PSEG

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

Hope Creek Generating Station

November 13, 1992

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 are being forwarded to you along with the summary of changes, tests, and experiments for October 1992 persuant to the requirements of 10CFR50.59(b).

Sincerely yours,

J. J. Hagan
General Manager -
Hope Creek Operations

RAR:ld
Attachments

C Distribution

The Energy People

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INDEX

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO.	50-354
UNIT	Hope Creek
DATE	11/13/92
COMPLETED BY	V. Zabielski
TELEPHONE	(609) 339-3506

MONTH October 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	0	17.	0
2.	0	18.	0
3.	0	19.	0
4.	0	20.	0
5.	0	21.	0
6.	0	22.	0
7.	0	23.	0
8.	0	24.	0
9.	0	25.	0
10.	0	26.	0
11.	0	27.	0
12.	0	28.	0
13.	0	29.	0
14.	0	30.	0
15.	0	31.	0
16.	0		

OPERATING DATA REPORT

DOCKET NO. 50-354
UNIT Hope Creek
DATE 11/13/92
COMPLETED BY V. Zabielski Vince
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period October 1992 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	0.0	5804.5	42,965.8
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	0.0	5742.0	42,316.6
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	0	18,508,363	134,505,506
10.	Gross electrical energy generated (MWH)	0	6,145,590	44,49[illegible],084
11.	Net electrical energy generated	0	5,860,258	42,511,807
12.	Reactor service factor	0.0	79.3	83.5
13.	Reactor availability factor	0.0	79.3	83.5
14.	Unit service factor	0.0	78.4	82.3
15.	Unit availability factor	0.0	78.4	82.3
16.	Unit capacity factor (using MDC)	0.0	77.7	80.2
17.	Unit capacity factor (Using Design MWe)	0.0	75.0	77.5
18.	Unit forced outage rate	0.0	2.2	4.8

19. Shutdowns scheduled over next 6 months (type, date, & duration):
None

20. If shutdown at end of report period, estimated date of start-up:
11/10/92

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 11/13/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH October 1992

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
8	10/1	S	745	C	4	Continuation of 4th Refueling Outage

Summary

REFUELING INFORMATION

DOCKET NO.	50-354
UNIT	Hope Creek
DATE	11/13/92
COMPILED BY	S. Hollingsworth
TELEPHONE	(609) 339-1051

October 1992

1. Refueling information has changed from last month:

   No X

2. Scheduled date for next refueling: 9/12/92

3. Scheduled date for restart following refueling: 11/11/92

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

   Yes No X

   B. Has the reload fuel design been reviewed by the Station Operating Review Committee?

   Yes X No

   If no, when is it scheduled?

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

6. Important licensing considerations associated with refueling:

   - Same fresh fuel as current cycle: no new considerations

7. Number of Fuel Assemblies:

A.	Incore	764
B.	In Spent Fuel Storage (prior to refueling)	760
C.	In Spent Fuel Storage (after refueling)	1008

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: (does not allow for full-core offload) 11/4, 2010 (EOC16)

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

October 1992

The 4th Refueling Outage began on September 12 and continued throughout the month of October. As of October 31, the unit was planned to be back on line on November 10.

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

OCTOBER 1992

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.

DCP	Description of Safety Evaluation
4EC-1010/05	This DCP connected permanent power feeds from lighting panels to existing convenience lights and receptacles in the Nuclear Steam Supply System panels in the Main Control Room. This DCP enhances the working environment for the performance of maintenance and surveillance testing in either the main Control Room or the lower control equipment room. The operability of the safety-related panels is not affected by this DCP. The installation meets seismic and electrical separation criteria. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-1021/01 4EC-1021/02 4EC-1021/03 4EC-1021/04	These DCPs removed some snubbers or replaced them with struts. The affected snubbers were in the Main Steam Lines and their associated Safety Relief Valve piping lines. The removal of the snubbers and the conversion to struts decreases the chance for the piping system to be in an unanalyzed condition due to snubber failure, increases the reliability of safety related equipment, and reduces station man-rem. All of the analyses to reduce the snubber population were performed per the ASME Boiler and Pressure Vessel Code requirements. They meet the design intent of the UFSAR, including the postulated pipe break criteria. Therefore, these DCPs do not involve any Unreviewed Safety Questions.
4EC-1021/05 4EC-1021/06	These DCPs removed some snubbers or replaced them with struts. The affected snubbers were in the Reactor Recirculation lines, the Residual Heat Removal lines, and their related components and equipment. The removal of the snubbers and the conversion to struts decreases the chance for the piping system to be in an unanalyzed condition due to snubber failure, increases the reliability of safety related equipment, and reduces station man-rem. All of the analyses to reduce the snubber population were performed per the ASME Boiler and Pressure Vessel Code requirements. They meet the design intent of the UFSAR, including the postulated pipe break criteria. Therefore, these DCPs do not involve any Unreviewed Safety Questions.

