PSEG
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

MAR 01 2012

PSEG Nuclear LLC

10 CFR 50.90

LR-N12-0067
LAR H12-01

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

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

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

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) requests an amendment to the facility operating license listed above. In accordance with 10 CFR 50.91(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 Hope Creek Generating Station. The proposed changes are administrative in nature and fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design. The affected TS sections are: the Index, 3.1.3.5, Table 3.3.2-1, Surveillance Requirement (SR) 4.4.3.2.2, 3.6.2.1, SR 4.8.4.1, Table 3.8.4.1-1, SR 4.9.8, 6.9.1.4, 6.9.1.5, and 6.10.1. The affected FOL sections are: Conditions 2.C.(4)a., 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), 2.C.(13), 2.C.(21), and 2.C.(22).

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 License Amendment Request (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.

If you have any questions or require additional information, please do not hesitate to contact Ms. Emily Maguire at (856) 339-1023.

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

Executed on 3/1/12
(Date)

Sincerely,

John F. Perry
Site Vice President – Hope Creek Generating Station

Attachments:
1. Evaluation of Proposed Changes
2. Technical Specification and Facility Operating License Pages with Proposed Changes

cc: W. Dean, Administrator, Region I, USNRC
R. Ennis, Project Manager, USNRC
NRC Senior Resident Inspector, Hope Creek
P. Mulligan, Manager IV, NJBNE
K. Yearwood, Commitment Tracking Coordinator, Hope Creek
L. Marabella, Corporate Commitment Tracking Coordinator

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 10 CFR 50.90, PSEG Nuclear LLC (PSEG) requests an amendment to the facility operating license NPF-57.

The proposed changes correct editorial items in the Technical Specifications (TS) and Facility Operating License (FOL) for Hope Creek Generating Station. The proposed changes are administrative in nature and fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design. The affected TS sections are: the Index, 3.1.3.5, Table 3.3.2-1, Surveillance Requirement (SR) 4.4.3.2.2, 3.6.2.1, SR 4.8.4.1, Table 3.8.4.1-1, SR 4.9.8, 6.9.1.4, 6.9.1.5, and 6.10.1. The affected FOL sections are: Conditions 2.C.(4)a., 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), 2.C.(13), 2.C.(21), and 2.C.(22).

2.0 PROPOSED CHANGES

Item	Description	Section / Page	TS / FOL	Action
1.	The word "BASES" was introduced into the header in error with the issuance of Amendment 134. (Reference 1)	Index xi	Header	Delete the word "BASES" from the header.
2.	Action a. concludes with: "Otherwise, be in at least HOT SHUTDOWN **with** the next 12 hours." The word "with" is an error and should be corrected to "within."	3/4 1-9	3.1.3.5 Action a.	Correct typographical error: change "with" to "within"
3.	Valve Actuation Group 1 is listed as one of the Valve Actuation Groups Operated By Signal for Trip Function 1.a.1), Low Low, Level 2 Reactor Vessel Water Level; Trip Function 1.b., Drywell Pressure - High; Trip Function 1.c., Reactor Building Exhaust Radiation – High; and Trip Function 1.d, Manual Initiation. The Valve Actuation Group 1 valves actuated by the signals listed above were the MSIV Sealing System (MSIVSS) inboard supply valves HV-5834A, HV-5835A, HV-5836A and HV-5837A. These valves were removed after issuance of	3/4 3-11	Table 3.3.2-1 Trip Functions 1.a.1); 1.b.; 1.c.; and 1.d.	Delete Valve Group 1 from Trip Functions 1.a.1); 1.b.; 1.c.; and 1.d.

