HOPE CREEK GENERATING STATION
MONTHLY OPERATING SUMMARY
MARCH 1988

Hope Creek entered the month of March in cold shutdown continuing its First Refueling Outage that commenced on February 13, 1988. The refueling outage continued throughout the month.

R-009
RAR:tlb

IE24
1/1

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 86-354

UNIT Hope Creek

DATE 4/15/88

COMPLETED BY M. Zapolski

TELEPHONE (609) 339-3738

MONTH March 1988

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	* unit in refueling	17	
2	outage for the entire	18	
3	month - all values are	19	
4	0.0 MWe-Net	20	
5		21	
6		22	
7		23	
8		24	
9		25	
10		26	
11		27	
12		28	
13		29	
14			
15			
16			

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 86-354

UNIT Hope Creek

DATE 4/15/88

COMPLETED BY M. Zapolski

REPORT MONTH March, 1988 TELEPHONE (609) 339-3738

NO.	DATE	TYPE F FORCED S SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/ COMMENTS
2	3/1	S	744.0	C	4	CONTINUATION OF REFUELING OUTAGE

SUMMARY

OPERATING DATA REPORT

DOCKET NO. 86-354
UNIT Hope Creek
DATE 4/15/88
COMPLETED BY H. Jensen
TELEPHONE (609) 339-5261

OPERATING STATUS

1. REPORTING PERIOD March 1988 GROSS HOURS IN REPORTING PERIOD 744

2. CURRENTLY AUTHORIZED POWER LEVEL (MWt) 3293
MAX. DEPEND. CAPACITY (MWe-Net) 1067 (1)
DESIGN ELECTRICAL RATING (MWe-Net) 1067 (1)

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	1045.0	8903.1
6. REACTOR RESERVE SHUTDOWN HOURS	0	0	0
7. HOURS GENERATOR ON LINE	0	1037.9	8783.0
8. UNIT RESERVE SHUTDOWN HOURS	0	0	0
9. GROSS THERMAL ENERGY GENERATED (MWH)	0	3,378,284	27,186,852
10. GROSS ELECTRICAL ENERGY GENERATED (MWH)	0	1,133,985	9,045,682
11. NET ELECTRICAL ENERGY GENERATED (MWH)	0	1,077,368	8,642,406
12. REACTOR SERVICE FACTOR	N/A	47.8	79.3
13. REACTOR AVAILABILITY FACTOR	N/A	47.8	79.3
14. UNIT SERVICE FACTOR	N/A	47.5	78.2
15. UNIT AVAILABILITY FACTOR	N/A	47.8	79.3
16. UNIT CAPACITY FACTOR (Using Design MDC)	0.0	46.2	72.1
17. UNIT CAPACITY FACTOR (Using Design MWe)	0.0	46.4	72.0
18. UNIT FORCED OUTAGE RATE	0	0	0

19. SHUTDOWNS SCHEDULED OVER NEXT 6 MONTHS (TYPE, DATE, & DURATION):
None

20. IF SHUT DOWN AT END OF REPORT PERIOD, ESTIMATED DATE OF STARTUP:
4/15/88

(1) August 1987 data is under management review.

REFUELING INFORMATION

COMPLETED BY: Chris Brennan

DOCKET NO.: 50-354
UNIT NAME: Hope Creek Unit 1
DATE: 4/15/88
TELEPHONE: 3193
EXTENSION: N/A

Month March 1988

1. Refueling information has changed from last month: First Report
   YES _____ NO _____

2. Scheduled date for next refueling: 11-04-89

3. Scheduled date for restart following refueling:

   12-18-89

4. A) Will Technical Specification changes or other license amendments be required?
   YES X NO _____

   B) Has the reload fuel design been reviewed by the Station Operating Review Committee?

   YES _____ NO X

   If no, when is it scheduled? 6-18-89

5. Scheduled date(s) for submitting proposed licensing action:
   7-18-89

6. Important licensing considerations associated with refueling:
   Information not presently available

7. Number of Fuel Assemblies:
   A) Incore 764
   B) In Spent Fuel Storage 232

8. Present licensed spent fuel storage capacity: 1108

   Future spent fuel storage capacity: 4006

9. Date of last refueling that can be discharged to spent fuel pool assuming the present licensed capacity: 12-18-89

SUMMARY OF CHANGES, TESTS, AND EXPERIMENTS

FOR THE HOPE CREEK GENERATING STATION

MARCH 1988

The following Design Change Packages (DCPs) 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.

