PSE&G

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

Nuclear Business Unit

FEB 13 1997

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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 1997** are being forwarded to you with the summary of changes, tests, and experiments that were implemented during **December 1996** pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

for MBB

Mark B. Bezilla
General Manager -
Hope Creek Operations

I/I

IE24

RP:LK:DS
Attachments

C Distribution

INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 02/10/97
COMPLETED BY: R. Phillips
TELEPHONE: (609) 339-2735

OPERATING DATA REPORT
OPERATING STATUS

1. Reporting Period January 1997 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	744.0	744.0	74467.1
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	744.0	744.0	73304.0
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	2410149	2410149	234278396
10.	Gross electrical energy generated (MWH)	820610	820610	77714823
11.	Net electrical energy generated (MWH)	789173	789173	74260892
12.	Reactor service factor	100.0	100.0	84.0
13.	Reactor availability factor	100.0	100.0	84.0
14.	Unit service factor	100.0	100.0	82.6
15.	Unit availability factor	100.0	100.0	82.6
16.	Unit capacity factor (using MDC)	102.9	102.9	81.2
17.	Unit capacity factor (using Design MWe)	99.4	99.4	78.5
18.	Unit forced outage rate	0.0	0.0	4.6

19. Shutdowns scheduled over next 6 months (type, date, & duration):

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

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OPERATING DATA REPORT
UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH JANUARY 1997

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
n/a						

DOCKET NO.: 50-354
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AVERAGE DAILY UNIT POWER LEVEL

MONTH JANUARY 1997

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	1067	17	1070
2	1062	18	1067
3	1056	19	1067
4	934	20	1063
5	1053	21	1068
6	1067	22	1063
7	1059	23	1061
8	1072	24	1066
9	1066	25	1058
10	1066	26	1070
11	1066	27	1064
12	1069	28	1060
13	1071	29	1070
14	1064	30	1065
15	1073	31	1069
16	1061		

DOCKET NO.: 50-354
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DATE: 01/10/97
COMPLETED BY: L. Kepley
TELEPHONE: (609) 339-1106

REFUELING INFORMATION

MONTH JANUARY 1997

1. Refueling information has changed from last month:

   Yes _ No X

2. Scheduled date for next refueling (RF07): 9/6/97

3. Scheduled date for restart following refueling: 11/5/97

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? To Be Determined for Cycle 8 COLR

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	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
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MONTHLY OPERATING SUMMARY

MONTH JANUARY 1997

- The Hope Creek Generating Station remained on-line for the entire month and operated at 100% power for the month of January 1997. There was one load reduction which is identified below.
- On January 4, 1997, at 0013 hours power was reduced to perform Control Rod Swap and monthly turbine valve testing. The unit was returned to 100% power at 2138 hours. The 24 hour average net power level was 87.5% (Design Electrical Rating).
- At the end of the month the unit had been on-line for 85 days.

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

MONTH JANUARY 1996

The following items completed during **December 1996** 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.

Design Changes Summary of Safety Evaluations

Replacement of Air-Operated Valves in the Safety Auxiliaries Cooling System (SACS), 4EC-3612, Pkgs 1 and 3. The Standby Diesel Generator (SDG) room cooler water supply valves in the SACS were replaced. The previous valves were carbon steel air actuated flexible wedge gate valves. The replacement valves are stainless steel air actuated ball valves which are more suitable to the application. UFSAR Figure 9.2-5 requires a change to show the type and material of the replacement valves and actuators. UFSAR Table 3.9-18 is being changed to show the valve/actuator replacement type. This replacement does not change the function of the valves or the function of the affected systems. The change does not alter the ability of the SACS, the SDG Room Recirculation System, or the SDGs to perform their intended safety function. The affected systems will function in accordance with the original design and licensing basis of the plant.

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

Replacement of Air-Operated Valves in the Safety Auxiliaries Cooling System (SACS), 4EC-3612, Pkg 32. The Core Spray (CS) pump room cooler water supply valves in the SACS were replaced. The previous valves were carbon steel air actuated flexible-wedge gate valves. The replacement valves are stainless steel air actuated ball valves which are more suitable to the application. UFSAR Figure 9.2-4 requires a change to show the type and material of the replacement valves/actuators. UFSAR Table 3.9-18

must be changed to show the valve/actuator replacement type. This replacement does not change the function of the valves or the function of the affected systems. The change does not alter the ability of the SACS, the Equipment Area Cooling System (EACS), or the CS pump room unit coolers to perform their intended safety functions. The affected systems will function in accordance with the original design and licensing basis of the plant.

