PSEG

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

Nuclear Business Unit

December 14, 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 **November 1995** are being forwarded to you with the summary of changes, tests, and experiments that were implemented during **November 1995** pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

Mark Reddemann
General Manager -
Hope Creek Operations

CLC

DL:DS:CC
Attachments

C Distribution

# INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 12/6/95
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

# AVERAGE DAILY UNIT POWER LEVEL

**MONTH NOVEMBER 1995**

| DAY | AVERAGE DAILY POWER LEVEL (MWe-Net) | DAY | AVERAGE DAILY POWER LEVEL (MWe-Net) |
|---|---|---|---|
| 1 | 931 | 17 | 0 |
| 2 | 925 | 18 | 0 |
| 3 | 913 | 19 | 0 |
| 4 | 931 | 20 | 0 |
| 5 | 928 | 21 | 0 |
| 6 | 920 | 22 | 0 |
| 7 | 919 | 23 | 0 |
| 8 | 910 | 24 | 0 |
| 9 | 912 | 25 | 0 |
| 10 | 733 | 26 | 0 |
| 11 | 0 | 27 | 0 |
| 12 | 0 | 28 | 0 |
| 13 | 0 | 29 | 0 |
| 14 | 0 | 30 | 0 |
| 15 | 0 | 31 | N/A |
| 16 | 0 | | |

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# OPERATING DATA REPORT
# OPERATING STATUS

1. Reporting Period November 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 | 241.7 | 6988.0 | 66923.9 |
| 6. | Reactor reserve shutdown hours | 0.0 | 0.0 | 0.0 |
| 7. | Hours generator on line | 241.6 | 6938.2 | 65941.6 |
| 8. | Unit reserve shutdown hours | 0.0 | 0.0 | 0.0 |
| 9. | Gross thermal energy generated (MWH) | 671780 | 22359904 | 210774249 |
| 10. | Gross electrical energy generated (MWH) | 226989 | 7397956 | 69825622 |
| 11. | Net electrical energy generated (MWH) | 210606 | 7072276 | 66725592 |
| 12. | Reactor service factor | 33.6 | 87.2 | 85.3 |
| 13. | Reactor availability factor | 33.6 | 87.2 | 85.3 |
| 14. | Unit service factor | 33.6 | 86.6 | 84.1 |
| 15. | Unit availability factor | 33.6 | 86.6 | 84.1 |
| 16. | Unit capacity factor (using MDC) | 28.4 | 85.6 | 82.5 |
| 17. | Unit capacity factor (using Design MWe) | 27.4 | 82.7 | 79.7 |
| 18. | Unit forced outage rate | 0.0 | 8.0 | 5.1 |

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Currently shutdown for Refueling Outage, RF06, began November 11, 1995

20. If shutdown at end of report period, estimated date of start-up:
    Start Up currently scheduled for February 6, 1996

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# OPERATING DATA REPORT
# UNIT SHUTDOWNS AND POWER REDUCTIONS

**MONTH NOVEMBER 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. | 11/11/95 | S | 478.4 This outage is still in progress. | C | 1 then 2 | Power was reduced to approximately 10%. The turbine was taken off-line, and then a manual scram inserted to start the 6th Refueling Outage |

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# REFUELING INFORMATION

**MONTH NOVEMBER 1995**

1. Refueling information has changed from last month:

   Yes X No __

2. Scheduled date for next refueling: 3/28/97 (a)

3. Scheduled date for restart following refueling: 5/28/97 (a)

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 (a)

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: (b)

| | |
|---|---|
| A. Incore (prior to current refueling outage) | 764 |
| B. In Spent Fuel Storage (prior to current refueling) | 1240 |
| C. In Spent Fuel Storage (after current 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)

**NOTES:**

(a) RF06 currently in progress. Dates are projected for RF07

(b) This data is for Cycle 7, Numbers will be revised at end of current refueling outage.

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# MONTHLY OPERATING SUMMARY

**MONTH NOVEMBER 1995**

The Hope Creek Generating Station began the month operating at 89.2%. The end of cycle coastdown continued until November 11, 1995 when the unit was taken off-line to begin the Sixth Refueling Outage. As of November 11, 1995 the unit had been on-line for 109 days.

