PSEG

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

Nuclear Business Unit

March 14, 1996

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Dear Sir:

MONTHLY OPERATING REPORT
HOPE CREEK GENERATION STATION UNIT 1
DOCKET NO. 50-354

In compliance with Section 6.9, Reporting Requirements for the Hope Creek Technical Specifications, the operating statistics for **February 1996** are being forwarded to you with the summary of changes, tests, and experiments that were implemented during **February 1996** pursuant to the requirements of 10CFR50.59(b).

Sincerely yours,

Mark Reddemann
General Manager -
Hope Creek Operations

DL:DS:CC
Attachments

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INDEX

DOCKET NO.: 50-354
UNIT: Hope Creek
DATE: 3/5/96
COMPLETED BY: D. W. Lyons
TELEPHONE: (609) 339-3517

OPERATING DATA REPORT
OPERATING STATUS

1. Reporting Period February 1996 Gross Hours in Report Period 696.
2. Currently Authorized Power Level (MWt) 3293
   Max. Depend. Capacity (MWe-Net) 1031
   Design Electrical Rating (MWe-Net) 1067
3. Power Level to which restricted (if any) (MWe-Net) None
4. Reasons for restriction (if any)

		This Month	Yr To Date	Cumulative
5.	No. of hours reactor was critical	0.0	0.0	66923.9
6.	Reactor reserve shutdown hours	0.0	0.0	0.0
7.	Hours generator on line	0.0	0.0	65941.6
8.	Unit reserve shutdown hours	0.0	0.0	0.0
9.	Gross thermal energy generated (MWH)	0.0	0.0	210774249
10.	Gross electrical energy generated (MWH)	0.0	0.0	69825622
11.	Net electrical energy generated (MWH)	0.0	0.0	66701000
12.	Reactor service factor	0.0	0.0	83.0
13.	Reactor availability factor	0.0	0.0	83.0
14.	Unit service factor	0.0	0.0	81.8
15.	Unit availability factor	0.0	0.0	81.8
16.	Unit capacity factor (using MDC)	0.0	0.0	80.3
17.	Unit capacity factor (using Design MWe)	0.0	0.0	77.5
18.	Unit forced outage rate	0.0	0.0	5.1

19. Shutdowns scheduled over next 6 months (type, date, & duration):
    Currently shutdown for Refueling Outage, RF06, began November 11, 1996
20. If shutdown at end of report period, estimated date of start-up:
    Criticality is scheduled for March 12, 1996
    Breaker closure is scheduled for March 15, 1996

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OPERATING DATA REPORT
UNIT SHUTDOWNS AND POWER REDUCTIONS

MONTH FEBRUARY 1996

NO.	DATE	TYPE F=FORCED S=SCHEDULED	DURATION (HOURS)	REASON (1)	METHOD OF SHUTTING DOWN THE REACTOR OR REDUCING POWER (2)	CORRECTIVE ACTION/COMMENTS
1.	02/01/96, 0000 hrs to 02/26/96, 1200 hrs	S S S S	11/95 - 478.4 12/95 - 744.0 01/96 - 744.0 02/96 - 612.0	C	4	Refueling Outage
2	02/26/96, 1200 hrs - end of period (still shutdown)	F	02/96 - 84.0	C	4 Unit was shutdown in November 1995	This is an unplanned (by INPO definition) forced extension of the Refueling Outage

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AVERAGE DAILY UNIT POWER LEVEL

MONTH FEBRUARY 1996

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	0
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	0
6	0	22	0
7	0	23	0
8	0	24	0
9	0	25	0
10	0	26	0
11	0	27	0
12	0	28	0
13	0	29	0
14	0	30	N/A
15	0	31	N/A
16	0		

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REFUELING INFORMATION

MONTH FEBRUARY 1996

1. Refueling information has changed from last month:

   Yes X No _

2. Scheduled date for next refueling: 4/5/97 (a)

3. Scheduled date for restart following refueling: 6/4/97 (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 FEBRUARY 1996

The Hope Creek Generating Station remained off-line the entire month of February 1996 for the sixth refueling outage. This resulted in total energy losses of 777780 MWHRS. As of February 29, 1996 the unit had been off-line for 111 days.

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

MONTH FEBRUARY 1996

The following items have been evaluated to determine:

1. If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or
2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or
3. If the margin of safety as defined in the basis for any technical specification is reduced.

The 10CFR50.59 Safety Evaluations showed that these items did not create a new safety hazard to the plant nor did they affect the safe shutdown of the reactor. These items did not change the plant effluent releases and did not alter the existing environmental impact. The 10CFR50.59 Safety Evaluations determined that no unreviewed safety or environmental questions are involved.

