PSEG

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

Nuclear Business Unit

October 12, 1995

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 **September 1995** are being forwarded to you with the summary of changes, tests, and experiments that were implemented during **September 1995** pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

Mark Reddemann
General Manager -
Hope Creek Operations

DL:RS:JC
Attachments

C Distribution

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DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 10/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

AVERAGE DAILY UNIT POWER LEVEL

MONTH SEPTEMBER 1995

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	1019	17	1010
2	1031	18	1039
3	1002	19	1040
4	1026	20	1036
5	1025	21	1004
6	1026	22	1031
7	1023	23	1048
8	1027	24	1039
9	1029	25	1041
10	1026	26	1038
11	1045	27	1040
12	1035	28	1038
13	1024	29	1041
14	1023	30	1037
15	1030	31	N/A
16	1043		

DOCKET NO.: 50-354
UNIT: Hope Creek
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OPERATING DATA REPORT
OPERATING STATUS

1. Reporting Period September 1995 Gross Hours in Report Period 720.

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	720.0	6001.3	65937.2
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	720.0	5951.6	64955.0
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	2362169	19357876	207772222
10.	Gross electrical energy generated (MWH)	774976	6403991	68831657
11.	Net electrical energy generated (MWH)	742281	6128221	65781537
12.	Reactor service factor	100.0	91.6	85.7
13.	Reactor availability factor	100.0	91.6	85.7
14.	Unit service factor	100.0	90.9	84.4
15.	Unit availability factor	100.0	90.9	84.4
16.	Unit capacity factor (using MDC)	100.0	90.7	82.9
17.	Unit capacity factor (using Design MWe)	96.6	87.7	80.1
18.	Unit forced outage rate	0.0	9.1	5.2

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Refueling Outage, November 11, 1995, 40 days (Under Review)

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

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OPERATING DATA REPORT
UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH SEPTEMBER 1995

NO.	DATE	TYPE F=FORCED S=SCHEDULE	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
1.		NONE				

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REFUELING INFORMATION

MONTH SEPTEMBER 1995

1. Refueling information has changed from last month:

   Yes _X_ No ___

2. Scheduled date for next refueling: 11/11/95

3. Scheduled date for restart following refueling: 12/20/95 (Under Review)

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? October 25, 1995

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 (prior to refueling)	1240
C. In Spent Fuel Storage (after refueling)	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)

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MONTHLY OPERATING SUMMARY

MONTH SEPTEMBER 1995

The Hope Creek Generating Station remained on-line for the entire month and operated at essentially 100% power from September 1, 1995 until 0237 hours on September 28, 1995 when the planned coastdown at the end of the cycle began. The following is a summary of events and activities that caused minor (<20%) deviations in megawatt output during the month of September 1995:

- Four planned power reductions for turbine valve testing occurred this month. During the power reduction on September 3, 1995, maintenance on the "A" Reactor Feed pump was performed.
- On September 7 and 22, 1995 reductions in electrical output occurred because of problems with feedwater heater dump valves.
- Power was reduced on September 20 and 21, 1995 for replacement and subsequent testing of a scram solenoid on control rod 46-11.

At the end of the month the unit had been on-line for 68 days.

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

MONTH SEPTEMBER 1995

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.

Deficiency Reports Summary of Safety Evaluations

- There were no changes, tests, or experiments in this category this month.

Design Change Summary of Safety Evaluations

- **4EC-3087 - MODIFICATION OF AIR SUPPLY TO DAMPERS IN TRUCK BAY**
  This modification installed a stainless steel air cylinder and associated hardware. This cylinder is continuously charged with instrument air. This air will be used to maintain Secondary Containment Integrity by keeping dampers 1GUHD-9450 A & B shut in the event of an instrument air failure with the truck bay door open. UFSAR Figure 9.4-5 will be updated to show this change.

  This modification does not change descriptions of the facility, its structures, components or systems contained in the UFSAR but simply enhances the operability of the dampers by installing a back-up air supply. The dampers will no longer be functionally dependent on the availability of the instrument air system. There are no safety-related equipment or components that could be a target for missiles, therefore II/I criterion is not a concern.

