PSE&G

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

Hope Creek Generating Station

February 14, 1991

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 January are being forwarded to you with the summary of changes, tests, and experiments for January 1991 pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

J. J. Hagan
General Manager -
Hope Creek Operations

RAR:ld
Attachments

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INDEX

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 2/14/91
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH January 1991

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 2/14/91
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period January 1991 Gross Hours in Report Period 744

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	0.0	29,781.5
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	0.0	0.0	29,293.1
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	0	0	92,542,408
10.	Gross electrical energy generated (MWH)	0	0	30,621,673
11.	Net electrical energy generated (MWH)	0	0	29,248,396
12.	Reactor service factor	0.0	0.0	82.5
13.	Reactor availability factor	0.0	0.0	82.5
14.	Unit service factor	0.0	0.0	81.2
15.	Unit availability factor	0.0	0.0	81.2
16.	Unit capacity factor (using MDC)	0.0	0.0	78.6
17.	Unit capacity factor (Using Design MWe)	0.0	0.0	75.9
18.	Unit forced outage rate	0.0	0.0	5.5

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    None
20. If shutdown at end of report period, estimated date of start-up:
    2/13/91

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 2/14/91
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH January 1991

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
1	1/1	S	744	C	4	3rd Refueling Outage

Summary

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek
DATE 2/14/91
COMPLETED BY S. Hollingsworth
TELEPHONE (609) 339-1051

MONTH January 1991

1. Refueling information has changed from last month:

   Yes No X

2. Scheduled date for next refueling: 12/26/90

3. Scheduled date for restart following refueling: 02/13/91

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 No X

   If no, when is it scheduled? not currently scheduled

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

6. Important licensing considerations associated with refueling:

   - Amendment 34 to the Hope Creek Tech Specs allows the cycle specific operating limits to be incorporated into the CORE OPERATING LIMITS REPORT; a submittal is therefore not required.

7. Number of Fuel Assemblies:

A.	Incore	764
B.	In Spent Fuel Storage (prior to refueling)	496
C.	In Spent Fuel Storage (after refueling)	760

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: July 22, 2007

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

JANUARY 1991

At the beginning of February, Hope Creek remained shutdown for the third refueling outage. The outage continued throughout the month.

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

JANUARY 1991

The following Design Change Packages (DCP's) 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 DCP's did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. The DCP's did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

DCP	Description of Design Change Package
4EC-3008	This DCP added rigging attachment points to allow for easier removal and replacement of valves and actuators. The valves and actuators addressed in this DCP are in the Containment Atmosphere Control System.
4EC-3039	This DCP upgraded the closed circuit television system that is used to observe the packaging and drum handling portions of the Solid Radwaste System. The DCP will enhance operation of this system through additional/upgraded cameras.
4EC-3130/01	This DCP installed two monorails to service the Reactor Water Cleanup System pumps and motors in the Reactor Building Reactor Water Cleanup Recirculation Pump Rooms. The monorails will enhance pump maintenance by providing the capability to lift either the pumps or the motors.
4EC-3148	This DCP installed a tide flex check valve at the end of each of the two Storm Drainage Outfall pipes. The check valves will prevent river water and sediment from flowing into the Storm Drainage System.
4EC-3149/02	This DCP added rigging attachment points to allow for easier removal and replacement of valves and actuators. The valves and actuators addressed in this DCP are in the Torus Water Cleanup, Safety Auxiliaries Cooling, and Containment Atmosphere Control Systems.
4HC-0195/04	This DCP replaced the inboard and outboard Containment Isolation Valves for the process sampling line off of the Nuclear Boiler System. The replacement valves have a longer life expectancy.
4HC-0212/02	This DCP upgraded the "B" Chilled Water Pump by replacing the motor and impeller. This upgrade will allow the Chilled Water System to operate with 2 pumps running and the other in standby.
4HC-0212/03	This DCP upgraded the "C" Chilled Water Pump by replacing the motor and impeller. This upgrade will allow the Chilled Water System to operate with 2 pumps running and the other in standby.

DCP	Description of Design Change Package
4HC-0212/04	This DCP installed two stainless steel pipe tie-ins to the Chilled Water System, one on the suction header of the Chilled Water Pumps and one on the discharge header. These tie-ins support the future installation of a mechanical filtering unit and a mixed bed demineralizer to cleanse the system of soluble impurities that reduce performance and equipment life.
4HC-0238/01	This DCP modified the Safety and Turbine Auxiliaries Cooling System piping and instrument tubing for the flow measurement used to monitor the Safety Auxiliaries Cooling System to Turbine Auxiliaries Cooling System flow. This DCP will reduce the flow measurement errors that have resulted in spurious actuations.