None of the DCPs created a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These DCPs did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

DCP	Description of Design Change Package
7124	This DCP installed a flange assembly consisting of two flanges and a blank plate to isolate branch valves leading from the High Pressure Coolant Injection System to the Residual Heat Removal Heat Exchangers. These mechanical isolations will prevent steam from leaking into the Residual Heat Removal System via steam condensing lines.
7202	This DCP added orifice plates in the Residual Heat Removal and Fuel Pool Heat Exchanger Loops to avoid cavitation of the butterfly valves. (Note: The Residual Heat Removal "A" Loop will be completed at a later time. The rest of the DCP has been installed). The addition of the orifice plates will provide an improved design for balancing the design flow rate in the heat exchanger flow paths.
4-HCO-86-0530	This DCP replaced under voltage relays in the Class 1E 4.16kv busses with solid state relays. The new relays will improve the trip time and implement Technical Specification Amendment 7.
4-HME-86-0781	This DCP added a pressure switch to the discharge piping of the Drywell Leak Detection - Radiation Monitoring System to detect high pressure conditions when the containment isolation valves close. The pressure switch will improve the long range performance of the sample pump by shutting it off so the motor will not overload when the containment isolation valves close.
4-EMP-86-0938	This DCP installed an air compressor and a high-pressure airline. The air compressor will be used to test respirators and SCBA units. It will also be used to fill the SCBA units.
4EC-1002/04	This DCP relocated the Source Range Monitor/Intermediate Range Monitor Preamp Panel, associated conduit and cable, and tubing, conduit, and cables for a differential pressure transmitter. This equipment interfered with the proposed Control Rod Drive Rebuild/Maintenance Facility.
4EC-1002/06	This DCP relocated junction boxes and associated devices under the vessel to make room for the Control Rod Drive Handling Machine Platform. This modification was in support of the Control Rod Drive Rebuild/Maintenance Facility.

DCP	Description of Design Change Package
4EC-1006	This DCP modified the interlocks on Residual Heat Removal valves to prevent the inadvertent draining of the Reactor Vessel to the Suppression Pool through the Shutdown Cooling Lines. The interlocks will prevent operators from opening the Residual Heat Removal Shutdown Cooling Valves when the Residual Heat Removal System is in the Shutdown Cooling Mode.
4EC-1030/01	This DCP installed rigging points for the removal of the "A" and "C" Core Spray Pumps and associated floor plugs. This will result in greater cost effectiveness when removing the pumps.
4EC-1030/04	This DCP installed rigging points for the removal of the "B" and "D" Core Spray Pumps and associated floor plugs. This will result in greater cost effectiveness when removing the pumps.
4EC-1030/06	This DCP provided permanent support lugs welded to the Drywell Personnel Airlock Barrel in the Reactor Building. It also provided a removable aluminum monorail bolted to the support lugs to expedite the removal of equipment and tools for maintenance. This will result in greater cost-effectiveness.
4EC-1030/08	This DCP provided permanent lifting arrangements for removing equipment and concrete floor plugs from the Reactor Building. This will result in greater cost-effectiveness.
4EC-1051	This DCP established continuous flow through the Off-Gas Radiation Monitoring System Sample Tank. This was accomplished by installing a vacuum pump to induce flow and to provide flow to overcome the water seal in the Sample Discharge Header to the condenser. This DCP makes a Temporary Modification permanent and continues to provide continuous flow through the Radiation Monitoring System Sampler, as required by Technical Specifications.