Item	Description	Section / Page	TS / FOL	Action
	Amendment No. 134 which revised the TS to delete the MSIV Sealing System. Consistent with the markups submitted in LAR H01-02 (ADAMS Accession No. ML011500295), the references to the MSIVSS valves were deleted from the table notation following Table 3.3.2-1, but were not deleted from the table itself. This appears to be an oversight that occurred during the License Amendment submittal. (Reference 1)			
4.	Surveillance requirements were extended for a one-time test interval to the first refueling outage for Pressure Isolation Valves (PIVs) that required an outage to test. PSE&G sent letter number NLR-N87047, dated 4/3/1987, to document this. This historical requirement is no longer applicable and the footnote should be removed.	3/4 4-12	SR 4.4.3.2.2.a footnote	Delete footnote
5.	Action c. states: "With **one** drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F." The word "one" was introduced in error during Amendment 133. It should be changed to "the" as it read originally. (Reference 2)	3/4 6-13	3.6.2.1 Action c.	Correct typographical error: change "one" to "the"
6.	The fourth sentence of surveillance requirement a.2. states: "The instantaneous element shall be tested by injecting a current in excess of **120% the** pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay."	3/4 8-25	SR 4.8.4.1.a.2	Correct typographical error: add the word "of" after 120%

Item	Description	Section / Page	TS / FOL	Action
	The word "of" is missing after 120%. This typographical error has been present since the original Hope Creek TS were issued.			
7.	Circuit Breaker No. 52-253012 powered the Recirc Pump Motor Hoist 1AH201 and Disconnect Switch 1AS204. This hoist was removed in 2007 and replaced with manually actuated model. The breaker was spared and de-termed, so LCO 3.8.4.1 no longer applies. The line associated with Circuit Breaker No. 52-253012* should be removed from Table 3.8.4.1-1, since the LCO no longer applies.	3/4 8-28	Table 3.8.4.1-1	Delete entire line containing Circuit Breaker No. 52-253012* from Table 3.8.4.1-1
8.	Surveillance requirement 4.9.8 read, in part: "The reactor vessel water level shall be determined to be at **least its** minimum required depth..." The word "at" is missing after at least. This typographical error has been present since the original TS were issued.	3/4 9-11	SR 4.9.8	Correct typographical error: add the word "at" after least.
9.	Amendment 161 removed the requirements for the monthly operating report and the annual occupational radiation exposure report but did not remove the requirement for an annual report for main steamline safety/relief valve (SRV) challenges. PSEG's request for amendment (ADAMS Accession No. ML050190207) was based on NRC approved Revision 1 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-369, "Removal of Monthly Operating Report and Occupational Radiation Exposure Report."	Administrative Controls 6-17 & 6-18	Annual Reports 6.9.1.4 & 6.9.1.5	Delete Annual Reports 6.9.1.4 and 6.9.1.5

Item	Description	Section / Page	TS / FOL	Action
	As noted in the NRC model Safety Evaluation for TSTF-369 (ADAMS Accession No. ML041690439), the NRC staff finds it acceptable to remove the requirement to report challenges to safety/relief valves along with the other reporting requirements. Omission of the SRV Reporting Requirements from the previous request for amendment appears to have been an oversight. (Reference 3)			
10.	Section 6.10 of the Administrative Controls is for Record Retention. Between section 6.10.1 and 6.10.2, there is an additional heading that reads, "Special Reports." It appears that the second heading for Special Reports is erroneous and should be deleted. This error has been present since the original TS were issued. The second heading for SPECIAL REPORTS is a typographical error that should be deleted.	Administrative Controls 6-21	Record Retention 6.10.1	Correct typographical error: Delete heading SPECIAL REPORTS under 6.10 RECORD RETENTION
11.	FOL 2.C.(4)a. was a one-time requirement to submit the inservice inspection program. This requirement was completed with the submission of PSE&G letter number NLR-N86235, dated 10/10/1986 (ADAMS Accession No. 8610200067). FOL 2.C.(4)a. is a historical requirement that was completed and should be deleted.	FOL page 4	FOL 2.C.(4)a	Delete FOL 2.C.(4)a.
12.	FOL 2.C.(8) required a one-time NRC approval of the solid waste control program. The approval was granted in a letter from the NRC dated 4/9/1987 (ADAMS Accession No. 8704150091).	FOL page 5	FOL 2.C.(8)	Delete FOL 2.C.(8)