The following Temporary Modification Requests (TMR's) 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 TMR's did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. The TMR's did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

TMR	Description of Temporary Modification Request
90-053	This TMR installed a power cable to provide a temporary power source to the 'A' Reactor Protection System Power Distribution Panel during the maintenance outage on the 4160 Volt Turbine Building Switchgear.
90-0 7	This TMR installed a power cable to provide temporary power to a 120 Volt Miscellaneous Instrument Power Supply during the maintenance outage on the 'A' 4160 Volt Class 1E Switchgear.
90-058	This TMR installed a power cable to provide temporary power to the Public Address System Power Supply during the maintenance outage on the 'A' 4160 Volt Class 1E Switchgear.
90-062	This TMR installed a power cable to provide temporary power to a 125 Volt Class 1E Battery Charger during the maintenance outage on the 'A' 4160 Volt Class 1E Switchgear.
90-063	This TMR installed a power cable to provide temporary power to the 'D' Class 1E 120VAC NSSS Computer Power Supply during the maintenance outage on the 'D' 4160 Volt Class 1E Switchgear.
90-064	This TMR installed a power cable to provide temporary power to a 24 Volt Battery Charger during the maintenance outage on the 'D' 4160 Volt Class 1E Switchgear.
90-065	This TMR installed a power cable to provide temporary power to a 'D' Class 1E 125 Volt Battery Charger during the maintenance outage on the 'D' 4160 Volt Class 1E Switchgear.
90-066	This TMR installed a power cable to provide temporary power to a 'D' Class 1E 125 Volt Battery Charger during the maintenance outage on the 'D' 4160 Volt Class 1E Switchgear.
91-001	This TMR added a connection to a temporary Air Compressor in the discharge line of the Service Air Compressor Aftercooler. The temporary compressors will be used to supply Service Air while the Service Air Compressors are out of service for a maintenance outage.

The following Deficiency Reports (DR's) 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 DR's did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. The DR's did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

DR	Description of Deficiency Report
HTE-90-082	This DR addresses the overtorquing of Fuel Channel Fastener bolts. The procedure states that the bolts should be torqued to 75 ± 5 inch-pounds. The torque wrench that was used on 58 fuel bundles was found to exhibit a torque of 81.37 inch-pounds. The DR authorizes the fuel bundles to be "used-as-is".
HMD-91-004 HMT-91-016 HMD-91-021	These three DR's were evaluated together. They address weld cracks on a hanger in the Recirculation System and linear indications on the outside surface of the weld between the Recirculation Loop Discharge pipe and the Residual Heat Removal return tee in both loops. The Safety Evaluations associated with these DR's state that the Hope Creek principal safety barriers were not seriously degraded and that the Recirculation System may be "used-as-is" while the plant is shutdown and operating in Residual Heat Removal Shutdown Cooling.
HMD-91-027	This DR addresses a Main Steam Drain Valve that has a seat contact area of less than 100%. The valve cannot be replaced at this time because of the unavailability of a spare valve. The drain valve may be "used-as-is" because there is another drain valve and a pipe cap in series, which meet the intent of the double-valve design.
RNR MC91-0032	This Receiving Nonconformance Report identifies a nonconformance with purchase specifications. The affected component is a butterfly valve for the Service Water System. The purchase order specified a leakage rate of 0 cc/min, which cannot be achieved by this valve. The valve may be "used-as-is" because the leakage rate is minimal.

The following procedure revisions 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 procedure revisions did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. The procedure revisions did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Procedure Revision	Description of Procedure Revision
HC.OP-IS.BC-0002(Q) Rev. 7	This Safety Evaluation addresses the use of temporary Measurement & Test Equipment instead of using permanently installed instrumentation as stated in the UFSAR. The use of the temporary Measurement & Test Equipment is consistent with ASME Section XI and Hope Creek's Inservice Test program.
HC.OP-SO.SE-0001(Q) Rev. 2	This procedure revision adds a section to defeat the downscale Rod Blocks when all fuel is removed from the Reactor Vessel. Rod Blocks are not required when there is no fuel in the Reactor Vessel.
HC.OP-SO.SF-0001(Q) Rev. 2	This procedure revision allows the Scram Discharge Volume High Level Rod Block to be bypassed when all fuel is removed from the Reactor Vessel. Rod Blocks are not required when there is no fuel in the Reactor Vessel.