COOPER NUCLEAR STATION
P.O. BOX 98, BROWNVILLE, NEBRASKA 68321
TELEPHONE (402)825-3811
FAX (402)825-5211

Nebraska Public Power District

NLS950014
March 1, 1995

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

Gentlemen:

Subject: Annual Operating Report
Cooper Nuclear Station
NRC Docket No. 50-298, DPR-46

In accordance with Paragraph 6.5.1.C of the Cooper Nuclear Station Technical Specifications, the Nebraska Public Power District, hereby submits the Cooper Nuclear Station Annual Operating Report for the period of January 1, 1994, through December 31, 1994.

We are enclosing one signed original for your use and, in accordance with 10 CFR 50.4 are transmitting one copy to the NRC Regional Office, and one copy to the NRC Resident Inspector for Cooper Nuclear Station.

Should you have any questions or comments regarding this report, please contact me.

Sincerely,

J. H. Mueller
Site Manager
Cooper Nuclear Station

JHM/tja:94an-rpt.ltr
Attachment

cc: NRC Regional Office
Region IV
Arlington, TX

NRC Resident Inspector
Cooper Nuclear Station

REIRS Project Manager (w/Personnel Man-Rem Report only)
Office of NRR - USNRC
Washington, DC

NPG Distribution w/o encl.
P. L. Ballinger w/encl.

Plant Manager w/1 Enclosure
Cooper Nuclear Station

JE47

Correspondence No: NLS950014

The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
"NONE"	

COOPER NUCLEAR STATION
BROWNVILLE, NEBRASKA

ANNUAL OPERATING REPORT
JANUARY 1, 1994 THROUGH DECEMBER 31, 1994

USNRC DOCKET 50-298

TABLE OF CONTENTS

I. PERFORMANCE CHARACTERISTICS

FUEL PERFORMANCE

Cycle 16 operation continued from January 1, 1994, through March 1, 1994. On March 2, 1994 the unit experienced an automatic scram due to high neutron flux caused by partial closure of the turbine governor valves resulting from a turbine control system malfunction. The unit was restarted on March 12, 1994, and startup continued until March 16, 1994. On March 16, 1994, the unit was manually scrammed to repair a Residual Heat Removal (RHR) valve. The unit was restarted on March 23, 1994 and continued operation until May 25, 1994. The unit was manually shutdown on May 25, 1994 to address Emergency Diesel Generator testing deficiencies and remained shutdown through December 31, 1994, to address this and other concerns.

Cycle 16 off-gas activity continued at essentially steady state levels with reactor coolant dose equivalent I-131 equilibrium values and off-gas release rates maintained well within the limits specified by the Cooper Nuclear Station (CNS) Technical Specifications. Comparisons of actual control rod densities predicted by computer program calculations at various core exposures indicated no reactivity anomalies of 1% or greater.

MSV AND MSRV FAILURES AND CHALLENGES

(Ref.: NUREG-0737, Action Item II.k.3.3)

There were no failures or challenges to the Safety Valves during 1994.

There were no failures or challenges to the Safety Relief Valves during 1994.

II. FACILITY CHANGES, TESTS, OR EXPERIMENTS REPORTABLE UNDER 10CFR50.59

REPORTABLE SPECIAL PROCEDURES (SPs)/ SPECIAL TEST PROCEDURES (STPs)

SP 92-051 and Amendment 1

TITLE: Spent Fuel Pool Cleaning Project

DESCRIPTION: The purpose of this Special Procedure and Amendment was to provide a guiding document for the use of the Chem Nuclear procedures to clean up the Spent Fuel Pool at CNS. This procedure incorporated QC, HP, Engineering, and Operations sign-offs into the Chem Nuclear procedures.

SAFETY ANALYSIS: This Special Procedure and Amendment provided guidance by the use of SORC approved Chem Nuclear procedures for performing work to clean up the Spent Fuel Pool. The margin of safety as defined in the basis for any Technical Specification was not reduced by this activity. The transport casks handling and loading were accomplished in accordance with the CNS Technical Specification 3.10-H (Spent Fuel Handling Casks) and CNS Procedure 10.26 (Working Over or in Reactor Vessel or Fuel Pool Requirements). This Special Procedure and Amendment did not authorize any deviation from approved CNS procedures and did not authorize any equipment alterations.

SP 92-068

TITLE: Filling Chem Nuclear Radwaste Containers

DESCRIPTION: This Special Procedure provided station and contractor personnel with the instructions for filling the various types and sizes of Chem Nuclear's Dry Active Waste (DAW) containers with equipment and components from the Spent Fuel Pool Clean Up Project.

SAFETY ANALYSIS: This Special Procedure provided guidance by the use of SORC approved Chem Nuclear procedures for filling various types of waste into containers from the Spent Fuel Pool Clean Up Project. The margin of safety as defined in the basis for any Technical Specification was not reduced by this activity. The transport casks handling and loading were accomplished in accordance with the CNS Technical Specification 3.10-H (Spent Fuel Handling Casks) and CNS Procedure 10.26 (Working Over or in Reactor Vessel or Fuel Pool Requirements). This Special Procedure did not authorize any deviation from approved CNS procedures and did not authorize any equipment alterations. The work performed by this Special Procedure was performed during a refueling outage while the plant was in a cold shutdown condition and primary containment integrity was not required.

SP 92-068 Amendment 1

TITLE: Removal, Packaging, and Shipping Old 3-55 Liners from the Spent Fuel Pool

DESCRIPTION: This Special Procedure Amendment provided station and contractor personnel with the instructions for hydrolyzing, removing, packaging, and shipping the old 3-55 liners from the Spent Fuel Pool Clean Up Project.

SAFETY ANALYSIS: This Special Procedure Amendment provided guidance by the use of SORC approved Chem Nuclear procedures for hydrolyzing, removing, packaging, and shipping the old 3-55 liners from the Spent Fuel Pool Clean Up Project. The margin of safety as defined in the basis for any Technical Specification was not reduced by this activity. The transport casks handling and loading were accomplished in accordance with the CNS Technical Specification 3.10-H (Spent Fuel Handling Casks) and CNS Procedure 10.26 (Working Over or in Reactor Vessel or Fuel Pool Requirements). This Special Procedure did not authorize any deviation from approved CNS procedures and did not authorize any equipment alterations. The work performed by this Special Procedure was performed during a refueling outage while the plant was in a cold shutdown condition and primary containment integrity was not required.

SP 92-140

TITLE: AP 101 Cask Loading and Handling Procedure

DESCRIPTION: This Special Procedure provided station and contractor personnel with an approved (SORC) set of guidelines that were used with the loading and handling of the AP 101 cask during the Spent Fuel Pool Clean Up Project.

SAFETY ANALYSIS: This Special Procedure provided guidance by the use of SORC approved Chem Nuclear procedures that were used for the loading and handling of the AP 101 cask during the Spent Fuel Pool clean up project. The margin of safety as defined in the basis for any Technical Specification was not reduced by this activity. The transport cask handling and loading were accomplished in accordance with the CNS Technical Specification 3.10-H (Spent Fuel Handling Casks) and CNS Procedure 10.26 (Working Over or in Reactor Vessel or Fuel Pool Requirements). This Special Procedure did not authorize any deviation from approved CNS procedures and did not authorize any equipment alterations. The work performed by this Special Procedure was performed during a refueling outage while the plant was in a cold shutdown condition and primary containment integrity was not required.

SP 93-258

TITLE: Low Low Set "B" Logic Test

DESCRIPTION: The Special Procedure provided instructions for performing testing of the Low Low Set (LLS) circuitry without inadvertent actuation of a Safety/Relief Valve. While performing CNS approved procedure 6.1.12 "ADS Reactor Pressure Permissive Calibration and Functional Logic Test, MS-71FRV lifted unexpectedly. The LLS "B" logic was declared inoperable. This Special Procedure temporary disabled MS-71FRV from the LLS "B" logic circuitry.

SAFETY ANALYSIS: This Special Procedure lifted a lead to prevent inadvertent actuation of MS-71FRV by the LLS "B" logic circuitry. This prevented manual actuation of MS-71FRV from the control room however, manual actuation of MS-71FRV from the Alternate Shutdown Room, and automatic actuation from overpressurization was still available. Lifting the lead did not present a new equipment safety concern, because the associated LLS circuitry was not adversely affected. Since the LLS function of MS-71FRV was declared inoperable, the Technical Specification LCO was entered and the station was within its licensing basis. Additionally, procedure 6.1.12 was successfully completed for the LLS "A" train prior to performance of this test. Therefore, the margin of safety was not reduced, and all accident analyses documented in the USAR remained bounding.

SP 94-208

TITLE: Miscellaneous Electrical System Surveillance Testing

DESCRIPTION: This Special Procedure provided instructions for station personnel to perform surveillance tests of various relays, and contacts associated with undervoltage trip and bus transfer functions.

SAFETY ANALYSIS: This Special Procedure involved exercising electrical equipment and components in the manner for which they are intended to be performed if called upon. As such, this Special Procedure did not increase the probability or the consequences of equipment or component malfunction. This Special Procedure was prepared to verify that the equipment and components would operate as expected with acceptable performance levels. The Special Procedure was performed while the plant was shutdown and in a cold condition. Although various plant equipment was disabled during performance of these tests, all Technical Specifications requirements were observed to verify that any LCO were not exceeded. Thus, this Special Procedure did not create the possibility of a new or different kind of accident, and the margin of safety was not reduced.

