PSEG

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

Hope Creek Generating Station

January 15, 1993

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 December are being forwarded to you with the summary of changes, tests, and experiments for December 1992 pursuant to the requirements of 10CFR50.59(b). The remainder of the Design Changes implemented during the Fourth Refueling Outage are also being submitted in this report.

Sincerely yours,

J. J. Hagan
General Manager -
Hope Creek Operations

RAR:ld
Attachments

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INDEX

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/93
COMPLETED BY V. Zablelski
TELEPHONE (609) 339-3506

MONTH December 1992

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1.	1051	17.	1060
2.	1076	18.	1082
3.	404	19.	1073
4.	0	20.	1056
5.	74	21.	1053
6.	678	22.	1068
7.	1065	23.	1069
8.	1067	24.	1056
9.	1071	25.	1080
10.	1069	26.	1069
11.	1066	27.	1066
12.	1055	28.	1070
13.	1088	29.	1067
14.	1047	30.	1066
15.	1053	31.	1056
16.	1083		

OPERATING DATA REPORT

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/93
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

OPERATING STATUS

1. Reporting Period December 1992 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	705.8	7094.3	44,255.6
6. Reactor reserve shutdown hours	0.0	0.0	0.0
7. Hours generator on line	692.4	6930.3	43,504.9
8. Unit reserve shutdown hours	0.0	0.0	0.0
9. Gross thermal energy generated (MWH)	2,221,937	22,216,076	138,213,218
10. Gross electrical energy generated (MWH)	751,120	7,395,560	45,748,054
11. Net electrical energy generated (MWH)	718,254	7,050,835	43,702,384
12. Reactor service factor	94.9	80.8	83.7
13. Reactor availability factor	94.9	80.8	83.7
14. Unit service factor	93.1	78.9	82.2
15. Unit availability factor	93.1	78.9	82.2
16. Unit capacity factor (using MDC)	93.6	77.9	80.1
17. Unit capacity factor (Using Design MWe)	90.5	75.2	77.4
18. Unit forced outage rate	6.9	2.6	4.8

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

OPERATING DATA REPORT

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/93
COMPLETED BY V. Zabielski
TELEPHONE (609) 339-3506

MONTH December 1992

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
10	12/3	F	51.6	H	2	Contract employee bumped cart into a MCC, causing Reactor Recirculation Pump M/G Set Vent Fans to trip, resulting in a double Recirc Pump trip. Control Operator manually scrammed the Reactor LER 354/92-013

Summary

REFUELING INFORMATION

DOCKET NO. 50-354
UNIT Hope Creek
DATE 1/15/93
COMPLETED BY S. Hollingsworth
TELEPHONE (609) 339-1051

MONTH December 1992

1. Refueling information has changed from last month:

   Yes No X

2. Scheduled date for next refueling: 3/5/94

3. Scheduled date for restart following refueling: 4/23/94

4. A. 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?

   Yes No X

   If no, when is it scheduled? 2/18/94

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

6. Important licensing considerations associated with refueling:

   - Highly likely that will use same or similar fresh fuel as current cycle: no new considerations.

7. Number of Fuel Assemblies:

A.	Incore	764
B.	In Spent Fuel Storage (prior to refueling)	1008
C.	In Spent Fuel Storage (after refueling)	1232 to 1264

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)
   (does not allow for full-core offload)

HOPE CREEK GENERATING STATION

MONTHLY OPERATING SUMMARY

DECEMBER 1992

Hope Creek entered the month of December at approximately 100% power. On December 3, a contract employee bumped a cart into a Motor Control Center, causing the Reactor Recirculation Pump Motor Generator Set Ventilation Fans to trip, resulting in a double Recirculation Pump trip. The Control Operator then manually scrammed the reactor in accordance with plant procedures. The unit was returned to service on December 5. As of December 31, the plant had been on line for 26 consecutive days.

