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PSEG
Nuclear LLC

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

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
TRAVERSING IN-CORE PROBE (TIP) SYSTEM
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

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

The proposed change will relocate Technical Specification (TS) 3.3.7.7 and 4.3.7.7, "Traversing In-Core Probe (TIP) System" and its Bases 3/4.3.7.7 to the Hope Creek UFSAR. Note (f) on TS Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements", will be modified to delete reference to the TIP system. This relocation is based on application of the NRC Final Policy Statement in the federal register on July 22, 1993 and 10CFR50.36. By letter dated June 11, 1996 (TAC Nos. M95285 and M95286), Limerick Station Unit 1 and 2 has received NRC approval of a similar request. Also, by letter dated October 29, 1996 (TAC Nos. M95156 and M95157), LaSalle County Station, Units 1 and 2 received NRC approval.

PSEG has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested change is provided in Attachment 1 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 2. The retyped Technical Specification pages are provided in Attachment 3.

PSEG requests approval of this proposed change by August 2005, with implementation within 60 days of receipt of the approved amendment.

A001

If you have any questions concerning this request, please contact Michael Mosier at (856) 339-5434.

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

Executed on 2/4/05

George P. Barnes

George P. Barnes
Site Vice President – Hope Creek

Attachments (3)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS TRAVERSING IN-CORE PROBE (TIP) SYSTEM

Table of Contents

1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station. The proposed change relocates Technical Specification (TS) 3.3.7.7 and 4.3.7.7, "Traversing In-Core Probe (TIP) System" and its Bases 3/4.3.7.7, and modify Note (f) of TS Table 4.3.1.1-1, "Reactor Protection System Instrumentation Surveillance Requirements".

2. PROPOSED CHANGE

Delete TS 3.3.7.7 and 4.3.7.7 and modify note (f) of TS Table 4.3.1.1-1 to delete reference to the TIP system.

3. BACKGROUND

In Section 50.36 of Title 10 of the Code of Federal Regulations (10 CFR 50.36), the Commission established the regulatory requirements related to the content of TSs. That regulation requires that the TSs include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in TSs.

The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), to determine which of the design conditions and associated surveillances should be located in the TSs as limiting conditions for operation. Four criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953):

1. installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

2. a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

3. a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The Commission's Final Policy Statement and documentation related to the revision of 10 CFR 50.36 acknowledged that implementation of these criteria may cause some requirements presently in TSs to be moved out of existing TSs to documents and programs controlled by licensees. Generic Letter 95-10 addresses the relocation of selected TS requirements related to instrumentation as a result of applying the 10 CFR 50.36 criteria. On reviewing typical TSs for nuclear power reactors, the staff determined that, in accordance with the 10 CFR 50.36 criteria, several specifications did not warrant inclusion in TSs. The staff also concluded that the instrumentation addressed by these specifications are not related to dominant contributors to plant risk. The following typical TSs are among the candidates for relocation to licensee-controlled documents:

- Incore Detectors (Movable Incore Detectors, Traversing Incore Probe)
- Seismic Monitoring Instrumentation
- Meteorological Monitoring Instrumentation
- Chlorine Detection System
- Loose-Part Detection System
- Explosive Gas Monitoring Instrumentation
- Turbine Overspeed Protection

The proposed change for Hope Creek is included as part of the first item listed above. Relocation of the TIP system is also included in the Improved Standard Technical Specifications for General Electric Plants, NUREG 1433.

4. TECHNICAL ANALYSIS

The 1993 Final Policy Statement allows for "line item improvements", and delineated four criteria whereby LCOs being considered for relocation, which do not meet any of the criteria, may be removed from TS and relocated to licensee controlled documents. The criteria are as follows:

Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

The TIP system is not used for detecting and indicating significant abnormal degradation of the primary pressure boundary. Any leakage of the portion of the TIP tubing in the reactor pressure boundary would be indicated in the control room similar to any other primary boundary leak (e.g., drywell pressure increase, increased sump flow rates).

Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The TIP system does not meet criterion 2 as it is used as a calibration tool for the LPRMs. The uncertainty of its measurements is included in the core monitoring methods.