SP 94-208A

TITLE: Miscellaneous Electrical System Relay/Contact and Functional Testing

DESCRIPTION: This Special Procedure provided instructions for station personnel to perform functional tests of various essential relays, and contacts associated with undervoltage trip and bus transfer systems and various other essential electrical functions.

SAFETY ANALYSIS: This Special Procedure involved exercising electrical equipment and components in the manner for which they are intended to be performed if called upon. This Special Procedure was prepared to verify that the relays, contacts, equipment and components would operate as expected with acceptable performance levels. This Special Procedure was performed while the plant was shutdown and in a cold condition. Although various plant equipment was disabled during performance of this test all Technical Specifications requirements were observed to verify that any LCO were not exceeded. Thus, the Special Procedure did not create the possibility of a new or different kind of accident, and the margin of safety was not reduced.

SP 94-208B

TITLE: Miscellaneous Electrical System Relay/Contact and Functional Testing

DESCRIPTION: This Special Procedure provided instructions for station personnel to perform functional tests of various relays, and contacts associated with Diesel Generator (DG) and HVAC essential contact functional testing.

SAFETY ANALYSIS: This Special Procedure involved exercising electrical equipment and components in the manner for which they are intended to be performed if called upon. As such, this Special Procedure did not increase the probability or the consequences of equipment or component malfunction. This Special Procedure was prepared to verify that the relays, contacts, equipment and components would operate as expected with acceptable performance levels. This Special Procedure was performed while the plant was shutdown and in a cold condition. Although various plant equipment was disabled during performance of this test all Technical Specifications requirements were observed to verify that any LCO were not exceeded. Thus, the Special Procedure did not create the possibility of a new or different kind of accident, and the margin of safety was not reduced.

SP 94-208C

TITLE: Core Spray and Residual Heat Removal Contacts Testing

DESCRIPTION: This Special Procedure provided instructions for station personnel to perform functional tests of various relays, and contacts associated with Residual Heat Removal System and Core Spray System. The tests were functional tests performed on the essential contacts of these systems.

SAFETY ANALYSIS: This Special Procedure involved exercising electrical equipment and components in the manner for which they are intended to be performed if called upon. This Special Procedure was prepared to verify that the relays, contacts, equipment and components would operate as expected with acceptable performance levels. This Special Procedure was performed while the plant was shutdown and in a cold condition. Although various plant equipment was disabled during performance of this test all Technical Specifications requirements were observed to verify that any LCO were not exceeded. Thus, the Special Procedure did not create the possibility of a new or different kind of accident, and the margin of safety was not reduced.

SP 94-208D

TITLE: Reactor Protection System (RPS) and High Pressure Coolant Injection (HPCI) Contact Testing

DESCRIPTION: This Special Procedure provided instructions for station personnel to perform functional tests of various relays, and contacts associated with RPS system and the HPCI system. The tests were functional tests performed the essential contacts of these systems.

SAFETY ANALYSIS: This Special Procedure involved exercising electrical equipment and components in the manner for which they are intended to be performed if called upon. As such, this Special Procedure did not increase the probability or the consequences of equipment or component malfunction. This Special Procedure was prepared to verify that the relays, contacts, equipment and components would operate as expected with acceptable performance levels. This Special Procedure was performed while the plant was shutdown and in a cold condition. Although various plant equipment was disabled during performance of this test all Technical Specifications requirements were observed to verify that any LCO were not exceeded. Thus, the Special Procedure did not create the possibility of a new or different kind of accident, and the margin of safety was not reduced.

STP 93-002 Amendment 1

TITLE: Determination of Fuel Pool Heatup Rate

DESCRIPTION: The purpose of this Special Test Procedure was to document the heat-up rate of the Spent Fuel Storage Pool so an accurate estimate could be performed to determine how long the system can be out of service to perform work. This Special test Procedure was required to complete work performed by Design Change DC 93-057 (SW/REC Modifications).

SAFETY ANALYSIS: The Special Test Procedure was used to determine how long the Spent Fuel Pool Cooling system could be shut down before an operating temperature of 120°F was reached. The Spent Fuel Pool Cooling system is non-essential and the temperature was monitored throughout the entire performance of this test. This test did not violate CNS procedural limits for the Spent Fuel Pool Cooling system and was operated with the design limits of the system. Therefore, this Special test Procedure posed no unreviewed safety question.

STP 93-062

TITLE: RHR Quad Heat Up Test

DESCRIPTION: The purpose of this Special Test Procedure was to collect temperature and air flow data in one RHR Quad room with one RHR pump operating. The test was performed with the hatch plugs removed, and the associated Fan Coil unit manually shut down to simulate loss of ventilation.

SAFETY ANALYSIS: This Special Test Procedure did not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety as evaluated in the USAR as a result of monitoring RHR Quad Temperature. The temperature was constantly monitored and the Test terminated prior to the temperature reaching the limiting temperature acceptable for operation of essential equipment in the RHR Quad. The temperature in the Reactor Building was also monitored and the test terminated before any temperature reached USAR limits during performance of this test. This test did not violate CNS procedural limits for the RHR Quad or Reactor Building and was operated with the design limits of the system. Therefore, this Special test Procedure posed no unreviewed safety question.

STP 93-158

TITLE: CS-MOV-MO5A Differential Pressure Testing

DESCRIPTION: The purpose of this Special Test Procedure was to perform a differential pressure test of CS-MOV-MO5A motor operated valve to obtain data which was subsequently used to provide assurance that a valve anomaly identified during Generic Letter 89-10 in-situ testing was not degrading. This Special Test Procedure was performed while the plant was at normal operating conditions and used the Core Spray A pump with the Core Spray Train A in a pump surveillance valve line up according to approved plant procedures.

SAFETY ANALYSIS: This Special Test Procedure did not change any existing failure modes. The test conditions of this test were within the original design basis conditions for the Core Spray system, and used an approved surveillance procedure to perform the test. This Special Test Procedure did not make any functional changes in system operation, components, or equipment. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change.

STP 94-061

TITLE: RVLC Loss of Water Level Lockout Test

DESCRIPTION: This Special Test Procedure provided instructions to perform a test of the loss of selected Reactor Pressure Vessel (RPV) level instrument, feedwater control signal failure trip, and to determine output limits and linearity of the level transmitters.

SAFETY ANALYSIS: This Special Test Procedure was performed while the plant was in a cold shutdown condition, thus the reactor feedwater pumps were not required to be operating. Generating feedwater control signal failures when the reactor feed pumps are not operating and the startup master controller in manual had no effect on RPV water level or the instruments that control RPV water level. The USAR allows testing and inspection of the feedwater control system and components during plant shutdown. Therefore, the margin of safety was not reduced, and all accident analyses documented in the USAR remained bounding.

STP 94-164

TITLE: Determination of Fuel Pool Heat Up Rate Testing

DESCRIPTION: The purpose of this Special Test Procedure was to remove the noncritical REC system from service which caused the loss of the Spent Fuel Pool cooling. This Special Test was required to complete work performed by Design Change DC 93-057, (SW/REC Modifications).

SAFETY ANALYSIS: The Special Test Procedure was used to remove the noncritical REC system from service thereby disabling the Spent Fuel Pool Cooling system. The Spent Fuel Pool Cooling system is non-essential and the temperature was monitored throughout the entire performance of this test, and the test would have been terminated prior to reaching an operating temperature of 120°F in the Spent Fuel Pool. This test did not violate CNS procedural limits for the Spent Fuel Pool Cooling system and was operated within the design limits of the system. Therefore, this Special Test Procedure posed no unreviewed safety question.

STP 94-167 and Amendment 1

TITLE: Control Room Envelope DP Versus Make-up Air Flow Test, and System Response Following Orifice Removal

DESCRIPTION: The Purpose of this Special Test Procedure was to gather the data necessary to determine make-up air flow versus envelope differential pressure for the control room envelope. Amendment 1 to the Special Test Procedure was to quantify the system response and flow rates with the orifice HV-RO-101 removed. System filter flow was limited to 660 cfm, as this is the present design limit of the filter unit.

SAFETY ANALYSIS: The only equipment important to safety that was affected by this Special Test Procedure and Amendment was the Control Room Emergency Filter System. This Special Test Procedure and Amendment did not require this system to operate outside of its capability. The plant was in cold shutdown during this test and no fuel was being moved. This Special test Procedure and Amendment did not modify, or change the operation of any safety-related equipment, require abnormal operation of any plant systems or procedures, or have any effect overall plant safety.

STP 94-187

TITLE: Control Room to Cable Spreading Room DP Balance Sensitivity Test

DESCRIPTION: The purpose of this Special Test Procedure was to verify whether the control room to cable spreading room differential pressure balance would change with 1) a simulated loss of electrical power, 2) resulting stops and starts of HVAC fans, and 3) closure of the fire dampers. This Special Test Procedure also determined if coupling the control room and cable spreading room together, by opening an exhaust duct penetration would be effective in reducing balancing problems.

SAFETY ANALYSIS: This Special Test Procedure gathered the necessary data to support the identification of the root cause for the failure of the control room pressure envelope. No system design parameters for the systems were exceeded, CNS operating procedures were used in the performance of this test, and the test did not require the use or operation of any plant system or component outside of its capability or design configuration. Therefore, this Special Test did not affect any safety-related plant system or component, and the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR was not created. All margins of safety as defined by Technical Specifications, USAR, and plant procedures were maintained.