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

Installation of Land Vehicle Barrier System, 5EC-3035, Pkg 1. This design change incorporates the vehicle barrier system into the Security Plan, drawings which are part of the Security Plan, and procedures which implement the Security Plan. The changes are related to operation, inspection, compensatory measures, etc. of the vehicle barrier system, as applicable. Amendment to Title 10CFR73.55(c)(7) and Regulatory Guide 5.68 requires licensees to establish vehicle control measures, including vehicle barriers, to protect against the use of a land vehicle as a means of transportation to gain unauthorized proximity to vital areas or to transport a bomb. As required by 10CFR73.55(c)(9), a formal Summary Description of vehicle control measures has been submitted to the NRC. The NRC and it's consultant, the Army Corps of Engineers, have reviewed the submittal and the analysis contained therein, and have accepted the submittal.

There are no anticipated operational transients or postulated design basis accidents previously evaluated which are considered applicable to the security system. The following design basis security threats and design basis events are considered applicable: assault design basis security threat, land vehicle design basis security threat, site drainage related to the probable maximum precipitation (due to the passive barriers), and tornado generated missiles. The passive barriers will not have any adverse impact on the existing site drainage conditions. The vehicle barrier system does not create any new tornado missiles that can adversely affect Category I structures or vital areas. The vehicle barrier system protects against the new land vehicle design basis security threat and does not degrade the existing security systems or interact with other systems. The security procedures address compensatory measures related to security system malfunctions.

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

Nuclear Services Building Construction, 5EC-3033, Pkg 1. This design change provides utility tie-ins for the Nuclear Services Building (NSB) constructed in the yard area between Salem and Hope Creek Generating Stations. The utilities are being provided from existing non-safety related systems. The utilities are: fire water and fire alarms, potable water, sanitary sewer, power, telecommunications, paging, yard lighting, storm water drain, and instrument irradiation facility. The design changes do not change the basic function, design basis or design criteria of the affected structures, systems or components. However, changes to tables and diagrams in the UFSAR are required. The NSB is functionally and physically separated from all existing systems, structures, and components which could affect reactor safety or be used to mitigate the consequences of previously analyzed accidents. In addition, there is no involvement with any existing radwaste treatment systems or safe shutdown fire protection systems. There are no adverse consequences of this facility's failure or system tie-in failure on the existing safety

related systems or structures due to design basis events such as wind, tornado, floods, or accidents involving nearby industrial transportation, and military facilities. Modifications to these systems for tie-ins are designed to the system's original criteria and will function as the original systems. There will be no change to these system's function or failure effects; the normal operation of or failure of any of the modifications to these systems will not create any new impact on or involvement with any accident previously evaluated. A potential fire in the NSB will not impact the safe shutdown capability of the plant due to the distance of the NSB to the safety related equipment/systems. The NSB could not fall on any safety related structure, nor jeopardize its structural integrity or safe shutdown capability. The NSB does not impact site flooding, since it does not present any new source of significant flood water. No new floating missiles concern is generated by the addition of the NSB. The underground fire protection distribution and fire alarm systems do not provide a safety related function and their use or failure would not increase the consequences of a malfunction of equipment important to safety beyond that previously evaluated. The tie-in with the plant fire protection main, potable water, sanitary sewer, telecommunications, electrical power, plant page and fire alarms systems do not interface with any safety related systems.

Therefore, this design 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 Notices   Summary of Safety Evaluations

Update Final Safety Analysis Report (UFSAR) Change to Reflect Hydrogen Water Chemistry (HWC) Source Terms, CN # 96-31. The UFSAR change updates the source term tables and the radiation zoning figure to reflect the increase in Nitrogen 16 (N-16) concentrations due to HWC operation at 22.5 scfm of hydrogen. There is no adverse impact to equipment due to operation of the HWC system or the associated increase in nitrogen source terms. There are no adverse equipment qualification impacts associated with the increase in radiation dose rates. The change will not degrade the original design basis for any safety related system. There are no important to safety systems or equipment that are affected by this change. Neither plant personnel nor the health and safety of the public are at risk when operating the HWC system at 22.5 scfm of hydrogen.

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.

Temporary Modifications   Summary of Safety Evaluations

Temporary Modification 96-036, Leak Repair of High Pressure Coolant Injection (HPCI)/Main Steam Drain Line, 1-AB-028-DBD-2". This temporary modification installed a leak repair clamp and temporary hanger on a through wall leak on line 1-AB-028-DBD-2" located in the torus room. The line is a non-safety related HPCI/Reactor Core Injection Cooling (RCIC) drain line which provides a flow path to the condenser during the standby mode of HPCI/RCIC. During operation, the flow path of the line is isolated and the steam side flow is to the respective system's barometric condenser. The temporary modification does not alter or change the function or design of the systems as described in the UFSAR. The clamp is designed to meet the design requirements of the

existing line. The temporary modification does not change the operating parameters associated with the Main Steam, HPCI, or RCIC systems.

