AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH December 1991

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	1080	17.	1070
2.	1063	18.	1055
3.	1052	19.	1073
4.	1058	20.	807
5.	1058	21.	836
6.	1058	22.	758
7.	953	23.	995
8.	658	24.	1061
9.	714	25.	1058
10.	625	26.	1061
11.	594	27.	1060
12.	620	28.	1051
13.	893	29.	1056
14.	1034	30.	1058
15.	1057	31.	1061
16.	1052		

OPERATING DATA REPORT

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/92
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period December 1991 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	744.0	7379.8	37,161.3
6. Reactor reserve shutdown hours	0.0	0.0	0.0
7. Hours generator on line	744.0	7281.5	36,574.6
8. Unit reserve shutdown hours	0.0	0.0	0.0
9. Gross thermal energy generated (MWH)	2,233,960	23,454,735	115,997,142
10. Gross electrical energy generated (MWH)	742,460	7,730,821	38,352,494
11. Net electrical energy generated (MWH)	711,146	7,394,865	36,651,549
12. Reactor service factor	100.0	84.2	84.2
13. Reactor availability factor	100.0	84.2	84.2
14. Unit service factor	100.0	83.1	82.9
15. Unit availability factor	100.0	83.1	82.9
16. Unit capacity factor (using MDC)	92.7	81.9	80.6
17. Unit capacity factor (Using Design MWe)	89.6	79.1	77.9
18. Unit forced outage rate	0.0	4.1	5.2

19. Shutdowns scheduled over next 6 months (type, date, & duration):
None

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

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/92
COMPLETED BY S. Hollingsworth
TELEPHONE (609) 339-1051

MONTH December 1991

1. Refueling information has changed from last month:

   Yes No X

2. Scheduled date for next refueling: 9/12/92

3. Scheduled date for restart following refueling: 11/11/92

4. A. Will Technical Specification changes or other license amendments be required?

   Yes No X

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

   Yes No X

   If no, when is it scheduled? not scheduled (on or prior to 7/24/92)

5. Scheduled date(s) for submitting proposed licensing action: N/A

6. Important licensing considerations associated with refueling:

   - Same fresh fuel as current cycle: no new considerations

7. Number of Fuel Assemblies:

A.	Incore	764
B.	In Spent Fuel Storage (prior to refueling)	760
C.	In Spent Fuel Storage (after refueling)	1008

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: 11/4, 2010 (EOC16)

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/92
COMPLETED BY V. Zabielski
TELEPHONE (509) 339-3506

MONTH December 1991

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
9	12/7	F	0	A	4	Condenser Air Leak due to crack in Steam Seal Evaporator Inlet Relief Piping.
10	12/20	F	0	A	4	Full Recirc Runback caused by failure the 'C' Primary Condensate Pump Lube Oil Supply Line.
11	12/23	F	0	A	4	Moisture Separator Leak.

Summary

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

December 1991

Hope Creek entered the month of December at approximately 100% power. On December 7, power was reduced because of a Condenser Leak caused by a crack in the Steam Seal Evaporator Inlet Relief Piping. Power was restored to approximately 100%. On December 20, there was a full Recirc Runback caused by a failure of the 'C' Primary Condensate Pump Lube Oil Supply Line, which reduced power. The unit remained at reduced power until a Moisture Separator Manway leak was repaired on December 23. On December 23, power was restored to approximately 100%. The unit operated for the remainder of the month without experiencing any shutdowns or any other reportable power reductions. On December [illegible], the plant completed its 234th day of continuous power operation. This surpasses the station's previous record of 221 days.

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

DECEMBER 1991

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.

DCP	Description of Safety Evaluation
4EC-3112/10	This DCP added conduit, power distribution equipment, and valve pit sump pump discharge piping to the yard area. This DCP also installs lighting, power distribution, and sump pumps with discharge piping in Service Water System Valve Pits. Heat Tracing with pipe insulation is being added to a fire protection line that was originally buried. The cathodic protection system components in a valve pit are being relocated to preclude damage during construction and maintenance of the valve pit. No Unreviewed Safety Questions were involved because it is all non-safety related equipment that does not interface with the Station Service Water System or any electrical system Technical Specification.
4EC-3115	This DCP replaced existing carbon steel piping in the Service Water System with new material of piping class HZD, 6% molybdenum stainless steel. The new piping is more resistant to corrosion and erosion. This DCP also added larger capacity Service Water Dewatering Drain Headers and a larger capacity Service Water Drain Tank to shorten the time required for dewatering and to preclude the manual cycling of the dewatering pump. The new Service Water piping has been designed to the criteria as the existing piping. The new piping is more resistant to corrosion and erosion. This DCP does not affect the safety-related function of the Service Water piping. Therefore, no Unreviewed Safety Questions were involved.
4EC-3204	This DCP installed a spring support in place of a vertical rigid pipe support in the High Pressure Coolant Injection System. Substituting the spring support in place of the rigid pipe support will allow pipe movement caused by thermal bowing during system warmup. No Unreviewed Safety Questions are involved with this DCP because the new analyzed pipe stresses are lower than the previous pipe stresses. Also, all piping, supports, and components are within the allowable tolerances.

