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PSEG
Nuclear LLC

10 CFR 50.90

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LCR H05-01, Rev. 1

JAN 16 2008

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

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

Subject: Supplement to License Amendment Request for Extended Power Uprate

Reference:
1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, September 18, 2006
2) Letter from USNRC to William Levis (PSEG Nuclear LLC), October 16, 2007
3) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, March 22, 2007

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

Amendment No. 172 (Reference 2) removed values for turbine first stage pressure associated with P_{bypass} from the TSs. P_{bypass} is the reactor power level below which the turbine stop valve closure and the turbine control valve fast closure reactor protection system trip functions and the end-of-cycle recirculation pump trip are bypassed automatically. This change was also included in the proposed TS changes in Reference 1. Attachment 1 to this letter provides revised marked up TS pages reflecting issuance of Amendment No. 172.

Attachment 2 updates PSEG's response to NRC request for additional information (RAI) 5.3 (Reference 3) to reflect weld overlay repairs performed during the most recent refueling outage.

A001
LRR

Attachment 3 revises the description of changes to the reactor recirculation system runback logic.

PSEG has determined that the information contained in this letter and attachments does not alter the conclusions reached in the 10CFR50.92 no significant hazards analysis previously submitted.

There are no regulatory commitments contained within this letter.

Should you have any questions regarding this submittal, please contact Mr. Paul Duke at 856-339-1466.

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

Executed on 1-16-08
(date)

Sincerely,

George P. Barnes

George P. Barnes
Site Vice President
Hope Creek Generating Station

Attachments (3)
1. Revised Markup of Technical Specification Pages
2. Updated Response to Request for Additional Information 5.3
3. Supplement to Request for License Amendment

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

ATTACHMENT 1

Hope Creek Generating Station

Facility Operating License NPF-57
Docket No. 50-354

Extended Power Uprate

Revised Markup of Technical Specification Pages

The following pages reflect changes to the Technical Specifications included in License Amendment No. 172:

TS Page
3/4 3-5
3/4 3-47

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn*.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per the Trip System are 4 APRMS, 6 IRMS and 2 SRMS.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

24%

(k) Also actuates the EOC-RPT system.

* Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure	2[b]
2. Turbine Control Valve-Fast Closure	2[b]

[a] A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

[b] This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

24%

ATTACHMENT 2

Hope Creek Generating Station

Facility Operating License NPF-57
Docket No. 50-354

Extended Power Uprate

Updated Response to Request for Additional Information

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt).

In Reference 2, the NRC requested additional information concerning PSEG's request. PSEG's original response to RAI 5.3 describing welded overlays was provided in Reference 3. During the most recent refueling outage, PSEG performed an additional weld overlay. The updated response to RAI 5.3 is provided below.

5) Piping & NDE Branch (CPNB)

5.3 Identify all flawed components including overlay repaired welds that have been accepted for continued service by analytical evaluation based on American Society of Mechanical Engineers (ASME), Section XI rules. Discuss the adequacy of such analysis considering the effect of the EPU on the flaws.

<u>Updated Response</u>
Hope Creek plant has three welded overlays on ASME Section XI flawed components. There are no ASME Section XI flawed components that have been accepted for continued service by analytical evaluation. The three overlays are on the reactor vessel core spray nozzle to safe end weld (N5B); the reactor vessel recirculation inlet nozzle to safe end weld (N2K); and the reactor vessel recirculation inlet nozzle to safe end weld (N2A).

The core spray overlay was verified adequate for EPU operation. At the core spray nozzle location, there is a slight (0.2%) change in temperature, but no change in pressure or flow due to EPU. Hence, the change in temperature has an insignificant effect on P + Q stresses and the fatigue usage for EPU.

The N2K recirculation inlet overlay was verified adequate for EPU operation. At the recirculation inlet nozzle location, there is a slight increase in pressure (1.1%), a slight decrease in temperature (-0.2%), and an increase in recirculation flow (3.4%). The pressure and temperature operating conditions used in the overlay analysis bound the EPU temperature and pressure conditions. For the flow increase, the change in the heat transfer coefficient used in the analysis is 2.7%, which is considered insignificant.

The design inputs for the N2A recirculation inlet overlay bound the EPU operating conditions.

The three overlays on the Hope Creek plant reactor vessel nozzles are adequate for EPU conditions.

References

1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, September 18, 2006

2) Letter from USNRC to William Levis (PSEG Nuclear LLC), February 23, 2007

3) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, March 22, 2007

ATTACHMENT 3

Hope Creek Generating Station

Facility Operating License NPF-57
Docket No. 50-354

Extended Power Uprate

Supplement to Request for License Amendment

In Reference 1, PSEG Nuclear LLC (PSEG) requested an amendment to Facility Operating License NPF-57 and the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS) to increase the maximum authorized power level to 3840 megawatts thermal (MWt). Attachment 4 to Reference 1, NEDC-33076P, Rev. 2, "Safety Analysis Report for Hope Creek Constant Pressure Power Uprate," described planned changes to the reactor recirculation system (RRS) runback logic.

The planned changes included revising the Secondary Condensate Pump (SCP) permissive setpoint so that the RRS runback logic would be armed when feedwater (FW) flow is more than 85% of the EPU rated FW flow. However, PSEG subsequently elected to maintain the SCP permissive setpoint at approximately the same FW flow (in Mlb/hour) as the pre-EPU setpoint. Thus, after EPU implementation, the RRS runback logic for both PCP and SCP trips will be armed when FW flow is greater than approximately 75% of EPU rated FW flow.

The instrument setpoint changes implemented for EPU are shown below. For EPU, the PCP and SCP permissive nominal setpoints are equal to the percent flow span when FW flow is 75% of rated flow. The setpoints provided previously in Table 5-3 in NEDC-33076P, Revision 2, included allowances for instrument loop uncertainties which are not required, given the flow capacity remaining after the trip of a single PCP or SCP. Changes from the information provided in NEDC-33076P, Revision 2, are marked by a revision bar in the margin.

Parameter / Device	CPPU Nominal Setpoint
Primary Condensate Pump 75% Permissive [2] (% FW flow span)	62.8
Secondary Condensate Pump 75% Permissive [2] (% FW flow span)	62.8

2. The Digital FW Control System processes the signals from both FW flow transmitters for the permissives for the Primary and Secondary Condensate Pumps. The setpoints are revised to set both the PCP and SCP trips at the same flow rate. Since both RRS runbacks will be intermediate runbacks at CPPU, the setpoints were chosen to be comparable to the current FW flow rates for the SCP trip.

These changes to the RRS permissive setpoints do not effect the results or conclusions of the analyses described in Reference 2 for trip of a reactor feed pump, PCP or SCP (RAI Responses 7.16, 7.17 and 7.18).

References

1) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, September 18, 2006

2) Letter from George P. Barnes (PSEG Nuclear LLC) to USNRC, June 22, 2007