PSEG

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

Nuclear Business Unit

February 13, 1996

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

Sincerely yours,

Mark Reddemann
General Manager -
Hope Creek Operations

DL:DS:CC
Attachments

C Distribution

The power is in your hands.

JE24

INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 2/5/96
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

OPERATING DATA REPORT
OPERATING STATUS

1. Reporting Period January 1996 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	0.0	0.0	66923.9
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	0.0	0.0	65941.6
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	0.0	0.0	210774249
10.	Gross electrical energy generated (MWH)	0.0	0.0	69825622
11.	Net electrical energy generated (MWH)	0.0	0.0	66708563
12.	Reactor service factor	0.0	0.0	83.7
13.	Reactor availability factor	0.0	0.0	83.7
14.	Unit service factor	0.0	0.0	82.5
15.	Unit availability factor	0.0	0.0	82.5
16.	Unit capacity factor (using MDC)	0.0	0.0	81.0
17.	Unit capacity factor (using Design MWe)	0.0	0.0	78.2
18.	Unit forced outage rate	0.0	0.0	5.1

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Currently shutdown for Refueling Outage, RF06, began November 11, 1996
20. If shutdown at end of report period, estimated date of start-up:
    Criticality is scheduled for February 26, 1996
    Breaker closure is scheduled for February 29, 1996

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OPERATING DATA REPORT
UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH JANUARY 1996

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.	01/01/96	S	11/95 - 478.4 12/95 - 744.0 01/96 - 744.0 This outage is still in progress.	C	4 Unit was shutdown in November 1995	Refueling Outage

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AVERAGE DAILY UNIT POWER LEVEL

MONTH JANUARY 1996

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	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		

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REFUELING INFORMATION

MONTH JANUARY 1996

1. Refueling information has changed from last month:

   Yes _ No X

2. Scheduled date for next refueling: 4/2/97 (a)
3. Scheduled date for restart following refueling: 6/1/97 (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 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 (prior to current refueling outage)	764
B. In Spent Fuel Storage (prior to RF06)	1240
C. In Spent Fuel Storage (after RF06)	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)

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

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MONTHLY OPERATING SUMMARY

MONTH JANUARY 1996

The Hope Creek Generating Station remained off-line the entire month of January 1996 for the sixth refueling outage. This resulted in planned energy losses of 831420 MWHRS. As of January 31, 1996 the unit had been off-line for 82 days.

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

MONTH JANUARY 1996

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.

Test Summary of Safety Evaluation

- **4EX-3510; PACKAGE 1, INSTALLATION OF EXPERIMENTAL FILLS IN COOLING TOWER** This test installed several new modular plastic fills and structural supports in the Hope Creek Cooling Tower. Fills were placed where there was sufficient space for installation without removing existing structural supports or asbestos cement board fill sheets. The new modular plastic fills were installed to evaluate their performance and susceptibility to biological fouling during power operation. The cooling capability of the Hope Creek Circulating Water system will not be affected because of the small amount of fill installed. The cooling tower is described only in a general sense in the Hope Creek UFSAR. This test does not change that description or the Design Parameters listed in UFSAR Table 10.4-4, therefore there is no change to the plant as described in the UFSAR. However, a test to accumulate data concerning bio-fouling is not described in the UFSAR. There are no previously anticipated operational transients or design basis accidents affected by this proposal. The Circulating Water System is not considered in the Accident Analysis because it is not required to be operational during the events described in Chapter 15 of the UFSAR.

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

Design Changes **Summary of Safety Evaluations**

- **4EC-3411; PACKAGE 3, RHR LOOP 'D' TO LOOP 'B' CROSS-TIE** This design change will allow RHR pump "D" to be aligned to utilize the heat removal capacity of Heat Exchanger "B." This is a change to the facility as described in the UFSAR Sections 5.4.7.1, 5.4.7.2.1, and 7.3.1.1.4 and Figures 5.4-13 and 6.3-12. This package changes the position of valve 1BCV-571 from normally open (as stated in package 2 of this Design Change) to normally closed and maintains valve 1BCV-570 as normally closed during power operation, when the cross-tie is not inservice. This change does not affect the performance or capability of the RHR System or any other system required to automatically actuate and mitigate the consequences of anticipated operational transients or postulated design basis accidents previously evaluated in the UFSAR. This change adds interconnecting piping and two manual isolation valves between RHR pumps' B and D discharge headers. Both valves will be normally closed. Installation of the proposed modification will not affect the failure mode or failure probability of any equipment important to safety.