  Therefore, this DCP 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-SO.HC-0007(R), REV 2 - SOLID RADWASTE OPERATION WITH TEMPORARY SOLIDIFICATION/DEWATERING/ & FILTRATION SYSTEMS** This revision gives instructions for use of a portable liquid Radwaste treatment skid. It provides directions for use of the Radwaste Building truck bay connections for temporary filtration equipment for liquid wastes. All wastes processed by this temporary system will be contained in the Radwaste system sumps and tanks. UFSAR Section 11.4.2.6 states that "permanent flanged connections are provided on the south wall of the SWMS truck bay to enable processing of concentrates, filter media, waste sludge and/or resin slurries by a portable dewatering/ solidification system." This procedure directs the use of these connections for filter processing of liquid waste. UFSAR Change Notice 95-33 has been developed to revise the Process Control Program and the UFSAR.

  A design review to ensure compliance with Regulatory Guide 1.143 has been performed and the procedure found acceptable. No plant operating parameters are affected by this change. Monitoring of the equipment while it is in service will ensure that any liquid leakage would be isolated quickly using local isolation valves. Any leakage would be contained in the truck bay. When processing is not in progress the system will be isolated and vented. This will prevent siphoning of liquids from the tank. This proposal does not involve changes to control of releases, compromise sampling for liquid releases nor affect the consequences of an unexpected or uncontrolled release.

  The solid and liquid radwaste processing systems have no safety-related functions and do not directly communicate mechanically or electrically with any safety-related systems directly. None of the anticipated or postulated design bases accidents are applicable to this proposal. The ability of the floor drains in the Radwaste Building truck bay and flooding levels analyzed for these rooms are unaffected by this revision.

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

- **SP-7, REVISION 7 - PERSONNEL ACCESS CONTROL** and **SP-9, REVISION 6 - CONTROL OF PACKAGES AND MATERIAL** These revisions will result in a change to the Salem-Hope Creek Security Plan. They are being made because a Final Rule published in the Federal Register, Volume 60, No. 173 deleted the requirement of 10CFR73.55(d)(8) for controlling personnel and material access to the primary reactor containment from within an adjacent vital area during periods of high traffic such as major maintenance and refueling outages.

  These changes affect only security processes by eliminating an administrative control in accordance with NRC regulations. No plant equipment or parameters are affected. Because of this scope, there are no associated credible failure modes and there can be no malfunctions or changes in frequency or consequences of malfunctions of equipment important to safety caused by this revision. Persons and materials are searched upon entry to the protected area. Access is controlled through vital area portals into the Hope Creek Reactor Building. The Security Program is designed to prevent purposeful acts of radiological sabotage. There is no Security accident analysis. Therefore, no operational transients, postulated design bases accidents can be affected as a result of this revision. And there can be no increase in the probability or consequences of an accident as a result of this change.

  Therefore, this procedure revision 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 Evaluations

- **HCR.8-0001, REVISION 0 - HOPE CREEK RELOAD SAFETY EVALUATION FOR CYCLE 6 EXTENSION** The purpose of this safety evaluation is to evaluate the operation of HCGS Cycle 6 beyond its previously reviewed and approved Reload Design Basis. Operation beyond the original effective full power capability will be accomplished by means of a power coast down. Once full power capability is lost, the coast down is expected to proceed at approximately 3% rated thermal power per week to approximately 83% power at the end of the coast down if no load management is initiated. Operation during this time will not require any new or different functions from structures, systems, or components or the need for new procedures. No new changes in the operating practices are required due to the power coast down.

  At PSE&G's request, an analysis for the period of extended operation was performed by General Electric using NRC approved methods. The analysis has been used as part of the evaluations to verify that all specified fuel design limits will be met during the coast down period. The analysis results have been reviewed and accepted as part of the design engineering evaluations of the coast down. No changes to systems, structures or components described in the UFSAR will be made, therefore no new failure modes are introduced and UFSAR existing failure modes remain applicable. No procedure revisions are required. The analysis indicated the need to implement a new more restrictive MCPR LCO than the current one. This limit will be incorporated into the NSSS computer in accordance with existing procedures.