STP 94-196

TITLE: Measurement of Control Room Differential Pressures to Adjacent Areas

DESCRIPTION: The purpose of this Special Test Procedure was to provide measured data on the actual differential pressures that exist between the control room and adjacent areas. The quantification of leakage was determined by using tracer gas. This type of testing measurement was required to accurately quantify the inleakage to the control room envelope.

SAFETY ANALYSIS: This Special Test Procedure did not modify or change the operation of any equipment which could initiate an accident, nor did the performance of the test distract or impact control room operators. Performance of this test operated all equipment as it would be called upon to operate during an accident, and did not subject any other components or systems to abnormal conditions. This test required system lineups identical to those performed by approved CNS operating procedures. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this Special Test Procedure.

STP 94-197

TITLE: Control Room Envelope Pressure Boundary Barrier Test

DESCRIPTION: The purpose of this Special Test Procedure was to determine air leakage into or out of the control room pressure envelope of air ducts, rooms, buildings, and other miscellaneous portions of the control room envelope pressure boundary. Quantification of total envelope leakage was determined using tracer gas. This type of testing measurement was required to accurately quantify the inleakage to the control room envelope.

SAFETY ANALYSIS: This Special Test Procedure did not require the use or operation of any plant system or component outside of its capability or design configuration, and therefore did not affect any safety-related plant system or component. Performance of this test operated all equipment as it would be called upon to operate during an accident, and did not subject any other components or systems to abnormal conditions. This test required system lineups identical to those performed by approved CNS operating procedures. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this Special Test Procedure.

STP 94-199

TITLE: Control Room Envelope Unfiltered Inleakage Test

DESCRIPTION: The purpose of this Special Test Procedure was to determine unfiltered air leakage into the control room pressure envelope from adjacent air ducts, rooms, buildings, and outside atmosphere. Positive pressure in the control room envelope relative to adjacent buildings does not preclude unfiltered inleakage, as local areas of relative negative pressure can exist.

SAFETY ANALYSIS: This Special Test Procedure did not modify or change the operation of any equipment which could initiate an accident, nor did the performance of the test distract or otherwise adversely impact control room operators. Performance of this test operated all equipment as it would be called upon to operate during an accident, and did not subject any other components or systems to abnormal conditions. This test required system lineups identical to those performed by approved CNS operating procedures. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this Special Test Procedure.

STP 94-277

TITLE: Determination of Leakage on RF-CV-14CV

DESCRIPTION: The purpose of this Special Test Procedure was to determine and quantify the leakage through RF-CV-14CV. The leakage was indeterminate using the CNS approved Local Leak Rate Test (LLRT) procedure 6.3.1.1.

SAFETY ANALYSIS: This Special Test Procedure was performed while the plant was in a cold shutdown condition. This test did not permanently change the operation or configuration of any system or component, nor did this test permanently alter any system parameters. No systems were operated outside their design limits with performance of this test. Since no system design parameters were exceeded, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR was not created. All margins of safety as defined by Technical Specifications, USAR, and plant procedures were maintained.

REPORTABLE DESIGN CHANGES

DC 88-126

TITLE: Permanent Trailer Power

DESCRIPTION: The purpose of this Design Change was to document the permanent installation of equipment installed by Temporary Design Change (TDC) 87-022. This change supplied trailer power from the non-essential 12.5 kV overhead line.

SAFETY ANALYSIS: The 12.5 kV system and components that deliver power to the trailers are not important to safety and have no interaction with any system or components that are important to nuclear safety. Failure of any of the components that supply trailer power will not affect any component or system important to nuclear safety. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change.

DC 90-174B Amendment 2

TITLE: Service Water Pump Gland Water Flow Requirements

DESCRIPTION: The purpose of this Amendment was to make the necessary changes required to ensure the newly defined minimum gland water flow requirements are maintained for the Service Water pumps. There were no other changes to the original Design Change or Amendment 1 that were reported in the 1993 Annual Report. The changes are a result of pump manufacturers recommendations regarding minimum gland water flow rate to the Service Water pumps and allowable elapsed time with no gland water injection. This Amendment resulted in increasing the flow indicating switch setpoint, increasing normal injection flow, and decreasing the time delay relay setpoint for the Fire Protection backup.

SAFETY ANALYSIS: The operational changes that were made to the gland water system for the Service Water pumps does not reduce the performance of the pumps, nor did this modification affect their ability to perform the safety function as described in the USAR. Increasing the setpoint for the flow will improve the plant design by providing assurance that normal conditions for SW pump gland flow have been achieved. Changing the time delay setting for the backup system will improve the design due to earlier initiation of the backup system. The changes combine to ensure continual gland water injection in accordance with pump manufacturers recommendations. These changes did not reduce the performance of the SW Pumps nor did the modifications affect their ability to perform the safety function as described in the USAR. As such, this modification did not increase the probability of an accident occurrence, create the possibility of a new accident, or decrease the margin of safety as defined in the basis for any Technical Specifications.

DC 91-005

TITLE: Water Treatment Aerator Tank and Circulator Pumps Removal

DESCRIPTION: The purpose of the Design Change was to remove the Water Treatment aerator tank, circulator pumps and associated hardware. This equipment has not been used at CNS except for the initial Water Treatment Plant operation. This Design Change also documented the work performed under Plant Temporary Modification 93-055, which disconnected the water supply.

SAFETY ANALYSIS: No new or different accident evaluated in the USAR was created as a result of the equipment that was removed with this Design Change. No malfunctions of equipment previously evaluated in the USAR involved the portion of the Water Treatment system affected by this Design Change. No operational aspects of any safe shutdown systems were affected. Therefore, the ability of affected systems to perform their safety function was unchanged.

ESC 91-101

TITLE: Replacement of Relays RHR-REL-K70A and RHR-REL-K75B

DESCRIPTION: This Equipment Specification Change replaced existing General Electric CR 2820 relays with Agastat E7012 time delay relays. The existing CR 2820 relays exhibited significant upward setpoint drift. The new Agastat E7012 relays have, by examination of maintenance history and surveillance test data, proven to be more reliable.

SAFETY ANALYSIS: The function of the relays affected by this change is to provide a time delay such that the RHR pumps 1A and 1D start in the desired time frame. This change did not alter the function of the relays it simply replaced the existing relays with a more reliable model. The relays maintain the setpoint limit as established in the Technical Specifications. Therefore, the consequences of an accident have not changed from those described in the USAR. Therefore, this ESC did not create an unreviewed safety question nor have any adverse effect on nuclear safety.

<u>DC 92-043</u>

TITLE: SRV Solenoid Valve Design Change

DESCRIPTION: This Design Change documented the replacement of solenoid valves on the Main Steam Safety Relief Valves (SRVs). The installation of the upgraded solenoid valves and manifolds occurred at the testing facility, and after completion of testing were removed for shipping. Upon return to CNS the solenoids and manifolds were reassembled on the SRVs. The installed locations of the SRVs, per setpoint requirements, were not changed by this Design Change.

SAFETY ANALYSIS: The new solenoids, manifolds, and air operator upgrades were based on vendor recommendations. Since they perform the same function as the original equipment, the existing failure modes as described in the USAR remain unchanged. Additionally, no operating or accident parameters were changed. This change did not decrease the reliability or alter the operating characteristics of the SRVs. The ability of the SRVs to perform their intended safety function was not degraded. The performance of the solenoids, manifolds, and air operators are identical to those performed by the original equipment and operating procedures. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this Design Change.

<u>DC 92-114</u>

TITLE: Power to Trash Monitor

DESCRIPTION: The purpose of this Design Change was to document the implementation of Temporary Design Change 88-286, which installed a 480 V three phase feeder and ground cable to provide power to the trash monitor van.

SAFETY ANALYSIS: The trash monitor van does not perform any safety functions; the permanent power to the trash monitor van does not degrade nuclear safety at CNS. This change did not introduce any new failure modes that could affect safety-related components. The probability of occurrence or consequences of an accident important to safety previously evaluated in the USAR was not increased because no safety-related equipment was impacted. This activity did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the USAR.

<u>DC 92-126</u>

TITLE: Reactor Feedwater Check Valve Modification

DESCRIPTION: The purpose of this Design Change was to resolve leakage and testing problems associated with the Reactor Feedwater Check Valves. This Design Change installed vent and drain lines in the Reactor Feed (RF) piping to allow for improved control of the draining process required for performance of Local Leak Rate Testing of the RF check valves. This change also modified the hinge pins assemblies and soft seat protrusion height to improve the valve seals.

SAFETY ANALYSIS: The new safety related vent and drain lines that were installed consist of passive components whose only function is to maintain the RF pressure boundary and are unlikely to fail. All modifications to the check valves were certified by the original manufacturer and verified by our engineering staff to ensure that they would meet or exceed the original design requirements. The previous design functions of the systems were not changed as a result of the modifications. The possibility of line breaks in the new drain and vent lines is encompassed by the high energy pipe break of the larger RF line previously evaluated in the USAR. Therefore, implementation of the Design Change did not constitute an unreviewed safety question.

DC 93-021

TITLE: Installation of Loop Seal in Rupture Seal Drain Line

DESCRIPTION: This Design Change (DC) documented the work performed by SORC approved Maintenance Work Request (MWR) 93-1204, for Engineering Work Request (EWR) 93-021, which were reported in the 1993 Annual report. This DC modified the 10" rupture seal drain line in the Fuel Pool Cooling piping. A loop seal was installed in this drain line to provide a barrier between Secondary Containment and the Radwaste Building. The addition of the loop seal prevents the possibility of an open leak path between the Radwaste Building and the bellows seal.