Design Changes Summary of Safety Evaluations

- **4EC-3449; PACKAGES 1 - 5, REPLACEMENT OF DEGRADED PIPE AND FITTINGS WITH CHROME ALLOY STEEL** This design change replaces various non-safety related carbon steel pipe fittings and segments with chrome alloy steel materials. Some replacements will have heavier wall material. The systems affected include Feedwater, Main Steam, Condensate, Extraction Steam, Seal Steam, Heater Drains, Moisture Separator Drains and Condenser Air Removal.. The replacement piping and materials have been designed to meet or exceed the design requirements of the original specifications. Except for the material, the replacements are identical in form, fit and function to the existing fittings. The UFSAR will be revised to reflect this change in material. This change will return the plant to its design configuration and restore excess design margin. It will reduce the likelihood of leakage due to erosion/ corrosion damage and thus will enhance the reliability of the affected systems. Because overall piping configuration as analyzed has not been changed, no additional postulated pipe breaks need to be considered. The installation and testing of the change was performed to the same criteria as the existing piping.

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

- **4EC-3500; PACKAGE 1, MAIN TURBINE TRIP LOGIC** This design change modifies the Level 8 High Water Trip logic for the Main Turbine. The change revises and/ or eliminates parts of the circuit where a single failure could cause or prevent a trip. It does this by revising the trip logic from a two out three trip logic that uses normally closed contacts that open to cause a main turbine trip to a two out of three logic that uses normally open contacts that close to cause a main turbine trip. It, also, eliminates an interposing relay and a confirmatory contact both from Bailey system that have postulated failures that could cause or prevent a main turbine trip. In the new logic there is no single failure that can either cause or prevent a main turbine trip. The removal of the Bailey components and the revision to the logic to eliminate the possibility of a trip prevented by single failures provide net gains in reliability and safety. The functionality of the OPS Test Panel is, also, improved as the circuits for this had to be redesigned. Because the main turbine trip circuit is more reliable and has less components there is no increase in the probability or consequences of a malfunction of equipment or an accident previously evaluated in the UFSAR. This change provides a Main Turbine Trip Circuit in which no single failure or component in test will result in a trip nor will it prevent a valid trip signal from actuating the trip coil.

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

Design Changes Summary of Safety Evaluations (continued)

- **4EC-3528; PACKAGE 1, CONDENSATE PRE-FILTER TIE-INS** The installation of a Condensate pre-filter to reduce the amount of iron in systems at Hope Creek will be accomplished in two phases. The Phase 1, encompassed by this design change, is the outage related installation of required pre-filter system tie-ins to the Condensate and supporting systems. Phase 2 is the non-outage installation of the pre-filters and supporting equipment. This package includes tie-ins to the Condensate, Condensate demineralizer and compressed air system piping, relocation of mechanical and electrical commodities which currently interfere with the proposed pre-filter skid equipment path, and addition of a shield wall to minimize radiation exposure in the turbine generator pedestal foundation area. By implementing these portions during RF06, the remainder of the system can be installed non-outage and can be available for use prior to RF07.

  Major items accomplished include:

  - Rerouting of the 32" Condensate line and adding two 36" tap connections and a 36" bypass line each with a manual gate valve for shutoff.
  - Addition of a taps and isolation valves on the Service Air header and Condensate Demineralizer Waste header for future use with the Condensate Pre-Filter system.
  - Addition of the shield wall at the south end of Condenser C.
  - Rerouting and relocating equipment and wiring associated with the Secondary Condensate pump and Chilled Water system.

  UFSAR Figures 9.3-3, sheet 1; 9.2-14, sheet 1; 10.4.5, sheet 1; 12.3-16, sheet 1; 12.3-17, sheet 1; 12.3-52, sheet 1; 12.3-53; 12.3-62; 12.3-63 will be revised to reflect this change.

  All systems and components affected by this change are classified as not important to safety, are in non-safety related systems and are located rooms in the Turbine Building that do not contain any safety-related equipment, therefore there are no credible failure modes which impact plant safety. None of the systems or components affected by this change are identified as a cause for anticipated operational transients or design basis accidents in the UFSAR. The design of the modifications is in accordance with the original design requirements of the affected systems, structures and components (SSCs). Analysis has been performed to verify that the structural integrity and functional requirements of the SSCs are not adversely affected by the rerouting of piping and wires, changing of pipe sizes and other physical rearrangements.

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

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

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

  PACKAGE 07, HPCI SYSTEM VALVES, 1BJHV-F006 (V-001) AND 1BJHV-8278 (V-059), DISC WEEP HOLE This Design Change Package installs weep holes in the reactor side of the discs of 1BJHV-F006 (V-001) and 1BJHV-8278 (V-059). UFSAR Figure 6.3-1 will have a note added indicating the valve is unidirectional because of the weep hole on the reactor side.