Item	Description	Section / Page	TS / FOL	Action
	FOL 2.C.(8) is a historical requirement that was completed and should be deleted.			
13.	As noted in section 13.3.3, NUREG-1048, Supplement 5, the NRC staff required license condition 2.C.(9) to ensure formal approval of offsite plans in a timely manner. FOL 2.C.(9) required that Hope Creek obtain approval from the Federal Emergency Management Agency (FEMA) for the state of New Jersey and Delaware Radiological Emergency Response Plans in accordance with 44 CFR 350. FEMA approved the plan for the state of New Jersey, as documented in a letter from FEMA Director James L. Witt to Christine T. Whitman, Governor of New Jersey, dated August 3, 1998. FEMA approved the plan for the state of Delaware, as documented in a letter from FEMA Director Julius W. Becton, Jr., to Pierre S. DuPont, IV., Governor of Delaware, dated June 5, 1986. FOL 2.C.(9) is a historical requirement that was completed and should be deleted.	FOL page 5	FOL 2.C.(9)	Delete FOL 2.C.(9)
14.	FOL 2.C.(10) required that changes to the Initial Startup Test Program be reported to the NRC. This requirement was completed through a series of correspondence, including PSE&G letter number NLR-N87045 (ADAMS Accession No. 8703190427), dated 3/16/1987. FOL 2.C.(10) is a historical requirement related to the Initial Startup Test Program that is no longer applicable and should be deleted.	FOL page 5	FOL 2.C.(10)	Delete FOL 2.C.(10)

Item	Description	Section / Page	TS / FOL	Action
15.	FOL 2.C.(12) contains two requirements related to the Detailed Control Room Design Review, which were relevant during initial startup. The first requirement, to submit the Detailed Control Room Design Review Summary Reports II and III, was completed, as documented in PSE&G letter numbers NLR-N86321 (ADAMS Accession No. 8611170204) and NLR-N87061 (ADAMS Accession No. 8705040039), dated 11/12/1986 and 4/23/1987, respectively. The second requirement, to provide temporary zone markings on safety-related instruments in the control room prior to exceeding five percent power, was also completed, as documented by PSE&G letter number NLR-N86097 dated 7/28/1986 (ADAMS Accession No. 8607310120). FOL 2.C.(12) is a historical requirement that is no longer applicable and should be deleted.	FOL page 6	FOL 2.C.(12)	Delete FOL 2.C.(12)
16.	FOL 2.C.(13) contains requirements related to the Safety Parameter Display System (SPDS) that were required to be completed prior to the earlier of 90 days after the restart from the first refueling outage or July 12, 1988. These SPDS requirements were completed, as documented in PSE&G letter number NLR-N88162, dated 10/13/1988 (ADAMS Accession No. 8810200072). FOL 2.C.(13) is a historical requirement that is no longer applicable and should be deleted.	FOL page 6	FOL 2.C.(13)	Delete FOL 2.C.(13)
17.	FOL 2.C.(21) contains Vibration	FOL page 10	FOL 2.C.(21)	Delete FOL 2.C.(21)

Item	Description	Section / Page	TS / FOL	Action
	Acceptance Criteria for SRVs specifically related to Extended Power Uprate (EPU), which was completed in 2008 via Amendment 174. The requirement to provide the Level 1 main steam safety relief valve vibration acceptance criteria to the NRC prior to increasing power above 3339 MWt was completed via PSEG letter number LR-N08-0123, dated 5/19/2008 (ADAMS Accession No. ML081480508). FOL 2.C.(21) is a historical requirement related to EPU implementation that is no longer applicable and should be deleted. (Reference 4)			
18.	FOL 2.C.(22) contains requirements related to EPU regarding the Steam Dryer. These requirements were completed with the implementation of EPU, as documented in PSEG letter numbers LR-N08-0199, dated 9/2/2008 (ADAMS Accession No. ML082540328), LR-N08-0224, dated 10/27/2008 (ADAMS Accession No. ML083100826), LR-N08-0123, dated 5/19/2008 (ADAMS Accession No. ML081480508), LR-N08-0111, dated 5/8/2008 (ADAMS Accession No. ML081360491), and LR-N09-0167, dated 7/30/2009 (ADAMS Accession No. ML092230344). FOL 2.C.(22) is a historical requirement related to EPU implementation that is no longer applicable and should be deleted. (Reference 4)	FOL pages 10 - 13	FOL 2.C.(22)	Delete FOL 2.C.(22)

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 and fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design.

