PSE&G

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

Hope Creek Generating Station

April 14, 1994

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 March are being forwarded to you with the summary of changes, tests, and experiments that were implemented during March 1994 pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

R. J. Hovey
General Manager -
Hope Creek Operations

DR:WS:JC
Attachments

C Distribution

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The Energy People

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OPERATING DATA REPORT

DOCKET NO. 50-354
UNIT Hope Creek
DATE 04/08/94
COMPLETED BY V. Zabielski VZ
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period March 1994 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	99.6	1515.6	54338.6
6. Reactor reserve shutdown hours	0.0	0.0	0.0
7. Hours generator on line	97.5	1513.5	53456.0
8. Unit reserve shutdown hours	0.0	0.0	0.0
9. Gross thermal energy generated (MWH)	293011	4941193	170904563
10. Gross electrical energy generated (MWH)	98980	1666820	56630774
11. Net electrical energy generated (MWH)	87330	1594181	54121865
12. Reactor service factor	13.4	70.2	85.1
13. Reactor availability factor	13.4	70.2	85.1
14. Unit service factor	13.1	70.1	83.9
15. Unit availability factor	13.1	70.1	83.9
16. Unit capacity factor (using MDC)	11.4	71.6	82.3
17. Unit capacity factor (Using Design MWe)	11.0	69.2	79.5
18. Unit forced outage rate	0.0	0.0	4.3

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

20. If shutdown at end of report period, estimated date of start-up:
4/23/94

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 04/08/94
COMPLETED BY V. Zabielski VCZ
TELEPHONE (609) 339-3506

MONTH March 1994

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	3/5	S	646.5	C	1 then 2	Power was reduced to approximately 10%, then Rx was manually SCRAMed to start 5th refueling outage.

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 4/11/94
COMPLETED BY V. Zabielski VCZ
TELEPHONE (609) 339-3506

MONTH March 1994

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	1056	17.	0
2.	1051	18.	0
3.	1045	19.	0
4.	784	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		

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek 1
DATE April 11, 1994
COMPLETED BY V. Zabielski VZ
TELEPHONE (609) 339-3506

MONTH March 1994

1. Refueling information has changed from last month:

   Yes X No

2. Scheduled date for next refueling: 9/16/95[a]

3. Scheduled date for restart following refueling: 10/31/95[a]

4. A. 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?

   Yes X[b] 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:
   N/A

7. Number of Fuel Assemblies:[b]

A. Incore	764
B. In Spent Fuel Storage (prior to refueling)	1008
C. In Spent Fuel Storage (after refueling)	1240

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 offload)
   (Assumes 244 bundle reloads every 18 months until then)
   (Does not allow for smaller reloads due to improved fuel)

NOTES: (a) RFO5 currently in progress. This data refers to RFO6.
(b) This data refers to CYL6.

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

March 1994

Hope Creek entered the month of March at approximately 98% power. The unit was ramping down for a scheduled refueling outage which started on March 5, 1994. As of March 4, the plant had been on line for 88 consecutive days.

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

March 1994

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	Summary of Safety Evaluation
4HC-0220	This Safety Evaluation discusses the design change that replaces eight butterfly valves with valves of an improved design. The new valves utilize a metal to metal seat design which will improve the seating capabilities of the valves over the current soft seats. The installation of the valves will require that Figure 9.2-5 sheet 1 of the UFSAR be revised to reflect the installation of these valves. The design change will not change the Safety Auxiliary Cooling System (SAC's) description, Design Bases, or Safety Evaluations described in Sect 9.2.2.1 and 9.2.2.3 of the UFSAR. Therefore, this DCP does not increase the probability or consequences of an accident previously described in the SAR and does not involve any Unreviewed Safety Question.
4EC-3343	This Safety Evaluation discusses a design change that removes the Conax environmental seals on various Rosemount transmitters and replaces them with qualified Namco quick disconnects. This DCP does not in any way affect or change the design function of the transmitter circuits. The change in type of environmental seal is considered a change to the facility. However, the SAR description of each of the systems impacted by this DCP is not detailed to the level that would require a change. Therefore, this DCP does not increase the probability or consequences of an accident previously described in the SAR and does not involve any Unreviewed Safety Question.
4EC-3410	This Safety Evaluation discusses the installation of the REM-Light Wet Transfer Equipment, consisting of the Service Pole Caddy, Main Steam Line Plug Assembly, and Dryer/Separator Sling Assembly and Hook Boot. It demonstrates that they meet or exceed the safety requirements imposed on the original equipment and therefore does not adversely impact the plant structures, equipment or systems.

