PSEG Nuclear LLC
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PSEG
Nuclear L.L.C.

10 CFR 50.90

LR-N09-0001
January 05, 2009

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

Hope Creek Generating Station
Facility Operating License No. NPF-57
NRC Docket No. 50-354

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

Subject: License Amendment Request to Eliminate Unnecessary Reporting Requirements in the Operating License and the Administrative Controls Section of the Technical Specifications

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG), hereby requests amendments to Facility Operating Licenses (FOLs) listed above for Hope Creek Generating Station (HCGS) and for Salem Generating Station, Units 1 and 2. The proposed amendments would delete the license conditions that require reporting of violations of other requirements (e.g., conditions listed in Section 2.C) in the operating licenses. The proposed amendments would also delete similar reporting requirements in the Administrative Section of Technical Specifications (TS).

The changes are consistent with the notice published in the Federal Register on November 4, 2005 as part of the consolidated line item improvement process (CLIIP).

Attachment 1 provides a description of the proposed change and confirmation of applicability. The marked up Operating License and Technical Specification pages for the proposed changes are provided in Attachment 2.

There are no regulatory commitments in this letter or attachments.

PSEG requests approval of the proposed license amendments by January 05, 2010, with implementation to be completed within 60 days.

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NRR

These proposed changes have been reviewed by the Plant Operations Review Committee, and the Nuclear Safety Review Board in accordance with PSEG procedures. In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated New Jersey official.

If you have any questions or require additional information, please contact Mr. Paul Duke at 856-339-1466.

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

Executed on January 05, 2009
(date)

Sincerely,

Christine T. Neely
Director - Regulatory Affairs
PSEG Nuclear LLC

Attachments (2)
1. Description and Assessment
2. Marked Up Operating License and Technical Specification Pages

cc: S. Collins, Regional Administrator – NRC Region I
R. Ennis, Project Manager - USNRC
NRC Senior Resident Inspector - Hope Creek
NRC Senior Resident Inspector - Salem
P. Mulligan, Manager IV, NJBNE

ATTACHMENT 1

License Amendment Request

Hope Creek Generating Station
NRC Docket No. 50-354

Salem Generating Station - Unit 1 and Unit 2
NRC Docket Nos. 50-272 and 50-311

Description and Assessment

Subject: License Amendment Request to Eliminate Unnecessary Reporting Requirements

Description and Assessment

1.0 INTRODUCTION

The proposed amendment would delete Section 2.F of Facility Operating License No. NPF-57 for Hope Creek Generating Station and Section 2.I of Facility Operating License No. DPR-75 for Salem Generating Station, Unit 2. The facility operating license sections being deleted require reporting of violations of the requirements in Section 2.C of the facility operating licenses. The proposed amendment also deletes Technical Specification (TS) 6.9.3 for Hope Creek, Salem Unit 1 and Salem Unit 2. TS 6.9.3 contains a reporting requirement similar to those being deleted from the Hope Creek and Salem Unit 2 Facility Operating Licenses (FOLs).

The availability of this operating license improvement was announced in the Federal Register on November 4, 2005 as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

Consistent with the CLIIP Notice of Availability (70 FR 67202), the following changes are proposed:

Hope Creek
The proposed amendment consists of deleting Section 2.F of Facility Operating License NPF-57 and TS 6.9.3. The current requirements of the license condition are as follows:

> *Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, PSEG Nuclear LLC shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and(e).*

The existing conditions in Section 2.C that are subject to the current reporting requirement consist of the following:

(1) *Maximum Power Level*

> *PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of 3840 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.*

(2) *Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 174, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into the license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) *Inservice Testing of Pumps and Valves (Section 3.9.6, SSER No.4)*

This License Condition was satisfied as documented in the letter from W. R. Butler (NRC) to C. A. McNeill, Jr. (PSE&G) dated December 7, 1987. Accordingly, this condition has been deleted.