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

DECEMBER 1992

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-3002/01	This DCP replaced 2" schedule 80 pipe with 2" schedule 40 pipe. The pipe affected by this DCP is downstream of the Drywell Condensate Cooler Flow Elements and improves their functional capability by modifying the sensors drain lines. The pipe replacement enhances the performance of the flow elements. There is no other physical or functional interconnection with any operating equipment. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3008/03	This DCP replaced two Containment Atmosphere Control System Containment Isolation Valves with upgraded butterfly valves. The new butterfly valves are designed with metal seats for tight shutoff, extended wear, and minimum maintenance. The valves were supplied with upgraded actuators to accommodate the higher torque requirements. The components installed via this DCP were designed to the same criteria as the original design. The piping configuration and the moderate energy line design have not changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3008/05	This DCP replaced three Containment Atmosphere Control System Containment Isolation Valves with upgraded butterfly valves. The new butterfly valves are designed with metal seats for tight shutoff, extended wear, and minimum maintenance. The valves were supplied with upgraded actuators to accommodate the higher torque requirements. The components installed via this DCP were designed to the same criteria as the original design. The piping configuration and the high energy line design have not changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3111/01	This DCP replaced Service Water spool pieces with 6% molybdenum stainless steel spool pieces. It also replaced four butterfly valves, and added new operators, a drain connection, thermowells, and an inspection port. The new Service Water 'A' and 'C' pump discharge piping, thermowells, drain valves, connection, and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4EC-3111/02	This DCP replaced carbon steel piping in the Service Water System with 6% molybdenum stainless steel piping. The affected piping was the Service Water 'A' Safety Auxiliaries Cooling System Supply, Service Water Reactor Auxiliaries Cooling System Supply and Discharge, and the Service Water 'B' Safety Auxiliaries Cooling System Supply. The manual and motor operated butterfly valves were also replaced with upgraded butterfly valves and operators. This DCP also included minor piping configuration changes. The new Service Water piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3111/03	This DCP replaced carbon steel piping in the Service Water System with 6% molybdenum stainless steel piping. It also replaces the Safety Auxiliaries Cooling System Heat Exchanger Flow Balancing Valves and Isolation Valves. Additionally, minor pipe re-routing and valve relocation were included to eliminate cavitation. This DCP also provided for the yard isolation of the Service Water 'A' Safety Auxiliaries Cooling System and improved isolation and shutoff at the secondary containment penetration isolation valve. The new Service Water piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3111/04	This DCP diverted Service Water from the Cooling Tower Basin and Cooling Tower Bypass Line to a manhole in the yard. This DCP permitted the draining of the Cooling Tower Basin and piping in order to perform the Service Water System piping modifications. The emergency bypass line has been designed to provide the required Service Water flow through each of the Service Water and Safety Auxiliaries Cooling System loops and ensure that adequate cooling water is available for each loop. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4EC-3114/02	This DCP replaced 'A' Service Water Strainer Backwash spool pieces with 6% molybdenum stainless steel spool pieces. It also replaced a butterfly valve. The new 'A' Service Water Strainer Backwash discharge piping and valve were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3114/04	This DCP replaced 'C' Service Water Strainer Backwash spool pieces with 6% molybdenum stainless steel spool pieces. It also replaced a butterfly valve. The new 'A' Service Water Strainer Backwash discharge piping and valve were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3114/09	This DCP replaced 'A' and 'C' Service Water and Travelling Screens and Screenwash System spool pieces with 6% molybdenum stainless steel spool pieces. It also replaced two butterfly valves and re-routed booster pump suction piping. The new 'A' and 'C' Service Water Travelling Screenwash Spray Water Booster Pumps Suction Piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4EC-3114/13	This DCP replaced an 'A' Service Water Travelling Screen Spray Wash spool piece with a 6% molybdenum stainless steel spool piece. It also replaced a butterfly valve, instrument shut off valves, and flow instrumentation. The new 'A' Service Water Travelling Screenwash Piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3114/14	This DCP replaced a 'C' Service Water Travelling Screen Spray Wash spool piece with a 6% molybdenum stainless steel spool piece. It also replaced a butterfly valve, instrument shut-off valves, and flow instrumentation. The new 'C' Service Water Travelling Screenwash Piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3114/15	This DCP replaced a 'B' Service Water Travelling Screen Spray Wash spool piece with a 6% molybdenum stainless steel spool piece. It also replaced a butterfly valve, control valves, instrument shut-off valves, and flow instrumentation. The new 'B' Service Water Travelling Screenwash Piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The overall piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4EC-3114/16	This DCP replaced a 'D' Service Water Travelling Screen Spray Wash spool piece with a 6% molybdenum stainless steel spool piece. It also replaced a butterfly valve, control valves, instrument shut-off valves, and flow instrumentation. The new 'D' Service Water Travelling Screenwash Piping and replacement valves were designed to the same criteria as the original piping. The new material reduces the likelihood of leakage due to erosion or corrosion. The piping configuration as analyzed for the moderate energy line design has not been changed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3121/01 4EC-3121/02	This DCP installed a hardened torus vent pursuant to NRC Generic Letter 89-16. It also installed a radiation monitor, a flow sensor for the radiation monitor, a remote manual outboard isolation valve, an in-line rupture disc downstream of the isolation valve, and a flange downstream of the rupture disc. The installation of the hardened torus vent piping has no affect on Emergency Core Cooling System operability. The torus vent is a system upgrade to reduce the risk of overpressurizing the primary containment. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3145/01	This DCP installed two Control Rod Drive Friction Testing Stations. Each Friction Testing Station consists of two electrical solenoid valves, a differential pressure transducer, and an electrical control box. The Friction Testing Stations will reduce personnel radiation exposure. This DCP only affects the non-safety related operation of the Control Rod Drives and will not have any impact adverse on plant safety. The presence of the Friction Testing Stations will not have any adverse impact on the function of the Control Rod Drive System. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3182/09	This DCP changed the power supply short circuit protection of field wires on 1E instrument loops by replacing fuses with resistors. This DCP will increase the reliability of the transmitters and reduce the amount of maintenance required due to oxidation within fuse assemblies. Replacing fuses with resistors does not change the function or performance of the transmitters. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4EC-3292/01	This DCP replaced wire rope slings in the special lifting device for the Reactor Well plugs and the Dryer/Separator Pool plugs with synthetic slings. This DCP will result in a reduction of personnel radiation exposure during plug movement activities. The synthetic slings and associated hardware meet the NUREG 0612 single failure proof criteria and the UFSAR design criteria. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4EC-3347	This DCP replaced the spring pack in the actuator for selected motor operated valves. The spring pack, in conjunction with the actuator torque switches, provide a means to control and limit actuator output torque. The replacement of the spring pack allows the actuator to provide the required torque to stroke the valve while assuring the maximum torque does not exceed the limiting available torque value of the actuator. Changing the sizing of the spring pack does not change the function of the valve and does not reduce the provided margin of safety. The DCP enhances the valves performance by improving the regulation of the actuator's output torque between calculated limits. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0204/25	This DCP installed a test box and four two-position keylock selector switches into a Reactor Protection System Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of scram due to leads coming into contact with a Reactor Protection Circuit or a ground. The function of the test switches is the same as the existing test switches that are currently used in the Nuclear Steam Supply Shutoff System. This DCP adheres to all established design and installation standards applicable to the Reactor Protection System and the Nuclear Steam Supply Shutoff System. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4HC-0204/26	This DCP installed two test boxes and six two-position keylock selector switches into a Reactor Protection System Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of scram due to leads coming into contact with a Reactor Protection Circuit or a ground. The function of the test switches is the same as the existing test switches that are currently used in the Nuclear Steam Supply Shutoff System. This DCP adheres to all established design and installation standards applicable to the Reactor Protection System and the Nuclear Steam Supply Shutoff System. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0204/27 4HC-0204/28	Each of these DCPs installed two three-position keylock selector switches into a Residual Heat Removal/Core Spray Relay Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and connecting a temporary switch into the Core Spray pump circuit which reduces the risk of short circuit. The test switches installed via this DCP will improve surveillance procedure performance and reduce the risk of a lifted lead coming in contact with a with an adjacent circuit or ground. The DCP does not prevent the automatic function or cooling capacity of the remaining Core Spray pumps to provide cooling water to the reactor. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0204/29	This DCP installed one two-position keylock selector switch into a High Pressure Coolant Injection System Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of a High Pressure Coolant Injection System actuation due to leads coming into contact with an adjacent circuit or ground. The test switch performs the same isolation function as physically lifting a lead in the panel. This DCP increases the proficiency of performing the surveillance procedure and adheres to all established design and installation standards applicable to Engineered Safety Features/Emergency Core Cooling Systems. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4HC-0204/30	This DCP installed a relay and two two-position keylock selector switches into the Inboard Isolation Valve Relay Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of an actuation, short circuit, or electrical shock. The isolation function of these test switches is equivalent to the current method of lifting leads. The installation of the test switches does not affect the integrity of the valve closure circuits or inhibit the ability of the Residual Heat Removal valves to open or to be manually closed. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0204/31	This DCP installed a relay and seven two-position keylock selector switches into the Outboard Isolation Valve Relay Panel. Utilization of the selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of an actuation, short circuit, or electrical shock. The test switches will be utilized to isolate the automatic initiation signals to the closure circuitry of Residual Heat Removal valves to prevent isolation of the valves during the performance of surveillance procedures. The isolation function that these test switches provide is equivalent to the previous method of lifting leads. Therefore, this DCP does not involve an Unreviewed Safety Questions.
4HC-0204/32	This DCP installed three three-position keylock selector switches into a Residual Heat Removal/Core Spray Panel. Utilization of the new selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of a short circuit, electrical shock, or damage to the Residual Heat Removal or Core Spray Pump 'D' control circuits. The isolation function of these test switches is equivalent to the previous method of lifting leads. The ability of the Core Spray and Residual Heat Removal Systems to provide cooling water to the reactor in the event of an accident is not affected by this DCP. This DCP increases the proficiency of performing the surveillance procedure and adheres to established design and installation standards applicable to Engineered Safety Features/ Emergency Core Cooling Systems. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4HC-0204/33	This DCP installed three three-position keylock selector switches into a Residual Heat Removal/Core Spray Panel. Utilization of the new selector switches during surveillance testing eliminates the need for lifting leads and reduces the risk of a short circuit, electrical shock, or damage to the Residual Heat Removal or Core Spray Pump 'C' control circuits. The test switches installed by this DCP provide an equivalent method to lifting leads. The ability of the Core Spray and Residual Heat Removal Systems to provide cooling water to the reactor in the event of an accident is not affected by this DCP. This DCP increases the proficiency of performing the surveillance procedure and adheres to all established design and installation standards applicable to Engineered Safety Features/Emergency Core Cooling Systems. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0204/34	This DCP installed protective guards around pushbuttons on the Main Vertical Board, and the local Feedwater Pump Control Panels, as well as the pistol-grip Motor Generator Set Transfer Switch on the Control Rod Test Instrument Panel. The DCP will reduce the possibility of a unit scram or power reduction caused by inadvertent operation of the pushbuttons or switch. The protective guards are do not perform any function other than to reduce the possibility of an inadvertent operation of the pushbuttons or switch. They do not limit the capability of the pushbuttons or switch to perform their safety functions and are installed in accordance with seismic and human factors guidelines. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0205/08	This DCP installed two Calibration Volume Chambers, eight Calibration Volume Chamber Isolation Valves, and tubing to the high and low vent taps of feedwater flow transmitters. This reduces the potential for air pockets to form during the performance of calibration procedures. The Calibration Volume Chambers were installed with double isolation valves to assure that system pressure and temperature are not applied during normal plant operation. They were installed in accordance with the applicable design criteria for non-safety related instrumentation in the turbine building. Therefore, this DCP does not involve any Unreviewed Safety Questions.

