PSEG

Public Service Electric and Gas Company P.O. Box 236 Hancocks Bridge, New Jersey 08038-0236

Nuclear Business Unit

November 12, 1995

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

Dear Sir:

MONTHLY OPERATING REPORT
HOPE CREEK GENERATION STATION UNIT 1
DOCKET NO. 50-354

In compliance with Section 6.9, Reporting Requirements for the Hope Creek Technical Specifications, the operating statistics for **October 1995** are being forwarded to you with the summary of changes, tests, and experiments that were implemented during **October 1995** pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

Mark Reddemann
General Manager -
Hope Creek Operations

DL:RS:JC
Attachments

C Distribution

180002

INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

AVERAGE DAILY UNIT POWER LEVEL

MONTH OCTOBER 1995

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	1021	17	995
2	1019	18	988
3	1016	19	983
4	1012	20	975
5	992	21	968
6	988	22	974
7	1004	23	972
8	998	24	954
9	1007	25	973
10	1007	26	963
11	1005	27	956
12	1000	28	949
13	994	29	945
14	993	30	943
15	990	31	938
16	998		

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

OPERATING DATA REPORT
OPERATING STATUS

1. Reporting Period October 1995 Gross Hours in Report Period 745.

2. Currently Authorized Power Level (MWt) 3293
   Max. Depend. Capacity (MWe-Net) 1031
   Design Electrical Rating (MWe-Net) 1067

3. Power Level to which restricted (if any) (MWe-Net) None

4. Reasons for restriction (if any)

		This Month	Yr To Date	Cumulative
5.	No. of hours reactor was critical	745.0	6746.3	66682.2
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	745.0	6696.6	65700.0
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	2330248	21688124	210102470
10.	Gross electrical energy generated (MWH)	766976	7170966	69598633
11.	Net electrical energy generated (MWH)	733449	6861669	66514985
12.	Reactor service factor	100.0	92.5	85.8
13.	Reactor availability factor	100.0	92.5	85.8
14.	Unit service factor	100.0	91.8	84.5
15.	Unit availability factor	100.0	91.8	84.5
16.	Unit capacity factor (using MDC)	95.5	91.2	83.0
17.	Unit capacity factor (using Design MWe)	92.3	88.1	80.2
18.	Unit forced outage rate	0.0	8.2	5.1

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Refueling Outage, November 11, 1995, Duration Under Review

20. If shutdown at end of report period, estimated date of start-up:
    N/A

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

OPERATING DATA REPORT
UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH OCTOBER 1995

NO.	DATE	TYPE F=FORCED S=SCHEDULE	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
1.		NONE				

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

REFUELING INFORMATION

MONTH OCTOBER 1995

1. Refueling information has changed from last month:

   Yes X No __

2. Scheduled date for next refueling: 11/11/95

3. Scheduled date for restart following refueling: Under Review

4A. Will Technical Specification changes or other license amendments be required?

   Yes __ No X

B. Has the Safety Evaluation covering the COLR been reviewed by the Station Operating Review Committee (SORC)?

   Yes __ No X

   If no, when is it scheduled? November 15, 1995

5. Scheduled date(s) for submitting proposed licensing action:

   Not required.

6. Important licensing considerations associated with refueling:

   N/A

7. Number of Fuel Assemblies:

A. Incore	764
B. In Spent Fuel Storage (prior to refueling)	1240
C. In Spent Fuel Storage (after refueling)	1472

8. Present licensed spent fuel storage capacity: 4006
   Future spent fuel storage capacity: 4006

9. Date of last refueling that can be discharged to spent fuel pool assuming the present licensed capacity: 5/3/2006 (EOC13)

(Does allow for full-core off-load)
(Assumes 244 bundle reloads every 18 months until then)
(Does not allow for smaller reloads due to improved fuel)

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

MONTHLY OPERATING SUMMARY

MONTH OCTOBER 1995

The Hope Creek Generating Station remained on-line for the entire month operating at reduced power as the planned coastdown at the end of cycle 6 continued. This coastdown reduced the maximum attainable unit power during the month from approximately 99.3% on October 1, 1995 to 89.2% on October 31, 1995. The following is a summary of events and activities that caused minor (<20%) deviations in megawatt output during the month of October 1995:

- Power was reduced for turbine valve surveillances on October 1, 8 and 15, 1995.
- Core flow was reduced on October 5, 1995 because of a safety concern involving high reactor recirculation flows and reactor vessel and recirculation piping vibrations. The reduction in core flow caused a reduction in reactor power. Based on an engineering evaluation core flow was returned to approximately 103.5% and power restored October 6, 1995. While power was reduced, the 3A though 6A feedwater heaters were removed for a planned outage and returned to service.
- Power was reduced late on October 5, 1995 and restored early on October 6, 1995 for the removal of the 3A through 6A feedwater heaters for planned maintenance.
- Power was reduced October 22, 1995 for the removal of the 3C through 6C feedwater heaters for planned maintenance.
- Because of control problems the B Reactor Feed Pump was manually tripped on October 24, 1995. Power was restored within the hour.