  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.

- **4EC-3463; PACKAGES 1, 2 & 3, EXTRACTION STEAM REPLACEMENT** This design change replaces plain carbon steel pipe in portions of the low pressure feedwater heater extraction steam lines with carbon steel pipe that has a metallurgically bonded alloy cladding. This is being done as part of the Hope Creek iron reduction program. The new material was selected because of its high corrosion and erosion resistance. The piping insulation will be changed from calcium silicate to blanket type Fiberglas. This is being done to increase reusability and ease reinstallation. The piping is designed to the same criteria as the original installation and, therefore, there is no affect on failure modes, operational transients, design basis accidents and equipment malfunctions. The extraction steam system is not a high energy fluid system. The change does not affect the pipe break accident scenarios described in UFSAR Sections 3.6 and 15.6.5.

  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.

- **4EC-3485; PACKAGE 1, ADDITION OF REACTOR VESSEL CAVITY FLOOD-UP TRANSMITTER** This design change adds a new transmitter and relocates an indicator on the "A" Channel instrument rack shown on UFSAR Figure 5.4-4. Both devices are located downstream of excess flow check valves, therefore, there is no impact on the Reactor coolant and Primary Containment pressure boundaries. The new transmitter will be used when the reactor vessel head is removed and the cavity is flooded. This transmitter provides a means for Control Room Operators to trend the reactor cavity water level when the N027 transmitter is made inoperable by removal of the reactor head. The new indicator is electrically, environmentally, and seismically equivalent to the N027 transmitter. This Design Change does not affect the seismic qualification, fire protection, electrical separation or interaction studies of the instrument rack. The addition of a qualified transmitter to a qualified instrument rack does not affect the probability or consequences of an accident, or malfunction of equipment important to safety. Nor does it introduce any new accidents or 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.

Design Changes Summary of Safety Evaluations (continued)

- **4EC-3517; PACKAGE 1, REPLACEMENT OF HPCI/RCIC DRAIN LINE** This design change replaced the carbon steel line and fittings from the "Tee" joining the Main Steam and HPCI/RCIC drains to the condenser with a corrosion/erosion resistant material. No parameters are affected by this change. The piping is designed to the same criteria as the original installation and, therefore, there is no affect on failure modes, operational transients, design basis accidents and equipment 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.

- **4EC-3579, DRILL A WEEP HOLE IN SELECTED FLEXIBLE WEDGE AND DOUBLE DISC GATE VALVES** This design change drills weep holes in the reactor side of valve discs to provide a relief path for the bonnet to prevent pressure locking. Industry documents describe events where flexible wedge or double disc gate valves have failed to open due to pressure locking. This occurs when a closed flexible wedge or double disc gate valve has high pressure fluid trapped in the valve bonnet cavity followed by system depressurization or when the bonnet is pressurized due to thermal expansion of the entrapped fluid. Under these conditions the opening capability of the motor operator may not be sufficient to overcome the force on the valve disc. This change modifies valves at Hope Creek with active safety functions identified as susceptible to pressure locking. Packages will be prepared to implement the modification on the each susceptible valve or group of valves and will be reported when complete.

  The weep holes enhance valve operability since they preclude pressure locking. The vendor has confirmed the structural adequacy of the valves is not affected by drilling a weep hole. Valve function is unchanged and remains as described in the UFSAR and the Technical Specifications. No new failure modes from those considered in the UFSAR are added. Ability of the valves to open under accident conditions is enhanced. The weep holes will not prevent the valves from performing their primary containment isolation function. Post-modification testing will include Appendix J, Type C Leak Rate testing.