(4) *Inservice Inspection (Section 6.6, SER; Sections 5.2.4.3 and 6.6.3, SSER No. 5)*

a. *PSE&G shall submit an inservice inspection program in accordance with 10 CFR 50.55a(g)(4) for staff review by October 11, 1986.*

b. *Pursuant to 10 CFR 50.55a(a)(3) and for the reasons set forth in Sections 5.2.4.3 and 6.6.3 of SSER No. 5, the relief identified in the PSE&G submittal dated November 18, 1985, as revised by the submittal dated January 20, 1986, requesting relief from certain requirements of 10 CFR 50.55a(g) for the preservice inspection program, is granted.*

(5) *Solid State Logic Modules*

PSEG Nuclear LLC shall continue, for the life of the plant, a reliability program to monitor the performance of the Bailey 862 SSLMs installed at Hope Creek Generating Station. This program should obtain reliability data, failure characteristics, and root cause of failure of both safety-related and non-safety-related Bailey 862 SSLMs. The results of the reliability program shall be maintained on-site and made available to the NRC upon request.

(6) *Fuel Storage and Handling (Section 9.1, SSER No. 5)*

a. *No more than a total of three (3) fuel assemblies shall be out of approved shipping containers, NRC-approved dry spent fuel storage systems, fuel assembly storage racks or the reactor at any one time.*

b. *The above three (3) fuel assemblies as a group shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and the storage rack array.*

c. *Fresh Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three (3) containers high.*

(7) *<u>Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)</u>*

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) 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.

(8) *<u>Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)</u>*

PSEG Nuclear shall obtain NRC approval of the Class B and C solid waste process control program prior to processing Class B and C solid wastes.

(9) *<u>Emergency Planning (Section 13.3, SSER No. 5)</u>*

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(10) *<u>Initial Startup Test Program (Section 14, SSER No. 5)</u>*

Any changes to the Initial Startup Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(11) *Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)*

The facility shall not be operated with reduced feedwater temperature for the purpose of extending the normal fuel cycle unless analyses supporting such operation are submitted by the licensee and approved by the staff.

(12) *Detailed Control Room Design Review (Section 18.1, SSER No. 5)*

a. *PSE&G shall submit for staff review Detailed Control Room Design Review Summary Reports II and III on a schedule consistent with, and with contents as specified in, its letter of January 9, 1986.*

b. *Prior to exceeding five percent power, PSE&G shall provide temporary zone markings on safety-related instruments in the control room.*

(13) *Safety Parameter Display System (Section 18.2, SSER No. 5)*

Prior to the earlier of 90 days after restart from the first refueling outage or July 12, 1988, PSE&G shall add the following parameters to the SPDS and have them operational:

a. *Primary containment radiation*
b. *Primary containment isolation status*
c. *Combustible gas concentration in primary containment*
d. *Source range neutron flux*

(14) *Additional Conditions*

The Additional Conditions contained in Appendix C, as revised through Amendment No. 135, are hereby incorporated into this license. PSEG Nuclear LLC shall operate the facility in accordance with the Additional Conditions.

(15) *PSE&G to PSEG Nuclear LLC License Transfer Conditions*

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 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.*

2) *Investments in the securities or other obligations of PSE&G or affiliates thereof, or their successors or assigns, shall be prohibited. In addition, except for investments tied to market indexes or other non-nuclear sector mutual funds, investments in any entity owning one or more nuclear power plants shall be prohibited.*

3) *No disbursements or payments from the trust shall be made by the trustee until the trustee has first given the NRC 30 days notice of the payment. In addition, no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the Director, Office of Nuclear Reactor Regulation.*

4) *The trust agreement shall not be modified in any material respect without prior written notification to the Director, Office of Nuclear Reactor Regulation.*