SAFETY ANALYSIS: This Design Change enhanced plant design by separating the Reactor Building from the Radwaste Building HVAC systems, therefore ensuring design basis building pressure relationships are maintained during all operating modes. The modification enhanced the Secondary Containment system and operation by implementing a positive means of sealing the Fuel Pool Cooling Reactor Building/Radwaste Building penetration. Installation of the loop seal eliminated a potential leak path of Secondary Containment; therefore, the performance and reliability of any plant system were not degraded. No new failure modes or effects of systems or components were created. There were no changes to system operating characteristics, and no new possible accidents or equipment malfunctions were created.

DC 93-034

TITLE: HV and RW AOV Supply Air Upgrade

DESCRIPTION: This Design Change replaced the pilot operated solenoid valves with direct acting solenoid valves in the HV and RW systems. This change also installed filters, and replaced the shutoff valves with less restrictive ball valves in the two systems. The purpose of the Design Change was to upgrade the supply air system to resolve closure problems associated with several Air Operated Valves (AOVs) located in the HV and RW systems.

SAFETY ANALYSIS: The modifications and changes implemented by this Design Change did not create the possibility for an accident of a different type previously evaluated in the USAR. Previously designed functions of the affected systems remain unchanged following implementation of this change and no new functions were created. Since there was no change in any system components or operating characteristics, the effect on overall plant safety was not changed.

ESC 93-094

TITLE: Gamma Tip Detector Replacement

DESCRIPTION: This Equipment Specification Change (ESC) replaced the existing Gamma Tip detectors with new detectors. The existing detectors exhibited a higher than desired failure rate. The new detectors have design and process improvements to increase the lifetime of the detectors.

SAFETY ANALYSIS: The new Gamma Tip sensing element and drive cable assemblies are mechanically and functionally the same as the old Gamma Tip assemblies. The detectors were replaced with new detectors that have the same form, fit, and function as the old detectors. As such, this equipment specification change did not increase the possibility of an accident occurrence, create the possibility of a previously evaluated accident occurrence, or decrease the margin of safety as defined in the Technical Specifications.

DC 93-112

TITLE: Control Room Envelope Loop Seals

DESCRIPTION: The purpose of the Design Change was to install loop seals on the equipment drain and floor drains in the Cable Spreading Room. Installation of the loop seals on the open equipment drains and floor drains enhanced the ability of the Control Room emergency ventilation system to maintain a positive pressurization of the Control Room envelope.

SAFETY ANALYSIS: This Design Change involved only passive components, designed to seismic Class II over I criteria, whose function is to improve the capability of the Control Room Envelope emergency ventilation system to maintain positive pressurization. The new loop seals installed by this Design Change do not cause a malfunction of any equipment as described in the USAR. Water seal maintenance is assured by periodic performance of a surveillance procedure. No changes to the operation of the plant were involved as a result of this Design Change. No functional changes were made to the affected systems, and all design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the basis of the Technical Specifications, USAR and plant procedures were maintained.

DC 93-253, MWR 93-4433

TITLE: Installation of Restricting Orifices in Core Spray Low Side Sample Lines

DESCRIPTION: This SORC approved MWR authorized the installation of two 0.25" restricting orifices in the 1" Core Spray instrument lines for CS-DPIS-43A and 43B (low side) which penetrate primary containment. These restricting orifices are required within primary containment instrument lines that are in direct communication with the primary coolant to limit the flow in the event of an instrument line break. These Core Spray instrument lines fall into this category. This MWR is documented under Design Change 93-253.

SAFETY ANALYSIS: This MWR did not alter the normal operation or function of any affected system. This Design Change improved plant design by providing a restriction of flow of fluid to maintain the offsite doses to substantially below 10CFR100 levels in the event of an instrument line break as analyzed in the USAR. This modification did not affect the Core Spray system. The orifices are passive components which continue to perform their safety function of maintaining Core Spray pressure integrity. No new failure modes or effects of systems or components were created. There were no changes to system operating characteristics, nor changes to any setpoint, and no new possible accidents or equipment malfunctions were created.

DC 94-029

TITLE: MS-MOV-MO77 Replacement and Relocation

DESCRIPTION: The purpose of this Design Change was to replace and relocate motor operated valve MS-MOV-MO77. Replacement was necessary due to the high failure rate of this valve during local leak rate tests, and relocation was deemed appropriate to reorientate the position of this valve to a vertical plane, which will minimize the inclusion of grease into the electrical department due to normal seepage

SAFETY ANALYSIS: The replacement and relocation of the essential Main Steam Valve and motor operator did not reduce the performance or reliability of the Main Steam Valve, nor did the modifications affect its ability to perform the safety function as described in the USAR. The implementation of the Design Change did not have any adverse affect on the Primary Containment or the Main Steam system. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a Main Steam and Containment integrity. No functional changes were made to any of the systems affected by this modification, and all other design criteria were maintained in accordance with applicable codes and standards. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-046

TITLE: Core Spray and Main Steam Torus Attached Piping (TAP) Modifications

DESCRIPTION: This Design Change removed an existing restriction orifice from both loops of the 10 inch Core Spray test return lines and replaced them with an orifice which is less susceptible to cavitation. This Design Change also modified a bleed steam support on the HPCI Turbine exhaust line. This modification consisted of replacing anchors with anchors which have a higher load capacity and are less susceptible to slippage induced by vibratory loads.

SAFETY ANALYSIS: Replacement of the restriction orifices in the Core Spray Test return lines removed a source of excessive vibration which in turn, eliminated undesirable pipe stress levels and displacements. The new restriction orifices reduce the cavitation to non-destructive levels. Pipe stress analyses of each CS loop demonstrate that the piping and pipe supports are within design code allowable stress. The new orifices reduce cavitation to within acceptable levels without affecting the pressure, flow, and temperature requirements of the CS system. Replacement of anchors on the HPCI steam support do not in any way impact HPCI performance or reliability. The replacement insures proper HPCI operation by providing anchors which reduce slippage caused by vibrations.

DC 94-059

TITLE: HV-AC-(AC-IS-1A) Compressor Replacement

DESCRIPTION: The purpose of the Design Change was to replace the compressor unit, and to modify the existing condenser unit electrical power wiring on the Service Water Pump Room "A" Condensing unit (HV-AC-(AC-IS-1A).

SAFETY ANALYSIS: Proper fuse coordination and isolation was provided by this Design Change to ensure proper response of load shedding, and component protection. The changes implemented were made in accordance with recommendations and guidance provided in the 1993 National Electric Code. This system has no impact on any safety-related function for CNS, nor did this change introduce any new failure modes that could affect safety-related components. Since there was no change in any safety system components or operating characteristics, the effect on overall plant safety was not changed.

DC 94-102

TITLE: SGT System Cross-tie Valve Modification/Heater Setpoint Change

DESCRIPTION: This Design Change authorized changing the position of the Standby Gas Treatment (SGT) system crosstie valve (SGT-V-49) from sealed open to locked partially open, and revised the SGTS system setpoint to ensure humidity control during two fan operation by providing a more realistic setpoint based on SGT system operating characteristics.

SAFETY ANALYSIS: This modification enhanced the ability of the SGT system to control the decay heat removal cooling flow and flow through the non-operating train in the event of a loss of instrument air or electrical power. This was accomplished by locking partially open the crosstie valve position to restrict air passage to that required for decay heat removal in an idled SGT system train following an accident . The setpoint change was included to ensure that the SGTS system heaters are initiated during two fan operation. The SGTS is dependent on the heaters to reduce the intake air relative humidity to 70% or lower. Humidity control is provided to ensure charcoal absorber efficiency. This change resulted in increased SGTS efficiency. This Design Change did not reduce the performance of the SGTS nor did the modifications affect its ability to perform the safety function as described in the USAR. As such, this modification did not constitute an unreviewed safety question or impact the margin of safety as defined in the basis for any Technical Specifications.

DC 94-166

TITLE: Essential DB-50 Breaker Shunt Trip Modification

DESCRIPTION: This Design Change modified twelve essential Westinghouse 480 volt DB-50 switchgear breakers by removing the auxiliary undervoltage trip device from the breakers and installed a shunt trip device. The shunt trip devices were electrically connected into the logic containing the time delay relays that initiates a time delay upon loss of voltage to the 480 volt switchgear bus. This will prevent the Diesel Generators from overloading during an accident scenario.

SAFETY ANALYSIS: This Design Change improved plant design by providing for better reliability on the 480 volt load shedding of non-essential loads during a transfer to the Diesel Generators. Installation of the new shunt trip devices and associated relay logic will ensure the non-essential loads are de-energized prior to the diesel generators powering the buses. Therefore, this design will ensure the diesel generators are maintained in the analyzed loading conditions. The capabilities and performance of the onsite and offsite power supplies remain as specified in the USAR. Use of a different trip device on the switchgear breakers in conjunction with the new logic of time delay relays will not degrade the performance and capability of the switchgear buses as described in the USAR. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR was created, therefore, no unreviewed safety question was created as a result of this change.

DC 94-201

TITLE: Battery Rooms Exhaust Fans and Non-Essential Control Building HVAC Trip

DESCRIPTION: This Design Change (DC) installed an interlock to trip the battery room exhaust fans, and the non-essential control building HVAC units, on a start of the essential control building HVAC fan.

The DC was required because the essential control building HVAC is balanced for operation when the battery room exhaust and non-essential control building HVAC fans are not running.