DCP	Description of Safety Evaluation
4EC-1054/01 4EC-1054/02 4EC-1054/03 4EC-1054/04	These DCPs modified the Emergency Diesel Generator Starting Air skid supply lines from the Air Dryers to the Air Receiver Tanks. The DCP includes the installation of Liquid Drain Tanks to collect any condensate in the supply lines and flush it through the floor drains. The installation of new piping, fittings, and drain tanks will improve system performance and reliability because the condensate water will no longer collect in the receivers, thereby reducing corrosion. The addition of the new valves will improve the availability of the Emergency Diesel Generators because the Starting Air Receiver Tanks can be fed from any compressor, allowing a compressor outage without affecting the operability of an Emergency Diesel Generator. Therefore, these DCPs do not involve any Unreviewed Safety Questions.
4EC-3022/01 4EC-3022/02 4EC-3022/03 4EC-3022/04	These DCPs installed quick disconnects for temperature switches in the Emergency Diesel Generators. The installation of the quick disconnects will improve the maintainability of the temperature switches. The temperature switches provide an alarm and have no control function. There is no change in control circuitry or setpoints of any instruments that are important to safety. Therefore, these DCPs do not involve any Unreviewed Safety Questions.
4EC-3104/01	This DCP provided Control Room overhead annunciation of Main Turbine and Feedwater Turbine sensor failure alarms that are fed from the sensors that input into the two out of three trip logic for the turbines. The sensors currently feed computer points only. The plant computer Control Room Integrated Display System and the annunciator system are not Class 1E and do not perform any safety-related functions. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Safety Evaluation
4EC-3285/01	This DCP installed a double o-ring gask-o-seal on the discharge flange of a pressure relief valve in the Core Spray system. The installation of a double o-ring gask-o-seal disc on the discharge flange does not affect the function of the relief valve. The disc seal is a passive component whose primary safety function is to mitigate the consequences of an accident by allowing Type B leak testing in lieu of Type A Integrated Leak Rate Testing following valve maintenance. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3316/01	This DCP relocated the Radiation Monitoring console from near the Control Room back panel into the Control Room "horseshoe". Moving the Radiation Monitoring console does not involve any functional change. The console is not essential for safe shutdown of the plant and serves no active emergency function during an accident. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3343/01 4EC-3343/03	These DCPs removed environmental seals on various transmitters and replaced them with environmentally qualified quick disconnects. The quick disconnects help to minimize stay time in the Radiological Control Area. These DCPs did not change the design function or the qualification of the system. Therefore, they did not involve any Unreviewed Safety Questions.
4EC-3374/01	This DCP replaced control rods. The nuclear life of the replaced control rods would have expired prior to the next refueling outage. The nuclear and mechanical design of the new control rods is equal to or exceeds the design requirements of the original equipment; therefore, this DCP does not involve any Unreviewed Safety Questions.

DR	Description of Deficiency Report
HTE 92-010	This DR addresses the installation of schedule 40 pipe instead of schedule 80 pipe at several Station Service Water 1 inch and 1 ½ inch root valve lines. Analysis of the lines indicate that the schedule 40 pipe could withstand the design basis earthquake. Also, if the schedule 40 pipe failed, the Safety Auxiliaries Cooling System Heat Exchanger Room is equipped with flooding alarms that indicate in the Control Room. The pipe was replaced with schedule 80 pipe during the 4th Refueling Outage. Therefore, the use of schedule 40 pipe did not involve any Unreviewed Safety Questions.
HTE 92-160	This DR addresses Scram Outlet Valves in the Control Rod Drive system that were inadvertently rebuilt with the incorrect seat material. This DR justifies the continued operation of these valves for two fuel cycles. Analysis shows that the incorrect seat material would remain in acceptable condition for 144 scrams involving extreme conditions. Therefore, this DR does not involve any Unreviewed Safety Questions.
HQA 92-203	This DR documents the failure a material supplier to adhere to the testing requirements of ASME Section II for a blind flange installed in the 'A' Station Service Water Common Supply Header. This DR allowed the flange to be used-as-is temporarily, but required its replacement prior to the end of the 4th Refueling Outage. The material tests required by ASME Section II have shown that the chemical composition of the blind flange is acceptable. The deficiency identified concerns the failure to perform the mechanical tests on a test specimen that was heat treated along with the finished product. A test sample was analyzed that came from the same heat code and heat number as the finished product; however, this sample was heat treated separately from the finished blind flange. The test results from this test sample were acceptable. Therefore, this DR does not involve any Unreviewed Safety Questions.