  PACKAGE 01, CORE SPRAY INJECTION VALVE, 1BEHV-F005A/B (1BEV-007/00[illegible] DISC WEEP HOLE This Design Change Package installs a weep hole in the disc of 1BEHV-F005A/B. UFSAR Figure 6.3-7 will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  PACKAGE 02, RHR/LPCI LOOP A INJECTION VALVE, 1BCHV-F017A (V-113), DISC WEEP HOLE This Design Change Package installs a weep hole in the disc of 1BCHV-F017A. UFSAR Figure 5.4-13, Sheet 2, will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  PACKAGE 04, LPCI LOOP B INJECTION VALVE, 1BCHV-F017C (V-101), DISC WEEP HOLE This Design Change Package installs a weep hole in the disc of 1BCHV-F017C. UFSAR Figure 5.4-13, Sheet 2, will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  PACKAGE 06, RCIC FEEDWATER ISOLATION VALVE, 1BDHV-F013 (V-005), DISC WEEP HOLE This Design Change Package installs a weep hole in the disc of 1BDHV-F013. UFSAR Figure 5.4.8 will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  Therefore, these packages of the DCP do not increase the probability or consequences of an accident previously described in the UFSAR and do not involve any Unreviewed Safety Questions.

Design Changes **Summary of Safety Evaluations (continued)**

- **4HE-0300, REWORK CLOSED MOTOR OPERATED VALVE (MOV) POSITION INDICATIONS ON VARIOUS MOVs** This design change modifies the motor operators for various valves. A separate package will be prepared for each valve and reported when it is implemented.

  In summary, limit switch LS-7 will be reset to provide unambiguous indication of MOV position by maintaining illumination of the valve open light until the valve has reached the full closed position. To allow the resetting of LS-7, limit switch LS-5, which is presently used to bypass torque switch WS-18 on initial opening, will be jumpered to bypass WS-18 during normal operations. This is necessary because LS-5 and LS-7 are integral to the same rotor, R2. Since torque switch WS-18 will be bypassed, this change, also, eliminates the potential for the torque switch to prevent valve opening in response to remote-manual initiation. The jumper will be removed during testing and limit switch calibration to minimize the potential for mechanical damage to the valve, which is possible when conducting limit testing.

  Discussion of torque switches in the UFSAR, Section 6.3.4.1, is part of a general discussion of testing performed on ECCS components, and does not establish criteria for determining when torque switches are required. However, most of these torque switches are shown on figures in the UFSAR and therefore, consequently these are changes that involve a change to the facility as described in the UFSAR.

  The revision of the LS-7 setting and the deletion of the open torque feature do not adversely affect the design basis or operation of the affected MOV, impact its ability to perform required functions, or modify the MOV in such a manner as to cause the valve to be susceptible to failure modes different from those considered in the UFSAR.

  PACKAGE 09, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F007B This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Minimum Flow Valve 1BCHC-F007B. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 7 of 13, which will be revised to reflect this change. The valve is a primary containment isolation valve. The valve is open during all normal plant operating modes. The valve is opened and verified open during plant start up when preparing the RHR System. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 11, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F017B This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Loop "B" Low Pressure Coolant Injection (LPCI) Injection Valve, 1BCHV-F017B. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 9 of 13, which will be revised to reflect this change. The valve is a primary containment isolation valve and a reactor coolant system pressure isolation valve. The valve is normally closed. It receives an automatic ECCS actuation signal to open. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 18, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F021A This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Loop "A" Containment Spray Valve, 1BCHV-F021A. The torque switch is shown in UFSAR Figure 7.3-8, Sheets 7 & 8 of 13, which will be revised to reflect this change. The valve is normally closed for proper Emergency Core Cooling System and Containment Isolation alignment. It is open when required for the Containment Spray mode of ECCS/RHR operation.

- **4HE-0300, REWORK CLOSED MOTOR OPERATED VALVE (MOV) POSITION INDICATIONS ON VARIOUS MOVs (continued)**

  PACKAGE 20, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F027A This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Pump "A" Suppression Pool Spray Valve, 1BCHV-F027A. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 12 of 13, which will be revised to reflect this change. The valve is normally closed for proper ECCS and Containment Isolation alignment. It is open when required for Suppression Pool Spray mode of operation. The modification will not prevent the MOV from performing its function.

  PACKAGE 21, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F027B This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Pump "B" Suppression Pool Spray Valve, 1BCHV-F027B. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 11 of 13, which will be revised to reflect this change. The valve is normally closed for proper ECCS and Containment Isolation alignment. It is open when required for Suppression Pool Spray mode of operation. The modification will not prevent the MOV from performing its function.