SAFETY ANALYSIS: This Design Change modified the logic associated with the essential control building HVAC system. By installing a trip in the Battery Room exhaust and non-essential control building HVAC fans, when the essential control building HVAC actuates the original safety design of the system was restored. This essential control building HVAC system performs the same function as the Battery Room exhaust, therefore, this function is maintained. This Design Change did not introduce any failure modes that could affect essential plant operations or nuclear safety. This Design Change did not reduce the effectiveness of the Essential Control Building HVAC system or interfere with the safety function of the Emergency bypass system. Implementation of this design change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

<u>DC 94-209</u>

TITLE: Drywell Personnel Airlock LLRT Valves

DESCRIPTION: This Design replaced the existing, non-traceable Local Leak Rate Test connections with qualified connections and valves on the Drywell Airlock's exterior bulkhead. The appropriate caps/plugs was also provided for these test connections.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to ensure containment integrity. The valves and test connections were leak tested to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

<u>DC 94-212</u>

TITLE: Torus Penetration X-218 Modification

DESCRIPTION: This Design Change permanently capped penetration X-218 with a 2" socket welded cap. This penetration was identified as having wiring for thermocouples and contained non-testable, gasketed joints. This equipment was abandoned in place in 1977. Completion of this Design Change will restore the penetration to its original design.

SAFETY ANALYSIS: This Design Change replaced an open penetration with a welded cap. The new equipment (cap) was installed in accordance with applicable codes and CNS procedures to maintain the safety function of the Torus. The penetration was leak tested and will continue to be tested per the Integrated Leak Rate Test to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

<u>DC 94-212A</u>

TITLE: Electrical Penetration X-209 A-D Modifications

DESCRIPTION: The purpose of this Design Change was to remove existing manual isolation valves and hardware for penetrations X-209 A & C and replace them with new flanged testable penetrations. The penetrations for X-209 B & D were permanently capped.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a level of safety commensurate with the containment system. This Design Change improved the reliability of the containment integrity. The penetrations were leak tested and will continue to be tested per the Integrated Leak Test to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-212B

TITLE: Penetrations X-43 and X-44 Testable Flanges

DESCRIPTION: The purpose of the Design Change was to replace the existing blind flanges off penetrations X-43 and X-44 with flanges incorporating a testable, double O-ring scheme. This change enables these joints, which are portions of the Primary Containment boundary, to be tested periodically in accordance with 10CFR50, Appendix J.

SAFETY ANALYSIS: This Design Change improved reliability and integrity of the Primary Containment, because, the new, flanged piping joints have temperature and pressure ratings well in excess of the design parameters established for Primary Containment. This Design Change did not decrease the margin of safety as defined by the CNS Technical Specifications because, it enables additional surveillance testing of components important to safety. The penetrations were Local Leak Rate tested and will continue to be tested to ensure penetration integrity. No functional changes were made to the affected systems, and all design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the Technical Specifications, USAR and plant procedures were maintained.

DC 94-212C

TITLE: REC-702MV and REC-709MV LLRT Valves and Test Connections

DESCRIPTION: This Design Change provided for the installation of two 8" block valves outboard of REC-702MV and REC-709MV, and two 1" vent valves inboard of the block valves, and two 1/2" test connection valves inboard of penetrations X-23 and X-24 in the drywell. These modifications will allow REC-702MV and REC-709MV to be Local Leak Rate Tested.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity or the REC system. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a level of safety commensurate with the Containment and REC systems. This Design Change improved the reliability of the containment integrity. The valves and test connections were leak tested and will continue to be tested per the Integrated Leak Rate Test to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-212D

TITLE: Penetrations X-21 and X-22 Upgrades

DESCRIPTION: The purpose of this Design Change was to enhance the isolation scheme for both the Service Air and Instrument Air headers, upstream of penetrations X-21 and X-22, respectively, and to provide test connections as a means for periodically local leak rate testing (LLRT) the containment isolation valves (CIV's) per 10CFR50 Appendix J requirements.

SAFETY ANALYSIS: The Design Change did not degrade CNS in reference to personnel, equipment, or nuclear safety design basis or margins. System or component performance were unaffected by the changes that were made. The penetrations were Local Leak Rate tested and will continue to be tested to ensure penetration integrity. This Design Change did not require abnormal operation of any plant systems or procedures, and did not introduce any new plant operational modes or equipment operation. Therefore, the effect on overall plant safety was not changed.

DC 94-212E

TITLE: Primary Containment Integrity Issues - Instrument Valves and Caps

DESCRIPTION: The Design Change installed valves at test connections for instruments in direct communication with Primary Containment. This Design Change also replaced unneeded tees located in instrument lines which communicate with Primary Containment with unions, elbows, or installed a welded cap; the change additionally cut and placed a welded cap on instrument lines that were previously spared out.

SAFETY ANALYSIS: The implementation of this Design Change did not have any adverse affect on the Primary Containment integrity. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a level of safety commensurate with the containment system. This Design Change improved the reliability of the containment integrity, and equipment and nuclear safety were not compromised by implementation. No functional changes were made to the affected systems, and all design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the Technical Specifications, USAR and plant procedures were maintained.

DC 94-212F

TITLE: Primary Containment Nitrogen Supply Penetrations

DESCRIPTION: The purpose of this Design Change was to modify several nitrogen/air supply penetrations to meet Primary Containment isolation requirements and maintain applicable system operational requirements. These modifications will allow for testing these penetrations in accordance with 10CFR50 Appendix J.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity or the Nitrogen/air supply systems. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards. This Design Change improved the reliability of the containment integrity. The penetrations were leak tested and will continue to be tested to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-212H

TITLE: Post Accident Sampling System Modifications and Penetration X-51F Upgrade

DESCRIPTION: The purpose of this Design Change was to replace the existing, Post Accident Sampling System (PASS), containment atmosphere sample isolation valves (located on penetration X-51F) with two new qualified containment isolation valves. This change also installed the proper equipment for the necessary configuration to allow periodic LLRT of the penetration.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a level of safety commensurate with the containment system. This Design Change improved the reliability of the containment integrity. The penetration was leak tested and will continue to be tested per the Integrated Leak Rate Test to ensure penetration integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-212J

TITLE: Redesignation of III-N and IV-P Lines Needed to Effect Primary Containment Isolation

DESCRIPTION: The purpose of this Design Change was to designate segments of III-N and IV-P piping from the penetration to the outer most containment isolation valve or barrier as II-N, as well as reconciling code and installation differences to assure that the piping quality is equivalent to that of the primary containment. Piping that could not be verified from existing design documentation (which should be classified as II-N) was also included in the scope of this DC.

SAFETY ANALYSIS: The implementation of the Design Change did not have any adverse affect on the Primary Containment integrity. Repair work was performed in accordance with applicable codes and standards to assure a level of safety commensurate with the containment system. This Design Change improved the reliability of the containment integrity. This Design Change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

DC 94-212M

TITLE: TIP Purge Line Modification

DESCRIPTION: This purpose of this Design Change was to assure proper containment isolation for the TIP purge line. This was accomplished by replacing a non-qualified check valve which comprised a portion of the Primary Containment Boundary with a qualified check valve.

SAFETY ANALYSIS: The implementation of this Design Change did not have any adverse affect on the Primary Containment integrity and did not introduce any failure modes that could affect essential plant operations or nuclear safety. The new equipment was designed, fabricated, installed and tested in accordance with applicable codes and standards to assure a level of safety commensurate with the containment system. This Design Change improved the reliability of the containment integrity, and equipment and nuclear safety were not compromised by implementation. No functional changes were made to the affected systems, and all design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the Technical Specifications, USAR and plant procedures were maintained.

DC 94-222

TITLE: Honeywell Transmitter Replacement

DESCRIPTION: This Design Change replaced one Honeywell differential pressure transmitter and two Honeywell pressure transmitters with equivalent qualified essential Rosemount transmitters. The existing Honeywell transmitters could not be relied upon to maintain Primary Containment for an extended period of time. Valves were also added to the differential pressure transmitter to facilitate calibration.

SAFETY ANALYSIS: The only safety function affected by this Design Change was the Primary Containment Boundary. This safety function was not diminished in any way because the calibration valves and replacement transmitters are essential and will maintain the integrity of the Primary Containment Boundary. Implementation of this design change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

DC 94-223

TITLE: HPCI Pressure Switch (68 A/B/C/D) Replacement

DESCRIPTION: This Design Change replaced the existing four HPCI pressure switches (68 A/B/C/D) with environmentally qualified newer models to achieve better accuracy. Tighter accuracy was required to maintain Technical Specification setpoint requirements.

SAFETY ANALYSIS: This Design Change simply changed out existing pressure switches in the HPCI system with more accurate switches that meet all requirements established for the switch application. The setpoints of the new switches were changed to ensure that the new switches would meet the Technical Specification requirements of isolating the HPCI system when pressure was reduced to 100 psig or lower, and to have the HPCI operable when pressure was greater than or equal to 113 psig. This change did not impact the performance of the HPCI system in any way. All existing accident scenarios and the USAR evaluation in response to those accidents remain bounding. Subsequent to this Design Change, Technical Specification Amendment No. 166 was received, increasing the HPCI operabilitypressure limit from greater than or equal to 113 psig to greater than or equal to 150 psig.