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

Procedures Summary of Safety Evaluations

HC.OP-AB.ZZ-0148(Q) Rev 3, "Turbine Auxiliaries Cooling System (TACS) Malfunction". This procedure revision changes the operator actions in HC.OP-AB.ZZ-0148(Q) to allow the operators to fail open 1EGHV-2522E&F, the Safety Auxiliary Cooling System to TACS isolation valves, if no large TACS break has been identified. The 1EGHV-2522E&F valves isolate SACS from TACS in case of a double guillotine shear of the TACS piping. This is beyond the criteria required for moderate energy piping by the Branch Technical Position APCSB 3-1 and MEB 3-1. The isolation of SACS from TACS during design basis accidents is accomplished via the 1EGHV-2522A-D valves. If the 1EGHV-2522E&F valves are failed open in accordance with the procedure revision, they would not shut if there were a TACS pipe break as described in the UFSAR. The revision to the procedure is not changing any of the actions required for isolation of the safety related portions of the system from the non-safety related portions of the system. The procedure change is providing guidance to the operators on how to recover the TACS system if no large TACS pipe break has occurred to assure that no condition exists which would challenge a safety function of the system (i.e. isolation of SACS from TACS). The SACS system will continue to function as designed. The credible failure mode associated with this change is the original design accident where the non-seismic piping may shear during a design basis Loss of Power/Loss of Coolant Accident (LOP/LOCA). During this accident, the isolation of SACS from TACS would be performed by the 1EGHV-2522A-D valves. The closure time of these valves is sufficient because during this accident, there is no power available to the SACS pumps for the first 45 seconds and the valves shut in 22 seconds.

Therefore, this procedure 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 Evaluation

Offsite Dose Calculation Manual (ODCM) Rev 15. The changes to the ODCM are editorial in nature and will not effect any systems, structures, or components that are important to safety. The changes made to the ODCM update the document to reflect current operating parameters and programs. The changes will serve to enhance the ODCM as a reference document for the Hope Creek Generating Station. The changes to the ODCM will not reduce the accuracy or reliability of any radiation monitor setpoint, dose calculation methodology, or Technical Specification limitation. The changes will not alter the original design or accident analyses that were performed as part of the licensing basis.

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

Deficiency Reports Summary of Safety Evaluations

There were no changes in this category implemented during December 1996.

It was discovered that the following procedure revisions were omitted from previous monthly operating reports.

NC.NA-AP.ZZ-0036(Q), "Control of Information System and Telecommunications Services" Rev. 2. This revision involved a change to a Nuclear Administrative Procedure and did not involve any physical changes to the facility. This change did not concern plant operation or accidents in the UFSAR.

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

NC.NA-AP.ZZ-0061(Q) "Significant Event Response Team Management", Rev. 2. The changes associated with this revision primarily served to simplify the previously approved procedure and include the following: a) modified the purpose of the procedure to provide guidance for SERT Managers for completing independent assessments; b) reduced the level of detail; c) identified a SERT position of Information Coordinator; and d) added an attachment listing SERT action items to be considered. This procedure controlled the SERT process. It did not direct the operation of equipment, either safety related or non safety related and did not affect any equipment or process controlled by Technical Specifications.

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

This procedure has since been eliminated.

NC.NA-AP.ZZ-0027(Q), "Inservice Inspection Program", Rev. 1. The revision summary is as follows: 1) deleted requirement for SORC review of ISI program submittals to the NRC; 2) made editorial changes to reflect the organization changes; and 3) reformatted the procedure.

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

NC.NA-AP.ZZ-0024(Q), "Radiation Protection Program", Rev. 5. This is a major revision of the procedure which consolidates the following procedures into one procedure: 1) NC.NA-AP.ZZ-0007(Q) Rev. 2 - ALARA Program; 2) NC.NA-AP.ZZ-0024(Q) Rev. 4 - Radiation Protection Program; and 3) NC.NA-AP.ZZ-0029(Q) Rev. 2 - Radioactive Material Control Program. These changes do not relate to design criteria, specifications, operation of the fuel cladding, RCS boundary, or containment, and do not concern any margin of safety as defined in the Bases of the Technical Specifications.

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