  All the anticipated operational transients (AOOs) and postulated accidents were reviewed for being potentially affected by cycle extension. The consequences of events that could lead to challenging shutdown margin requirements or stand by liquid control system capability are less restrictive than those analyzed previously for Cycle 6 due to the core being in a less reactive state during coastdown.

  Since no new functions or conditions are required of systems, structures or components during the coast down period, the probability of having an accident previously evaluated in the UFSAR is unaffected. No new initiating events are introduced by coast down operation. Only two AOOs, Turbine trip without bypass and Feedwater controller failure, were identified for analysis to quantify plant response during the coast down. Both AOOs were evaluated at end of coast down exposure and power as abounding point. The results of the analysis were acceptable. Therefore, no increase in consequence is incurred and no modifications to MAPLHGR or LHGR Tech Specs are required.

  Therefore, operation of HCGS Cycle 6 beyond its previously reviewed and approved Reload design basis will not increase the probability or consequences of an accident previously described in the UFSAR and does not involve an Unreviewed Safety Question.

<u>Other Summary of Safety Evaluations (continued)</u>

- **<u>UFSAR CHANGE NOTICE CN 89 - 07, INCORPORATION OF DCP 4EC-1055</u>** This change notice evaluates and justifies fire load changes and room descriptions to UFSAR Section 9A resulting from DCPs 4EC-1055 and 4EC-1002. This change adds a new room, 4410A, incorporates fire loading from a cable tray and combustible loading information for rooms 4326, 4328, 4333, 4410, 4408, and 4410A into the appropriate sections of the UFSAR. These DCPs increased the fire loading in Fire Area RB1 from 21.7 to 22.0 minutes and in Fire Area RB5 from 24.1 to 24.2 minutes. Both of these are well below the 1 hour and three hour fire barriers encompassing the areas. Since the fire load increase from the modifications does not raise the total fire load above the fire barrier rating for these areas, the increased fire loads do not create a credible for the fire area boundaries.

  With fire loads remaining within the design parameters of the fire barriers the probability and consequences of an accident (fire), and malfunctions of equipment as previously evaluated in the UFSAR remains unchanged. The Fire Hazard Analysis does not assess fire risk in terms of likelihood but rather on the basis that a fire will occur, and that damage to equipment important to safety in that area happens. Thus, the fire load increases in areas RB1 and RB5 do not affect the basis of the Fire hazards Analysis. Since the fire load in RB1 and RB5 remains less than one hour, fire barrier integrity is maintained and the probability and consequences of occurrence of malfunction of equipment due to fire spread beyond the area is unchanged.

  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.

- **<u>UFSAR CHANGE NOTICE CN 89 - 001, FIRE PROTECTION OPERATIONAL TESTING AND INSPECTION REQUIREMENTS</u>** This change notice clarifies in Section 9.5.1.4.2 that the National Fire Protection(NFPA) Codes are used as guidance rather than as strict standards for periodic inspections and tests of fire protection equipment and systems. Inspections and tests are performed in accordance with the Fire Protection Program. The test and inspection intervals recommended by the NFPA codes may be varied because of equipment inaccessibility or performance history. Changes to the program do require a 50.59 review. Inaccessibility of equipment was recognized in the original Fire Protection Technical Specifications. The UFSAR already explicitly recognizes that NFPA standards and codes are used as guidelines for testing of fire protection equipment in non-safety related areas of the plant.

  Per Generic Letter 86-10, "accident" is a postulated fire. This change only clarifies the standards involved with inspection and testing requirements. Inspection and testing deals with mitigating a fire once it has started. Therefore, this change does not increase the probability or consequences of an accident previously evaluated in the UFSAR. Fire Protection equipment is neither safety-related nor important to safety. Therefore, there are no malfunctions of equipment important to safety that were previously evaluated in the UFSAR applicable to this change. Because of the nature of the change and the classification of the system the probability and consequences of a malfunction of equipment important to safety are unaffected. This change does not affect the analysis of the consequences of a fire.