  PACKAGE 22, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F017A This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Loop "A" Low Pressure Coolant Injection (LPCI) Injection Valve, 1BCHV-F017A. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 9 of 13, which will be revised to reflect this change. The valve is a primary containment isolation valve and a reactor coolant system pressure isolation valve. The valve is normally closed. It receives an automatic ECCS actuation signal to open. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 24, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F017C This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Loop "C" Low Pressure Coolant Injection (LPCI) Injection Valve, 1BCHV-F017C. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 9 of 13, which will be revised to reflect this change. The valve is a primary containment isolation valve and a reactor coolant system pressure isolation valve. The valve is normally closed. It receives an automatic ECCS actuation signal to open. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 26, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F008 This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Shutdown Cooling Outboard Isolation Valve, 1BCHV-F008. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 7 of 13, which will be revised to reflect this change. This valve is a containment isolation valve and a reactor coolant system pressure isolation valve and is therefore normally closed. This valve receives an automatic closure signal to ensure RHR suction is isolated from the Reactor Coolant Pressure Boundary. This valve does not receive any automatic opening signal. The modification will not prevent the MOV from performing its function.

- **4HE-0300, REWORK CLOSED MOTOR OPERATED VALVE (MOV) POSITION INDICATIONS ON VARIOUS MOVs (continued)**

  PACKAGE 27, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F009 This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Shutdown Cooling Outboard Isolation Valve, 1BCHV-F009. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 7 of 13, which will be revised to reflect this change. This valve is a containment isolation valve and a reactor coolant system pressure isolation valve and is therefore normally closed. This valve receives an automatic closure signal to ensure RHR suction is isolated from the Reactor Coolant Pressure Boundary. This valve does not receive any automatic opening signal. The modification will not prevent the MOV from performing its function.

  PACKAGE 31, REWORK CLOSED MOV POSITION INDICATION ON 1BDHV-F013 This Design Change Package installs the limit switch modifications on the Reactor Core Isolation Cooling (RCIC) Feedwater Isolation Valve, 1BDHV-F013. The torque switch is shown in UFSAR Figure 7.4-2, Sheet 4 of 9, which will be revised to reflect this change. This valve is a containment isolation and therefore is normally closed to isolate RCIC form Feedwater Loop "B." The valve automatically opens on a RCIC actuation signal. The modification will not prevent the MOV from performing its function.

  PACKAGE 32, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F005A This Design Change Package installs the limit switch modifications on the Core Spray Loop "A" inboard injection valve, 1BEHV-F005A. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 3 of 3, which will be revised to reflect this change. This valve is a containment isolation valve and a reactor coolant system pressure isolation valve and is therefore normally closed. This valve is closed and verified closed during plant start up when preparing the Core Spray system for standby operation. This valve receives an automatic signal to open in response to a Core Spray initiation signal, in conjunction with a Reactor Vessel low pressure signal. The modification will not prevent the MOV from performing its function.

  PACKAGE 34, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F001A This Design Change Package installs the limit switch modifications on the Core Spray Pump "A" Suction Valve, 1BEHV-F001A. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 2 of 3, which will be revised to reflect this change. This valve is located outside the Primary Containment and is a Containment Isolation Valve. This valve has no automatic safety function. This valve is opened and verified open during plant start up when preparing the Core Spray system for standby operation The valve is open during all normal plant operating modes and all design basis accident scenarios. The modification will not prevent the MOV from performing its function.

  PACKAGE 36, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F001C This Design Change Package installs the limit switch modifications on the Core Spray Pump "C" Suction Valve, 1BEHV-F001C. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 2 of 3, which will be revised to reflect this change. This valve is located outside the Primary Containment and is a Containment Isolation Valve. This valve has no automatic safety function. This valve is opened and verified open during plant start up when preparing the Core Spray system for standby operation The valve is open during all normal plant operating modes and all design basis accident scenarios. The modification will not prevent the MOV from performing its function.

Design Changes Summary of Safety Evaluations (continued)

- **4HE-0300, REWORK CLOSED MOTOR OPERATED VALVE (MOV) POSITION INDICATIONS ON VARIOUS MOVs (continued)**

  PACKAGE 48, REWORK CLOSED MOV POSITION INDICATION ON 1FCHV-F007 This Design Change Package installs the limit switch modifications on the Reactor Core Isolation Cooling (RCIC) turbine steam supply inboard isolation valve, 1FCHV-F007. The torque switch is shown in UFSAR Figure 7.4-2, Sheet 1 of 9, which will be revised to reflect this change. This valve is located inside the containment and is a containment isolation valve. The valve is opened during plant start up when the RCIC steam supply low pressure isolation can be reset to supply steam from the "A" Main Steam line to the RCIC Turbine steam supply line. The valve is open during normal operations. The valve automatically closes upon receipt of an RCIC automatic isolation signal. The modification will not prevent the MOV from performing its function.