Criterion 3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The TIP system's direct accident/transient function is the containment isolation function of the TIPs when they are penetrating primary containment. This system function is not related to the calibration function covered by the subject specification. The TIP system's function is to serve as a calibration tool for the LPRMs. The calibration results are utilized by the core monitoring system to develop uncertainty input values.

Criterion 4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The Traversing-in-Core Probe (TIP) System is used for calibration of the LPRM detectors and update of the parameters that incorporate LPRM and TIP data into the thermal limit calculations. This LCR moves the requirements dealing with these functions to the UFSAR. The remaining TIP-related functions in TS such as those dealing with containment isolation are unaffected. The TIP system (1) is not used to prevent degradation of the reactor coolant pressure boundary, (2) is not a condition of a DBA or transient analysis that is based upon the integrity of the fission product barrier, and (3) is not a portion of the primary success path of a safety sequence and analysis. In addition, NEDO-31466 does not identify any probabilistic risk assessment concerns with the TIP system.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

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

   Response: No.

   The proposed change will relocate the requirements from the Technical Specifications (TS) to the Hope Creek Updated Final Safety Analysis Report (UFSAR) a licensee-controlled document. The relocated requirements will be retained in licensee-controlled documents, which will be maintained under the requirements of the provisions of 10CFR50.59. Since any changes to licensee controlled documents are required to be evaluated per 10CFR50.59, no increase in the probability or consequences of an accident previously evaluated is allowed.

   In addition, these proposed change will not affect any equipment important to safety, in structure or operation. This change will not alter operation of process variables. Structures, systems or components as described in the UFSAR or licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

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

   Response: No.

   The proposed change does not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will not reduce the margin of safety since they have no impact on safety analysis assumptions. Any future changes to the TIP system requirements will be evaluated under 10CFR50.59. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

5.2 Applicable Regulatory Requirements/Criteria

The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), to determine which of the design conditions and associated surveillances should be located in the TSs as limiting conditions for operation. Four criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953). The Hope Creek TIP system does not meet any of the four criterion. Therefore, this information can be relocated to owner controlled documents.

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

6. ENVIRONMENTAL IMPACT EVALUATION

PSEG has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or a surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7. REFERENCES

7.1 Code of Federal Regulations, 10CFR 50.36.
7.2 Improved Standard Technical Specifications for General Electric Plants, NUREG 1433.
7.3 Hope Creek, Updated Final Safety Analysis Report.
7.4 Hope Creek, Technical Specifications.
7.5 Issuance of Amendments- June 11, 1996 (TAC Nos. M95285 and M95286), Limerick Station, Units 1 and 2.
7.6 Issuance of Amendments- October 29, 1996 (TAC Nos. M95156 and M95157), LaSalle County Station, Units 1 and 2.
7.7 Relocation of Selected Technical Specifications Requirements Related to Instrumentation, Generic Letter 95-10.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSES NFP-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License NPF-57 are affected by this change request:

Technical Specification	Page
Index	ix
Index	xviii
Table 4.3.1.1-1	3/4 3-8
3/4.3.7.7	3/4 3-89
B3/4.3.7.7	B3/4 3-5

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 4.3.1.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter/Trip Unit	S	Q(k)	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) DELETED
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH), ~~using the TIP system.~~
(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).
(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.
(i) This item intentionally blank
(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(k) Verify the tripset point of the trip unit at least once per 92 days.
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7 The traversing in-core probe system shall be OPERABLE with:

a. Five movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all five detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, MCPR, or MFLPD.

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the LPRM calibration function.

DELETED

~~*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.~~

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.2 DELETED

3/4.3.7.3 DELETED

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown monitoring instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. For a discussion of SPIRAL RELOAD and SPIRAL UNLOAD and the associated flux monitoring requirements, see Technical Specification Bases Section 3/4.9.2. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial neutron flux distribution of the reactor core.

DELETED

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INSTRUMENTATION

3.3.7.7 DELETED

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.2 DELETED

3/4.3.7.3 DELETED

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3/4.3.7.7 DELETED