DCP	Description of Design Change Package
4EC-1057	This DCP added a control valve operator to each Chiller Condenser Turbine Auxiliaries Cooling System Outlet Butterfly Valves in the Turbine Building Chillers. A pressure controller will sense refrigerant pressure and provide a control signal to each valve operator to throttle Turbine Auxiliaries Cooling System cooling water through the chiller condensers. This will prevent chiller trips due to seasonal temperature fluctuations.
4EC-1058/01	This DCP installed fiberglass re-inforced plastic enclosures around the Service Water Travelling Screen components. The enclosures are utilized to eliminate water spray around the motor areas.
4EC-1075	This DCP relocated Heating, Ventilation, and Air Conditioning Moisture Sensors and Transmitters from the Computer Room to the Control Room Return Air Ducts. This modification will provide a more representative indication of the Control Room moisture level and improve the operability of the humidification/dehumidification systems.
4EC-1082/03	This DCP corrected discrepancies identified during the Control Room Design Review Process. The specific discrepancies corrected by this DCP are as follows: 1) the alarms in the Control Room were divided into sections, with each section having a separate sound, 2) the secondary condensate feed pumps and valves were sequenced the same way as the primary condensate feed pumps and valves, 3) installed a more descriptive push button configuration, and 4) corrected the Hydrogen-Oxygen Analyzer Recorder Scale.
4EC-1082/04	This DCP aligned the Intermediate Range Monitor recorders with the Intermediate Range Monitor range switches. This discrepancy was identified during the Control Room Design Review process and is part of a commitment to improve human factors in the Control Room.
4EC-1082/05	This DCP provided for indication of Reactor Vessel metal/flange temperatures in the Main Control Room. This information is used to monitor thermal stress and was identified as a necessity during the Control Room Design Review process.

DCP	Description of Design Change Package
4EC-1082/06	This DCP modified the Reactor Recirculation Flow Indicating Controller to make the indicated output "direct acting" with the valve position. The previous configuration required the operator to increase the station output to close the recirculation flow control valve. This discrepancy was identified during the Control Room Design Review process and is part of a commitment to improve human factors in the Control Room.
4EC-1082/07	This DCP provided a warning light in the Control Room to indicate when the Bailey Logic Cabinet has detected a "Containment High Pressure" and/or a "Reactor Vessel Level Low Low" signal and latched in a "half-trip" of the Primary Containment Isolation System Loss of Coolant Accident Level 2 Isolation Signal. The need for this warning light was identified during the Control Room Design Review process and is part of a commitment to improve human factors in the Control Room.
4EC-1085	This DCP extended the ductwork in the High Pressure Coolant Injection Pipe Chase Room to allow more efficient mixing of cooling air in the room. The previous maximum temperatures exceeded the FSAR maximum temperature commitments and have accelerated the rate of equipment degradation. This modification will rectify both of these problems.
4EC-1086	This DCP installed a demineralizer to the Reactor Auxiliaries Cooling System. This demineralizer will maintain low conductivity water to provide an acceptable corrosion treatment as recommended by General Electric.
4EC-1087	This DCP installed a demineralizer to the Safety and Turbine Auxiliaries Cooling System. This demineralizer will maintain low conductivity water to provide an acceptable corrosion treatment as recommended by General Electric.
4HC-0002	This DCP installed shutdown range reactor level indication on the Remote Shutdown Panel. This will provide the operator with accurate level indication during shutdown conditions and allows the operator to determine water level above +60", which is required for certain operational sequences.

DCP	Description of Design Change Package
4HC-0014	This DCP modified the Safety and Turbine Auxiliaries Cooling System Accumulators by installing a floating roof and a grating inside the accumulators and a diffuser at the Nitrogen Inlet Nozzle. The floating roof will act as a barrier between the nitrogen and the water inside of the accumulators, minimizing the mixing of water and nitrogen. The grating will serve as a vortex breaker and minimize turbulence inside of the accumulators. The diffuser will diffuse the nitrogen jet impinging upon the floating rod, therefore preventing an unbalanced force on the floating roof.
4HC-0026	This DCP modified the Reactor Water Clean Up System to preclude the low flow lock-out of the demineralizers with subsequent backwashing and precoating by ensuring holding pump flow is established prior to tripping the recirculation pumps and closing the isolation valves. This modification will reduce liquid radwaste inventory and processing costs and increase the reliability and availability of the Reactor Water Clean Up system.

The following Temporary Modification Requests (TMRs) 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.