The proposed changes to the Index page xi Header, 3.1.3.5 Action a., 3.6.2.1 Action c., SR 4.8.4.1.a.2., SR 4.9.8, and 6.10.1 are typographical errors that were either present from initial issuance or were introduced inadvertently through license amendment requests. The proposed change to the SR 4.4.3.2.2.a. footnote is to delete a historical requirement that has expired.

The proposed change to Table 3.3.2-1 Trip Functions 1.a.1); 1.b.; 1.c.; and 1.d. is to remove Valve Group 1 from the listed trip functions. Prior to the issuance of Amendment 134, the Valve Actuation Group 1 valves actuated by the signals listed above were the MSIV Sealing System (MSIVSS) inboard supply valves HV-5834A, HV-5835A, HV-5836A and HV-5837A. These valves were removed after issuance of Amendment No. 134, which revised the TS to delete the MSIV Sealing System. Consistent with the markups submitted in LAR H01-02 (ADAMS Accession No. ML011500295), the references to the MSIVSS valves were deleted from the table notation following Table 3.3.2-1, but were not deleted from the table itself. This appears to be an oversight during the License Amendment process.

The proposed change to Table 3.8.4.1-1 removes the LCO and Surveillance Requirements for Circuit Breaker No. 52-253012, which powered the Recirc Pump Motor Hoist 1AH201. In 2007 the hoist was replaced with a manually actuated model, and the breaker was spared and de-termed. The current TS require that all primary containment conductor overcurrent protection devices shown in Table 3.8.4.1-1 shall be operable in Operational Conditions 1, 2, and 3. If any of the 480 volt circuit breakers are inoperable, they must be removed from service by disconnecting the breaker, and they must be maintained disconnected under administrative control. The current plant design meets the intent of the current TS requirement because the 52-253012 breaker is currently disconnected, and since it is now spared, will remain disconnected. Therefore, the LCO for primary containment penetration conductor overcurrent protection devices is no longer applicable. Circuit Breaker No. 52-253012 and the related information should be removed from Table 3.8.4.1-1.

To support the use of TSTF-369, a model safety evaluation was published titled "Notice of Availability of Model Application Concerning Technical Specifications Improvement to Eliminate Requirements to Provide Monthly Operating Reports and Occupational Radiation Exposure Reports Using the Consolidated Line Item Improvement Process" (ML041690439). In the model safety evaluation, the NRC stated that they had previously approved the elimination of reporting requirements for TS challenges to safety/relief valves with the acceptance of TSTF-258, "Changes to Section 5.0, Administrative Controls." The model safety evaluation made the following statement: "the staff's acceptance of TSTF-258 and subsequent approval of plant-specific adoptions of TSTF-258 is based on the fact that the information on challenges to relief and safety valves is not used in the evaluation of the MOR data, and that the information needed by the NRC is adequately addressed by the reporting requirements in 10 CFR 50.73, 'Licensee event reports'." Therefore, it is acceptable to remove the requirement in the Administrative Controls Section 6.9.1.4 and 6.9.1.5 to report safety/relief valve challenges annually.

The proposed FOL changes involve deleting historical requirements related primarily to initial plant start-up and operation through the first cycle and refueling outage (Conditions 2.C.(4)a., 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), 2.C.(13)), or actions related to Extended Power Uprate (EPU) (2.C.(21) and 2.C.(22)). 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 its 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 Hope Creek Operating License. The proposed changes correct editorial items in the Technical Specifications (TS) and Facility Operating License (FOL) for Hope Creek Generating Station. The proposed changes are administrative in nature and fall into one of four categories: (1) correct typographical errors, (2) delete historical requirements that have expired, (3) make editorial changes to correct errors and/or omissions from previous license amendment requests, or (4) update component lists to reflect current plant design. The affected TS sections are: the Index, 3.1.3.5, Table 3.3.2-1, Surveillance Requirement (SR) 4.4.3.2.2, 3.6.2.1, SR 4.8.4.1, Table 3.8.4.1-1, SR 4.9.8, 6.9.1.4, 6.9.1.5, and 6.10.1. The affected FOL sections are: Conditions 2.C.(4)a., 2.C.(8), 2.C.(9), 2.C.(10), 2.C.(12), 2.C.(13), 2.C.(21), and 2.C.(22).