At the end of the month the unit had been on-line for 99 days.

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 11/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS FOR THE HOPE CREEK GENERATING STATION

MONTH OCTOBER 1995

The following items have been evaluated to determine:

1. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

3. If the margin of safety as defined in the basis for any technical specification is reduced.

The 10CFR50.59 Safety Evaluations showed that these items did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These items did not change the plant effluent releases and did not alter the existing environmental impact. The 10CFR50.59 Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Procedure Summary of Safety Evaluation

- **HC.OP-SO.CC-0001(Z), REV 13 - GENERATOR GAS CONTROL SYSTEM OPERATION** This revision changes the valves used for filling the generator with hydrogen. The procedure has valves 01KH-V018 and 01KH-V019 normally closed. UFSAR Figure 10.2.3 shows these valves normally open. UFSAR Change request 95-043 will be submitted to revise the UFSAR. Also, included in this UFSAR change to Figure 10.2.3 is the change of position for 1CC-C035 and 1CC-C037 from normally open to normally closed. These valves isolate the automatic make-up valve that would keep the generator full of hydrogen. Hope Creek does not use automatic make-up of hydrogen for safety reasons. In the event of a fire, auto make-up would feed the fire with hydrogen.

  The credible failure mode is hydrogen, a combustible gas, escaping from a leaking valve. The valves and the storage tanks are located outside and therefore pose minimum risk for the plant. The isolation is being changed to the supply side of the pressure control valve to guarantee a leak free installation inside the plant. There are no transients or design basis accidents associated with either the Main Generator - Gas Control or Service Gas systems. The hydrogen storage skid is not associated with nor does it interface with safety-related or important to safety components.

  Therefore, this procedure revision does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

Deficiency Report Summary of Safety Evaluation

- **DR #950929231 - PRIMARY CONTAINMENT INSTRUMENT GAS SKID MOISTURE SEPARATORS** This evaluation supports the "Use-As-Is" disposition of Deficiency Report # 950929231 to operate the PCIG system with the existing moisture separators on each skid. The moisture separators are oversized for the application. In order to be effective these moisture separators require a minimum flow of 145 SCFM, this application has a flow of only 35 SCFM. With this lower flow the necessary centrifugal forces are not established within the separator and the moisture removal process does not occur. During normal operation, the system will continue to operate within the UFSAR design specification of air with a dew point ≤ 35F.

  During the worst case Design Basis LOCA, the suction of the PCIG from the secondary containment is assumed to be 148F, 100% humidity air for 100 days. Under these conditions the separators will function to some extent and the drying towers will be overworked but the skid will remain in operation and the air will have an elevated dew point. A large portion of this water will be separated in the coolers because of the temperature drops. This water will exit the system via a drain line. More water will be removed by the in-line filters and the drying towers but eventually these will be saturated. At this point the moisture content of the outlet air will equal that of the inlet air. It has been calculated that the air receivers on the system could hold the entire volume of water expected during the 100 day post-LOCA period.

  There are no new failure modes associated with this change. During normal operation the air supplied will still meet the UFSAR design specification. The degraded nature of the separators does not adversely affect the ability of the PCIG to perform its normal, LOP or post-LOCA functions. There are no new malfunctions of equipment expected because of this condition. Analysis of operation of the SRVs and MSIV Sealing Steam system with higher water content in the instrument gas has been performed and found acceptable.

  Therefore, the "Use-As-Is" disposition on this Deficiency Report does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

Other Summary of Safety Evaluations

- **UFSAR CHANGE NOTICE CN 94-10, CORRECTING THE DESCRIPTION OF BREAKER OPERATION FROM FAULT RELAY ACTUATION** This change notice corrects incomplete or erroneous information in UFSAR Sections 8.2 and 8.3 concerning transformers and transformer feeder faults and undervoltages of normal voltage sources. The change ensures the UFSAR reflects the current engineering documents and is consistent with the Technical Specifications and the design basis of the Electrical Distribution System. No physical changes are being made in the plant and no system parameters are affected by the change. The facility always did meet the requirements of the Technical Specifications and would have responded properly to electrical faults. There is no change to the probability or consequences of accidents or malfunctions of equipment previously evaluated in the UFSAR because the change only corrects and clarifies information in the UFSAR. The change cannot create an accident or malfunction of a different type because it only corrects and clarifies information in the UFSAR.