None of the TMRs created a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These TMRs did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Safety Evaluation	Description of Temporary Modification Request (TMR)
88-0010	This TMR provided a temporary power source to a Non-1E 20kv Uninterruptable Power Supply Inverter during the maintenance outage of the "A" 4.16kv switchgear. This modification is to be in use only when the "A" 4.16kv switchgear is inoperable and the plant is in operational condition 5.
88-0011	This TMR provided a temporary power source to a Non-1E Battery Charger during the maintenance outage of the "B" 4.16 kv switchgear. This modification is to be in use only when the "B" 4.16 kv switchgear is inoperable and the plant is in operational condition 5.
88-0021	This TMR blocked off the smoke detectors in the Reactor Building, elevation 256' 6" so that normal operation of the Polar Crane did not cause false nuisance alarms. A Fire Watch was posted as a compensatory measure during this modification.
88-0022	This TMR provided a 480 volt Non-1E temporary power source to a 125 volt DC battery charger during the "A" 4.16 kv Class 1E bus outage. This modification is to be used only when "A" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0023	This TMR provided a 480 volt Non-1E temporary power source to a 125 volt DC battery charger during the "B" 4.16 kv Class 1E bus outage. This modification is to be used only when "B" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0025	This TMR provided a 480 volt Non-1E temporary power source to a 125 volt DC battery charger during the "D" 4.16 kv Class 1E bus outage. This modification is to be used only when "D" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0027	This TMR provided a 480 volt Non-1E temporary power source to a 125 volt DC battery charger during the "D" 4.16 kv Class 1E bus outage. This modification is to be used only when "D" Channel is inoperable and the plant is in operational conditions 4,5, and *.

Safety Evaluation	Description of Temporary Modification Request (TMR)
88-0028	This TMR provided a 480 volt Non-1E temporary power source to a 120 volt AC Class 1E Public Address System Inverter during the "A" 4.16 kv Class 1E bus outage. This modification is to be used only when "A" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0030	This TMR provided a 480 volt Non-1E temporary power source to a 120 volt AC Nuclear Steam Supply System Computer Inverter during the "D" 4.16 kv Class 1E bus outage. This modification is to be used only when "D" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0035	This TMR installed blind flanges for isolation to allow removal of a valve in the "B" Safety Auxiliaries Cooling System. The valve was removed for repairs and re-installed. At that time, the blind flanges were removed.
88-0038	This TMR provided temporary 120 volt AC power to a 24 volt DC Battery Charger during the "B" 4.16 kv Class 1E bus outage. This modification is to be used only when "B" Channel is inoperable and the plant is in operational condition 4,5 or *.
88-0046	This TMR provided a 480 volt Non-1E temporary power source to a Class 1E Fuel Pool Cooling Pump during the "A" 4.16 kv Class 1E bus outage. This modification is to be used only when "A" Channel is inoperable and the plant is in operational conditions 4,5, and *.
88-0057	This TMR provided a 480 volt Non-1E temporary power source to a 120 volt AC distribution panel during the "A" 4.16 kv Class 1E bus outage. This modification is to be used only when "A" Channel is inoperable and the plant is in operational conditions 4,5, and *.

Safety Evaluation	Description of Temporary Modification Request (TMR)
88-0065	The "A" Hydrogen Analyzer Calibration Gas Relief Valve leaks through, emptying the bottle. In order to prevent the loss of gas, the bottle could be valved closed. This would cause annunciation in the Control Room. This TMR installed a jumper across the low pressure switch that causes the annunciation when the gas bottle is valved closed.

The following Deficiency Requests (DRs) 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.

None of the DRs created a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These DRs did not change the plant effluent releases and did not alter the existing environmental impact. The Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Safety Evaluation	Description of Deficiency Report (DR)
88-0031	An Emergency Diesel Combustion Air High Temperature Switch was discovered to have a nicked signal wire, which caused the switch to create a ground. The repair process for this nicked wire consists of butt splicing a new wire to the switch and applying the Raychem process. This repair returns the switch to its original design.
88-0047	During the performance of a time response test on a Reactor Water Level Transmitter, the response time was outside the acceptable value. The transmitter may be used "as is" because its response time does not raise the response time for the loop above its Technical Specification Requirement.
88-0051	An ASME weld preheat of 69°F was used instead of the procedurally required 70°F. The welds may be used "as is" because this did not violate the applicable ASME code section.
88-0052	An ASME weld preheat of 63°F was used instead of the procedurally required 70°F. The welds may be used "as is" because this did not violate the applicable ASME code section.
88-0060	This DR deals with two instances of potentially lost parts. These parts are a glove, or a wad of tape, or a polyethylene bag, and a pushbutton from a switch. All of the potentially lost parts have been analyzed for the following: 1) the potential for fuel bundle flow blockage and the subsequent fuel damage, 2) the potential for control rod interference, and 3) the potential for corrosion or other chemical reaction with reactor materials. Safe reactor operation would not be compromised by the presence of the lost objects.