  PACKAGE 49, REWORK CLOSED MOV POSITION INDICATION ON 1FCHV-F008 This Design Change Package installs the limit switch modifications on the Reactor Core Isolation Cooling (RCIC) turbine steam supply outboard isolation valve, 1FCHV-F008. The torque switch is shown in UFSAR Figure 7.4-2, Sheet 1 of 9, which will be revised to reflect this change. This valve is located outside the containment and is a containment isolation valve. The valve is opened during plant start up when the RCIC steam supply low pressure isolation can be reset to supply steam from the "A" Main Steam line to the RCIC Turbine steam supply line. The valve is open during normal operations. The valve automatically closes upon receipt of an RCIC automatic isolation signal. The modification will not prevent the MOV from performing its function.

  PACKAGE 56, REWORK CLOSED MOV POSITION INDICATION ON 1KPHV-5834A This Design Change Package installs the limit switch modifications on the Main Steam Isolation Valve (MSIV) Inboard Seal Gas Supply Valve, 1KPHV-5834A. The torque switch is shown in UFSAR Figure 7.3-17, Sheet 1 of 3, which will be revised to reflect this change. This valve is a normally closed containment isolation valve. The valve may be remote manually opened following a LOCA to reduce untreated leakage of the containment atmosphere resulting from possible loss of MSIV leaktightness. The valve receives an automatic containment isolation signal. There is no automatic opening signal for this valve. The modification will not prevent the MOV from performing its function.

  PACKAGE 59, REWORK CLOSED MOV POSITION INDICATION ON 1KPHV-5837A This Design Change Package installs the limit switch modifications on the Main Steam Isolation Valve (MSIV) Inboard Seal Gas Supply Valve, 1KPHV-5837A. The torque switch is shown in UFSAR Figure 7.3-17, Sheet 1 of 3, which will be revised to reflect this change. This valve is a normally closed containment isolation valve. The valve may be remote manually opened following a LOCA to reduce untreated leakage of the containment atmosphere resulting from possible loss of MSIV leaktightness. The valve receives an automatic containment isolation signal. There is no automatic opening signal for this valve. The modification will not prevent the MOV from performing its function.

  Therefore, these packages of the DCP do not increase the probability or consequences of an accident previously described in the UFSAR and do not involve any Unreviewed Safety Questions.

Design Changes   Summary of Safety Evaluations (continued)

- **4HE-0083, INSTALLATION OF VENT VALVES ON SELECTED PIPING HEADERS**

  PACKAGE 03, INSTALLATION OF VENT VALVES ON SELECTED LOCATIONS OF THE RESIDUAL HEAT REMOVAL HEADERS This Design Change Package installs two vent valves between valves 1BCV-182 and 1BCV-F041A. These valves will improve draining of the system. Normal operation is not affected because the valves will be capped closed during normal operation and there is no reason to reposition them. The installation meets all applicable design, material and construction standards, therefore, the design basis of the RHR system is not changed. None of the operational transients or postulated design basis accidents evaluated in the UFSAR are applicable to this modification. UFSAR Figure 5.4-13 will be revised to show this change.

  PACKAGE 04, INSTALLATION OF VENT VALVES ON SELECTED LOCATIONS OF THE RESIDUAL HEAT REMOVAL HEADERS This Design Change Package installs two vent valves between valves 1BCV-183 and 1BCV-111. These valves will improve draining of the system. Normal operation is not affected because the valves will be capped closed during normal operation and there is no reason to reposition them. The installation meets all applicable design, material and construction standards, therefore, the design basis of the RHR system is not changed. None of the operational transients or postulated design basis accidents evaluated in the UFSAR are applicable to this modification. UFSAR Figure 5.4-13 will be revised to show this change.

  PACKAGE 07, INSTALLATION OF VENT VALVES ON SELECTED LOCATIONS OF THE CORE SPRAY HEADERS This Design Change Package installs two vent valves between valves 1BEV-005 and 1BEHV-F006A. These valves will improve draining of the system. Normal operation is not affected because the valves will be capped closed during normal operation and there is no reason to reposition them. The installation meets all applicable design, material and construction standards therefore, the design basis of the Core Spray system is not changed. None of the operational transients or postulated design basis accidents evaluated in UFSAR are applicable to this modification. UFSAR Figure 6.3-7 will be revised to show this change.

