# INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 1/5/96
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

# AVERAGE DAILY UNIT POWER LEVEL

**MONTH DECEMBER 1995**

| 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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# OPERATING DATA REPORT
## OPERATING STATUS

1. Reporting Period December 1995 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 restriction (if any)

| | | This Month | Yr To Date | Cumulative |
|---|---|---|---|---|
| 5. | No. of hours reactor was critical | 0.0 | 6988.0 | 66923.9 |
| 6. | Reactor reserve shutdown hours | 0.0 | 0.0 | 0.0 |
| 7. | Hours generator on line | 0.0 | 6938.2 | 65941.6 |
| 8. | Unit reserve shutdown hours | 0.0 | 0.0 | 0.0 |
| 9. | Gross thermal energy generated (MWH) | 0.0 | 22359904 | 210774249 |
| 10. | Gross electrical energy generated (MWH) | 0.0 | 7397956 | 69825622 |
| 11. | Net electrical energy generated (MWH) | 0.0 | 7063919 | 66717235 |
| 12. | Reactor service factor | 0.0 | 79.8 | 84.5 |
| 13. | Reactor availability factor | 0.0 | 79.8 | 84.5 |
| 14. | Unit service factor | 0.0 | 79.2 | 83.3 |
| 15. | Unit availability factor | 0.0 | 79.2 | 83.3 |
| 16. | Unit capacity factor (using MDC) | 0.0 | 78.2 | 81.7 |
| 17. | Unit capacity factor (using Design MWe) | 0.0 | 75.6 | 79.0 |
| 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 17, 1996

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# OPERATING DATA REPORT
# UNIT SHUTDOWNS AND POWER REDUCTIONS

**MONTH DECEMBER 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. | 12/01/95 | S | 11/95 -478.4<br>12/95 - 744.0<br>This outage is still in progress. | C | 4<br>Unit was shutdown in November 1995 | Refueling Outage |

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# REFUELING INFORMATION

**MONTH DECEMBER 1995**

1. Refueling information has changed from last month:

   Yes X No __

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 DECEMBER 1995**

The Hope Creek Generating Station remained off-line the entire month of December 1995 for the sixth refueling outage. This resulted in planned energy losses of 831420 MWHRS. As of November 30, 1995 the unit had been off-line for 20 days.

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

**MONTH DECEMBER 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

- **4HM-0177, INSTALLATION OF CARPETING & SHEET VINYL FLOORING IN CHEMISTRY DEPARTMENT OFFICES & LABORATORIES** This design change added carpeting and sheet vinyl flooring to areas of the Hope Creek Chemistry Department offices and laboratories. The purpose of the new flooring systems is to reduce noise levels, Chemistry technician fatigue, and improve the appearance of the facility. The carpeting meets the requirements of the ASTM E84 Tunnel Test and, in combination with the padding, adds only six minutes fire loading as in-situ combustibles. This increase is considered negligible. This combined total is within the fire hazards analysis loading for the subject areas. UFSAR Section 9A will be revised to reflect the change.

  There are no operational transients or postulated design basis accidents associated with this change. The presence of the flooring has no affect on the probability of a fire starting as it cannot combust without an ignition source. The Fire Hazards Analysis (FHA) does not assess fire risk in terms of likelihood but rather bases the analysis on the premise that a fire will occur, and damage to equipment within the fire areas happens. As such, fire load increases within fire areas do not affect the bases of the FHA provided the fires can be contained within the boundary of the fire area. Since the fire load in the affected areas remains within the design, fire barrier integrity will be maintained and the probability of occurrence of a malfunction of equipment due to fire spread remains unchanged.

  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)

- **4HE-0075; PACKAGE 2, REMOVAL AND REPLACEMENT OF FEEDER CABLES** This design change replaced the existing 500 MCM feeder cable BL1D0542G to the distribution panel 1BD417 with a 350 MCM cable as a result of Electrical Calculation E-1.4(Q), Revision 3. This cable change increased the circuit resistance and reduced the fault current contribution to within the interrupting rating of the device. This change will greatly improve the reliability of the equipment. The calculation had determined that the interrupting rating of the bus and branch circuit breakers was lower than the available short circuit current due to a fault. UFSAR Figure 8.3-8, sheets 1 & 2 show the distribution panel feeder cable size and will be revised as a result of this change.

  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.

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

  **PACKAGE 02, INSTALLATION OF VENT VALVES ON SELECTED LOCATIONS OF THE CORE SPRAY HEADERS** This Design Change Package installs 1 inch vent and drain valves in various locations in the Core Spray System. These valves will improve the venting and draining of the system and reduce the draining time involved. The design basis of the Core Spray system will not be changed. The valves will be capped closed during normal operation and there is no reason to reposition them. UFSAR Figure 6.3-7 (P&ID M52-1) will require revision to address these changes to plant configuration.