PSEG has evaluated the proposed changes to the TS and FOL, using the criteria in 10 CFR 50.92, and determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards 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, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

   The proposed changes do not have any impact on structures, systems and components (SSCs) of the plant, and no affect on plant operations. The proposed changes do not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. The proposed changes to the technical specifications do not result in the addition or removal of any equipment but update component lists to reflect equipment that was previously removed or abandoned. Therefore, these proposed changes do 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, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

   The proposed changes do 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 changes 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 editorial in nature that correct typographical errors, or delete historical requirements that have expired. These changes do not affect the intent of any TS requirements.

   The proposed changes incorporate 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, and 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 10 CFR 50.92 (c), and, accordingly, a finding of no significant hazards consideration is justified.

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change recordkeeping, reporting, or administrative procedures or requirements, or would change the format of the license or make editorial, corrective, or other minor revisions.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(10). 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: "Hope Creek Generating Station – Issuance of Amendment Re: Increase in Allowable Main Steam Isolation Valve (MSIV) Leakage Rate and Elimination of MSIV Sealing System (TAC No. MB1970)", Amendment 134, dated October 3, 2001. (ML012600176)

(2) Letter from NRC to PSEG: "Hope Creek Generating Station – Issuance of Amendment Re: Vacuum Breaker Technical Specification Changes (TAC No. MB0323)", Amendment 133, dated October 3, 2001. (ML011730396)

(3) Letter from NRC to PSEG: "Hope Creek Generating Station, Issuance of Amendment to Eliminate Requirements to Provide Monthly Operating Reports and Annual Occupational Radiation Exposure Reports (TAC No. MC6283)", Amendment 161, dated January 11, 2006. (ML060050496)

(4) Letter from NRC to PSEG: "Hope Creek Generating Station – Issuance of Amendment Re: Extended Power Uprate (TAC No. MD3002)", Amendment 174, dated May 14, 2008. (ML081230581)

Technical Specification and Facility Operating License Pages with Proposed Changes

The following Technical Specification and Facility Operating License pages for **Renewed Facility Operating License No. NPF-57** are affected by this change request:

Technical Specification	Page
Index Header	xi
3.1.3.5	3/4 1-9
Table 3.3.2-1	3/4 3-11
4.4.3.2.2.a	3/4 4-12
3.6.2.1.c	3/4 6-13
4.8.4.1.a.2	3/4 8-25
Table 3.8.4.1-1	3/4 8-28
4.9.8	3/4 9-11
6.9.1.4 6.9.1.5	6-17 6-18
6.10.1	6-21

Facility Operating License	Page
2.C.(4)a	4
2.C.(8)	5
2.C.(9)	5
2.C.(10)	5
2.C.(12)	6
2.C.(13)	6
2.C.(21)	10
2.C.(22)	10, 11, 12, 13

Hope Creek Technical Specification Pages with Proposed Changes

Renewed Facility Operating License No. NPF-57

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS ~~BASES~~

REACTIVITY CONTROL SYSTEMS

CONTROL ROD SCRAM ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.1.3.5 Each control rod scram accumulator shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 5*.

ACTION:

--- NOTE ---

Separate condition entry is allowed for each control rod

---

a. In OPERATIONAL CONDITIONS 1 or 2:

1. With one control rod scram accumulator inoperable and reactor pressure ≥ 900 psig, within 8 hours,

   a) Restore the inoperable accumulator to OPERABLE status, or

   b) Declare the associated control rod scram time "slow"***, or

   c) Insert the associated control rod, declare the associated control rod inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN ~~with~~ within the next 12 hours.