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

**MONTH NOVEMBER 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.

## Design Changes Summary of Safety Evaluations

- **4EC-3317, STRAINER LIFT MODIFICATIONS** This design change installs structural steel components in the B/D Bay of the Service Water Intake Structure (SWIS) to enable separate rigging of Service Water strainer items. UFSAR Section 9.1.5 will be revised to describe and evaluate the new trolley beams. There are no new credible failure modes introduced by this change. The only credible failure is a heavy load drop and it is evaluated in UFSAR Section 9.1.5.3.3. The strainer items being lifted weigh less than the heaviest load lifted in the SWIS. Since the larger load drop has been evaluated for the UFSAR and found acceptable, a strainer load drop would, also, be acceptable. Therefore, this change did not increase the consequences of an accident previously evaluated in the UFSAR.

  This change will not degrade the performance of the Service Water system or increase challenges to the system function. This is due to the use of Seismic II/I criteria of rigging steel, the fact the new loads are lighter than previously analyzed loads, and the use of original design specifications and construction practices.

  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.

**Procedures** **Summary of Safety Evaluations**

- **HC.NA-AP.ZZ-0024(Q), REV 5, RADIATION PROTECTION PROGRAM** 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. This revision implements a commitment in letter NLR N94202 to the NRC, Reply to Notice of Violation, Inspection Report 50-354/94-20, Section F. Re-examination of Root Cause Investigation, Corrective Action 1, "The existing requirements contained in upper tier radiation protection related administrative procedures will be consolidated into one procedure, and the remaining procedure will be strengthened with additional guidance (on removal and release of contaminated equipment from the RCA)." These changes do relate to design criteria, specifications, operation of equipment important to safety, the fuel cladding, RCS boundary, or containment, and do not address any margin of safety.

  NC.NA-AP.ZZ-0007(Q) is specifically mentioned in UFSAR Sections 12.1.3.1.1, 12.1.3.5, 12.5.1.1, 12.5.3.8. SAR Change Notice 94-38 is currently in preparation to provide a comprehensive update to the Hope Creek Radiation Protection Program. The incorporation of NC.NA-AP.ZZ-0007(Q) into NC.NA-AP.ZZ(Q)-0024 does not change the intent of any of the UFSAR sections.

  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.

- **HC.OP-SO.BC-0001(Q), REV 20, RESIDUAL HEAT REMOVAL SYSTEM OPERATION** This revision provides an alternate method of vessel makeup when the Control Rod Drive (CRD) is not in operation and Shutdown Cooling is in operation. Two manual isolation valves between the Condensate Storage and Transfer system (AP) and shutdown Cooling pump suction will be opened to provide a flow path. Water will be added to the vessel using the normal shutdown cooling return path. An operator will be stationed at the valves, in communication with the Control Room, during the evolution in case prompt action is needed.

  A review of UFSAR Chapter 15 reveals that reactor coolant inventory increase, decrease and loss of shutdown cooling are the only applicable accidents. An increase in inventory is the desired result of the operation. Uncontrolled increase in inventory has no adverse consequences to the public and main steam line flooding is not credible because of MSIV configuration. The only credible method to lose shutdown cooling because of this change is to draw air into the system due to either a pipe break or draw down of AP. This is not likely because the suction pressure of the RHR pumps is greater than atmospheric pressure. Decrease in inventory can be postulated by rupture of the Reactor Coolant pressure boundary which could be extended to the AP system when the valves are open. Both the AP and RHR systems are rated at 150 psig. Any malfunction of either system would result in the operator closing either or both of the manual valves. If this action does not occur, the RHR shutdown cooling valves operate on a low vessel level signal and LPCI and Core Spray are designed to keep the core covered and protect the public.

  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.