5) *The trustee, investment advisor, or anyone else directing the investments made in the trust shall adhere to a "prudent investor" standard, as specified in 18 CFR 35.32(3) of the Federal Energy Regulatory Commission's regulations.*

c. *PSEG Nuclear LLC shall not take any action that would cause PSEG Power LLC or its parent companies to void, cancel, or diminish the commitment to fund an extended plant shutdown as represented in the application for approval of the transfer of this license from PSE&G to PSEG Nuclear LLC.*

(16) <u>*Mitigation Strategy*</u>

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) *Fire fighting response strategy with the following elements:*

1. *Pre-defined coordinated fire response strategy and guidance*
2. *Assessment of mutual aid fire fighting assets*
3. *Designated staging areas for equipment and materials*
4. *Command and control*
5. *Training of response personnel*

(b) *Operations to mitigate fuel damage considering the following:*

1. *Protection and use of personnel assets*
2. *Communications*
3. *Minimizing fire spread*
4. *Procedures for implementing integrated fire response strategy*
5. *Identification of readily-available pre-staged equipment*
6. *Training on integrated fire response strategy*
7. *Spent fuel pool mitigation measures*

(c) *Actions to minimize release to include consideration of:*

1. *Water spray scrubbing*
2. *Dose to onsite responders*

(17) *The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.*

(18) *Upon implementation of Amendment No. 173 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Surveillance Requirement 4.7.2.2.a, in accordance with TS 6.16.c.(i), the assessment of CRE habitability as required by Specification 6.16.c.(ii), and the measurement of CRE pressure as required by Specification 6.16.d, shall be considered met. Following implementation:*

a. *The first performance of Surveillance Requirement 4.7.2.2.a, in accordance with Specification 6.16.c.(i), shall be within the specified frequency of 6 years, plus the 18 month allowance of Surveillance Requirement 4.0.2, as measured from July 29, 2001, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.*

b. *The first performance of the periodic assessment of CRE habitability, Specification 6.16.c(ii), shall be 3 years, plus the 9 month allowance of Surveillance Requirement 4.0.2, as measured from July 29, 2001, the date of the most recent successful tracer gas test, as stated in the December 9, 2003 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.*

c. *The first performance of the periodic measurement of CRE pressure, Specification 6.16.d, shall be within 18 months, plus the 138 days allowed by Surveillance Requirement 4.0.2, as measured from April 5, 2006, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.*

(19) *Leak rate tests required by Surveillance Requirement 4.6.1.2.a and 4.6.1.2.h to be performed in accordance with the Primary Containment Leakage Rate Testing Program are not required to be performed until their next scheduled performance, which is due at the end of the first test interval that begins on the date the test was last performed prior to implementation of Amendment No.174.*

(20) *<u>Top Guide Beams</u>*

Until there is more detailed guidance regarding the inspections of the top guide beams or the issue is resolved by the BWRVIP generically, the *following license condition applies to Hope Creek to preclude the loss of the component's intended function:*

Enhanced visual testing (EVT-1) of the top guide grid beams will be performed in accordance with GE SIL 554 following the sample selection and inspection frequency of BWRVIP-47 for CRD guide tubes. That is, inspections will be performed on 5 percent of the population within six years, and 10 percent of the total population of cells within twelve years. The sample locations selected for examination will be in areas that are exposed to the highest fluence. This inspection plan will be implemented beginning with the first RFO following EPU operation.

(21) Vibration Acceptance Criteria for SRVs

PSEG Nuclear LLC shall provide the Level 1 main steam safety relief valve vibration acceptance criteria to the NRC staff prior to increasing power above 3339 MWt.

(22) Steam Dryer

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer).