  Therefore, this UFSAR change does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

Other Summary of Safety Evaluations (continued)

- **UFSAR CHANGE NOTICE CN H88-32, INCREASED FIRE LOAD DUE TO FIBERGLASS LADDERS IN THE POWER BLOCK** - This change notice revises the Fire Hazards Analysis to include the affect of fiberglass ladders permanently stored in the power block on brackets installed by DCP 4HM-0119. Both storage at the maximum capacity of the brackets and temporarily locating the ladders in other places in the power block while in use were considered.

  Calculations of combustible loads and fire severity performed assuming each permanent storage area was at its maximum complement revealed that the increase in combustible loading is well within the design capability of the existing fire protection features. The text of UFSAR Section 9A.1.8 will be revised to acknowledge that the permanent storage areas for fiberglass ladders have been established with an insignificant affect on fire loading. Use of the ladders throughout the plant will be controlled in accordance with plant procedures. Fire load values, as identified in UFSAR Section 9A, are not functional entities and therefore have no failure modes associated with them. Failure modes do not apply to fire barriers when fire load values do not exceed the design rating of the barriers. There are no operational transients or postulated design basis accidents associated with this change. The presence of the ladders has no affect on the probability of a fire starting as they cannot combust without an ignition source. Per Generic Letter 86-10, "accident" is a postulated fire.

  The Fire Hazards Analysis (FHA) does not assess fire risk in terms of likelihood but rather bases the analysis on the premise that a fire will occur, and damage to equipment important to safety within the fire areas happens. As such, fire load increases within fire areas do not affect the bases of the FHA provided the fires are contained within the boundary of the fire area. Since the fire load in all affected areas remains within the design, fire barrier integrity will be maintained and the probability of occurrence of a malfunction of equipment due to fire spread remains unchanged.

  Therefore, this UFSAR change does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

- **UFSAR CHANGE NOTICE CN 87-173 UPDATE UFSAR TABLE 7.5-1** This change notice corrects erroneous information in UFSAR Table 7.5-1, "Displayed Parameters Important to Safety," and ensures the table reflects the applicable engineering documents. The information being inserted is consistent with the Technical Specifications and the design basis of the Post Accident Monitoring Instrumentation as provided to meet Reg Guide 1.97. No physical changes are being made in the plant and no systems or parameters are affected by the change. The facility always did meet the requirements of Reg Guide 1.97. The operators would have used the correct instrumentation regardless of errors in the Table. There are no credible failure modes. There is no change to the probability or consequences of accidents or malfunctions of equipment previously evaluated in the UFSAR because the change only corrects erroneous information and ensures the table reflects the applicable engineering documents, the Technical Specifications, and the design basis of the Post Accident Monitoring Instrumentation. The change cannot create an accident or malfunction of a different type because it only corrects and clarifies information in the UFSAR.

  Therefore, this UFSAR change does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

Other Summary of Safety Evaluations (continued)

- **EXAMINATION REQUIRED BY NUREG-0619, FEEDWATER NOZZLE SAFE-END, BLEND RADIUS, AND BORE INSPECTION SCHEDULE** The Hope Creek In Service Inspection (ISI) Long Term Plan (LTP) is being revised to move the performance of an ultrasonic (UT) examination of the feedwater nozzle safe-end, blend radius, and bore from RF06 to RF07. This change to the Hope Creek ISI LTP is a change to a procedure described in the UFSAR, as per the Hope Creek ISI LTP SER, dated December 11, 1987. Implementation of this change will reduce occupational exposure, outage time and costs, allow additional time for the for the BWR Owners' Group to develop a generic resolution of the issue of feedwater nozzle inspections. No plant modifications are involved in this change. There are no credible failure modes. If a leak did occur it would be detected by the leak detection system well in advance of a pipe failure, which is consistent with the leak before break concept.

  If a failure did occur, it is within the evaluated design basis of the plant. Failure of the feedwater line is evaluated in UFSAR Section 6.3. Extending the interval does not change the assumptions associated with the analysis of pipe breaks contained in chapters 6 and 13 of the UFSAR. The change in inspection interval does not affect the probability of the event because the inspections are still being performed within 16 years, which is the time for postulated crack growth to 1" (General Electric calculation NEDC-32480-P) The malfunction of equipment important to safety would be a failure of the reactor coolant pressure boundary, which is the feedwater line inside the drywell. This failure is already evaluated as part of our licensing basis. This proposal does not change any failure mechanism or frequency previously evaluated. The UFSAR licensing basis accident analysis concludes that he results of pipe breaks, including the DBA feedwater line breaks within the dry well are acceptable. The design basis piping failure remains the recirculation system pipe break, which bounds a failure of the feedwater nozzle safe-end. Therefore there is no increase in the consequences of a malfunction of equipment important to safety previously evaluated in the UFSAR.

  Therefore, this UFSAR change does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

Design Changes Summary of Safety Evaluations

- There were no changes, tests, or experiments in this category this month.

Temporary Modifications Summary of Safety Evaluations

- There were no changes, tests, or experiments in this category this month.