DC 94-262

TITLE: Control Room Pressurization Enhancements

DESCRIPTION: The purpose of this Design Change was to install a new essential, qualified UL listed fire/smoke dampers in the penetration between the Control Room and the Cable Spreading Room. This penetration was opened to couple the two rooms to reduce balancing problems between the two rooms and to equalize the pressure in the two rooms. This Design Change also replaced a existing transformer and installed other electrical components for proper system operation.

SAFETY ANALYSIS: The modifications performed by the Design Change only enhanced the ability of the Control Room Emergency Bypass Filter system to maintain Control Room habitability post accident. This system is an accident mitigation system that cannot affect the probability of an accident. No operational aspects of any safe shutdown systems were affected. Therefore, the ability of affected systems to perform their safety function was unchanged. As such, this modification did not increase the probability of an accident occurrence, create the possibility of a new accident, or decrease the margin of safety as defined in the Technical Specifications.

DC 94-263

TITLE: Fuse Modification for Diesel Generator Engine Control Panel

DESCRIPTION: This Design Change authorized the replacement and modification of the overcurrent protection (breaker removal/fuse replacement) for the Diesel Generator (DG) Engine Control Panels. This change also provided proper coordination for the DG2 Engine Control Panel controls for both normal operation and operation during a fire in an alternate shutdown area. Additionally, this change replaced the 10 amp fuses for the Speed Regulator Potentiometer Motor with 5 amp fuses to ensure proper coordination with upstream fuses.

SAFETY ANALYSIS: Existing electrical separation was maintained with implementation of the Design change. The modifications and fuse replacement performed by this change provided for adequate circuit coordination margins for all affected circuits. This fuse replacement ensures the DG Engine Control Panels will have proper fuse coordination during normal, accident, and alternate shutdown conditions to enhance the reliability of the DG's. The Diesel Generators retained their designed safety shutdown features and their emergency operation/function remains as specified in the USAR.

DC 94-267

TITLE: RCIC-MO14 Change to DC Power

DESCRIPTION: This Design Change installed a DC motor for RCIC-MO14. This change also changed the power feeder cables and control cables to the motor to ensure that the RCIC system is completely independent of AC power to meet the design requirements of the system.

SAFETY ANALYSIS: The Design Change did not change the function of the RCIC system or the function of RCIC-MO14. This modification simply changed the power source for the subject valve to bring the system into the original configuration and licensed condition. No safety design basis or functional requirements of any systems were affected. Therefore, this modification did not change the existing safety analysis for Cooper Nuclear Station, nor increase the probability or consequences of an accident as analyzed in the CNS USAR.

DC 94-268

TITLE: RHR-V-168 and RHR-V-169 Modifications

DESCRIPTION: This Design Change removed the existing 3/4" RHR test connection valve RHR-V-169 in order to reduce vibration concerns. A 3/4" piece of pipe with end cap was installed in place of this valve. This new RHR test connection configuration is within Appendix J requirements for instrument line with an administratively controlled cap.

SAFETY ANALYSIS: Removal of the RHR-V-169 valve did not present a safety concern or create an unreviewed safety question. This is due to the fact that the test line modification alleviated vibration concerns, and that this modification did not alter the system flow path, nor change the operational state of any system. Use of the administratively controlled cap does not degrade the performance of this system as described in the USAR. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change.

DC 94-272

TITLE: Fuse Modification for Elapse Time Meters

DESCRIPTION: This design Change modified the existing Control Power Circuits for the Control Room Emergency Bypass Fan and Standby Gas Treatment Exhaust Fans. This modification installed fuses in the power circuit to the elapse time meter.

SAFETY ANALYSIS: The elapse time meters are non-essential equipment in an essential circuit, and as such needed to have a protective device to provide for isolation in case of a short circuit to protect the essential portion of the circuit. This change ensures that the safety function capability of both Control Room Emergency Bypass fan and Standby Gas Treatment fans remains intact. This change maintained the proper fuse coordination and enhanced the electrical separation by providing an isolation device between non-essential and essential components. Thus implementation of this Design Change did not constitute an unreviewed safety question.

DC 94-288

TITLE: Standby Liquid Control System Heat Tracing Upgrade

DESCRIPTION: This Design Change upgraded and enhanced the suction piping and pump head heat tracing capabilities for the Standby Liquid Control System. Additionally, this change authorized upgrades and enhancements to the suction piping and pump head insulation system, and ambient temperature monitoring capabilities .

SAFETY ANALYSIS: No new or different accident evaluated in the USAR was created as a result of the heat tracing upgrade and enhancements implemented with this Design Change. No malfunctions of equipment previously evaluated in the USAR involved the heat tracing capabilities of the Standby Liquid Control System. No operational aspects of any safe shutdown systems were affected. This change merely enhanced and upgraded an existing plant function and provided additional procedure steps to ensure equipment operability. Therefore, the ability of affected systems to perform their safety function was unchanged.

DC 94-302

TITLE: HV-FCU-(HV-DG-1C) and HV-FCU-(HV-DG-1D) Circuit Modification

DESCRIPTION: This Design Change modified the control circuits of the Diesel Generator fan coil units HV-FCU-(HV-DG-1C) and HV-FCU-(HV-DG-1D). The contacts of the Diesel Generator High Pressure Carbon Dioxide System (CO_2) were electrically relocated within the circuits of the DG fan coil units. In addition, an emergency start relay contact and an isolation switch contact from the respective DG's was added in parallel with the pressure switch contact in each circuit, thus providing an emergency start of the DG fan coil units.

SAFETY ANALYSIS: This Design Change improved plant design by ensuring that a fire, seismic event, or inadvertent actuation of the DG CO_2 system would not render both divisions of the essential DG fans inoperable. The CO_2 pressure switch contacts were electrically relocated within the DG fan coil circuits and an emergency start contact along with an isolation switch contact was installed to bypass the pressure switch. Furthermore, by adding another emergency relay and using this relay to initiate other relays in the circuit the cable voltage drop was reduced. This ensures that there will be sufficient voltage available to pick up the emergency start relays. The modified DG essential fan coil units' circuitry will ensure that a common mode failure of the CO_2 pressure switches will not disable the essential DG fans thereby maintaining the safety function and capability of the Diesel Generators. Electrical separation was maintained with implementation of the Design Change. The modifications performed by this change provided for adequate circuit separation, protection, and isolation for all affected circuits. The Diesel Generators retained their designed safety shutdown features and their emergency operation/function remains as specified in the USAR.

DC 94-326

TITLE: Control Room Damper Logic Upgrade, RFC Mounting Bracket, and HPCI Test Jacks

DESCRIPTION: This Design Change modified the existing control logic for the Emergency Bypass Filter System, by providing a direct interlock from the radiation monitor to close the air intake valve, and pantry exhaust valve, and to open the bypass intake valve. This Design Change also replaced a mounting bracket for RFC-AMP-AML124, and installed test jacks to facilitate testing of HPCI-REL-K35.

SAFETY ANALYSIS: This modification simplified the system, but still maintained the original design requirements. The functional characteristics of the replaced components is duplicated by the new components. The performance of the new control logic does not impact any safety system nor impacts any system that is relied upon to perform a safety-related function. Installation of the bracket and test jacks do not compromise any system operation or safety function. This Design Change did not introduce any failure modes that could affect plant operations or nuclear safety. Implementation of this design change did not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

DC 94-330

TITLE: LLRT Test Connections for CS-18CV and CS-19CV

DESCRIPTION: This Design Change redesignated CS-18CV and CS-19CV as Primary Containment Isolation valves, as well as installed four new, 3/4" test connections for these valves. These test connections are required to support LLRT of the subject valves which are now required to be tested to these requirements as a result of their redesignation.

SAFETY ANALYSIS: The Design Change did not impact the design margins of either the Core Spray system or the Primary Containment nor did this change alter any normal or accident operational parameters of any safety or non-safety related systems. This change did not introduce any new failure modes to the plant. This change solely redesignated valves that were already part of the plant configuration and installed test ports to ensure the valves would perform and meet LLRT requirements. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change.

DC 94-332

TITLE: Residual Heat Removal Minimum Flow Bypass Valve Modifications

DESCRIPTION: This Design Change was performed to change the normal operating position of Residual Heat Removal (RHR) minimum flow bypass valves RHR-MOV-MO16A/B from closed to open. The existing LPCI configuration uses normally-closed LPCI minimum flow bypass valves. Therefore, if a design basis LOCA were postulated with a concurrent loss of off site power and a single onsite 125 volt DC power failure, it is possible that the RHR pumps could fail due to deadhead operation which would result in LPCI unavailability. This was an unanalyzed condition, therefore, the valves were changed from normally closed to normally open.

SAFETY ANALYSIS: The performance of the RHR LPCI is reduced somewhat since lower flows to the vessel are postulated. The GE analysis GENE-J1102439-1, has shown that the worst case accident for CNS remains as the discharge break and single failure of the LPCI injection valve (i.e. only two Core Spray pumps injecting). The accident scenarios that rely on LPCI injection, even with the conservative assumption that both minimum valves fail open, do not result in an increase to the worst case accident scenario. An evaluation of the current licensing core heatup analysis shows that the current licensing peak cladding temperature (PCT) is not adversely impacted by the updated input values and input procedures. There is sufficient margin in the analysis to accommodate the effects of the LPCI flow reduction resulting from the open minimum flow bypass valves without affecting the current licensing basis PCT. Therefore, the current fuel MAPLHGR limits remain valid for CNS, and CNS continues to meet the requirements of 10CFR50.46.