  Therefore, these packages of the DCP do not increase the probability or consequences of an accident previously described in the UFSAR and do not involve any Unreviewed Safety Questions.

- **4HE-0258; PACKAGE 02, SW STRAINER SMALL BORE VENT AND DRAIN LINE PIPING REPLACEMENT** This design change replaces the one and two inch SW strainer small bore vent and drain cement lined piping with 6% molybdenum austenitic stainless steel piping which meets the original design requirements and is of the same schedule as the original piping. This design change upgrades the existing small bore piping as part of PSE&G's Service Water piping upgrade effort. Galvanic corrosion of dissimilar metals will be prevented through the use of Maloney insulating kits with Garlock gaskets. Therefore, it does not introduce any new credible failure scenarios, or adversely impact the existing failure modes. The system P&IDs in the UFSAR will have to be revised to reflect the new material. (NOTE - Completion of 4HE-0258, Package 01 was reported last month)

  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

- **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, Control Room Supply System; D7.5, Environmental Design Criteria; D5.1, Design Criteria for Control Room; and D3.5, Control Area HVAC Specification to reflect a temperature range of 72°F ± 6°F and a Relative Humidity range of 20% to 60% for the nominal expected environmental conditions 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. These values are compatible with existing plant and equipment design. The currently listed environmental conditions for the Control Room were extracted from PNO-A61-4270-0015, the General Electric specification. GE has identified these values as the expected conditions as a result of calculations. They do not represent the limiting conditions for operation. The Control Room Emergency Filtration (CREF) charcoal absorbers are the most limiting components with respect to relative humidity requirements in the Control Room Area. The relative humidity requirement for the charcoal absorbers will be met a control room relative humidity of 60%. The operating parameters of the Control Room Supply (CRS) System or any other control room associated equipment will not be changed. This change incorporates limits consistent with the original design. It does not alter or degrade the present operating modes of the system.

  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 96-07, HOPE CREEK UFSAR SECTION 9.3.37.1 REVISION** This change notice revises UFSAR Section 9.3.3.7.1, Instrumentation Application, to reflect the installed instrumentation. The text is being changed from "local radiation processor" and "processor" to "programmable controller." The UFSAR currently describes the sump pumps as being controlled by a local radiation processor (LRP). However, the start and stop control functions described in UFSAR Section 9.3.3.7.1 are actually controlled by a programmable controller. There are no affects on the operational transients or postulated design basis accidents previously evaluated in the UFSAR because neither the Drywell Floor Drain nor Drywell Equipment Sump Pumps are credited in the Hope Creek UFSAR as mitigating the consequences of an accident. The plant drainage systems have no safety-related function. The failure of a programmable controller would have the same result as the failure of an LRP. This change does not affect the control scheme described in the UFSAR.

  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 96-08, EDG OPERATION - MANUAL MAKEUP TO THE DG LUBE OIL MAKEUP TANK** This change will allow making up to the Emergency Diesel Generator (EDG) Lube Oil (LO) Makeup Tanks locally via the installed conservation vent fitting at the top of the tank. UFSAR Section 9.5.7 is being revised to allow filling of the Lube Oil Makeup Tank from other than the installed LO fill piping. This procedure change will not introduce any new failure modes in relation to the credible accidents/ malfunctions described in the UFSAR. This proposal does not alter the operational or postulated design basis previously evaluated in the UFSAR. This procedure change does not degrade the performance of the EDGs nor does it affect any design specification of the EDGs and associated equipment. No physical changes are made to any system, structure or component.

  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 HCR.80002, HOPE CREEK FUEL POOL COOLING AND CLEANUP SYSTEM UFSAR CHANGE** This change will revise UFSAR Section 9.1.3 to add the faulted configuration of only one pump and one heat exchanger in service to the description. The current UFSAR Section 9.1.3 describes only the faulted configuration of one pump and two heat exchangers in service. Both these configurations were reviewed and approved by the NRC in the HCGS SER as part of the design bases for the Fuel Pool Cooling and Cleanup (FPCC) System. This failure results in a fuel pool temperature of 174°F. The change will not affect any plant configuration or procedures as described in the UFSAR. The failure associated with this change is the failure of one loop of the FPCC. This failure has been previously evaluated and accepted by the NRC. This change does not introduce any new failures. There are no transients or accidents previously evaluated in the UFSAR applicable to the FPCC. Malfunctions to the FPCC System which would cause a loss of one loop of FPCC are listed in UFSAR Table 9.1-3.