  Therefore, this package of the DCP does not increase the probability or consequences of an accident previously described in the UFSAR and does not involve any Unreviewed Safety Questions.

- **4HE-0097; PACKAGE 04, TURBINE BUILDING FIRE HOSE STATION VALVES** This design change replaces the adjustable pressure restricting valves on the 78 fire hose stations in the turbine building. The replacement was done to provide Site Protection with the maximum allowable pressure at each hose station. The increase in available pressure and flow will be maintained throughout the entire standpipe system. There is no change to the Fire Protection program or the ability of the facility to achieve and maintain safe shutdown in the event of a fire. The new valves meet all design, material, and construction standards applicable to fire hose stations. Because the replacement valves do not change the functionality of the fire hose stations and meet all the design requirements, they do not degrade or prevent actions assumed in the UFSAR.

  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-0126; PACKAGE 01, ADDING FUSES TO MASTER TRIP SOLENOID COILS A AND B** This design change adds fuses in line with the 24 VDC power supply to the Master Trip Solenoid coils A and B. This will prevent interruption of 24 VDC power to other Electro-Hydraulic Control system circuits in case either of the two Master Trip Solenoid coils is short circuited. UFSAR Figure 10.2.11 will be revised to reflect this change. A blown fuse is the only credible failure mode. The blown fuse will protect the wiring from overheating and the power supply from interruption. Because this change does not affect the operability of the Master Trip Solenoid coils, there are no operational transients or postulated design basis accidents applicable to 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-0140; PACKAGE 01, SSE DRAIN LINE 1-CA-020 CHANGE MATERIAL TO STAINLESS STEEL** This design change replaces carbon steel line 1-CA-020 of the Steam Seal Evaporator (SSE) with stainless steel piping. The line provides a constant blowdown from the SSE to the main condenser. The stainless steel piping will provide greater corrosion and erosion resistance. This change does not introduce any new failure modes or change existing failure modes of the Steam Seal system. This change does not change the operation of or parameters for the Steam Seal system. The Steam Seal system does not have any safety-related functions nor does its failure compromise any safety-related system or component or prevent the safe shutdown of the plant.

  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-0193; PACKAGE 1, HPCI TURBINE OIL RESERVOIR DRAIN** This design change replaces the drain plug on the HPCI turbine oil reservoir with approximately 3 feet of 1 inch carbon steel piping, a lock closed globe valve and a threaded end cap. The piping and valve will be designed, procured and installed in accordance with ASME Section III, Class 2 requirements. The assembly is Seismic Category I. These requirements meet or exceed the requirements for existing auxiliary piping and components on the HPCI pump turbine skid. The valve may be used for maintenance activities involving oil changes, drawing samples, or draining accumulated moisture to prevent buildup. UFSAR Figure 6.3-2 will be revised to show this change. This change extends a passive boundary for retention of the oil supply for the HPCI turbine. Since none of the accidents evaluated in the UFSAR are initiated by a loss of oil from the HPCI turbine, this activity will not increase the probability of occurrence of an accident. The probability of occurrence of a loss of turbine oil, the only malfunction considered applicable to this change, is not increased. This is because the additional surface area is small compared to the existing reservoir, the piping is designed in accordance with the original specifications, and there are two closures on the end of the pipe, a valve and an end cap. The consequences of a loss of oil would not be increased.

  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-0202; PACKAGE 1, INSTALL CHECK VALVE (1ACV-068) ON MT SEALING STEAM FIRST LEAKOFF LINE (1AC-1-1/2"-VBT-060) FROM BYPASS CONTROL VALVES (1ABXV-1007 A THROUGH J) TO MT CROSS-AROUND HEADER (1AC-42"-VFT-020)** This design change installs a single lift check valve on the Main Turbine Sealing Steam Common First Leakoff Line. The Common First Leakoff Line discharges main steam from the stems and bushings of the Bypass Control Valves to one of the Main Turbine Cross Around Headers. The new valve prevents an undesired backflow of steam from that cross-around header through the Bypass Control Valves to the Main Condenser when the valves are closed. The backflow causes wear on valve stems and bushings. The Turbine Bypass system has no safety-related function nor does failure of the system compromise any safety-related system, structure or component (SSC) or prevent a safe shutdown of the plant. The only credible failures are for the new check valve to either fail to open or to seat. If the new check valve fails to seat, the backflow that currently exists will continue and will cause wear on the valve stems and bushings. Since the Bypass valves are rarely opened and the chance of the check valve failing closed is unlikely, the probability of both conditions existing is even more unlikely. Installation of the check valve will minimize wear on the stems and bushings of the Bypass Control valves. There is no credible malfunction of equipment important to safety considered applicable to this proposal.