2. With two or more control rod scram accumulators inoperable and reactor pressure ≥ 900 psig,

   a) Within 20 minutes of discovery of this condition concurrent with charging water pressure < 940 psig, restore charging water header pressure to ≥ 940 psig otherwise place the mode switch in the shutdown position**, and

   b) Within one hour, declare the associated control rod scram time "slow"***, or

   c) Within one hour insert the associated control rods, declare the associated control rods inoperable and disarm the associated control valves by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

---

* At least the accumulator associated with each withdrawn control rod. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods.

*** Only applicable if the associated control rod scram time was within the limits of Table 3.1.3.3-1 during the last scram time Surveillance. Rods that are already considered "slow" should be declared inoperable and fully inserted.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE ACTUATION GROUPS OPERATED BY SIGNAL	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low Low, level 2	~~1,~~ 2, 8, 9, 12, 13, 14, 15, 17, 18	2	1, 2, 3	20
2) Low low Low, Level 1	10, 11, 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	~~1,~~ 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18	2[j]	1, 2, 3	20
c. Reactor Building Exhaust Radiation - High	~~1,~~ 8, 9, 12 13, 14, 15, 17, 18	3	1, 2, 3	28
d. Manual Initiation	~~1,~~ 8, 9, 10 11, 12, 13, 14, 15, 16, 17, 18	1	1, 2, 3	24
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	19[c]	2	1, 2, 3 and *	26
b. Drywell Pressure - High	19[c]	2[j]	1, 2, 3	26
c. Refueling Floor Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	29
d. Reactor Building Exhaust Radiation - High	19[c]	3	1, 2, 3 and *	28
e. Manual Initiation	19[c]	1	1, 2, 3 and *	26

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric gaseous radioactivity in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage),

b. Monitoring the drywell floor and equipment drain sump flow rate in accordance with the Surveillance Frequency Control Program, and

c. Monitoring the drywell air coolers condensate flow rate in accordance with the Surveillance Frequency Control Program, and

d. Monitoring the drywell pressure in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

e. Monitoring the reactor vessel head flange leak detection system in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage), and

f. Monitoring the drywell temperature in accordance with the Surveillance Frequency Control Program (not a means of quantifying leakage).

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to the IST Program and verifying the leakage of each valve to be within the specified limit:

a. In accordance with the Surveillance Frequency Control Program, ~~**~~ and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies specified in the Surveillance Frequency Control Program.

~~** P.I.V. leak test extension to the first refueling outage is permissible for each RCS P.I.V. listed in Table 3.4.3.2-1, that is identified in Public Service Electric & Gas Company's letter to the NRC (letter No. NLR-N87047), dated April 3, 1987, as needing a plant outage to test. For this one time test interval, the requirements of Section 4.0.2 are not applicable.~~

LIMITING CONDITION FOR OPERATION (continued)

ACTION: (Continued)

3. With the suppression chamber average water temperature greater than 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

the

c. With ~~one~~ drywell-to-suppression chamber bypass leakage in excess of the limit, restore the bypass leakage to within the limit prior to increasing reactor coolant temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.1 The suppression chamber shall be demonstrated OPERABLE:

a. By verifying the suppression chamber water volume to be within the limits in accordance with the Surveillance Frequency Control Program.

b. In accordance with the Surveillance Frequency Control Program in OPERATIONAL CONDITION 1 or 2 by verifying the suppression chamber average water temperature to be less than or equal to 95°F, except:

   1. At least once per 5 minutes during testing which adds heat to the suppression chamber, by verifying the suppression chamber average water temperature less than or equal to 105°F.

   2. At least once per hour when suppression chamber average water temperature is greater than 95°F, by verifying:

      a) Suppression chamber average water temperature to be less than or equal to 110°F.

c. At least once per 30 minutes in OPERATIONAL CONDITION 3 following a scram with suppression chamber average water temperature greater than 95°F, by verifying suppression chamber average water temperature less than or equal to 120°F.

d. By an external visual examination of the suppression chamber after safety/relief valve operation with the suppression chamber average water temperature greater than or equal to 177°F and reactor coolant system pressure greater than 100 psig.

e. In accordance with the Surveillance Frequency Control Program by a visual inspection of the accessible interior and exterior of the suppression chamber.