  This change will, also, change the reference for the location of the decay heat information from Table 9.1-18 to Table 9.1-17 in the last paragraph of UFSAR Section 9.1.3.3. This second change is a typographical correction

  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# 95-060, SSWS LOOP "A" & LOOP "B" TEMPORARY DISCHARGE PIPING TO MH50 & MH51** This Temporary Modification diverts Service Water normally flowing to the Cooling Tower Basin or the Tower Bypass Line to manholes in the yard during operational conditions 4 or 5. This Modification will permit maintenance on yard piping for the Service Water System. The SSWS emergency bypass line has been designed to facilitate the passage of the required SSWS flow through each of the SSWS/ SACS loops in the event of a failure of the yard discharge piping. Based on an analysis of the piping geometry in the temporary piping, piping stresses as a result of a dynamic event will not exceed allowable stresses. The margins of safety as discussed in the Technical Specifications will not be impacted by the utilization of the temporary piping.

  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# 96-001, INSTALLATION OF TEMPORARY SERVICE WATER STRAINER BACKWASH DISCHARGE LINES** This Temporary Modification removes the Service Water strainer effluent discharge from the normal flow path shown on UFSAR Figure 9.2-2 and diverts the flow back to the river via two of the Unit II spare bays. This is being done to allow replacement of piping on the traveling screens and strainers in accordance with Design Change 4HE-0207. There are no operational transients or postulated design bases accidents that are applicable to this proposal. The backwash function is not altered by this proposal. The backwash function does not impact any radiological system. There are no malfunctions of equipment important to safety that are considered applicable to this proposal. A complete failure of this temporary modification will not cause an accident or change single failure criteria for any safety-related component or system.

  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.

Deficiency Report Summary of Safety Evaluation

- **DEFICIENCY REPORT 960115095, DIESEL GENERATOR JACKET WATER COOLING HEADER, 1A-G-400 ORIFICE EROSION - ORIFICE SIZE EVALUATION AND TESTING PORTION ONLY** This Deficiency Report disposition results in re-installation of orifices in the Diesel Generator "A" (A EDG) Jacket Water Cooling system that are at variance with the design documentation for testing and evaluation. This test will verify their acceptability with the design criteria. The orifices represent the sizes found installed in the EDG during maintenance. The parameters affected are the relative cooling water flows to the EDG engine and inlet turbocharger. Failure of the orifices to provide the required flow (orifice too small) could result in inadequate cooling. Failure of the orifice to properly restrict flow (orifice too large) could result in pump failure from runout. The activity will be performed while the a EDG is considered inoperable for maintenance. While the unit is in Operational Condition 4 or 5, only two of four diesels are required to be operable. The functional performance of the Jacket Water Cooling System will not be altered by this re-installation.

  Therefore, this deficiency report disposition 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

- **HCR.8-0003, HOPE CREEK RELOAD SAFETY EVALUATION FOR CYCLE 7** This safety evaluation addresses the impact of Hope Creek reload core design on the cycle 6 operation. There will be a discharge of 232 fuel assemblies at end of cycle 5 operation. The safety analyses for the reload are reviewed.

  The limiting transients previously evaluated in Chapter 15 of the UFSAR have been re-analyzed for the Hope Creek cycle 6 / reload 5 design. A new set of core operating limits is generated and documented in the Core Operating Limits Report which is referenced in the Technical Specification. This analysis was performed with methods previously reviewed and approved by the NRC staff for reload design and analysis purposes.

  The replaced fuel bundles have the same nuclear design and minor axial zoning change. The minor axial zoning change has economical benefits while maintaining the same nuclear and thermal-hydraulic characteristics. The zoning change takes advantage of the large shutdown margin available for cycle 6 and therefore relax the zoning requirement for shutdown consideration.

  Based on the GE projected cycle performance, all aspects of the fuel thermal-mechanical bases have been reviewed. The projected cycle 6 core design and operation will not violate any fuel thermal-mechanical design criteria.

  Therefore, this Safety Evaluation 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

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