  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-0219; PACKAGE 1, FUEL POOL GATE SEALS (CONNECTING SPENT FUEL STORAGE POOL TO REACTOR WELL)** This design change replaces the existing flexible hose and associated fittings downstream of valve 1KAV7068, with stainless steel tubing and fittings. The tubing and fittings are located inside Service Box #8 adjacent to the Spent Fuel Pool at Elevation 201'. The tubing and fittings are classified as non-seismic since once installed inside the service box they are incapable of generating gravitational missiles. There are no safety-related consequences associated with the postulated rupture of the tubing and fittings described in the UFSAR. UFSAR Figure 9.1-5, sheet 2 of 2, will be revised to reflect this change. Because the new stainless tubing and fittings have improved wear resistance and are more suitable for the intended range of service conditions, the integrity of the pressure boundary is improved which enhances the reliability of the gate seals. The improved pressure boundary integrity reduces the potential for loss of air from the gate seals due to failure of hose and fittings. Failure of the gate seals is bounded by the accident analyses in Chapter 15 of the UFSAR. The proposed modification does not alter the operating characteristics of the gate seals.

  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-0224; PACKAGES 1 - 8, MSIV OPEN LIMIT SWITCH CALIBRATION** This design change modifies the setting for Limit Switch LS1 on all eight Main Steam Isolation Valves (MSIVs), 1ABHV-F022A, B, C, and D AND 1ABHV-F028A, B, C, and D from the existing settings of 90% open to the new setting of 87% open. These LS1 settings are only used for Control Room indication of MSIV movement. UFSAR Section 5.4.5 will be revised to reflect this change. Also, several General Electric (GE) Elementary Diagrams specifically listed in UFSAR Tables will be revised. The new setting is required to ensure through partial stroke testing that the MSIVs are not mechanically bound and are capable of closure. GE recommends a minimum of 1" valve stem travel during partial stroke testing to ensure the valve is capable of closure. This change which is patterned off a change at another GE BWR provides a minimum of 1.43" valve stem travel. This change does not affect any system, design or operational parameter. MSIV performance parameters and operational characteristics are not affected by the new LS1 settings. No credible failure modes are associated with this change. Plant control systems are not impacted by the new setting. None of the operational transients and postulated accidents of the UFSAR Section 15 are applicable to this change. Revising the LS1 settings provides positive Control Room indication that the MSIVs are not mechanically bound and are capable of closure.

  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-0239; PACKAGE 01, ABB CONTROL ROD MATERIAL CHANGE** This design change changes the material used in control rod blades from SS304L to SS 316L. The SS 316L has higher resistance to intergranular stress corrosion cracking (IGSCC) and irradiation assisted stress corrosion cracking (IASCC) than the SS 304L. An added benefit comes from the lower cobalt content of the SS 316L material. This will reduce the amount of gamma radiation in the plant from the Cobalt-60 isotope. UFSAR Section 4.6.2.3.1.1 will have to be revised and a reference added to several other sections to reflect the change in material. The existing procedures on friction testing and scram timing are applicable to the ABB Control Rods. The volume of the control rod handle is one of the parameters needed to calculate the maximum subcritical banked withdrawal position for the Emergency Procedure Guideline (EPG). The volume of the ABB Control Rod handle is conservative relative to the value used in the existing calculations. This material change will not affect any systems or parameters since the geometry, dimensions, neutronics properties and rod worth of the improved ABB Control Rods are the same as those being replaced. The overall rod worth and power shaping of the control rods was evaluated. The new Control Rods have a higher rod worth and no unacceptable power distribution was observed.

  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-0263; PACKAGE 01, RIVER WATER LEVEL INSTRUMENTATION CHANGEOUT** This design change will permanently install the instrument used to monitor the river level just prior to entering the Service Water Structure currently in place at 0EPLIT-2220-2 under a Temporary Modification and install the same type of device in 0EPLIT-2220-1 to. This change will provide the Control Room Operators with a reliable means to measure the river level. UFSAR Figure 9.2-2 will be revised to reflect this change. In the event the instruments fail the Technical Specification requirements to monitor the river level could be met using other methods. This is a change in manufacture and model only, the new components are of the same type and have the same output but with increased accuracy as the old units. These instruments are non-Q and do not provide any control functions nor are they connected to any safety-related equipment.

