PSE&G

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038

Hope Creek Generating Station

December 21, 1994

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

HOPE CREEK GENERATING STATION
DOCKET NO. 50-354
UNIT NO. 1
SPECIAL REPORT 94-003-00

This Special Report is being submitted pursuant to the requirements of Hope Creek Generating Station Facility Operating License NPF-57.

Sincerely,

R.J. Hovey
General Manager -
Hope Creek Operations

LAA/

Attachment
SORC Mtg. 94-085
C Distribution

IE22 1/1

SPECIAL REPORT		
FACILITY NAME (1) HOPE CREEK GENERATING STATION	DOCKET NUMBER (2) 0 5 0 0 0 3 5 4	PAGE (3) 1 OF 5

TITLE (4): Condition prohibited by the Facility Operating License - Operation of the facility in excess of the licensed power level.

EVENT DATE (5)			NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	*	NUMBER	*	REV	MONTH	DAY	YEAR	FACILITY NAME(S)	DOCKET NUMBER(S)
1 1	2 1	9 4	9 4	-	0 0 3	-	0 0	1 2	2 1	9 4		

OPERATING MODE (9)	1
POWER LEVEL %	1 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10CFR: (CHECK ONE OR MORE BELOW) (11)

	20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)	XX	OTHER (Specify in Abstract below and in Text)
	20.405(a)(1)(iii)		50.73(a)(2)(i)(B)		50.73(a)(2)(viii)(A)		
	20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
	20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
Lou Aversa, Senior Staff Engineer - Technical	6 0 9 3 3 9 3 3 8 6

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE NOTED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS?	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS?
B									

SUPPLEMENTAL REPORT EXPECTED? (14)	YES	x	NO		DATE EXPECTED (15)	MONTH	DAY	YEAR
						0 1	3 1	9 5

ABSTRACT (16)

On Monday, November 21, 1994, at 2100 hours, a Nuclear Control Operator (NCO-RO licensed) performing a review of an hourly Periodic Core Monitor Report (P1 edit) noticed that a failed sensor, associated with Control Rod Drive System flow, was indicated on the edit. The sensor is utilized as an input to the thermal heat balance calculation that determines reactor thermal power level. The current P1 edit indicated that reactor thermal power was 3292 Megawatts Thermal (MWt). The sensor was restored and a new P1 edit indicated reactor power was 3307MWt. Since this was above the licensed power level of 3293 MWt, reactor power was reduced to 3288MWt, by reducing reactor recirculation flow. A review of the previous P1 edits for the day revealed previous shift average power level had exceeded the license power limit. A second over power condition was discovered on December 12, 1994, when preliminary results of a feedwater flow calibration test indicated a non-conservative feedwater flow input into the process computer. As a result of the preliminary analysis reactor operation has been limited to 3277 MWt. The root cause of the initial event was attributed to personnel errors. The results of an evaluation of the feedwater flow discrepancy will be reported in a supplemental report. Personnel involved in the initial event have been counseled and disciplined. The event has been reviewed with other maintenance department personnel for lessons learned. Interim actions for the second event include limitations on reactor power and Nuclear Instrument calibrations until the feedwater flow discrepancy evaluation is complete.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor (BWR/4)
Plant Computer - RJ EIIS Identifier (ID)

IDENTIFICATION OF OCCURRENCE

TITLE (4): Condition prohibited by the Facility Operating License - Operation of the Facility in excess of the licensed power level.

Event 1 Date: 11/21/94
Event Time: 1400 to 2100
Event 2 Discovery Date: 12/12/94
This SR was initiated by Incident Report No. 94-216 and 94-235

CONDITIONS PRIOR TO OCCURRENCE

Plant in OPERATIONAL CONDITION 1 (Power Operation)
Reactor Power 100% of rated, 1108 MWe

DESCRIPTION OF OCCURRENCE

On Monday, November 21, 1994, at 2100 hours, a Nuclear Control Operator (NCO-RO licensed) performing a review of an hourly Periodic Core Monitor Report (P1 edit) noticed that a failed sensor, associated with Control Rod Drive System flow, was indicated on the edit. The sensor is utilized as an input to the thermal heat balance calculation that determines reactor thermal power level. The current P1 edit indicated that reactor thermal power was 3292 Megawatts Thermal (MWt). The sensor was restored and a new P1 edit indicated reactor power was 3307MWt. Since this was above the licensed power level of 3293 MWt, reactor power was reduced to 3288MWt, by reducing reactor recirculation flow. A review of the previous P1 edits for the day revealed the sensor had failed at 1349 hours. The 1400 hour P1 indicated the sensor was failed and that core thermal power was 3282MWt. A review of the P1 edits for the previous 12 hour shift determined that reactor power had been increased after the 1400 hour P1 to 3293MWt and was maintained at 3293 for the period the sensor was failed. It was determined that licensed reactor average power level was exceeded for the preceding 12 hour shift period (100.09%). Reactor power was reduced to 3285 MWt for the remainder of the current 12 hour shift to ensure the 12 hour average licensed power level would not be exceeded.