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

4EX-3441

This Safety Evaluation addresses procedural guidance for operation in natural circulation based upon GE's analysis. Since the present analysis applies only to the fifth refueling outage, procedural guidance will be provided by temporary operations procedure THC.OP-IO.ZZ-0010(Q) "Natural Circulation Decay Heat Removal". Later when the scope of the analysis is broadened, the procedure will be revised and then made permanent.

Natural circulation will be initiated when the reactor has been shutdown for at least 10 days. Fuel Pool Cooling and Cleaning and Reactor Water Cleanup are in operation. These in combination with natural circulation provides sufficient heat removal capacity to maintain average reactor coolant temperature less then 140°F. Throughout the duration of natural circulation operation at least one RHR Shutdown Cooling Loop and/or at least one recirculation Pump will be available to be put in operation.

Additionally, this evaluation discusses the Test DCP 4EX-03441 which is being conducted, concurrently with natural circulation operation. It will include test instrumentation to monitor reactor coolant temperature at various locations in the reactor vessel, reactor cavity, and spent fuel pool. This will directly monitor reactor coolant temperature at the inlet and outlet of the core. The readings will be compared to indications from permanently installed instrumentation, thus providing highly refined correlations between average coolant temperature and process temperatures.

Each UFSAR section in which natural circulation is discussed was reviewed to determine the impact that would be caused by operation in natural circulation and performance of the proposed test. Based on that review it was concluded that the performance of natural circulation operation and the proposed test will not reduce the margin of safety.

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

4HE-0070 This safety evaluation discusses the installation of a Portafab Building for snubber testing activities related to both Hope Creek and Salem In-Service Inspection Snubber Testing Programs. The location of the facility is at the north end of the turbine building 137 elevation. UFSAR Figure 9.5-5 required updating to show this installation.

This enclosure is located in the Unit Two side of Hope Creek (Cancelled Area). The UFSAR does not discuss the effects of the Cancelled Plant Area adjacent to Safety Related Areas. Section 3.7.2.9.1 of the UFSAR addresses interaction of non-seismic category 1 structures with seismic category 1 structures and does state that structural separation be provided.

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

Procedure	Summary of Safety Evaluation
HCP.8-0002	This Safety Evaluation discusses a revision of a plant procedure VHC.RE.FR.ZZ-0011(Q) "Vacuum Sipping of BWR Fuel Assemblies", to perform fuel inspection using a sipping process. The first time this fuel inspection technique was used was RFO3. During RFO3 the spent fuel pool was partially racked for spent fuel storage. The pool is now fully racked. The structure (spent fuel pool slab system) has been evaluated for the increased load resulting from the sipping operations and it has been determined that the design fuel load exceeds the actual fuel loads, therefore the fuel pool sipping will not compromise the pool slab. The sipping equipment is not seismically qualified, however consequences of the sipping equipment failure during a seismic event is bounded by that of the UFSAR Fuel Handling Accident Analyses. Installation of the sipping equipment and fuel handling during the sipping campaign will be governed by existing plant procedure. Therefore, this procedure revision does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.
HC.RA-IS.ZZ-0008	This Safety Evaluation addresses the procedure for the Primary Containment Integrated Leak Rate Test (CILRT). The procedure provides detailed instruction to achieve conditions required for the performance of the CILRT. The procedure performs temporary changes to the facility and includes restoration steps. Independent verification of each temporary change and subsequent restoration is required. Therefore, this Procedure revision does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.

NC.NA-AP.ZZ-0023

This safety Evaluation discusses the Nuclear Administrative Procedure to control "Scaffolding and Transient Loads".

Changes to the procedure include:

Entering all scaffold into the Scaffold Control Log. Previously only scaffolds erected in safety related areas were logged.

Seismic Restraint requirements can be waived provided that a Limited Condition of Operability (LCO) has been entered for the associated equipment.

Requirements for a pre-startup walkdown which involves an inspection of the safety related areas of the plant prior to entering into mode 2.