  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-0268, PACKAGE 1, FEEDWATER HEATER DRAIN CAPS** This Design Change Package replaces high energy, non-safety-related, non-seismic vent and drain pipes from the 3A, 4A, 5A, B & C, and 6A Feedwater Heaters (FWH) and Condensate System piping to their respective equipment drains with capped sections of piping. UFSAR Figure 10.4-5 will be revised to reflect this change. The change reduces the length of the lines but does not alter the root valve design, operating characteristics or the size and service conditions in the vent and drain lines. Each modified line remains within its FWH Room inside the Turbine Building and does not adversely affect the existing UFSAR assessments of postulated pipe ruptures and their consequences. Since the caps are not pressure boundary components, their failure is assume to generate missiles. Because all of the caps are located within FWH rooms any missile generated cannot affect safety-related structures, systems, components or functions in accordance with the existing UFSAR transient and accident analyses. Any leakage will continue to be processed as potentially radioactive consistent with the existing design. The reconfigured lines do not adversely affect the equipment and floor drains, Turbine Building HVAC Systems, radiation monitoring and protection features.

  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-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 01, REWORK CLOSED MOV POSITION INDICATION ON 1ABHV-F016 This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Minimum Flow Valve 1ABHV-F016. The torque switch is shown in UFSAR Figure 7.3-4, Sheet 1 of 3, which will be revised to reflect this change. The valve has a safety-related function to close to provide inboard primary containment isolation of the Main Steam Isolation valve above seat drain lines. The valve is open to drain the main Steam Isolation Valve above seat drain area until approximately 15% load on reactor start up. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 25, REWORK CLOSED MOV POSITION INDICATION ON 1BCHV-F017D This Design Change Package installs the limit switch modifications on the Residual Heat Removal System (RHR) Loop "D" Low Pressure Coolant Injection (LPCI) Injection Valve, 1BCHV-F017D. The torque switch is shown in UFSAR Figure 7.3-8, Sheet 9 of 13, which will be revised to reflect this change. The valve is a primary containment isolation valve and a reactor coolant system pressure isolation valve. The valve is normally closed. It receives an automatic ECCS actuation signal to open. The modification will not prevent the MOV from performing its safety function.

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

PACKAGE 42, REWORK CLOSED MOV POSITION INDICATION ON 1BGHV-F001 This Design Change Package installs the limit switch modifications on the Reactor Water Clean Up System inlet inboard isolation valve, 1BGHV-F001. The torque switch is shown in UFSAR Figure 7.3-10, which will be revised to reflect this change. The valve is a Containment Isolation Valve. This valve is opened and/or verified open during plant start up. The valve is open during all normal plant operating modes. The valve will automatically close in response to a number of events. This closure limits the loss of reactor coolant, the release of radioactive material, and prevents the removal of liquid reactivity control material if SLC is operating. The modification will not prevent the MOV from performing its safety function.

PACKAGE 43, REWORK CLOSED MOV POSITION INDICATION ON 1BGHV-F004 This Design Change Package installs the limit switch modifications on the Reactor Water Clean Up System inlet outboard isolation valve, 1BGHV-F004. The torque switch is shown in UFSAR Figure 7.3-10, which will be revised to reflect this change. The valve is a Containment Isolation Valve. This valve is opened and/or verified open during plant start up. The valve is open during all normal plant operating modes. The valve will automatically close in response to a number of events. This closure limits the loss of reactor coolant, the release of radioactive material, and prevents the removal of liquid reactivity control material if SLC is operating. The modification will not prevent the MOV from performing its safety function..

PACKAGE 46, REWORK CLOSED MOV POSITION INDICATION ON 1BJHV-F012 This Design Change Package installs the limit switch modifications on the High Pressure Coolant Injection (HPCI) Pump minimum flow bypass isolation valve, 1BJHV-F012. The torque switch is shown in UFSAR Figure 7.3-2, Sheet 2 of 6, which will be revised to reflect this change. The valve is normally closed and is a Containment Isolation Valve. It opens during HPCI initiation when the HPCI pump discharge pressure is 125 psig and HPCI pump discharge flow is less than 350 gpm. When HPCI flow exceeds 350 gpm the valve automatically closes. Valve, also, automatically closes if either the HPCI Turbine stop valve or the HPCI Turbine steam supply valve is full closed. The modification will not prevent the MOV from performing its function.

PACKAGE 47, REWORK CLOSED MOV POSITION INDICATION ON 1BJHV-8278 This Design Change Package installs the limit switch modifications on the High Pressure Coolant Injection (HPCI) Pump discharge to feedwater isolation valve, 1BJHV-8278. The torque switch is shown in UFSAR Figure 7.3-2, Sheet 2 of 6, which will be revised to reflect this change. The valve is normally closed and is a Containment Isolation Valve. There are no normal operating conditions when the valve would be operated. The safety-related function of this valve is to open automatically upon HPCI initiation. The modification will not prevent the MOV from performing its safety function.

