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JAN 0 8 2001

LR-N01-007
LCR H00-009

PSEG
Nuclear LLC

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

Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
EMERGENCY CORE COOLING SYSTEMS SURVEILLANCE REQUIREMENTS
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

In accordance with 10CFR50.90, PSEG Nuclear LLC 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 change revises the Technical Specifications to reduce the acceptable surveillance test values for core spray flow contained in section 4.5.1.b.1.

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, and a statement of environmental considerations is provided in Attachment 2. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 3.

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

Should you have any questions regarding this request, please contact Mr. John Nagle at 856-339-3171.

Sincerely,

Mark B. Bezilla
Vice President – Technical Support

Affidavit
Attachments (3)

A 001

JAN 0 8 2001

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REF: LR-N01-007
LCR H00-09

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

Mark B. Bezilla, being duly sworn according to law deposes and says:

I am Vice President – Technical Support of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Hope Creek Generating Station, Unit 1, are true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 8 day of January, 2001

Notary Public of New Jersey

My Commission expires on _____

SHERI L. HUSTON
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/08/2003

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REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

BASIS FOR REQUESTED CHANGE

PSEG Nuclear LLC, under Facility Operating License No. NPF-57 for the Hope Creek Generating Station, requests that the Technical Specifications (TS) contained in Appendix A to the Operating License be amended as proposed herein: to revise TS 4.5.1.b.1, Core Spray System Flow.

REQUESTED CHANGE, PURPOSE AND BACKGROUND:

PSEG is requesting a change to the Hope Creek Technical Specifications (TS) to change the acceptance values for Core Spray subsystem flow contained in TS 4.5.1.b.1 from the current value of "at least 6350 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure" to "at least 6150 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥ 105 psi above suppression pool pressure".

Current Design Basis: The Core Spray System is required to mitigate the consequences of a decrease in Reactor Coolant Inventory. Following a decrease in Reactor Coolant Inventory event, the Core Spray System must be capable of delivering 6350 gpm/loop with a reactor vessel pressure at least 105 psi greater than the suppression pool pressure and less than 8030 gpm/loop with no pressure differential between the reactor vessel and the drywell.

Figure 6.3-9 of the UFSAR shows the Core Spray system flow capability of the CS pumps with respect to a differential pressure between the drywell (suppression pool) and the reactor vessel. This curve is used within the Appendix K LOCA analysis. The values are 6250 gpm at 105 psid. A delivered Core Spray flow of 6350 gpm at 105 psid to the reactor vessel ensures that 6250 gpm is delivered to the fuel when accounting for 100 gpm due to an assumed shroud by-pass.

Background: The mechanical calculations for Hope Creek are being systematically upgraded. As part of this effort a Hydraulic Model was developed for the Core Spray system using the software program Proto-Flo™. This new analysis includes conservatisms such as instrumentation uncertainty that were not accounted for in the original calculations and reflects the current system configuration. This new analysis identified a situation where the Core Spray System performance differed from the system performance described in the SAR. The Core Spray System may no longer have any operating margin to the flows currently specified in Figures 6.3-5, 6.3-8, and 6.3-9 of the UFSAR under worst-case conditions with the pumps at their maximum allowable degraded condition as allowed by the TS surveillance acceptance criteria. The core spray system remains operable and is capable of delivering sufficient flow to the reactor following a design bases accident based on current pump performance; however, the operating margin was significantly reduced by the new pump acceptance criteria. Revising

the limits based on a required flow of 6150 gpm in lieu of 6350 gpm at 105 psid will regain operating margin.

JUSTIFICATION OF REQUESTED CHANGES:

The Core Spray System is required to mitigate the consequences of a decrease in Reactor Coolant Inventory, specifically the Main Steam Line Break, the Loss of Coolant Accident (LOCA), and the Feedwater Line Break. Following a decrease in Reactor Coolant Inventory event, the Core Spray System must be able to deliver flow to the pressurized reactor vessel.