In a second event, on Monday, December 12, 1994, preliminary results of a feedwater flow calibration test indicated a non-conservative feedwater flow input into the process computer resulting in a lower than actual calculated thermal power. As a result of the preliminary analysis, reactor operation has been limited to 3277 MWt. Limitations have been placed on Nuclear Instrument calibrations to ensure margins to Technical Specification limits are not exceeded. General Electric

DESCRIPTION OF OCCURRENCE

Company is performing a validation of the test data. An evaluation will be performed to determine final corrective action(s). The investigation results and corrective actions will be presented in a supplement to this special report.

ANALYSIS OF OCCURRENCE

The Nuclear Steam Supply System (NSSS) Process Computer is used to monitor core performance and provide indication of core thermal power. The NSSS computer utilizes inputs from field devices in the Control Rod Drive (CRD) System, Feedwater System, Reactor Water Cleanup (RWCU) System, Reactor Recirculation System and Reactor pressure to calculate the thermal heat balance. The computer monitors the output of each device for a reasonable value and whether the device output is within valid limits. In the event the output becomes invalid (failed sensor) the computer will substitute the last known reasonable value into the thermal heat balance calculation, and print a coded number for the failed sensor on the hourly P1 edit.

A root cause investigation of the first event determined that the CRD flow input error to the process computer resulted when the CRD flow transmitter calibration was performed without the required notification being made to the Reactor Engineering Department. The CRD flow transmitter is calibrated using a generic calibration procedure for this type of transmitter. As this generic procedure can be used for transmitters in a wide variety of systems or applications, specific notes and precautions regarding the function of the transmitters are specified on the Instrument Control Data Card (ICD Card) for each transmitter. The generic calibration procedure refers the user to the ICD Card for any special instructions regarding the device being calibrated. The ICD card for this transmitter required a notification to the Reactor Engineering group as this transmitter affects the heat balance. The Instrument and Controls technicians performing the calibration read the note, however, they assumed the notification was performed by the job supervisor during his pre-job briefing with the nuclear shift supervisor. The briefing did not cover this item. When the technicians began the calibration, the transmitter began trending downscale. As the transmitter output decreased, but prior to reaching the failed sensor value, the computer updated with a lower than normal flow value. When the output reached the failed sensor value, the computer used the last known reasonable value as a substitute although this value was significantly lower than actual CRD flow.

The NCOs monitor core thermal power via the hourly P1 edits provided by the process computer. The P1 edit provides operational data regarding core performance, thermal limits and indication of instrument failures that are utilized in the calculation of the data. The Operators are required to review the edit for indication of failed instruments and notify Reactor Engineering. The edit following the start of the calibration did indicate the failed flow sensor for the CRD flow. The

ANALYSIS OF OCCURRENCE

Operators who reviewed the edit did not identify the failed CRD flow sensor until 2200 hours. Reactor power indication also changed following the flow sensor failure. The NCO assumed this power reduction indicated on the P1 edit resulted from a Xenon transient that was in progress from a power reduction taken the previous day for turbine valve testing. The NCO increased reactor power to 3293 MWt from the indicated 3282 MWt provided on the 1400 hour P1 edit. At 1500 hours the P1 indicated that reactor power was still low (3286MWt) and the NCO again increased reactor power to 3293MWt.

SAFETY SIGNIFICANCE

This event posed minimal safety significance. Reactor power was maintained within uncertainties used in the Hope Creek Accident and Transient Analysis throughout this event.

PREVIOUS OCCURRENCES

There has been one previous occurrence of an overpower condition. This event was attributed to a span error on the feedwater flow transmitters. See LER 88-024-00.

APPARENT CAUSE OF OCCURRENCE

The root cause of the first event was procedural non-compliance. The technicians performing the transmitter calibration did not follow the ICD card instructions. Contributing to this event was operators not performing an adequate review of the P1 edit and not accounting for unexpected power changes.

The root cause of the second event will be provided in the supplement to this report following completion of the investigation.

CORRECTIVE ACTIONS

Personnel involved in this event have been and will be counseled and or disciplined as deemed appropriate.

The event was discussed with all controls shop personnel emphasizing the use of the ICD card "notes" section and the procedural adherence requirements associated with it and the importance of performing all steps of a procedure as written.

Planning will identify and annotate the preventive maintenance work orders for devices which input to the core thermal power calculation.

CORRECTIVE ACTIONS

A list of items that the NCO is expected to review on each P1 edit has been provided by Reactor Engineering to the Operations Department.

Reactor Engineering will provide periodic reports, to Operations, containing expected Xenon response and affects of cycle exposure for planned transients.

Operators training will be provided on utilization of the above reports.

An interim action to limit reactor power will remain in effect until the evaluation of the feedwater flow deviation is completed.

Limitations have been placed on Nuclear Instrument calibrations to ensure margins to Technical Specification limits are not exceeded.

Sincerely,

RJH

R.J. Hovey
General Manager -
Hope Creek Operations

SORC Mtg. 94-085
Recommended approval: Yes
C Distribution