PACKAGE 53, REWORK CLOSED MOV POSITION INDICATION ON 1FDHV-F002 This Design Change Package installs the limit switch modifications on the High Pressure Coolant Injection (HPCI) turbine "C" steam supply inboard isolation valve, 1FDHV-F002. The torque switch is shown in UFSAR Figure 7.3-2, Sheet 4 of 6, which will be revised to reflect this change. The valve is located inside the containment and is a primary containment isolation valve. This valve is opened during plant start up when the HPCI Steam Supply Low Pressure isolation can be reset to supply steam from the "C" Main Steam Line to the HPCI Turbine. The valve remains open during normal operation. The valve shuts upon receipt of a HPCI automatic isolation signal. The modification will not prevent the MOV from performing its safety function.

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

  PACKAGE 54, REWORK CLOSED MOV POSITION INDICATION ON 1FDHV-F003 This Design Change Package installs the limit switch modifications on the High Pressure Coolant Injection (HPCI) turbine "C" steam supply outboard isolation valve, 1FDHV-F002. The torque switch is shown in UFSAR Figure 7.3-2, Sheet 4 of 6, which will be revised to reflect this change. The valve is located outside the containment and is a primary containment isolation valve. This valve is opened during plant start up when the HPCI Steam Supply Low Pressure isolation can be reset to supply steam from the "C" Main Steam Line to the HPCI Turbine. The valve remains open during normal operation. The valve shuts upon receipt of a HPCI automatic isolation signal. The modification will not prevent the MOV from performing its safety function.

  PACKAGE 57, REWORK CLOSED MOV POSITION INDICATION ON 1KPHV-5835A This Design Change Package installs the limit switch modifications on the Main Steam Isolation Valve (MSIV) Inboard Seal Gas Supply Valve, 1KPHV-5835A. The torque switch is shown in UFSAR Figure 7.3-17, Sheet 1 of 3, which will be revised to reflect this change. This valve is a normally closed containment isolation valve. The valve may be remote manually opened following a LOCA to reduce untreated leakage of the containment atmosphere resulting from possible loss of MSIV leaktightness. The valve receives an automatic containment isolation signal. There is no automatic opening signal for this valve. The modification will not prevent the MOV from performing its function.

  PACKAGE 58, REWORK CLOSED MOV POSITION INDICATION ON 1KPHV-5836A This Design Change Package installs the limit switch modifications on the Main Steam Isolation Valve (MSIV) Inboard Seal Gas Supply Valve, 1KPHV-5836A. The torque switch is shown in UFSAR Figure 7.3-17, Sheet 1 of 3, which will be revised to reflect this change. This valve is a normally closed containment isolation valve. The valve may be remote manually opened following a LOCA to reduce untreated leakage of the containment atmosphere resulting from possible loss of MSIV leaktightness. The valve receives an automatic containment isolation signal. There is no automatic opening signal for this valve. The modification will not prevent the MOV from performing its function.

UFSAR Change Notices Summary of Safety Evaluations

- **UFSAR CHANGE NOTICE CN 96-10, STACS & RACS MINIMUM DESIGN TEMPERATURE REDUCTION** This change notice lowers the Safety and Turbine Auxiliary Cooling System (STACS) and Reactor Auxiliaries Cooling System (RACS) minimum design temperatures from 65°F to 32°F and 40°F respectively. UFSAR Sections 1.8.1.97.5.7 and 9.2.2.2 and Tables 6.2-2, 7.5-1, 9.2-3, 9.2-9 will be revised to reflect this change. All equipment and components cooled by these systems were identified. Then the design and material specifications of the equipment and components and the operational affects on the processes were reviewed for the impact of the lower temperatures. All STACS and RACS are capable of performing their design functions without adverse impact from the lower temperatures. The lower temperature will not cause service induced degradation above and beyond the previous temperature. The STACS and RACS will still be able to perform their intended design functions. There are no credible failure modes associated with this change. Over 30 events from UFSAR Tables 15.9-2 and 15.9-4 were considered applicable to this proposal and reviewed for the affect of this change. This change does not compromise any UFSAR accident analysis assumptions or design limits. This change does not modify any equipment or components. This change does not cause the initiation of any accident nor create any new failure mechanisms.