  **PACKAGE 06, INSTALLATION OF VENT VALVES ON SELECTED LOCATIONS OF THE RESIDUAL HEAT REMOVAL HEADERS** This Design Change Package installs 1 inch vent and drain valves in various locations in the RHR System. These valves will improve the venting and draining of the system and reduce the draining time involved. The design basis of the RHR system will not be changed. The valves will be capped closed during normal operation and there is no reason to reposition them. UFSAR Figure 5.4-13 (P&ID M51) will require revision to address these changes to plant configuration.

  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, 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.

  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.

- **4HE-0274; PACKAGE 2, CHANGE RHR/ CORE SPRAY PERMISSIVE FROM INDIVIDUAL SERIES CIRCUITS TO A JOINT ONE OUT OF TWO TAKEN TWICE CIRCUIT** This design change rearranges the four 7-8 contacts of the 27AY auxiliary relays in the 4160V Class 1E Channel B bus undervoltage relaying logic from two individual series circuits to a one out of two taken twice circuit. This change makes closing the permissive circuit more reliable by allowing both the RHR and core spray pump motors to be operable in the event that one of the 27AY auxiliary undervoltage relays fails or is out of service while still protecting the pump motors during an actual LOP or undervoltage event. Figure 9A-3 in the UFSAR will be changed to show the rearrangement of the undervoltage auxiliary relay contacts and the transfer switch contact for the RSP. This change does not affect operation of the RHR system as described in the UFSAR.

  Rearranging the contacts for the RHR/ Core Spray permissive does not create the possibility of any new failures. Currently, loss of one of the auxiliary relays results in the loss of one of the permissives. This change increases the reliability of the closing permissive circuits by allowing RHR and Core Spray to operate if any one of the four auxiliary undervoltage relays fails or is out of service.

  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)**

- **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 33, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F005B** This Design Change Package installs the limit switch modifications on the Core Spray Injection Valve, 1BEHV-F005B. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 3 of 3, 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 and receives an ECCS signal to open. The MOV will continue to be operated in accordance with existing plant accident analyses. The modification will not prevent the MOV from performing its safety function.

  **PACKAGE 35, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F001B** This Design Change Package installs the limit switch modifications on the Core Spray Pump 'B' Suction Valve, 1BEHV-F001B. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 2 of 3, which will be revised to reflect this change. The valve is located outside the Primary Containment and is a containment isolation valve. The valve has no automatic safety function. The valve is verified as open during plant start up.

  **PACKAGE 37, REWORK CLOSED MOV POSITION INDICATION ON 1BEHV-F001D** This Design Change Package installs the limit switch modifications on the Core Spray Pump 'D' Suction Valve, 1BEHV-F001D. The torque switch is shown in UFSAR Figure 7.3-6, Sheet 2 of 3, which will be revised to reflect this change. The valve is located outside the Primary Containment and is a containment isolation valve. The valve has no automatic safety function. The valve is verified as open during plant start up.

  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)**

- **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 the valve discs to provide a relief path for the bonnet in order to prevent pressure locking. Numerous industry documents describe conditions 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 a 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 additional force on the valve disc. Engineering evaluation EE: H-11-ZZ-MEE-0864 identifies those valves at Hope Creek with active safety functions that are susceptible to pressure locking. Several packages will be prepared to implement the modification on the susceptible valves. Each package will be reported when it is implemented.

  Addition of the weep holes will enhance valve operability since they will preclude pressure locking. There will be no change valve structure or function. The valves will continue to function as described in the UFSAR and the Technical Specifications. to failure modes different from those considered in the UFSAR. Ability of the valve to open under accident conditions will be enhanced. These valves are, also, primary containment isolation valves the addition of a weep hole on the reactor side will not prevent the valve from performing this function. Post-modification testing will include Appendix J, Type C Leak Rate testing. The vendor has evaluated and confirmed the structural adequacy of the valves is not affected by drilling a weep hole.

  **PACKAGE 3, RHR/LPCI LOOP B INJECTION VALVE, 1BCHV-F017B (V-016), DISC WEEP HOLE** This Design Change Package installs a weep hole in the disc of 1BCHV-F017B. UFSAR Figure 5.4-13, Sheet 1, will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  **PACKAGE 5, RHR/LPCI LOOP D INJECTION VALVE, 1BCHV-F017D (V-004), DISC WEEP HOLE** This Design Change Package installs a weep hole in the disc of 1BCHV-F017D. UFSAR Figure 5.4-13, Sheet 1, 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.