This Design Change was performed when those portions of the affected systems were not required to perform their safety function as governed by the limiting conditions of operation of the Technical Specifications. The modifications performed by this Design Change did not in any way degrade the Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. Performance of this Design Change did not present an unreviewed safety question or require a change to the Technical Specifications as defined in 10CFR50.59. Therefore, this modification did not change the existing safety analysis for Cooper Nuclear Station, nor increase the probability or consequences of an accident as analyzed in the CNS USAR.

DC 94-335

TITLE: Nutherm Starter Wiring and RFC Flow PMIS Signal Resistor Modification

DESCRIPTION: This Design Change modified the wiring on five Nutherm starters and tested four other starters to ensure they provided for a loss of position indication in the event of thermal overload. This change also relocated the PMIS RFC flow signal resistors to increase the accuracy of the feedwater flow indicator circuit.

SAFETY ANALYSIS: The design basis for the starters were not changed or altered as a result of the Design Change. The starters were rewired to provide a loss of position indication in the event of a thermal overload, this will alert operators that there is a problem with the valves. The operation and function of the valves remain as they were prior to making the wiring modification. The Equipment Qualification of the starters was not affected in any way. Electrical separation was maintained with the change. The basis of design for the RFC system also was not altered by this Design Change. The relocation of the PMIS signal input resistors will increase the accuracy of the data displayed by the PMIS and did not affect the operation or function of the RFC system in any way.

DC 94-373

TITLE: Intake Structure Guide Wall Modification

DESCRIPTION: The purpose of this Design Change was to provide a flow path from the Missouri River to the Intake Structure Forebay (area between Intake Structure Bays and Guide Wall) which will ensure Service Water Pump operability during the non-navigational season. This flow path will provide the water flow required for operation of one Service Water Pump when river level is between 872 feet MSL and 865 feet MSL. Above 872 feet MSL, water flows over the top of the guide wall ensuring Service water system operability. This Design Change provided a water flow path which restored the Service Water system to the original design basis requirement, which is to be capable of delivering flow at a low river water level of 865 feet MSL.

SAFETY ANALYSIS: Providing an opening in the Intake Structure guide wall did not present a safety concern or create an unreviewed safety question. This is due to the fact that the modification brought the service water intake structure back to its original design basis requirements, and that this modification did not alter the system flow path, nor change the operational state of any system. A gate that slides between rails mounted on the guide wall, will close the hole during navagational season when icing conditions cannot occur, the gate will then be removed from the rails. Use of the administratively controlled gate does not degrade the performance of this system as described in the USAR. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change. Therefore, by operating the plant in accordance with the nuclear safety operational requirements as specified in Technical Specifications, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety was not increased, nor was the margin of safety as defined in the basis of any Technical Specifications reduced. No physical changes to plant nuclear safety systems were implemented by this Design Change and all safety systems will continue to be operated in their normal (as designed) configuration.

DC 94-377

TITLE: Isolation of Non-Essential Loads from Essential Loads

DESCRIPTION: This Design Change modified the existing control power circuits for distribution panel CCP1A, and CCP1B. This change installed fuses in the power circuits in a configuration which isolates the Equipment Qualified and Essential loads from the non-essential loads or devices. The fuse installation also provided proper coordination for the CCP1A and CCP1B, circuits that were affected.

SAFETY ANALYSIS: Existing electrical separation was maintained with implementation of the Design Change. The modifications and fuse installation performed by this change provided for adequate circuit protection and coordination margins for all affected circuits. This fuse installation for the CCP1A, and CCP1B circuits will provide for proper coordination between the upstream circuit fuses thereby ensuring the safety function capability of the affected circuits. Portions of these circuits involved non-essential equipment and components in an essential circuit, and as such needed to have a protective device to provide for isolation in case of a short circuit to protect the essential portion of the circuit. This change ensures that the safety function capability of both the CCP1A, and CCP1B circuits involved remains intact. This change established the proper fuse coordination and enhanced the electrical separation by providing an isolation device between non-essential and essential components. Thus, implementation of this Design Change did not constitute an unreviewed safety question.

DC 94-382

TITLE: Modification of HPCI Turbine Exhaust Drip Leg Drain Valve Controls

DESCRIPTION: The purpose of this Design Change was to modify the control logic of the HPCI Turbine Exhaust Drip Pot Drain Valves (HPCI-AOV-AO70/AO71), so that the system would not be vulnerable to a single failure that could cause a breach of Primary Containment. Additionally, the two SOV pilots for these valves were replaced with essential SOVs to ensure that they will not fail and affect the ability of the AOVs to maintain correct valve position.

SAFETY ANALYSIS: This Design Change did not affect any components that are Primary Containment boundary items. The equipment that was changed are not safety-related except that they must "not-fail". All HPCI and PC system functions were maintained with implementation of this change. All original safety functions were unchanged. No operational parameters or process conditions have been changed, and no new safety functions were added, the HPCI drain function now becomes a manual operation. This Design Change replaced non-essential items with essential ones. All design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the basis of the Technical Specifications, USAR and plant procedures were maintained.

DC 94-401

TITLE: REC-712MV and REC-713MV Modifications

DESCRIPTION: This Design Change replaced the motor for REC-712MV in order to provide additional torque thrust margins for the valve. In addition, this Design Change installed new overload heaters in the Motor Control Centers for REC-712MV, and REC-713MV to ensure that the motors would be adequately protected and that spurious tripping is avoided while allowing the valves to perform their design functions.

SAFETY ANALYSIS: The function of both REC-712MV and REC-713MV were not impacted nor changed in any way as a result of this Design Change. Implementation of this Design Change was performed when those portions of the affected system were not required to perform their safety functions, as governed by the Limiting Conditions of Operation as defined in the plant Technical Specifications. The testing and verification sections of this Design Change assured that the affected components would adequately perform their safety and design functions. All margins of safety as defined in any basis for Technical Specifications, USAR, and plant procedures were maintained. Plant equipment was operated in accordance with applicable CNS approved procedures, and Technical Specifications. Therefore, this modification did not constitute an unreviewed safety question.

REPORTABLE ACTIVITIES

LCR 94-0019 and LCR 94-0089

TITLE: USAR Change for REC Essential Heat Load

DESCRIPTION: These License Change Requests (LCRs) revised the specific REC essential services design cooling load of 1.7×10^6 Btu/hr to $.72865 \times 10^6$Btu/hr. These changes are referenced in approved Nuclear Engineering calculation NECD 94-021, and NECD 94-021, revision 1 which addresses REC heat load capabilities. The calculation provides the basis for the new heat load capabilities of the REC essential services.

SAFETY ANALYSIS: These License Change Requests only revised the existing reference for the REC cooling load from NECD 94-021 (1.1×10^6 Btu/hr) to NECD 94-021 Revision 1 ($.72865 \times 10^6$Btu/hr). This does not change the function, operation, or reliability of the ECCS pumps. A conservative heat load is utilized to ensure equipment functionality. This USAR change did not affect any hardware or system lineups. This USAR change did not change or degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

LCR 94-0022

TITLE: Clarifications to the Class IS Piping Sections of the CNS USAR

DESCRIPTION: The purpose of this License Change Request was to correct and clarify portions of the CNS USAR to document the following changes: 1) clarify barge impact criteria and indicate that response spectrum analysis could be used. 2) clarify the CNS high energy line break criteria for inside containment to reflect actual licensing basis criteria. 3) remove actual pipe stresses and the location of highest stresses so changing the USAR every time a piping system is reanalyzed can be avoided. 4) editorial change to the method of combining piping loads at Drywell penetrations and 5) correction of typographical errors and other minor clarifications and corrections. This information is located in approved NPPD calculations.

SAFETY ANALYSIS: The proposed changes to the USAR are only editorial and clarification changes, and do not alter the design basis allowable stress values, design basis loading conditions. The licensing basis requirements of the subject piping has not been altered and the ability of the piping to perform as required during any design basis event has not been changed. These changes did not require abnormal operation of any plant systems or procedures, and did not introduce any plant equipment alteration. Therefore, these changes did not increase the probability or consequences of an accident as analyzed in the CNS USAR.

LCR 94-0024

TITLE: Revise USAR Description of Second Level Undervoltage System

DESCRIPTION: The purpose of this USAR change and evaluation was to correct an error in the USAR text describing the operation of the second level undervoltage relay system. This change only corrects the wording to reflect the actual operation of the system. There were no systems, subsystems, components, or computer software affected by this change.

SAFETY ANALYSIS: The present wording of the USAR implies that the second level undervoltage system logic actually initiates load shedding of the 4000 volt motor breakers and non-essential MCC breakers. The new revised wording will correctly describe the second level undervoltage logic and further describes that the first level undervoltage logic will initiate either a trip of the 4000 volt motors or a trip of the non-essential MCC breakers depending on whether the second level undervoltage system transfers to the emergency transformer or the diesel generators. The capability of the second level undervoltage system will continue to perform its safety function as it has in the past. The description of this system is being changed to correctly define how the system actually operates. The existing failure modes and effects of the EE system remain applicable. All margins of safety defined by the basis of any Technical Specifications, USAR and plant procedures were maintained. Therefore, this LCR did not create an unreviewed safety question or have an adverse effect on nuclear safety.

LCR 94-0040

TITLE: USAR Change for Barge Impact Acceleration of 4.0 G

DESCRIPTION: The purpose of this USAR change was to remove mention of "4.0g" from section X-8.1.6 of the USAR when discussing barge impact loading. The original barge impact analysis for the Cooper Nuclear Station (CNS) Intake Structure states that Class I equipment experiences accelerations of 3.0 g. This is consistent with other sections in the USAR which address barge impact loading.