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

- **UFSAR CHANGE NOTICE CN 96-12, QUALITY ASSURANCE DURING THE OPERATIONS PHASE** This change notice revises UFSAR Chapter 17.2, Quality Assurance During the Operations Phase, to reflect the current organization structure and responsibilities. It, also, provides editorial enhancements and clarifications. Procedures and functions described in Sections 13.1, 13.4, and 17.2 and Technical Specification sections 6.5 and 6.8 will be affected. The changes are administrative in nature and have no affect on the plant's design basis, its equipment, malfunctions of equipment, failure modes, operational transients or postulated design basis accidents, assumptions contained in the plant safety analysis, or the physical design and/ or operation of the plant.

  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# 95-006, TEMPORARY TEMPERATURE CONTROL FOR THE "B" SERVICE WATER PUMP ROOM AND THE SERVICE WATER INTAKE SCREEN AREA** This Temporary Modification replaced the four permanently installed 15 kW heaters in the Traveling Screen Motor Room with two 30 kW heaters during a non-safety related bus outage. The temporary heaters were powered from a portable generator and did not affect the plant's electrical system. Calculation GQ-4 establishes a heat load for the room of 47 kW. The heaters in the Service Water Pump Room were, also, lost during the bus outage. No temporary heaters were required in that area because the heat generated by one running pump was sufficient to maintain the required temperature. Calculation GQ-3 evaluated this condition. The Service Water Intake Structure ventilation system supports the operation of the SW system by maintaining design basis temperatures. It has no direct impact on any operational transients or postulated design basis accidents. The ability of the SW system to mitigate the consequences of an accident was not affected. Failure of the temporary heaters would have affected the operable SW loop in the same manner as failure of the permanently installed heaters. This Temporary Modification was removed prior to the end of the period.

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

Deficiency Report Summary of Safety Evaluation

- **DEFICIENCY REPORT 951215101, SERVICE WATER TANK (10-T-544) REPAIR UNDER WORK ORDER #951215101** This Deficiency Report disposition allows for the removal of the Station Service Water Intake Structure (SWIS) pump room B & D supply and exhaust fans from service while work is performed on the SWIS head tank. The ventilation will be under administrative control during this time. This allows the SW system to remain operable during the repair. Removal of the SWIS ventilation during the repair affects the space temperature for all ESF electrical equipment located in the 'B" & 'D" compartment. The removal of the SWIS ventilation has only two credible failures, one freezing of ESF equipment in the area. Second, exceeding the continuous duty rating for all ESF electrical equipment if the temperature gets too high. UFSAR Section 7.3.1.1.11.6.4.b states that the SWIS-HVAC supply fans are normally operated in the AUTO-mode. Therefore, operation in manual is a change to the facility. Freezing will not occur because the expected heat loss of the area when the temperature is between 60°F and 104°F is 131,500 BTU/hour. The heat gain from the operating equipment is 146,790 BTU/hour. This is assuming the ambient temperature is between 5°F and 40°F. Operational administrative controls will ensure that the design temperatures are not exceeded. Proper response to the Control Room alarm will assure that the room conditions will remain within the design space temperatures. The SWIS ventilation equipment is available for use. Since the SWIS ventilation is still capable of maintaining the proper design space temperatures, this condition does not affect accidents previously evaluated in the UFSAR. The implementation of operational administrative controls during the removal of the SWIS ventilation will not degrade the performance of the safety system or component assumed to function in the accident analysis.

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

  NOTE: The operational Administrative control put in place was procedure THC OP-SO GQ-0002(Q), REV. 0, whose 10CFR50.59 is discussed later in this report.

Other Summary of Safety Evaluation

- **SALEM - HOPE CREEK SECURITY PLAN, REV 6** This revision to the Security Plan incorporates the requirements of 10CFR73.55 (c)(7) for implementing vehicle controls, including a vehicle barrier system to counter the revised design basis threat contained in 10CFR73.1 (a). This change increases the level of protection against execution of threat components involving a design basis vehicle. The NRC Office of NRR and its consultant have reviewed the vehicle control measures, the Vehicle Barrier System (VBS) construction and the safe standoff distance locations and found them acceptable. No plant parameters or systems are affected by this change. Security procedures have been revised to reflect changes in Sally port operations, emergency vehicle search and surveillance of the VBS. The Security Program is designed to prevent purposeful acts of radiological sabotage. There is no Security accident analysis. Therefore, no operational transients or postulated design bases accidents can be affected as a result of this revision. And there can be no increase in the probability or consequences of an accident as a result of this change.