DCP	Description of Design Change Package
4HC-0212/06	This DCP added a 100 gpm activated carbon filter and a mixed bed demineralizer system to the Chilled Water System. This DCP will improve the quality of Demineralized Water in the Chilled Water System. This DCP only affects the non-safety related section of the Chilled Water System. The Service Air and Demineralized Water load increases are intermittent and will not impede any safety functions of these systems. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0220/04	This DCP replaced four butterfly valves with upgraded valves. The valves are Safety Auxiliaries Cooling System Valves that are cross-tied to the 'A' Control Room Chillers and the 'A' Technical Support Center Chillers. The new valves have a better isolation capability. This DCP does not alter the function of the Safety Auxiliaries Cooling System. The new design is an improvement that minimizes the potential for an accident or malfunction. Therefore, this DCP does not involve any Unreviewed Safety Questions.
4HC-0234/01 4HC-0234/02 4HC-0234/03 4HC-0234/04	These DCPs remove temporarily installed Reactor Vessel Level Transmitters that were installed to collect data for troubleshooting spurious Loss of Cooling Accident/scram signals that were experienced during power ascension. The instrumentation that was removed was only used during power ascension. After the desired data was obtained, the instrumentation was electrically disconnected and valved out. The removal of the instrumentation does not affect the remaining operable level instruments. Therefore, these DCPs do not involve any Unreviewed Safety Questions.
4HE-0011/01 4HE-0011/02	These DCPs each removed a blind flange from between two valves in the Primary Containment Instrument Gas Header. A safety relief valve with an atmospheric discharge tail pipe was installed in place of the flange. These DCPs ensure the integrity of the piping system by protecting it with an overpressure protection device. The DCPs do not compromise the isolation actuation instrumentation channels or any safety related system or component. Therefore, these DCPs do not involve any Unreviewed Safety Questions.