## UFSAR Change Notices    Summary of Safety Evaluations

- **UFSAR CHANGE NOTICE CN 95-48, ELECTRICAL ISOLATION SYSTEM ENVIRONMENTAL CONDITIONS REFERENCE CHANGES** This change notice changes UFSAR Section 7.1.2.5.2, Electrical Isolation, to state the actual environmental conditions associated with the description of the devices. The UFSAR description states that the listed environmental conditions are the worst case whereas they are actually the normal conditions. The parameters will be changed as follows:

  | PARAMETER | CURRENT CONDITIONS | NEW CONDITIONS |
  |---|---|---|
  | Control Equipment Rooms Relative Humidity | 20% - 50% | 20% - 90% |
  | Main Control Room Relative Humidity | 40% - 50% | 20% - 60% |
  | Main Control Room Temperature Range | 74F - 78F | 66F - 78F |

  These values were extracted from the original design bases documents. These design parameters are within the normal operating specifications for plant equipment. The normal and accident modes of operation of plant HVAC systems will not be altered by this change. The electrical isolation devices described in UFSAR Section 7.1.2.5.2 were tested under the conditions stated. This change incorporates stated values consistent with the original design and within the operating parameters for associated plant equipment. The failure modes of the affected equipment have not changed and the redundant and diverse configuration of the associated 1E equipment is not altered.

  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 CN 95-49, TEMPERATURE AND HUMIDITY EXCURSIONS IN THE MAIN CONTROL ROOM (#5510) AND ASSOCIATED AREAS** This change notice revises UFSAR Section 9.4.1.1.1 to state a temperature range of 72F +/- 6F and a humidity range of 20% to 60% for the nominal expected environmental conditioned in the control room area. These values accurately reflect the requirements for normal plant operation and will not introduce any new effects to the plant or system operation. This change does not affect the operating parameters of the Control Room Supply (CRS) system or any control room associated equipment. Equipment performance is not altered by the proposed changes. Operation at the new environmental conditions will not result in exceeding the limits for the control room emergency filtration (CREF) system charcoal absorbers.

  These changes to the parameters associated with a system designed to mitigate the consequences of an accident are within the normal operating specifications of the equipment. No changes are being made to the design basis of the CRS.

  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 Notices Summary of Safety Evaluations (continued)

- **UFSAR CHANGE NOTICE CN 95-50, UFSAR SECTION 9.1, FUEL STORAGE AND HANDLING UPDATE** This change notice makes several administrative changes to UFSAR Section 9.1 concerning fuel storage and handling. These changes include:

  - The specified radiation level on the refueling bridge
  - The rigging methods for the nut rack to reference another UFSAR Section
  - Lifting of the fuel pool gates and fuel transfer chute
  - Correction of typographical error on number of NUREG-0612 on Table 9.1-14

  All parameters and systems affected are in the Fuel Handling and Reactor Servicing System (KE). No plant parameters are directly affected by this change. There are no process parameters affected by the proposal. The proposal updates descriptions of equipment used on the refuel floor and in the fuel pool.

  Previously analyzed failure modes are still applicable. These include the fuel drop accident analysis, UFSAR 15.7.4, loss of fuel pool inventory, UFSAR 9.1.2, and load drop analysis, UFSAR 9.1.4.3. The boundaries prescribed for the existing credible failure modes are not affected by the change. The change does not initiate physical change to any equipment important to safety nor are functions of equipment important to safety 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.

## Temporary Modifications Summary of Safety Evaluations

- **TM# 95-063, HOPE CREEK OUTAGE TELEPHONE INSTALLATION** This temporary modification will install cable runs in excess of 200 feet that are not installed in conduit. This is a change to the facility as UFSAR Section 9.5.2.3 describes telephone cable as being installed in rigid conduit. This telephone cable will be used to improve communications during the refueling outage. No operational transients are applicable because no transients are initiated by the telephone system. The Fire Hazard Analysis states that cable initiated fires are not credible. However, since the cable insulation is flammable it will be considered a transient combustible and will be controlled in accordance with NC.NA-AP.ZZ-0025(Q).

  This temporary modification will not affect any safety related equipment, radiation monitoring equipment, or radioactive waste systems. It will be physically separated from those systems and cables related to those systems during installation, use, and removal. The telephone cable is plenum rated and low smoking. A partial of complete loss of the telephone system is addressed in Section 9.5.2.3 of the UFSAR. Loss of AC to the telephone system is addressed in Section 15.4.3 of the UFSAR.