1. The following requirements are placed on initial operation of the facility at power levels above 3339 MWt to 3840 MWt for the power ascension:

a. PSEG Nuclear LLC shall monitor hourly the main steam line (MSL) strain gage data during power ascension above 3339 MWt for increasing pressure fluctuations in the steam lines.

b. PSEG Nuclear LLC shall hold the facility at 105 percent and 110 percent of 3339 MWt to collect data from the MSL strain gages required by Condition 1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall submit the evaluation to the NRC staff upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after submitted to the NRC.

c. If any frequency peak from the MSL strain gage data exceeds any of the Level 1 limit curves, PSEG Nuclear LLC *shall return the facility to a lower power level at which the limit curve is not exceeded. PSEG Nuclear shall resolve the uncertainties in the steam dryer analysis, evaluate the continued structural integrity of the steam dryer, and submit that evaluation to the NRC staff.*

d. *In addition to evaluating the MSL strain gage data, PSEG Nuclear LLC shall monitor reactor pressure vessel water level instrumentation and MSL piping accelerometers on an hourly basis during power ascension above 3339 MWt. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data (including consideration of the EPU bump-up factor), PSEG Nuclear LLC shall stop power ascension, evaluate the continued structural integrity of the steam dryer, and submit that evaluation to the NRC staff.*

2. *PSEG Nuclear LLC shall implement the following actions for the initial power ascension at power levels above 3339 MWt to 3840 MWt:*

a. *In the event that acoustic signals are identified that challenge the limit curves during power ascension above 3339 MWt, PSEG Nuclear LLC shall evaluate dryer loads and re-establish the limit curves based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency including application of 65 percent bias error and 10 percent uncertainty to all the SRV acoustic resonances.*

b. *After reaching 111.5 percent of 3339 MWt, PSEG Nuclear LLC shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit curves with the updated ACM load definition, which will be submitted to the NRC staff.*

c. *After reaching 115 percent of 3339 MWt, PSEG Nuclear LLC shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit curves with the updated ACM load definition, which will be submitted to the NRC staff.*

d. *During power ascension above 3339 MWt, if an engineering evaluation is required because a Level 1 acceptance criterion is exceeded, PSEG Nuclear LLC shall perform the structural analysis to address frequency uncertainties up to ±10 percent and assure that peak responses that fall within this uncertainty band are addressed.*

e. *PSEG Nuclear LLC shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139; and to identify the NRC Project Manager for the facility as the point of contact for providing power ascension testing information during power ascension.*

f. *PSEG Nuclear LLC shall submit the final EPU steam dryer load definition for the facility to the NRC staff upon completion of the power ascension test program.*

g. *PSEG Nuclear LLC shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC staff, including methodology for updating the limit curves, prior to initial power ascension above 3339 MWt.*

3. *PSEG Nuclear LLC shall prepare the EPU startup test procedure to include:*

   a. *the stress limit curves to be applied for evaluating steam dryer performance;*

   b. *specific hold points and their duration during EPU power ascension;*

   c. *activities to be accomplished during hold points;*

   d. *plant parameters to be monitored;*

   e. *inspections and walk downs to be conducted for steam, FW, and condensate systems and components during the hold points;*

   f. *methods to be used to trend plant parameters;*

g. *acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections;*

h. *actions to be taken if acceptance criteria are not satisfied; and*

i. *verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above 3339 MWt.*

PSEG Nuclear LLC shall provide the related EPU startup test procedure sections to the NRC staff prior to increasing power above 3339 MWt.

4. *The following key attributes of the program for verifying the continued structural integrity of the steam dryer shall not be made less restrictive without prior NRC approval:*

   a. *During initial power ascension testing above CLTP, each test plateau increment shall be approximately 5 percent of 3339 MWt;*

   b. *Level 1 performance criteria; and*

   c. *The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.*

   Changes to other aspects of the program for verifying the continued structural integrity of the steam dryer may be made in accordance with the guidance of NEI 99-04.