DCP	Description of Safety Evaluation
4EC-3211	This DCP modified the start logic to the Deep Well Pumps so the Nuclear Department Administrative Building will not run out of fresh water during normal conditions. A new low level switch contact was added to provide a signal to the Deep Well Pump Start Control circuitry to start the lead Deep Well Pump and send a signal to open the Domestic Water Storage Tanks Supply Valves. This will reduce the number of Deep Well Pump starts and ensure that both the Domestic Water Storage Tanks and the Nuclear Department Administrative Building Fresh Water Storage Tank are being filled. This DCP does not have an adverse effect on the operation of any of the affected systems, neither does it involve any safety-related systems or equipment. Therefore, no Unreviewed Safety Questions were involved.
4HM-0627	This DCP installed a 1" drain line and associated valves in the 'B' Primary Containment Instrument Gas Compressor Test Return Suction Line. This test return line is only used during an 18 month surveillance test. The drain valves will be closed during normal operation. The drain line and associated valves are passive during the use of the test return line. Being in the closed position during the test will not affect the Loss of Coolant Accident isolation requirements of the test return line. This DCP does not impact the normal operation of the system; therefore, no Unreviewed Safety Questions were involved.

TMR	Description of Safety Evaluation
91-055	This TMR disabled a high temperature switch input to the Offgas Panel Trouble Annunciator in the Main Control Room. At periods of low flow, heat from the reheater raised the temperature above the setpoint even though the exit temperature from the cooler condenser was at design conditions. This resulted in a nuisance alarm that has been eliminated by this TMR. Moisture and temperature are monitored downstream of this temperature element at the reheater. If any moisture or temperature problems exist, they will be detected and alarmed at that point; therefore, no Unreviewed Safety Questions were involved.
91-056	This TMR authorizes the use of a 10 amp fuse in the 125VDC control power supply feed to the 250VDC High Pressure Coolant Injection System Motor Control Center. The vendor document indicates that a 30 amp fuse should be used. It is not apparent if the 10 amp fuse was supplied by the vendor or inadvertently installed after the Motor Control Center was placed in operation. An analysis was performed that showed the 10 amp fuse to be of an adequate size for the High Pressure Coolant Injection System to perform its intended safety functions. Therefore, no Unreviewed Safety Questions were involved.
91-057	This TMR authorizes the use of a 10 amp fuse in the 125VDC control power supply feed to the 250VDC Reactor Core Isolation Cooling System Motor Control Center. The vendor document indicates that a 30 amp fuse should be used. It is not apparent if the 10 amp fuse was supplied by the vendor or inadvertently installed after the Motor Control Center was placed in operation. An analysis was performed that showed the 10 amp fuse to be of an adequate size for the Reactor Core Isolation Cooling System to perform its intended safety functions. Therefore, no Unreviewed Safety Questions were involved.

DR	Description of Deficiency Report
HTE 91-196	The Reactor Building Lightning Mast toppled from its mounting on top of the Secondary Reactor Containment Dome. It remained lying horizontally on top of the building, supported by the surrounding railing until it was removed. Various roof areas that could be impacted by the lightning mast were analyzed and found to be able to sustain impact from the lightning mast. Probability analysis indicates that only one direct lightning strike to a critical building is expected to occur every 5 years. It has also been qualitatively concluded that there are no increased risks associated with the lack of the lightning mast for the duration required for replacement. Therefore, no Unreviewed Safety Questions are involved if the new lightning mast is installed prior to 5/31/92.

Procedure Revision	Description of Safety Evaluation
VHC.MD-GP.ZZ-0224(Q) Rev. 0	This procedure is a vendor procedure that controls the use of the Valve Operation Test and Evaluation System for the Residual Heat Removal Discharge to Radwaste Isolation Outboard Valve. This is a non-intrusive analysis system used to evaluate the performance of motor-operated valves. The Valve Operation Test and Evaluation System is a non-intrusive analysis system that does not require any alterations to the valve. There is no safety impact caused by attaching the force sensor to the yoke of the valve. Because no alterations are made to either the characteristics or the internals of the valve, this procedure does not involve an Unreviewed Safety Question.

Tech Spec

UFSAR Section	Description of Safety Evaluation
9.2.2.4	This UFSAR Change Notice addresses the use of the 1983 edition through the summer of 1983 addenda of ASME Section XI for the Hope Creek Inservice Testing Program. Previously, the UFSAR referenced the 1977 edition through the summer of 1978 addenda. No Unreviewed Safety Questions were involved because Hope Creek has complied with the 1983 edition and summer addenda as previously submitted to the NRC in the Safety Evaluation Report. The two codes are the same concerning the performance of inservice testing.