TMR	Description of Safety Evaluation
92-037	This TMR removed the overload heaters from the breakers for the Reactor Water Cleanup Discharge to Condenser Valve and the Reactor Water Cleanup Discharge to Equipment Drain Valve. Removing the overload heaters from the breakers will prevent the valves from inadvertently opening during an Appendix R fire. Disabling these valves, along with the overhead annunciator, does not prevent their associated systems from performing their designed functions. Also, the UFSAR discusses the Appendix R requirement that the valves be disabled. Therefore, this TMR does not involve any Unreviewed Safety Questions.
92-038	This TMR installed electrical jumpers across the #2 Feedwater Heater High High Level Trip Switches. These switches cause spurious high level trip signals during low power levels. The jumpers were removed after the level signals stabilized. The Feedwater system is not safety related and is not required to be operable following a LOCA, other than for containment isolation. Failure of the Feedwater system does not compromise any safety related system or components. This TMR has no impact on the containment isolation function of the Feedwater system. Therefore, this TMR does not involve any Unreviewed Safety Questions.
92-039	This TMR changed the normal line-up of the test line for the Main Steam Isolation Valve Seal System return to the Primary Containment Instrument Gas System. It also added a hose to the drain leg of a test return line valve. The new hose will be used as a drain point. This line-up is controlled from the Control Room and all of the automatic functions of the valves are intact. The containment isolation valves for the Primary Containment Instrument Gas System and the Main Steam Isolation Valve Seal Steam Seal System are unaffected by this TMR. Therefore, this TMR does not involve any Unreviewed Safety Questions.

TMR	Description of Safety Evaluation
92-040	This TMR installed electrical jumpers across the High Bearing Oil Temperature Trip Switch for the 'A' Control Room Chiller. This TMR will allow the chiller to run with a defective module and/or thermistor until a replacement part can be procured and installed. The Control Area Chilled Water System is comprised of two 100% capacity redundant loops; therefore, the loss of any single component cannot result in a loss of cooling. Also, jumpering the trip circuit and providing for increased operator attention to oil temperature does not place the equipment in any additional jeopardy. Therefore, this TMR does not involve any Unreviewed Safety Questions.

Procedure Revision	Description of Safety Evaluation
HC.CH-TI.ZZ-0012(Q) Rev. 18	This procedure revision change consisted of the following changes: confirming samples and analyses are now required when a parameter exceeds an action level prior to initiating an "out-of-range" or "off-normal chemistry" notification, a section has been added for sampling and/or notification requirements for Action Statements, the In-Line Instrument Out of Service Form has been removed, and Action Levels for system parameters have been changed. 10CFR50.59 is applicable due to the changed specifications for the operation of the Fuel Pool Demineralizer. The Fuel Pool Demineralizer is to maintain an effluent SiO_2 of < 50 ppb. This limit was established prior to installing borosilicate spacers, which raised the influent levels, in the Fuel Pool. The stated limit is not reasonably achieved with the rated resin removal efficiencies. The proposed limit takes the removal efficiency into account to ensure acceptable demineralizer performance. Allowance is made for the fact that SiO_2 is a non-aggressive anion and no increased corrosion mechanism is expected. Therefore, this procedure revision did not involve Unreviewed Safety Questions.

UFSAR Section	Description of Safety Evaluation
Table 3.11-2	The table identifies safety-related systems and maximum conditions during which they may be required to operate or remain in a safe condition. The table is not referenced in the UFSAR text. The conditions are the maximum design basis event ambient temperatures, pressures, humidities, and radiological exposures in areas where the systems are located. The information contained in this table is provided in design documents in a more useful format. Therefore, this table is being deleted. No changes were made to any equipment. There is no effect on any system operation. The only change is to the UFSAR, no physical work has been performed. Therefore, this UFSAR change does not involve an Unreviewed Safety Question.
9.2.2.2	This UFSAR change documents that the use of demineralizers is the only method in use to minimize corrosion in the Safety and Turbine Auxiliary Cooling System. Based on past industry operating experience, the use of demineralized water has been recommended over chemical addition and provides sufficient corrosion inhibition. Therefore, this UFSAR change does not involve an Unreviewed Safety Question.