Procedure Revision	Description of Safety Evaluation
HC.IC-GP.SF-0001(Q) Rev 5	This procedure revision provides guidance that allows Control Rod withdrawal when the core is off-loaded. The guidance includes steps to jumper the Rod Position Information System input to the Reactor Manual Control System and to jumper the low power setpoint contacts to the Rod Sequence Control System. The Reactor Manual Control System, Refueling Interlocks, and the Rod Position Indication System are part of the controls that limit Control Rod motion during refueling. The signal to the Safety Parameters Display System provides Control Rod information to the operator. The margin of safety is to avoid an inadvertent criticality. This procedure is used only when there is no fuel in the core; therefore this procedure does not involve an Unreviewed Safety Question.
HC.MD-FR.KE-0003(Q) Rev 8	This procedure revision includes a temporary change to allow the removal of the Reactor Pressure Vessel Head Insulation package using slings and the Auxiliary Hook of the Polar Crane. The UFSAR indicates that the Reactor Pressure Vessel Strongback is used to lift the insulation package. Failure of one of the support slings or one of the Polar Crane redundant load wires would retain the load. Ample capacity and redundancy were specified, the consequences of a single failure using slings and the Auxiliary Hook of the Polar Crane are the same as the previous system. Therefore, this procedure does not involve an Unreviewed Safety Question.

Procedure Revision	Description of Safety Evaluation
HC.MD-GP.ZZ-0099(Z) Rev 0	This new procedure eliminates the need for a temporary modification when either Service Air Compressor is out of service during electrical bus and Turbine Auxiliaries Cooling System outages. It describes the steps required to install and remove temporary compressors. The Instrument Air system has no safety-related functions other than the integrity of the piping through the containment penetration. Failure of the system will not compromise any safety-related system or component or prevent a safe shutdown of the plant. Therefore, this procedure does not involve an Unreviewed Safety Question.
HC.OP-GP.PB-0002(Q) Rev 0	This new procedure establishes guidelines for the removal and return to service of the 4.16KV vital bus and provides direction for the installation of temporary power when maintenance is to be performed during outages. Malfunctions in the non-1E system (temporary power) is totally isolated from, and has no effect on, class 1E equipment that is vital to the safe shutdown of the plant. This procedure will only be implemented with the plant in Operating Conditions 4, 5, or *, when only two of the four vital channels are required to be operable. This procedure will only be implemented when channel 'B' is not required to be operable. Therefore, this procedure does not involve an Unreviewed Safety Question.

Procedure Revision	Description of Safety Evaluation
HC.OP-GP.PB-0003(Q) Rev 0	This new procedure establishes guidelines for the removal and return to service of the 4.16KV vital bus and provides direction for the installation of temporary power when maintenance is to be performed during outages. Malfunctions in the non-1E system (temporary power) is totally isolated from, and has no effect on, class 1E equipment that is vital to the safe shutdown of the plant. This procedure will only be implemented with the plant in Operating Conditions 4, 5, or *, when only two of the four vital channels are required to be operable. This procedure will only be implemented when channel 'C' is not required to be operable. Therefore, this procedure does not involve an Unreviewed Safety Question.
HC.SA-AP.ZZ-0049(Q) Rev 6	This procedure revision deletes HC.SA-AP.ZZ-0049(Q), which has been superseded by NC.NA-AP.ZZ-0049(Q). The UFSAR states that station administrative procedures provide station wide direction in areas that are common to all station departments. NC.NA-AP.ZZ-0049(Q) includes the majority of the administrative controls and department responsibilities previously contained in HC.SA-AP.ZZ-0049(Q). The remaining administrative controls and department responsibilities have been included in department level procedures. Transferring administrative controls and department responsibilities from one procedure to another does not impact the probability or consequences of any type of accident. Therefore, this procedure does not involve an Unreviewed Safety Question.

Procedure Revision	Description of Safety Evaluation
NC.NA-AP.ZZ-0025(Q) Rev 1	This procedure revision modifies the storage and use of combustible materials and changes the method for completing the transient combustible evaluation. This procedure does not change the evaluations made of the fire protection equipment or change the criteria or assumptions used to develop the Fire Hazard Analysis. It introduces steps for the prevention of fire and minimizes the impact of fire on the station. Therefore, this procedure does not involve an Unreviewed Safety Question.
NC.NA-AP.ZZ-0050(Q) Rev 1	This procedure revision enhances the Station Testing Program by the inclusion of guidance for the testing of major components after painting of movable parts, linkage, shafts, and springs. It also provides specific guidelines for testing of motor operated valves. This revision does not change any previously analyzed testing requirement nor does it change any testing method. Therefore, this procedure does not involve an Unreviewed Safety Question.