5. *During the first scheduled refueling outage after Cycle 15 and during the first two scheduled refueling outages after reaching full EPU conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 inspection guidelines.*

6. *The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage. The results of the power ascension testing to verify the continued structural integrity of the steam dryer shall be submitted to the NRC staff in a report within 60 days*

> *following the completion of all Cycle 15 power ascension testing. A supplement shall be submitted within 60 days following the completion of all EPU power ascension testing.*

The reporting requirement defined in HCGS Technical Specification (TS) 6.9.3 requires a report to the NRC for violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. Reports are required to be submitted to the NRC via the Licensee Event Report System within 30 days. As discussed in the Federal Register notices associated with the use of the CLIIP to remove the reporting requirement of Section 2.F of the facility operating license, this application also includes the deletion of TS 6.9.3.

<u>Salem</u>
The proposed amendment consists of deleting Section 2.I of Facility Operating License DPR-75 for Salem Unit 2 and TS 6.9.3 for Salem Unit 1 and Unit 2. The Salem Unit 1 Facility Operating License does not contain reporting requirements for violations of the Facility Operating License.

The current requirements of Section 2.I of the Salem Unit 2 FOL are as follows:

> *PSEG Nuclear LLC shall report any violations of the requirements contained in Section 2, Items C. (3) through C. (25), E..F.. and G of this license within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee, no later than the first working day following the violation, with a written-followup report within 14 days.*

The existing conditions in the Salem Unit 2 FOL Section 2.C that are subject to the current reporting requirement consist of the following:

(3) *<u>Special Low Power Test Program</u>*

> *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

(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 June30, 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.*

(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.

(11) *<u>Containment Isolation (Section 6.2.3, Supplements 4 and 5)</u>*

Within 90 days after issuance of the license, PSE&G shall demonstrate 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) *<u>Main Condenser (Section 14.0, Supplement 4)</u>*

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) *<u>River Traffic Accidents (Section 2.2.1, Supplement 1)</u>*

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) *<u>Waterhammer Test (Appendix C, A-1, Supplement 4 and Section 22.2, II.E.1.1. Supplement 5)</u>*

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) *<u>Control Rod Guide Thimble (Section 4.2.2, Supplement 4)</u>*

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

[Salem Unit 2 License Condition 2.C(15)(b) was previously deleted.]

(c) <u>*Pressure Isolation Valves (Section 5.3.2, Supplement 5)*</u>

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) <u>*Diesel Generator Reliability (Section 8.3.4, Supplement 5)*</u>

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) <u>*Containment Sump Model Test (Appendix C, A-43, Supplement4)*</u>

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) <u>*Under-Voltage Protection (Section 8.4.1, Supplement 4)*</u>

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 an 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) <u>*Short-Term Accident Analysis and Procedure Revision (Section 22.2, I.C.1. and I.C.8)*</u>

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

(c) <u>*Auxiliary Feedwater System Reliability Evaluation (Section 22.2, II.E.1.1)*</u>

(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) <u>*Upgrade Emergency Preparedness (Section 22.2, III.A.1.1 and Section 22.3, III.A.2)*</u>

(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 requiring 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 (Section22.3, 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 (Section22.3, 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;*

(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.

(i) *<u>Inadequate Core Cooling Instruments (Section 22.3, II.F.2)</u>*

PSE&G shall install and demonstrate the operability 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) *<u>Thermal Mechanical Report (Section 22.3, II.K.2.13)</u>*

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) <u>*Analysis of Voiding Potential (Section 22.3, II.K.2.17)*</u>

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) <u>*Sequential Auxiliary Feedwater Flow Analysis (Section 22.3, II.K.2.19)*</u>

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) <u>*Effect of Loss of Alternating-Current Power on Pump Seals (Section 22.3, II.K.3.25)*</u>

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) <u>*Revised Small-Break Loss-of-Coolant-Accident Methods (Section 22.3, II.K.3.30)*</u>

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

(o) <u>*Compliance With 10 CFR Part 50.46 (Section 22.3, II.K.3.31)*</u>

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) <u>*Emergency Support Facilities (Section 22.3, III.A.1.2)*</u>

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.