## Procedures Summary of Safety Evaluations

- **HC.OP-GP.HB-0003(Q), REV 0, REFUEL OUTAGE ALTERNATE FLOWPATH FOR DRYWELL SUMPS** This procedure provides for under vessel area drainage during outages when the normal discharge path is unavailable due to testing of the isolation valves or maintenance on the discharge piping for the Drywell Liquid Radwaste system. This procedure installs a fitting on the discharge of the Drywell drain sump pumps and allows effluent routing to the downcomers and the Torus via hoses. There is no detrimental affect on any plant equipment required during the refueling outage. All systems will be restored to design configuration and temporary equipment removed prior to entering Mode 3. Hoses, flanges and strainers used will be qualified to ANSI B31.1. Leakage will drain back to the sumps. There is no potential for an unmonitored leak path. The procedure will, also, jumper out the closed signal from the containment isolation valves to the sump pump controls. Normally, the pumps will not run if the valves are closed. This procedure will allow the pumps to operate with the valves closed. The jumpers are in the pump controls and therefore do not affect the design operating functions of the valves.

  The only system whose operation may be impacted is the Drywell liquid radwaste systems. UFSAR section 9.3.3 states there are no safety functions associated with this system. Since all equipment will be removed prior to entering Mode 3, primary Containment integrity will not be impacted in the modes where it is required.

  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.SE-PR.EC-0001(Q), REV 0, FUEL POOL COOLING HEAT EXCHANGER HEAT TRANSFER CAPACITY** This procedure provides for collection of fuel pool cooling heat exchanger performance data and determination of its heat transfer capability. This procedure will be performed to ensure the heat exchange capability is bounded by design assumptions.. This procedure will cause the following CRIDS points to be inoperative while performing the test:

  | | |
  |---|---|
  | A3175 | Fuel Pool Cooling Heat Exchanger Common Inlet |
  | A3176 | Fuel Pool Cooling AE202 Outlet Temperature |
  | A3177 | Fuel Pool Cooling BE202 Outlet Temperature |

  These points are shown on UFSAR Figure 9.1-5 and 9.2-4 and described in UFSAR Section 9.1.3.2.2.3 and 9.1.3.5. The temporary removal of these points is the only change to the facility as described in the UFSAR created by this procedure.

  This procedure is for data gathering only. The disabled CRIDS points provide indication only. The Fuel Pool Cooling system (FPCCS) is not discussed in any of the accidents described in UFSAR Chapter 15. UFSAR Section 9.1.3.4 states that the FPCCS does not perform a specific function in shutting down the reactor or in mitigating the consequences of an accident. The temporary loss of indication for the CRIDS points listed will not result in any unmonitored temperatures for the FPCCS heat exchangers. The affected parameters will be more precisely monitored during the performance of this procedure. UFSAR Table 9.1-3 evaluates the loss of fuel pool cooling. Any malfunction associated with this procedure is bounded by the analysis for a loss of fuel pool cooling because the implementation of this procedure is nonintrusive to the FPCCS and SACS.

  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 Notice Summary of Safety Evaluation

- **UFSAR CHANGE NOTICE CN 95-17, UFSAR CHANGE AS A RESULT OF REACTOR VESSEL SURVEILLANCE CAPSULE REPORT** This change notice implements the results of testing performed on the surveillance capsule removed from the Hope Creek reactor during the Fifth Refueling Outage. Testing of the capsule specimens was done in accordance with 10CFR50, Appendices G & H and Regulatory Guide 1.99, Revision 2. The results of the testing are contained in GE report GE-NE-523-A164-1294. Incorporating the methodology of RG 1.99, revision 2, new pressure-temperature operating limits (P-T curves) were calculated from the test results. Even though the most limiting materials are not contained in the surveillance capsules, the radiation damage sustained by the materials in this specimen was applied to all materials that make up the belt-line region of the reactor vessel. Next, the P-T curves were formulated and the most limiting curve for each of the required operating conditions was plotted. All results were checked against the criteria of RG 1.99, Revision 2 and found to be acceptable. Updating the current UFSAR with these P-T curves incorporates the most recent methodology that the NRC has approved for calculating how a reactor pressure vessel will react to radiation embrittlement. Use of these techniques to develop operating limits for the Hope Creek vessel will ensure vessel integrity, a major premise for all accident scenarios, is maintained for the life of the plant. This proposal will maintain the required design safety margin against vessel failure. The new operating limits resulting from this testing, ensure the reactor vessel will be operated in a manner that prevents brittle fracture.

  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

**TM# 95-052, INSTALL SINGLE JUMPER FOR D TRAVELING WATER SCREEN SPEED SWITCH** This temporary modification installs an electrical jumper that causes the Service Water traveling screen to operate in high speed only. UFSAR Section 7.3.1.1.33.1 states that the screen has 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. This Temporary Modification does not affect the ability of the Service Water system or the Ultimate Heat Sink to perform its 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, the screen will shift from fast speed to slow and an alarm will ring in the control room. There are no anticipated operational transients or postulated design basis accidents associated with this change.

The jumper is 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 this temporary modification 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, 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 Reports Summary of Safety Evaluations

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