PSEG

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

Hope Creek Generating Station

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

Sincerely yours,

J. J. Hagan
General Manager -
Hope Creek Operations

RAR:ld
Attachments

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AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 6/10/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH May 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	1054	17.	1050
2.	1042	18.	1049*
3.	1043	19.	1049*
4.	1044	20.	1057
5.	1059	21.	1048
6.	1049	22.	1040
7.	1053	23.	1036
8.	1072	24.	1036
9.	1052	25.	1056
10.	942	26.	831
11.	1055	27.	0
12.	1054	28.	0
13.	1043	29.	0
14.	1045	30.	0
15.	1053	31.	116
16.	1048		

* Due to an error in recording the meter readings, the exact average daily power levels for May 18 and 19 are unknown. The listed averages represent the average of the two-day total.

OPERATING DATA REPORT

DOCKET NO.	50-354
UNIT	Hope Creek
DATE	6/10/92
COMPLETED BY	V. Zabielski
TELEPHONE	(609) 339-3506

OPERATING STATUS

1. Reporting Period May 1992 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	647.7	3329.5	40,490.8
6. Reactor reserve shutdown hours	0.0	0.0	0.0
7. Hours generator on line	632.1	3285.3	39,860.9
8. Unit reserve shutdown hours	0.0	0.0	0.0
9. Gross thermal energy generated (MWH)	2,043,069	10,536,961	126,534,104
10. Gross electrical energy generated (MWH)	678,860	3,521,880	41,374,374
11. Net electrical energy generated	647,860	3,365,951	40,017,500
12. Reactor service factor	87.1	91.3	84.8
13. Reactor availability factor	87.1	91.3	84.8
14. Unit service factor	85.0	90.1	83.5
15. Unit availability factor	85.0	90.1	83.5
16. Unit capacity factor (using MDC)	84.5	89.5	81.3
17. Unit capacity factor (Using Design MWe)	81.6	86.5	78.5
18. Unit forced outage rate	15.0	3.3	5.0

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Refueling outage, 9/12/92, 60 days

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

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 6/10/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH May 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
4	5/26	F	111.9	A	1 & 2	Failed Drywell to Suppression Chamber Decay test: power was reduced to 21% and the Reactor was manually scrammed LER 354/92-006

Summary

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek
DATE 6/10/92
COMPLETED BY S. Hollingsworth
TELEPHONE (609) 339-1051

MONTH May 1992

1. Refueling information has changed from last month:

   Yes 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 No X

   If no, when is it scheduled? not scheduled (on or prior to 7/24/92)

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

May 1992

Hope Creek entered the month of May at approximately 100% power. A reactor shutdown was commenced at 1503 on May 26 per the requirements of Tech Spec 3.6.1.1 because the Drywell to Suppression Chamber Decay Test failed to meet its acceptance criteria. To comply with the Action Statement, a manual scram was initiated at 2213 with reactor power at 21%. The unit was brought back on line on May 31.

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

MAY 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.

TMR	Description of Safety Evaluation
92-012	This TMR installed electrical jumpers across the High Bearing Oil Temperature Trip Switch. This jumper permits the 'A' Control Room Chiller to run with a defective module and/or thermistor until a replacement part can be installed. The Control Area Chilled Water System is comprised of two 100% capacity redundant loops; therefore, the loss of any single component cannot result in a loss of cooling. Also, jumpering the trip circuit and providing for increased operator attention to oil temperature does not place the equipment in any additional jeopardy. Therefore, this TMR does not involve any Unreviewed Safety Questions.
92-013	This TMR installed electrical jumpers across the Feedwater Heater's High High Level Trip Switches. These switches cause spurious high level trip signals during low power levels due to inleakage in the reference leg. The jumpers are only required until the level signals stabilize. The Feedwater system is not safety related and is not required to be operable following a LOCA, other than for containment isolation. Failure of the Feedwater system does not compromise any safety related system or components. This TMR has no impact on the containment isolation function of the Feedwater system. Therefore, this TMR does not involve any Unreviewed Safety Questions.
92-014	This TMR removed the overload heaters from the breakers for the Reactor Water Cleanup Discharge to Condenser Valve and the Reactor Water Cleanup Discharge to Equipment Drain Valve. Removing the overload heaters from the breakers will prevent the valves from inadvertently opening during an Appendix R fire. Disabling these valves, along with the overhead annunciator, does not prevent their associated systems from performing their designed functions. Also, the UFSAR discusses the Appendix R requirement that the valves be disabled. Therefore, this TMR does not involve any Unreviewed Safety Questions.

Procedure Revision	Description of Safety Evaluation
HC.OP-GP.ZZ-0001(Q) Rev 0	This new procedure eliminates the possibility of a Residual Heat Removal Shutdown Cooling Isolation due to a loss of Reactor Protection System power by defeating the automatic isolation signals to the Shutdown Cooling Suction Isolation Valves and the Shutdown Cooling Return to Reactor Pressure Vessel Valves. This procedure will be used only during refueling with the Reactor cavity flooded, the Fuel Pool to Reactor cavity gates removed, and with management approval. The Shutdown Cooling mode of the Residual Heat Removal System is designed to be controlled by the operator from the Control Room. The design basis for the most limiting single failure is that Shutdown Cooling can be established by manual action. This procedure retains the ability to manually isolate the Shutdown Cooling Suction Isolation Valves. Elimination of the automatic isolation capability does not jeopardize the functional design basis of the Shutdown Cooling mode of the Residual Heat Removal System; therefore, there are no Unreviewed Safety Questions associated with this new procedure.