Incorporating supplementary guidelines for control and restraint of transient loads on the refueling floor (Reactor Building 201 elv.).

Increasing the clearances for horizontal (side to side) clearances from 1 to 3 inches depending on equipment to 4 inches for all equipment.

All other changes to the procedure involved procedural enhancements and editorial changes.

Review of the administrative controls of the program has determined that the proposed changes will not cause an increase to the consequences of a previously evaluated accident as described in the SAR. Although changes have been made to relax the seismic restraint requirements, these changes apply only in specific applications, and are implemented under Limited Condition of Operation (LCO) established and controlled by the Operations Department.

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

HC.MD-GP.ZZ-099

This Safety Evaluation discusses changes to the procedure "Installation of Temporary Air Compressors". This change allows for the temporary compressors to act as a backup to the station permanent compressors, therefore increasing the reliability of the system. Since the procedure requires the temporary compressors to meet the air quality standards of the permanent compressors there are no additional variables introduced. The loss of the temporary compressors would have the same effect as the loss of the permanent compressors, which is already evaluated.

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

Other	Summary of Safety Evaluation
SAR-CN-94-04	This Safety Evaluation discusses the revision to SAR Figure 9.1-32 sht 3 of 13 which shows the laydown area for the spent fuel pool slot plugs on the refueling floor (201 elv.). This change allows the placement of the plugs at any location on the east side of the refuel floor within the cross-hatched area. This revision is supported by an engineering calculation of the allowable floor loading. The analysis of the slot plugs as shown in the table 9.1-12 is unchanged. The margin of safety for the fuel pool slot plugs remains as shown in table 9.1-12 note (d). Therefore, this SAR Change Notice does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.
UFSAR CN-94-02	This Safety Evaluation discusses the lifting of the RPV Head Insulation Package using a sling arrangement and the Polar Crane Auxiliary Hook as an alternative method to the present described procedure of using the RPV Head strongback and the Polar Crane Main Hook. The Polar Crane is the specified equipment for lifting the Insulation Package and as such it has been analyzed for malfunction. This lift with either the Main or Auxiliary Hook is within the analysis of the crane and therefore is no increase to the consequences of malfunction. Therefore, this UFSAR Change Notice does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.

Temporary Modification	Summary of Safety Evaluation
94-003	This Safety Evaluation discusses a Temporary Modification which modifies the discharge path for the Drywell Floor Drain sump 1B-T-267 and Drywell Equipment Drain Sump 1A-T-267 during refueling outage operations. The Normal discharge path will not be available due to outage activities (LLRT, VOTES, Surveillances and potential repairs). The T-Mod will remove spool pieces to allow temporary hose to be connected and run to the Torus through a nearby downcomer. This T-Mod will also jumper out the closed signal from containment isolation valves 1HBHV-F003/4 and F019/020 to the sump controls. This T-Mod will not affect the level instrumentation or alarms associated with the Drywell Floor and Equipment Drain Sumps. The sump pump level controls will start and stop the pumps as designed. Therefore, this Temporary Modification does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.
94-004	This Safety Evaluation discusses a Temporary Modification to 1BBLT-N027-B021 which provides the control room indication of shutdown level 0-400 inches. The new transmitter and indicator scale will provide 0-550 inch range. Tech Specs requires the level in the vessel to be above 483.5 inches while moving fuel. The transmitter output is directed only to the control room indication and to a local alarm in panel 10C214 on 201'elv. in the Reactor Building. No Safety system actuation is associated with this instrument. Therefore, this Temporary Modification does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.

Deficiency Report	Summary of Safety Evaluation
HMD-94-023	This Safety Evaluation discusses a Deficiency Report which dispositions certain internal check valve parts which were found missing when the valve was opened per the Check Valve Inspection Program (SOER 86-03). The disposition is for restoration of the components to an operable status thru replacement or modification of parts and analysis of the system operation (RHR) with the lost parts. The basis for the determination is a General Electric Report titled" Lost Parts Analysis of Check Valve Parts for Hope Creek Generating Station". The GE Analysis addressed the potential for fuel bundle flow blockage, fuel damage, and operational restrictions. Therefore, this Deficiency Report does not increase the probability or consequences of an accident previously described in the SAR and does not involve an Unreviewed Safety Question.