  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.

Other Summary of Safety Evaluations (continued)

- **UFSAR CHANGE NOTICE CN H91 - 010, CORRECTIONS AND UPDATE OF FHA, AREA AB2** This change notice updates Table 9A-15, " Fire Hazards Analysis Tabulation" for Fire Area AB2 to be consistent with the description provided in Section 9A.6.3, "Electrical Access Area Division II, Fire Area AB2," Table 9A-4, "Fire Areas and Associated Room Numbers," and Calculation ELEC-22.

  With fire loads remaining within the design parameters of the fire barriers the probability and consequences of an accident (fire), and malfunctions of equipment as previously evaluated in the UFSAR remains unchanged. The Fire Hazard Analysis does not assess fire risk in terms of likelihood but rather on the basis that a fire will occur, and that damage to equipment important to safety in that area happens. Since the variations in calculated fire loads were not found to exceed the fire protection features currently described, e.g. design rating of the barriers and suppression systems, the changes do not create a credible failure mode for the fire area.

  Per Generic Letter 86-10, "accident" is a postulated fire. UFSAR Chapter 15 does not include operating transients or design basis events that deal with Fire Hazards Analysis. Since the fire is already assumed to occur this change cannot increase the probability of a fire. Since the variations in calculated fire loads were not found to exceed the fire protection features currently described if a fire were to occur, it would be limited to the fire area of origin and as such not increase the consequences of the fire. Per the General Criteria for conducting Fire hazards Analysis all components affected by the postulated fire are assumed to be inoperable or mis-operate, whichever is worse. Therefore, malfunctions of the components affected by the postulated fire as previously evaluated are considered applicable to this change. Since the fire is already assumed to occur and all equipment affected, this change cannot increase the probability of occurrence or consequences of a malfunction of equipment important to safety nor can it create an accident or malfunction of a different type in the affected area.

  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.

- **UFSAR CHANGE NOTICE 92-22, UPDATE UFSAR TABLE 7.5-1** This change notice updates UFSAR Table 7.5-1 to be consistent with Engineering Documents as identified in Human Engineering Discrepancy Report T127 and DCP 4EC-1082. This change will not have any affect on the plant since the change in UFSAR Table 7.5-1 corrects erroneous information and ensures the table reflects UFSAR Figure 6.2-29. The information being changed is consistent with the Technical Specifications and the design basis of the Post Accident Monitoring Instrumentation as provided to meet Regulatory Guide 1.97.

  No systems or parameters are affected by the change. Therefore, there are no new credible failure modes. No physical changes to the plant are being made. Correcting erroneous information (typos) in this table has no affect on the probability or consequences of an accident or malfunction of equipment important to safety.

  These changes to correct errors and make the table consistent with Engineering Documents and UFSAR Figure 6.2-29 has no affect on the design basis of the plant. 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.

Other Summary of Safety Evaluations (continued)

- **UFSAR CHANGE NOTICE 95-031, PART 1 - CLARIFICATIONS AND UPDATE OF SECTION 6.2.4, CONTAINMENT ISOLATION SYSTEM, AND 6.2.6, PRIMARY CONTAINMENT LEAK RATE TESTING** This change notice clarifies portions of the primary containment design and containment leak rate testing requirements by reclassifying the piping and flanges in the HPCI and RCIC Turbine exhaust vacuum breaker network associated with Containment penetration P204 and making editorial changes in Section 6.2. Licensee Event Report (LER) 89-013 identified the existing lack of specific guidance in the UFSAR on leak test requirements for the flanges on penetration 204. There is no change in Primary Containment design or leak rate testing requirements.