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

- **SALEM - HOPE CREEK SECURITY CONTINGENCY PLAN, REV 4** This revision to the Security Contingency Plan deletes Security Contingency Event 14, "Land Vehicle Bomb Alert." This is being done as part of the incorporation of 10CFR73.55 (c)(7) requirements. NRC Regulatory Guide 5.68, which provides guidance for implementing the rule, specifically states that the requirement of Generic Letter 89-03, which required establishment of a Land Vehicle Bomb Alert contingency, is superceded. This event can be deleted because of the low probability of advance notice of such an event and the increased assurance provided by the installation of the VBS of the ability to counter the components of the design basis security threat involving the malevolent use of a vehicle. No plant parameters or systems are affected by this change. The Security Program is designed to prevent purposeful acts of radiological sabotage. There is no Security accident analysis. Therefore, no operational transients or postulated design bases accidents can be affected as a result of this revision. And there can be no increase in the probability or consequences of an accident as a result of this change.

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

- **SALEM - HOPE CREEK SECURITY TRAINING AND QUALIFICATION PLAN, REV 3**
  This revision to the Security training and Qualification Plan -
  - Adds flexibility in establishing routes for physical fitness qualification runs
  - Extends physical examination requirements for unarmed members of the Security Force
  - Changes arms qualifications for Access Operations and Security Operations Supervisors
  - Allows for firearms instructors to certified by agencies other than the NRA
  - Raises the passing score for the security task concerning legal authority of security officers
  - Revises the badge issue process
  - Incorporates changes to Security organization titles.

  This change involves only minor changes to the composition of security force training and qualifications. The organization change merely changes title. No security force duties or training content are affected. No plant parameters or systems are affected by this change. The Security Program is designed to prevent purposeful acts of radiological sabotage. There is no Security accident analysis. Therefore, no operational transients or postulated design bases accidents can be affected as a result of this revision. And there can be no increase in the probability or consequences of an accident as a result of this change.

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

Procedures Summary of Safety Evaluations

- **THC.OP-SO.GQ-0002(Q), REV. 0, SERVICE WATER INTAKE STRUCTURE VENTILATION SYSTEM TEMPORARY OPERATING PROCEDURE** This procedure obtains temperature data to support operation of the Station Service Water Intake Structure (SWIS) pump room B & D supply and exhaust fans while they are temporarily degraded per the guidelines of Generic Letter 91-18 (GL 91-18) during repair of the Service Water (SW) head tank. This monitoring of temperature will allow the SW system to remain operable during the repair. Removal of the SWIS ventilation to repair SWIS Lubrication Tank will be controlled by the Nuclear Shift Supervisor (NSS) and will affect the space temperature for all ESF electrical equipment located in the "B" & "D" compartment. This procedure mitigates high temperatures that could result in exceeding the continuous duty factors for the equipment by providing manual actions in lieu of the normal automatic actions. UFSAR Section 7.3.1.1.11.6.4.b states that the SWIS-HVAC supply fans are normally operated in the AUTO-mode. During the repair, they will be in the STOP position. The procedure will ensure that the design temperatures are not exceeded. There are no previously evaluated operational transients or postulated design basis accidents affected by this proposal. Since the SWIS ventilation is still capable through operation in accordance with this procedure of maintaining the proper design space temperatures, this procedure does not affect accidents previously evaluated in the UFSAR. This mode of operation is specifically allowed by GL 91-18 which states that ventilation systems required to ensure operability in the summer may not be required in the winter. The implementation of operational administrative controls during the removal of the SWIS ventilation will not degrade the performance of the safety system or component assumed to function in the accident analysis..

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

- **HC.RP-RW.ZZ-0910(Q), REV. 0, STORAGE AND TRACKING OF RADIOACTIVE MATERIAL IN THE SPENT FUEL POOL** This procedure ensures that material stored in the Spent Fuel Pool (SFP) to which personnel have access is tracked, labeled, and stored by Radiation Protection. The procedure, also, ensures that access to such material is controlled by Radiation Protection. The UFSAR states "the spent fuel pool has been designed for, and now provides, a storage capacity for 3976 fuel assemblies, plus 30 multipurpose cavities for storage of control rods, control rod guide tubes, and defective fuel containers." It, also, states that "Control rod storage hangers on the Spent Fuel Pool walls provide storage for 62 control rods." No provision is made for storage of other material in the SFP. UFSAR Change Notice 96-11 is being submitted to revise the UFSAR to add a provision for storage of other material in the SFP. The implementation of this procedure does not direct movement of fuel or allow storage of items in the Spent Fuel racks and therefore does not increase the possibility of a fuel assembly being dropped. All movement of radioactive material is subject to Technical Specification 3.9.7 and is restricted by this procedure. Because the design of the SFP is such that the water level cannot drop below the top of the fuel racks, only activities which could cause a leak in the liner can increase the possibility of a loss of level below the top of the racks. The SFP is not subject to damage by radioactive material stored in it and bounded by the bundle drop accident.

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