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Nuclear Business Unit

MAR 15 2000

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

Gentlemen:

**REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
CORE ALTERATION DEFINITION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354**

In accordance with 10CFR50.90, Public Service Electric & Gas (PSE&G) Company hereby requests a revision to the Technical Specifications (TS) for the Hope Creek Generating Station (HC). In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed changes affect the TS definition of CORE ALTERATION to implement requirements similar to the Improve Technical Specifications. These changes are required to support replacement of the Source Range Monitor (SRM) and Intermediate Range Monitor (IRM) detectors and associated dry tubes during Hope Creek's ninth refueling outage (RFO9), scheduled to commence April 22, 2000. Since approval of these requested changes will have a significant impact on outage operations and schedule, PSE&G is requesting that these changes be reviewed in an expedited manner to support implementation when the outage commences. Similar changes were approved by the NRC for the LaSalle County Station, Units 1 and 2, in a Safety Evaluation Report dated, October 18, 1999.

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and a determination has been made that this request involves no significant hazards considerations. The basis for the requested change is provided in Attachment 1 to this letter. A 10CFR50.92 evaluation, with a determination of no significant hazards consideration, is provided in Attachment 2. The marked-up Technical Specification pages affected by the proposed changes are provided in Attachment 3.

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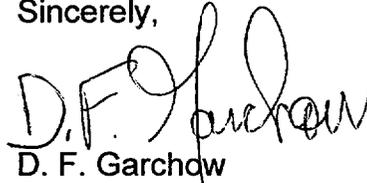
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Upon NRC approval of this proposed change, PSE&G requests that the amendment be made effective on the date of issuance, but allow an implementation period of three days to provide sufficient time for associated administrative activities.

Should you have any questions regarding this request, please contact Mr. James Priest at 856-339-5434.

Sincerely,



D. F. Garchow
Vice President – Technical Support

Affidavit
Attachments (3)

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**HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)**

BASIS FOR REQUESTED CHANGE:

Public Service Electric and Gas Company (PSE&G), under Facility Operating License No. NPF-57 for the Hope Creek Generating Station, requests that the TS contained in Appendix A to the Operating License be amended as proposed herein to revise the TS definition 1.7, CORE ALTERATION. The proposed change removes elements that do not significantly affect core reactivity included in the current definition of CORE ALTERATIONS. Although the scope of the proposed changes is similar to the Improved Standard Technical Specifications (ITS) as documented in NUREG-1433, Revision 1, "Standard Technical Specifications, General Electric Plants, BWR/4," they do not include the additional non-conservative changes associated with control rod movement found in the ITS. Therefore, the additional TS requirements that would otherwise be revised in a manner consistent to the ITS to permit removal of control rods from defueled cells are not required.

The proposed changes result in significant accrued cost reductions and productivity enhancements over the remaining operating life of the plant, allowing outage resources to be directed to other activities, which ultimately will result in improvements in plant maintenance, operations and overall safety. The proposed changes will also eliminate potential TS compliance problems when SRM and IRM detectors need to be replaced during plant outages, with no impact on the margin of safety.

DESCRIPTION OF PROPOSED CHANGES:

The current definition of CORE ALTERATION in TS Section 1.0, Definitions, Item 1.7, follows:

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

The request change revises the TS Definition 1.7, CORE ALTERATION as follows:

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) are not considered to be CORE ALTERATIONS. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

JUSTIFICATION OF REQUESTED CHANGES:

A CORE ALTERATION is a specific activity conducted while in Operational Condition 5, "Refueling," requiring additional controls to be in place. The following requirements are enforced while performing CORE ALTERATIONS:

- Shutdown margin as required by TS Section 3/4.1.1, "Shutdown Margin," must be maintained while in Operational Condition 5, and core alterations must be suspended if shutdown margin requirements cannot be met;
- Neutron monitoring and manual scram functions provided by the reactor protection system, required by TS Section 3/4.3.1, "Reactor Protection System Instrumentation," must be maintained while in Operational Condition 5, core alterations are required to be suspended except for replacement of local power ranger monitor (LPRM) strings provided SRM instrumentation is operable as required by TS Section 3.9.2, "Instrumentation;"
- TS Section 3.9.1, "Reactor Mode Switch," requires that the reactor mode switch shall be operable and locked in the shutdown or refuel position while in Operational Condition 5. If the mode switch is not locked in "Refuel," core alterations shall not be performed unless the refuel position interlocks are operable. Core alterations are to be suspended if the mode switch conditions can not be met, or if applicable, the refueling interlocks are inoperable;
- Two SRMs are required to be operable, with one SRM located in the quadrant where core alterations are being performed, and the other SRM located in an adjacent quadrant, per TS 3.9.2, "Instrumentation." If this condition is not met, all core alterations are to be suspended.

- TS Section 3.9.3, "Control Rod Position," requires all control rods to be inserted during core alterations, except control rods removed per TS 3.9.10.1, "Single Control Rod Removal," and TS 3.9.10.2, "Control Rod Removal;"
- TS Section 3.9.5, "Communications," requires that direct communications be maintained between the control room and the refueling platform personnel during core alterations;
- TS Section 3.9.10.1 and TS Section 3.9.10.2 require that SRMs and the reactor mode switch are operable, with adequate shutdown margin demonstrated, and that appropriate restrictions are place for control rods prior to the removal of a control rod.

The requirement for defining a specific activity (i.e., core alterations), in Operational Condition 5, is to ensure that additional controls are in place to protect against or mitigate a reactivity excursion or fuel assembly drop when moving reactivity control elements. The requirement to demonstrate shutdown margin ensures that the reactor will be maintained sufficiently subcritical to preclude an inadvertent criticality in the shutdown condition. The neutron monitoring instrumentation and the associated reactor protection system trips provide protection against a "reactivity excursion."

The requirements related to the reactor mode switch ensure that the restrictions on control rod withdrawal and refueling platform movement during refueling are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to the reactor internals and fuel assemblies, and exposure of personnel to excessive radiation.

The requirement that all control rods be inserted during core alterations ensures that fuel will not be loaded into a cell without a control rod.

The requirement for communications capability ensures that refueling platform personnel can be promptly informed of significant changes in facility status or core reactivity conditions during movement of fuel within the reactor pressure vessel.

TS requirements related to removal of control rods ensure that maintenance or repair of control rods is performed under conditions that limit the probability of inadvertent criticality.

In RFO9, the SRM and IRM detectors and their associated dry tubes will be replaced to enhance the material condition and reliability of those components. Movement of the SRMs and IRMs in a non-normal manner would constitute core alterations under the current TS definition. The proposed change permits these activities to be performed while not being considered CORE ALTERATIONS. This is expected to produce appreciable savings in outage critical path.

The proposed change to the definition of CORE ALTERATIONS limits the definition of movement to only those components that can affect core reactivity, primarily fuel assemblies and control rods. Unlike the ITS, the proposed changes conservatively do not permit the movement of control rods in defueled cells. The neutron monitoring requirement is applicable at all times while in Operational Condition 5. The proposed change recognizes that the movement of neutron monitoring fission chambers used in boiling water reactors does not significantly affect core reactivity, and places no restrictions on their movement/removal. This is consistent with the current TS in that normal movement of SRMs, IRMs and special movable detectors are not considered to be CORE ALTERATIONS.

Approval of the above changes will enable Hope Creek to more efficiently perform the SRM and IRM detector and dry tube replacements, while minimizing the dose to refueling platform personnel since either a full core off load would be avoided or the number of required refueling bridge tool changes will be reduced. Command and control of CORE ALTERATIONS is not impacted by the proposed changes. Direct communications will be maintained between the control room and the refueling platform personnel during CORE ALTERATIONS as required by TS Section 3.9.5. The refuel floor supervisor will continue to observe and supervise the removal and replacement of SRMs and IRMs, as well as other detectors and control blades, within the reactor pressure vessel.