  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.

- **TM# 95-065, RECONFIGURATION OF SHUTDOWN RANGE LEVEL** This temporary modification replaces Shutdown Range Reactor Level detector, 1BBLT-N027-B21, with a 0-100 psig Rosemount Model 1153 pressure transmitter. 1BBLT-N027-B021 has a maximum indication when normally configured of 372 inches. The reactor cavity level when fully flooded during refueling is 492 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 for indication and to a local alarm in panel 10C214 on 201' elevation. 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 UFSAR and does not involve an Unreviewed Safety Question.

- **TM# 95-049, INSTALL SINGLE JUMPER FOR A TRAVELING WATER SCREEN SPEED SWITCH**
- **TM# 95-050, INSTALL SINGLE JUMPER FOR B TRAVELING WATER SCREEN SPEED SWITCH**
- **TM# 95-051, INSTALL SINGLE JUMPER FOR C TRAVELING WATER SCREEN SPEED SWITCH**

  These temporary modifications install electrical jumpers that cause the Service Water traveling screens to operate in high speed only. UFSAR Section 7.3.1.1.33.1 states that the screens have two speeds of operation. This jumper is needed because of failed logic cards that may inhibit the high speed operation which is needed due to current environmental conditions. These Temporary Modifications do not affect the ability of the Service Water system or the Ultimate Heat Sink to perform their intended functions. The only credible failure mode is if the jumper becomes disconnected, contacts an electrical ground, faults a fuse and the screen stops rotating. If this occurs, an alarm will ring in the Control Room. If the fuse does not fault then the screen will shift from fast s[peed to slow and an alarm will, also, ring in the control room. There are no anticipated operational transients or postulated design basis accidents associated with this change.

  The jumpers are of the same size and type of materials currently installed in the circuit. There is no additional electrical load. Any impact to the Service Water system postulated by the installation or failure of these temporary modifications is bounded by the single train failure criteria. UFSAR Section 9A states that the loss of all four Service Water traveling screens is not an immediate concern for the safe shutdown of the plant

  Therefore, these temporary modifications do not increase the probability or consequences of an accident previously described in the UFSAR and do not involve an Unreviewed Safety Question.

## Other Summary of Safety Evaluations

- **UFSAR SECTION 13.7 SALEM; 13.6 HOPE CREEK, SALEM - HOPE CREEK SECURITY PLAN, REVISION 5** The following changes are being made by this revision: **Plan Section: Paragraph 1.1, pages 1-3** Management organization and title changes, reflecting the re-organization of the PSE&G Nuclear Business Unit. **Plan Section: Paragraph 2.2, page 7** Change specifies that station personnel such as Emergency Duty Officer and Senior Nuclear Shift Supervisors, who have security responsibilities under the Security Contingency Plan receive training appropriate to their duties through participation in Security drills and Emergency Preparedness drills and exercises having Security involvement. **Plan Section: Chapter 9, page 33** Deletes the requirement to post primary reactor containments during outages and major maintenance in accordance with the Finai Rule change to 10CFR73.55(d)(8), published in the Federal Register, Volume 60, No. 173, dated 9/5/95. These changes involve no plant equipment and do not reduce the security effectiveness of the Security Plan.

  The change to 10CFR73.55(d)(8) is based upon the facts that persons and materials are searched upon entry to the protected area, and access is controlled through vital area portals prior to persons being able to gain access to containment. Thus access to containment is already controlled. Furthermore, after the containment is secured following periods of heavy traffic, existing NRC requirements for walkdown inspections and security searches apply and assure the security of the containment. Since the Security Program is designed to prevent purposeful acts of radiological sabotage, there are no accident analyses applicable to it. However, the range of threats to the plants has been analyzed and addressed in Section 13.7 of the hope Creek UFSAR 13.7. Based on our own analysis and the NRC's published information this procedure change does not create the possibility of an accident or security threat of different type from any previously evaluated in the UFSAR.

  Therefore, this change to the Salem - Hope Creek Security Plan 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, tests, or experiments in this category th s month.