As a result of the issue discussed above (Background Section) an analysis was performed to determine the acceptability of changing the flow requirement contained in the current design bases and, hence changing the TS Surveillance Requirement to reflect the new flow value. This analysis included ECCS performance analyses (in accordance with approved Appendix K models) by both fuel vendors supplying fuel for the current core (GE and Westinghouse). The results of both of these analyses is that a reduction in Core Spray flow to 6050 gpm (6150 gpm minus 100 gpm for an assumed shroud by-pass) will result in a very small (less than ten (10) °F) increase in Peak Clad Temperature (PCT) in the most limiting case. Other parameters of interest remain bounded by current analyses. This increase in PCT is not considered to be significant using the guidance provided in 10 CFR 50.46(a)(3)(i), which states "Each applicant for or holder of an operating license or construction permit shall estimate the effect of any change to or error in an acceptable evaluation model or in the application of such a model to determine if the change or error is significant. For this purpose, a significant change or error is one which results in a calculated peak fuel cladding temperature different by more than 50°F from the temperature calculated for the limiting transient using the last acceptable model, or is an accumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50°F."

Conclusion

The change to PCT is not considered significant and the resultant PCT is maintained well below the 2200 °F safety limit and hence the requirements specified in GDC 35 are met.

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10CFR50.92 EVALUATION AND ENVIRONMENTAL ANALYSIS

PSEG Nuclear LLC 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 change revises TS 4.5.1.b.1, Core Spray System Flow to specify "at least 6150 gpm at against a test line pressure corresponding to a reactor vessel pressure of ≥ 105 psi above suppression pool pressure". This represents a change in minimum flow from the current value of 6350 gpm.

BASIS

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

The proposed change specifies revised surveillance values for the Core Spray System and does not alter any system or modify any operating procedures. The Core Spray pumps will remain able to perform their required safety related function in order to provide cooling to the reactor core. The revised surveillance value will not increase the consequences of accidents previously evaluated in the SAR.

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

The proposed change specifies revised surveillance requirements of the core spray system and makes no changes to the physical plant or operating procedures. No new accident scenarios, failure mechanisms or limiting single failures are created as a result of the proposed change in the core spray system surveillance value. The change does 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 change specifies surveillance requirements for the core spray system Analyses have determined that for operation at the new surveillance limit, fuel cladding oxidation and hydrogen generation remain within previously analyzed limits. There will

not be a significant increase in peak cladding temperature resulting from this change and that the limits specified in 10CFR50.46 continue to be met.

> 10CFR50.46 (b)(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200° F.
> (2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
> (3) *Maximum hydrogen generation.* The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operation. Thus, the proposed change, which revises the surveillance limit for the core spray system, does not involve a significant reduction in a margin of safety.

CONCLUSION

Based on the above, PSEG Nuclear has determined that the proposed change does not involve a significant hazards consideration.

ENVIRONMENTAL CONSIDERATION

This proposed revision to the Technical Specifications changes a requirement with respect to a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that the proposed changes involve no significant hazards consideration. The proposed changes do not involve a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite; and there is no significant increase in individual or cumulative occupational radiation exposure. Since the proposed changes conform to the criteria for licensing actions eligible for categorical exclusion specified in 10 CFR 51.22(c)(9), no environmental assessment or environmental impact statement is required.

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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:

Technical Specification	Page
4.5.1.b.1	3/4 5-4

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the core spray system, the LPCI system, and the HPCI system:

a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

c) Verify the RHR System cross tie valves on the discharge side of the pumps are closed and power, if any, is removed from the valve operators.

2. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when tested pursuant to Specification 4.0.5:

1. The two core spray system pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥105 psi above suppression pool pressure.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure corresponding to a reactor vessel to primary containment differential pressure of ≥20 psid.

3. The HPCI pump develops a flow of at least 5600 gpm against a test line pressure corresponding to a reactor vessel pressure of 1000 psig when steam is being supplied to the turbine at 1000, +20, -80 psig.**

c. At least once per 18 months:

1. For the core spray system, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.