Provisions to limit potential offsite exposures in the event of a significant release of radioactivity from loads transported over irradiated fuel will be maintained. Secondary containment will be implemented and controlled by station administrative procedures during core alterations and during movement of control blades over irradiated fuel. Secondary containment consists of a the Filtration, Recirculation and Ventilation System (FRVS) that provides a charcoal filter on the ventilation exhaust prior to discharging to the environment and associated radiation monitors that isolate secondary containment on high radiation.

The proposed changes do not impact the requirements for refueling evolutions associated with shutdown margin, core monitoring, and reactor protection system

operability. The existing TS requirements will also require the insertion of all insertable control rods when sufficient SRMs and IRMS are not operable. Secondary containment will continue to be required during CORE ALTERATIONS. There are no changes made to assumptions used the accident analyses. The SRM and IRM maintenance activities may be performed safely and without any undue risk to the public when conducted in accordance with the proposed changes.

ENVIRONMENTAL IMPACT:

The proposed TS changes were reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in the individual or cumulative occupational radiation exposures. Based on the foregoing, PSE&G concludes that the proposed TS changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

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10CFR50.92 EVALUATION

Public Service Electric & Gas (PSE&G) has concluded that the proposed changes to the Hope Creek Generating Station (HC) Technical Specifications do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

The proposed changes affect the TS definition 1.7, CORE ALTERATION. The proposed change removes elements that do not significantly affect core reactivity included in the current definition of CORE ALTERATIONS, similar to the Improved Technical Specifications. The proposed changes can result in significant accrued cost reductions and productivity enhancements over the remaining operating life of the plant, allowing outage resources to be directed to other activities, which ultimately will result in improvements in plant maintenance, operations and overall safety. The proposed changes will also eliminate potential TS compliance problems when SRM and IRM detectors need to be replaced during plant outages, with no impact on the regulatory margin of safety.

BASIS

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed TS change does not involve any physical changes to plant structures, systems or components (SSC) and there is no direct effect on plant operation. The proposed changes do not affect any accident initiators or precursors and do not change or alter the design assumptions for systems or components used to mitigate the consequences of an accident. The proposed changes do not impact the requirements for refueling evolutions associated with the shutdown margin, core monitoring and reactor protection system operability. There are no changes to parameters governing plant operation and no different or new types of equipment will be installed. These changes do not

impact any accident previously evaluated in the Updated Final Safety Analysis Report (UFSAR). Therefore, no increases in the probability of an accident or consequences will result due to this change.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes do not involve any physical changes to the design of any plant SSC. There are no changes to the parameters governing plant operation and no different or new type of equipment will be installed. There is no change in any method by which a safety related system performs its function. No new type of equipment is being introduced and installed equipment is not being operated in a new or different manner. There are no setpoints affected by the proposed action. This proposed action will not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. As such, no new failure modes are being introduced. There are no changes to assumptions in the accident analysis. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed changes contained in this submittal do not adversely affect existing plant safety margins or the reliability of the equipment assumed to operate in the safety analysis. The initial conditions and methodologies used in the accident analyses remain unchanged. Therefore, accident analyses results are not impacted. There are no resulting effects on plant safety parameters or setpoints. The proposal does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings, or a significant relaxation of the bases for the limiting conditions for operations. Therefore, these proposed changes do not cause a reduction in the margin of safety.

CONCLUSION

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

**HOPE CREEK GENERATING STATION
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TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

<u>Technical Specification</u>	<u>Page</u>
Definition 1.7	1-2

DEFINITIONS

CORE ALTERATION

REPLACE WITH INSERT A

- 1.7 ~~CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, TIPS, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.~~

CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY

- 1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.

CORE OPERATING LIMITS REPORT

- 1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

- 1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC-approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

- 1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

E-AVERAGE DISINTEGRATION ENERGY

- 1.12 \bar{E} shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

- 1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation set-point at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

INSERT A

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement) are not considered to be CORE ALTERATIONS. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.