  This revision is for clarification and editorial corrections in Containment design and Containment Leak Rate Sections of the UFSAR. Therefore, there are no malfunctions of equipment important to safety previously evaluated in the UFSAR applicable, nor are there changes in the probability or consequences of malfunctions previously evaluated. No plant parameters are affected.

  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.

- **UFSAR CHANGE NOTICE 95-031, PART 2 - INCORPORATION OF HOPE CREEK TECHNICAL SPECIFICATION AMENDMENT 76** This change notice eliminates Type C Leak Rate tests for Containment Isolation Valves (CIVs) in certain primary Containment penetrations that terminate below the minimum water level in the Torus. By testing these valves in accordance with the Hope Creek Inservice Test (IST) program and considering the water seal created by the 30 day water supply of the Torus compliance with the containment leakage criteria of 10CFR50, Appendix J and the off-site dose rate requirements of 10CFR100 is assured. IST will be performed in accordance with the ASME Boiler & Pressure Vessel Code, Section XI - Division I. The specific valves affected by this change were listed in Amendment 76 to the Hope Creek Technical Specifications which was reviewed and approved by the NRC and transmitted to PSE&G on August 1, 1995.

  The elimination of Type C Leak Rate testing does not increase the probability of exceeding the containment leakage criteria of 10CFR50, Appendix J. The Torus is designed to be filled with water during the postulated design basis accident and for 30 days post-accident, therefore the CIVs will remain water sealed during these times. This prevents the primary reactor containment atmosphere from impinging on the CIVs and precludes leakage out of the containment. Additionally, in accordance with Sections III.C.2 and III.C.3 of Appendix J, these CIVs need not be tested with air. Further, it is not necessary to test them with water, as the purpose of the water leak rate test is to assure a supply of sealing water for 30 days following the onset of an accident. As the Torus is postulated to always remain filled with water, no leak rate test is necessary to satisfy 10CFR50, Appendix J requirements. CIV operability is maintained by the Hope Creek IST program and leak tightness is maintained by the water seal from the Torus. compliance with General Design Criteria is maintained.

  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.

Other Summary of Safety Evaluations (continued)

- **UFSAR CHANGE NOTICE 95-23 - SECTION 17.2-1 A.1(a)** This change notice deletes the Security program Administrative Procedures from the Q-list in UFSAR Table 17.2.1, and reference to Regulatory Guide 1.17, Protection of Nuclear Plants Against Industrial Sabotage, which has been revoked by the NRC from UFSAR Section 17.2.2. There are no physical changes. Regulatory Guide 1.33, Appendix A cites Security and Visitor Control as activities that should be covered by written administrative procedures. There is no reference to the security system. Quality requirements applicable to Security are contained in 10CFR73.55(g)(4) and UFSAR Section 13.6, the Security Plan. Therefore, Q-listing of the Security Program is redundant. Its removal from the Q-list will have no effect upon the quality or QA oversight of the Security Program.

  No plant parameters are affected. The Security System is designed to prevent purposeful acts of radiological sabotage. There is no security accident analysis. Because, this proposal does apply to any plant equipment, there is no effect on malfunctions, probability or consequences, or types of failures of equipment important to safety.

  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 Modification Summary of Safety Evaluation

- **94-023 - RECYCLE LOOP FOR GEZIP SKID** This Temporary Modification installed a recycle loop to the GEZIP skid in accordance with General Electric recommendations to improve reliability of the system. The recycle loop uses a recirc pump in place of an injection pump. Other minor control circuit, tubing and pump changes were, also, performed. This change affects only the GEZIP system and has no operational impact on any other plant systems, structures or components.

  This change could cause a failure of the GEZIP system, or leaks could develop at the new fittings. This system is not required for normal plant operation or for safe shutdown of the unit. If problems occur the system will be shutdown, the problems corrected and the system restarted when conditions permit. Leakage would be handled by the floor drains. Leakage form the GEZIP was analyzed with the initial installation of the system. There are anticipated operational transients or postulated design bases accidents previously evaluated in the UFSAR applicable to this proposal.

  Therefore, 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.