient time for associated administrative activities and fulfillment of commitments.

The proposed change has been reviewed and approved by the Salem Station Operations Review Committee.

Should you have any questions regarding this request, please contact E. Villar at (856) 339-5456.

Sincerely,

D. F. Garchow

D. F. Garchow
Vice President - Operations

Affidavit
Attachments (2)

C Mr. H. Miller, Administrator - Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. R. Fretz, Licensing Project Manager - Salem
U. S. Nuclear Regulatory Commission
One White Flint North
Mail Stop 4D3
11555 Rockville Pike
Rockville, MD 20852

USNRC Resident Inspector Office (X24)

Mr. K. Tosch, Manager IV
Bureau of Nuclear Engineering
P. O. Box 415
Trenton, NJ 08625

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STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

D. F. Garchow, being duly sworn according to law deposes and says:

I am Vice President - Operations of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Salem Generating Station, Units 1 and 2, are true to the best of my knowledge, information and belief.

David F. Garchow

Subscribed and Sworn to before me

this 17th day of Aug., 2001

Jennifer M. Turner
Notary Public of New Jersey

My Commission expires on JENNIFER M. TURNER NOTARY PUBLIC OF NEW JERSEY My Commission Expires July 25, 2005

1.0 INTRODUCTION

Discussion and Assessment

The proposed License amendment would revise administrative Technical Specification 6.8.4 e "Post accident Sampling."

2.0 DESCRIPTION

The proposed TS change contained herein deletes Technical Specification (TS) Administrative Controls 6.8.4 e "Post accident Sampling" for Salem Units 1 and 2, and License Condition 2.C.25 d for Salem Unit 2, thereby eliminating the requirements to have and maintain the Post Accident Sampling System (PASS) at Salem Station.

The changes are consistent with NRC approved Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-366. The availability of this Technical Specification improvement was announced on October 31, 2000 in Federal Register, Vol. 65 No. 211, as part of the consolidated line item improvement process (CLIIP).

3.0 BACKGROUND

Westinghouse Owners Group (WOG) topical report WCAP-14986-A, Rev. 2, "Post Accident Sampling System Requirements: A Technical Basis," evaluated the PASS requirements to determine their contribution to plant safety and accident recovery. The topical report considered the progression and consequences of core damage accidents and assessed the accident progression with respect to plant abnormal and emergency operating procedures, severe accident management guidance, and emergency plans. WCAP-14986-A, Rev. 2, concluded that the current PASS samples specified in NUREG-0737, "Clarification of TMI Action Plan Requirements," may be eliminated.

4.0 TECHNICAL ANALYSIS

4.1 Applicability of Published Safety Evaluation

PSEG Nuclear LLC (PSEG) has reviewed the safety evaluation published as part of the CLIIP, including the supporting information provided to support TSTF-366 (i.e., WCAP-14986-A, Rev.2, "Post Accident Sampling System Requirements: A Technical Basis," submitted October 26, 1998, as supplemented by letters dated April 28, 1999, April 10, 2000, and May 22, 2000). PSEG has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to the Salem Generating Station Units 1 and 2, and justify this amendment for the incorporation of the changes to the Salem Unit 1 and 2 Technical Specifications (TS).

4.2 Optional Changes and Variations

PSEG is not proposing any variations or deviations from the technical specification changes described in TSTF-366 or the NRC model safety evaluation published on October 31, 2000.

The elimination of PASS does not result in any additional changes to the Salem Unit 1 and 2 Technical Specifications, or Technical Specification Bases section. Attachment 2 provides the information and changes to administrative TS 6.4.8 e, and Salem Unit 2 License Condition 2.C.25 d.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

PSEG has reviewed the proposed no significant hazards consideration determination published as part of the CLIIP. PSEG has concluded that the proposed determination presented in the notice is applicable to the Salem Units 1 and 2 and the determination is hereby incorporated, by reference to satisfy the requirements of 10 CFR 50.91 (a).

5.2 Verification and Commitments

As discussed in the notice of availability published in Federal Register, Vol. 65, No. 211, "Notice of Availability for Referencing in License Amendment Applications Model Safety Evaluation on Technical Specification Improvement to Eliminate Requirements on Post Accident Sampling Systems Using the Consolidated Line Item Improvement Process," dated October 31, 2000, PSEG provides the following plant-specific verifications and commitments:

1. PSEG will develop and maintain contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. A description of the contingency plans will be contained in appropriate plant procedures and implemented following the license amendment approval during the next scheduled refueling outages of each Salem Unit. Establishment of contingency plans is considered a regulatory commitment.

2. The capability for classifying fuel damage events at the Alert level threshold has been established at 2 - 5% fuel clad damage. This level of core damage is associated to radioactivity levels of 300 μCi/cc dose equivalent iodine. This capability is described in plant implementing

procedures and will be fully implemented following the license amendment approval during the next scheduled refueling outages of each Salem Unit. The capability for classifying fuel damage events is considered a regulatory commitment.

3. PSEG has established the capability to monitor (sample) radioactive iodines that have been released to offsite environs. This capability is described in plant implementing procedures and will be fully implemented following the license amendment approval during the next scheduled refueling outages of each Salem Unit. The capability to monitor (sample) radioactive iodines is considered a regulatory commitment.

6.0 ENVIRONMENTAL EVALUATION

PSEG has reviewed the environmental evaluation included in the model safety evaluation published on October 31, 2000 as part of the CLIIP. PSEG has determined that the staffs findings presented in that evaluation are applicable to the Salem Units 1 and 2.

7.0 REFERENCES

Standard Technical Specification Change Traveler TSTF-366, "Elimination of Requirements for a Post Accident Sampling System (PASS)."

Westinghouse Owners Group (WOG) topical report WCAP-14986-A,Rev.2, "Post Accident Sampling System Requirements: A Technical Basis,' July 2000."

Federal Register, Vol. 65, No. 21 1, "Notice of Availability for Referencing in License Amendment Applications Model Safety Evaluation on Technical Specification Improvement to Eliminate Requirements on Post Accident Sampling Systems Using the Consolidated Line Item Improvement Process, dated October 31, 2000."

ATTACHMENT 2

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

Technical Specification	Page
6.8.4 e	6-19

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

Facility Operating License	Page
2. C.25 d	18

Technical Specification	Page
6.8.4 e	6-19

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Postaccident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

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6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate

*It is acceptable if the licensee maintains details of the program in plant operation manuals.

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(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) Post-Accident Sampling (Section 22.3, II.B.3)

PSE&G shall complete actions needed to provide the capability to promptly obtain and perform radioisotopic and chemical analysis of reactor coolant and containment atmosphere samples under degraded core conditions without excessive exposure at the first outage of sufficient duration but no later than prior to startup following the first refueling outage.

(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.

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System Subcooling Margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring

e. Postaccident Sampling

A program* which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel

(ii) Procedures for sampling and analysis,

(iii) Provisions for maintenance of sampling and analysis equipment.

6.8.4.f. Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 47.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary containment leakage rate acceptance criterion is less than or equal to 1.0 L_a. During the first unit startup

*It is acceptable if the licensee maintains details of the program in plant operation manuals.