The existing conditions in the Salem Unit 2 FOL Section 2.E, F and G that are subject to the current reporting requirement consist of the following:

E. *The licensees shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54 (p). The plans, submitted by Letter dated May 19, 2006, are entitled: "Salem-Hope Creek Nuclear Generating Station Security Plan," "Salem-Hope Creek Nuclear Generating Station Security Training and Qualification Plan," and "Salem-Hope Creek Nuclear Generating Station Security Contingency Plan." The plans contain Safeguards Information protected under 10 CRF 73.21.*

F. *A temporary exemption from General Design Criterion 57 found in Appendix A to 10 CFR Part 50 is described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report, Supplement No. 5, Section 6.2.3.1. This Exemption is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. The exemption, therefore, is hereby granted and shall remain in effect through the first refueling outage as discussed in Section 6.2.3.1 of Supplement 5 to the Safety Evaluation Report. The granting of the exemption is authorized with the issuance of the Facility Operating License, dated May 20, 1981. The facility will operate, to the extent authorized herein, in conformity with the application as amended, the provisions of the Act, and the regulations of the Commission.*

G. *This license is subject to the following additional condition for the protection of the environment:*

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, PSEG Nuclear LLC shall prepare and record an environmental evaluation of such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated in the Final Environmental Statement or any addendum thereto, PSEG Nuclear LLC shall provide a written evaluation of such activities and obtain prior approval from the Director of Nuclear Reactor Regulation.

The reporting requirement defined in Salem Unit 1 and Unit 2 Technical Specification (TS) 6.9.3 requires a report to the NRC for violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire. Reports are required to be submitted to the NRC via the Licensee Event Report System within 30 days. As discussed in the Federal Register notices associated with the use of the CLIIP to remove the reporting requirement of Section 2.I of the Salem Unit 2 facility operating license, this application also includes the deletion of TS 6.9.3 for Salem Unit 1 and Unit 2.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on November 4, 2005 (70 FR 67202) and the Notice of Opportunity to Comment published on August 29, 2005 (70 FR 51098).

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on November 4, 2005 (70 FR 67202) and the Notice of Opportunity to Comment published on August 29, 2005 (70 FR 51098).

5.0 TECHNICAL ANALYSIS

PSEG has reviewed the safety evaluation (SE) published on August 29, 2005, as part of the CLIIP Notice of Opportunity to Comment. PSEG has concluded that the justifications presented in the SE prepared by the NRC staff are applicable to HCGS and to Salem Unit 1 and Unit 2 and justify this amendment of the facility operating license and TS for HCGS and for Salem Unit 1 and Unit 2.

6.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Opportunity to Comment published on August 29, 2005 (70 FR 51098).

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

PSEG has reviewed the proposed no significant hazards consideration determination published on August 29, 2005 (70 FR 51098), as part of the CLIIP Notice of Opportunity to Comment. PSEG has concluded that the proposed determination presented in the notice is applicable to HCGS and to Salem Unit 1 and Unit 2 and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

PSEG has reviewed the environmental evaluation included in the model SE published on August 29, 2005 (70 FR 51098), as part of the CLIIP Notice of Opportunity to Comment. PSEG has concluded that the NRC staff's findings presented in that evaluation are applicable to HCGS and to Salem Unit 1 and Unit 2 and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. PSEG is not proposing variations or deviations from the changes described in the NRC staff's model SE published on August 29, 2005 (70 FR 51098).