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value between 150% and 300% of the pickup of the long time delay trip element and verifying that the circuit breaker operates within the time delay bandwidth for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current in excess of 120% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. In accordance with the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

2. 480-VOLT MOLDED CASE CIRCUIT BREAKERS (Continued)

CIRCUIT BREAKER NO.	LOCATION	TYPES	SYSTEMS OR EQUIPMENT POWERED
52-252063	10B252	IM TM	Drywell Equip Drain Sump Pump 1AP267
52-252064	10B252	IM TM	Drywell Floor Drain Sump Pump 1CP267
52-252073	10B252	IM TM	Feedwater Inlet A Shutoff 1AE-HV-F011A
52-262021	10B262	IM TM	Drywell Cooler A Fan 1A2V212
52-262022	10B262	IM TM	Drywell Cooler B Fan 1B2V212
52-262031	10B262	IM TM	Drywell Cooler C Fan 1C2V212
52-262032	10B262	IM TM	Drywell Cooler D Fan 1D2V212
52-262041	10B262	IM TM	Drywell Cooler E Fan 1E2V212
52-262042	10B262	IM TM	Drywell Cooler F Fan 1F2V212
52-262051	10B262	IM TM	Drywell Cooler G Fan 1G2V212
52-262052	10B262	IM TM	Drywell Cooler H Fan 1H2V212
52-262063	10B262	IM TM	Drywell Equip Drain Sump Pump 1BP267
52-262064	10B262	IM TM	Drywell Floor Drain Sump Pump 1DP267
~~52-253012*~~	~~10B253~~	~~IM~~ ~~TM~~	~~Recirc Pump Motor Hoist 1AH201 Disconnect Switch 1AS204~~
52-253021	10B253	IM TM	Recirc Pump 1BP201 Suction Valve 1BB-HV-F023B
52-253031	10B253	IM TM	Recirc Pump 1BP201 Discharge Valve 1BB-HV-F031B
52-253053	10B253	IM TM	Reactor Vessel Head Vent Inboard Isolation 1BB-HV-F001

3/4.9.8 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 2 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least at its minimum required depth within 2 hours prior to the start of and in accordance with the Surveillance Frequency Control Program during handling of fuel assemblies or control rods within the reactor pressure vessel.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the USNRC Administrator, Region 1, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL ~~REPORTS~~

Deleted

6.9.1.4 ~~Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.~~

Deleted

6.9.1.5 ~~Reports required on an annual basis shall include:~~

~~a. Deleted~~

~~*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.~~

~~b. Documentation of all challenges to main steamline safety/relief valves.~~

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 The Annual Radiological Environmental Operating report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

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 DELETED

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.

Hope Creek Facility Operating License Pages with Proposed Changes

Renewed Facility Operating License No. NPF-57

(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)

Deleted

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.

---

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

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) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

Deleted

~~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) Emergency Planning (Section 13.3, SSER No. 5)

Deleted

~~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 50.54(s)(2) will apply.~~

(10) Initial Startup Test Program (Section 14, SSER No. 5)

Deleted

~~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 a rated thermal power feedwater temperature less than 329.6°F for the purpose of extending the normal fuel cycle.

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

Deleted

~~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)

Deleted

~~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 renewed 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 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 renewed 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,

(21) Vibration Acceptance Criteria for SRVs

Deleted

~~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

Deleted

~~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 stream 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 reestablish 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.

(23) Irradiated GE14i fuel bundles shall be stored at least four feet from the wall of the Spent Fuel Pool.

(24) PSEG Nuclear LLC may make changes to the programs and activities described in the UFSAR supplement, submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, provided the licensee evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

(25) Appendix A of NUREG-2102, “Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station,” dated June 2011, and the licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on May 19, 2011, describes certain future programs and activities to be completed before the period of extended operation. PSEG Nuclear LLC shall complete these activities no later than April 11, 2026, and shall notify the NRC in writing when implementation of these activities is complete.