10.0 REFERENCES

1. *Federal Register* Notice of Opportunity to Comment on Model Safety Evaluation on Elimination of Typical License Condition Requiring Reporting of Violations of Section 2.C of Operating License Using the Consolidated Line Item Improvement Process, August 29, 2005 (70 FR 51098)

2. *Federal Register* Notice of Availability of Model Application Concerning Elimination of Typical License Condition Requiring Reporting of Violations of Section 2.C of Operating License Using the Consolidated Line Item Improvement Process, November 4, 2005 (70 FR 67202)

ATTACHMENT 2

Hope Creek Generating Station
NRC Docket No. 50-354

Salem Generating Station - Unit 1 and Unit 2
NRC Docket Nos. 50-272 and 50-311

License Amendment Request to Eliminate Unnecessary Reporting Requirements

Markup of Proposed Facility Operating License and Technical Specification Page Changes

<u>Hope Creek FOL Page</u>

14

<u>Hope Creek TS Page</u>

6-21

DELETED

F. ~~Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, PSEG Nuclear LLC shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written followup within thirty days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).~~

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on April 11, 2026.

FOR THE NUCLEAR REGULATORY COMMISSION
- original signed by H.R. Denton -
Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A - Technical Specifications (NUREG-1202)
2. Appendix B - Environmental Protection Plan
Date of Issuance: July 25, 1986

DELETED

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report

~~6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.~~

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

Salem Unit 1 TS Page

6-24b

h. The following reporting requirements are applicable only for Refueling Outage 18 and the subsequent operating cycle:
The number of indications detected in the upper 17 inches of the tubesheet thickness along with their location, measured size, orientation, and whether the indication initiated on the primary or secondary side.

i. The following reporting requirement is applicable only for Refueling Outage 18 and the subsequent operating cycle:
The operational primary to secondary leakage rate observed in each steam generator during the cycle preceding the inspection and the calculated accident leakage rate for each steam generator from the lowermost 4 inches of tubing (the tubesheet is nominally 21.03 inches thick) for the most limiting accident. If the calculated leak rate is less than 2 times the total observed operational leakage rate, the 180 day report should describe how the calculated leak rate is determined.

SPECIAL REPORTS

DELETED

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

~~6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.~~

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

Salem Unit 2 FOL Page

24

Salem Unit 2 TS Page

6-24b

DELETED

H. If PSEG Nuclear LLC plans to remove or to make significant changes in the normal operation of equipment that controls the amount of radioactivity in effluents from the Salem Nuclear Generation Station, the NRC shall be notified in writing regardless of whether the change affects the amount of radioactivity in effluents.

I. ~~PSEG Nuclear LLC shall report any violations of the requirements contained in Section 2, Items C. (3) through C. (25), E., F., and G of this license within 24 hours by telephone and confirmed by telegram, mailgram, or facsimile transmission to the Director of the Regional Office, or his designee, no later than the first working day following the violation, with a written followup report within 14 days.~~

J. The licensees shall immediately notify the Commission of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

K. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended to cover public liability claims.

L. The licensee is authorized to defer certain eighteen-month surveillance items from the dates required by Technical Specifications 4.0.2(a) and 4.7.10.2(c). These surveillances shall be completed prior to startup following the first refueling outage. The provisions of Technical Specifications 4.0.2(b) and 4.7.10.2(c) are not changed. The affected items are identified in the Safety Evaluation accompanying Amendment No. 14 issued October 22, 1982 and this license change.

M. This license is effective as of the date of the issuance and shall expire at midnight April 18, 2020.

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

f. Total number and percentage of tubes plugged to date,

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

h. Information regarding the application of W* inspection methodology (applicable to tubes within the hot-leg region of the tubesheet); including the number of indications, the location of indications (relative to the BWT and TTS), the orientation (axial, circumferential, volumetric), the severity of each indication (e.g., near through-wall or not through wall), the tube side where the indication initiated (inside or outside diameter), the cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet, the condition monitoring and operational assessment main steam line leak rate (including aggregate calculated main steam line break leak rate from all other sources), and an assessment of whether the results were consistent with expectations regarding the number of flaws and flaw severity (and if not consistent, a description of the proposed corrective action).

SPECIAL REPORTS DELETED

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

~~6.9.3 Violations of the requirements of the fire protection program described in the Updated Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator of the Regional Office of the NRC via the Licensee Event Report System within 30 days.~~

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