SAFETY ANALYSIS: This USAR change has no impact on safety, it is merely correcting the USAR to be consistent with the original design basis. The original barge impact study identifies the barge impact acceleration to be 3.0g. This USAR change does not affect the original analysis parameters, such as barge size and mass, angle of impact, barge velocity, river level, etc. This change simply corrects the USAR to be consistent with other sections and the original design basis for barge impact acceleration. Therefore, this change did not modify, or change the operation of any safety-related equipment, require abnormal operation of any plant systems or procedures, or have any effect on overall plant safety.

LCR 94-0066

TITLE: USAR Change for RCIC Pump Suction Valve Auto Open Signal

DESCRIPTION: The Purpose of this USAR change was to remove the statement from USAR Table VII-3-1, which indicates that the Reactor Core Isolation Cooling Pump Suction Valve from the Suppression Pool opens on a high Suppression Pool level signal.

SAFETY ANALYSIS: There is no requirement stated in GE design Specification 22A1354 R4, for an automatic transfer of RCIC pump suction from the Emergency Condensate Storage Tank to the Suppression Pool. Automatic transfer of pump suction is not an RCIC system design basis. Additionally, the CNS Technical Specifications do not contain a requirement for automatic transfer of pump suction on high pool level. The RCIC is not part of the Core Standby Cooling System and operation of the RCIC system is not assumed in the LOCA accident analysis. Therefore, this USAR change does not increase the probability of an accident or increase the consequences of an accident evaluated in the USAR. Since this USAR revision does not reflect a change to the system, system operation, or the system design basis, it does not create the possibility of a different type of accident than any previously evaluated in the USAR.

LCR 94-0068

TITLE: Reorganization/Realignment of Reporting Channels in the Nuclear Power Group

DESCRIPTION: The purpose of this License Change Request was to revise the USAR to incorporate organizational changes to the Nuclear Power Group. The changes reflect the new reporting alignments of the Licensing Department and the Emergency Preparedness Department. The Licensing Department will report to the Senior Manager of Safety Assessment, and the Emergency Preparedness Department will report to the Senior Manager of Site Support.

SAFETY ANALYSIS: The changes in the Nuclear Power group management organization and alignment do not affect the design or operation of any system, structure or component. The changes are simply in management reporting requirements. These changes will continue to provide for qualified and responsible individuals in the management positions. The changes are considered to be an administrative change to the NPG organization which does not affect the performance of the organization or station to effectively respond to plant transients or emergencies. These changes revise reporting channels only.

LCR 94-0071

TITLE: RHR, HPCI, CS, and RCIC Surveillance Testing

DESCRIPTION: The purpose of this License Change Request was to document a USAR change to include statements in the USAR specifying that certain testing requires electrically isolating valves that could prevent the automatic transfer to the injection mode from the test mode if a system initiation signal was received while the system is in the test mode. These changes apply to the RHR, HPCI, CS, and RCIC systems when performing surveillance testing for system operability.

SAFETY ANALYSIS: The USAR presently states that the ECCS systems will be capable of automatically returning from the test mode to the operating mode if a system initiating signal is received while the system is in the test mode. Certain testing at CNS requires electrically isolating valves during testing which would prevent this auto return to the operating mode. The CNS Technical Specifications Limiting Conditions for Operation (LCOs) specify a minimum of seven days of continued reactor power operation if any ECCS systems or RCIC is found to be inoperable, provided the remainder of the systems are operable. Provided a system is declared inoperable while the test is being performed, and all other ECCS systems are operable, the short duration of the testing which inhibits auto return to the operating mode for that system under test, which is substantially less than the LCO specified time interval for continued reactor operation, would not reduce system reliability or plant safety.

Declaring a system inoperable and complying with the Technical Specifications LCO during testing does not increase the probability of occurrence or the consequences of an accident or malfunction as evaluated in the USAR, create the possibility for a different type of accident or malfunction than any previously evaluated in the USAR, or reduce the margin of safety as defined in the basis for any Technical Specification.

LCR 94-0073

TITLE: USAR Change for Core Spray (CS) Pump Suction Valve Auto Open Signal

DESCRIPTION: The purpose of this License Change Request (LCR) documented a USAR change to remove the reference to auto-open signal to the Core Spray (CS) Pump Suction Valves CS-MO-7A and CS-MO-7B upon Core Spray system initiation.

SAFETY ANALYSIS: There is no requirement stated in GE design Specification 22A1435 Revision 1 for an auto-open signal to the Core Spray Pump Suction Valve on Core Spray initiation. The design basis function of the valve is to provide the capability for remote manual isolation in the event of a line break in the Core Spray loop. The keylock switch currently installed ensures that the suction valve is open and therefore, the need for the auto-open signal is not necessary nor required. This is an administrative change to correct the USAR to the design basis configuration of the system. Since this USAR revision does not reflect a change to the system, system operation, or the system design basis, it does not create the possibility of a different type of accident than any previously evaluated in the USAR.

LCR 94-0077

TITLE: Removal of USAP Table VIII-4-2

DESCRIPTION: The purpose of this License Change Request was to delete USAR Table VIII-4-2 "Periodic Tests Auxiliary Power System." This table was provided from IEEE 308-1970, which lists suggested test frequencies for auxiliary power systems. The incorporation of these suggested test frequencies into the USAR during original licensing of the plant now makes them rigid requirements. This was not the intent of the IEEE 308 guidance, and the current CNS maintenance practices and schedules ensure that the CNS equipment meet the intent of IEEE 308-1970 for auxiliary power equipment.

SAFETY ANALYSIS: The intent of periodic testing is to test equipment at regular intervals to detect deterioration of the systems. IEEE 338-1987, Criteria for the periodic surveillance testing at Nuclear Power Stations, allows for changes in testing intervals based upon several factors such as equipment performance history, corrective actions associated with failures, plant design changes with equipment and detection of significant changes in failure rates. The current CNS Maintenance Program does meet the intent of IEEE 308-1978 in periodically testing equipment to detect the deterioration of a system toward an unacceptable condition.

The CNS Maintenance Program for auxiliary power systems was reviewed and the current maintenance practices and preventative maintenance schedules at CNS ensure that this detection would be accomplished and that an unacceptable condition would be identified. The frequency of testing at CNS is based upon several factors including vendor recommendations, operating history, corrective action responses to equipment failures and equipment environment. The Maintenance Program established for CNS adequately test systems and equipment for deteriorating conditions. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was increased or created as a result of removing this Table from the USAR.

LCR 94-0078

TITLE: USAR Change to Tables V-2-7 and VII-3-1

DESCRIPTION: The purpose of the License Change Request (LCR) was to remove from the approved CNS surveillance procedure 6.3.1.1, certain valves and flanges which are presently included in this procedure but do not fall under the requirements of 10CFR50 Appendix J. This LCR also revises the USAR by identifying those valves that do not require Appendix J testing, on Table V-2-7 "Testable Primary Containment Valves" and Table VII-3-1, "Process Pipeline Penetrating the Primary Containment."

SAFETY ANALYSIS: There are two reasons why the leak rate testing of certain valves could be deleted from Appendix J testing as outlined by the surveillance procedure and identified in the USAR. Some of the valves are separated by at least two containment isolation valves from the containment penetration. Other valves and flanges which are containment isolation barriers are located on lines which are water sealed. This change was a document only change and did not involve any physical modifications to the plant nor any change in the safety classification of any system or component. No functional changes were made to the affected systems, and all design criteria in accordance with applicable governing codes, standards, and practices were maintained. All margins of safety as defined by the basis of the Technical Specifications, USAR and plant procedures were maintained.

LCR 94-0086

TITLE: Reorganization of the Nuclear Power Group (NPG) - Addition of Site Manager

DESCRIPTION: This License Change Request documented a USAR revision for the NPG organizational changes that were modified by the addition of the Site Manager position and the responsibilities of this position.

SAFETY ANALYSIS: Changes in the NPG organization structure do not affect the design or operation of any plant system, structure, or component described in the USAR accident analysis. The organization changes that took place affect only which position will perform the required responsibilities. The person filling the new position is qualified to perform the assigned tasks and responsibilities. This change is considered to be an administrative change to the NPG organization which does not affect the performance of the organization or the station to effectively respond to plant transients or emergencies.

LCR 94-0091

TITLE: USAR Change for REC Cross-tie Valves REC-19 and REC-21

DESCRIPTION: The purpose of this License Change Request was to revise the USAR to indicate that REC cross-tie valves REC-19 and REC-21 will be normally closed when both REC heat exchangers are in service, and normally open otherwise. This change was required to prevent the diversion of a portion of the total REC flow through an uncooled heat exchanger in the event of a loss-of-coolant accident (LOCA) with a concurrent loss-of-offsite power (LOOP) and a failure of Division II emergency power.

SAFETY ANALYSIS: This change only affects REC system operations when both REC heat exchangers are in service. The reconfiguration has only a small percentage reduction in heat exchanger capability with 3 REC pumps running and two heat exchangers in service. This only occurs during normal power generation and does not affect heat removal following an accident. The reconfiguration does not subject the REC system to pressure, flow, or temperature conditions which exceed the original design specification of the system, nor does it affect the maximum heat transfer capability of the system under normal operating conditions. This change does not affect the ability of the system to perform its function while sustaining a single failure. The reconfiguration actually enhances the operability of the REC system under the limiting accident condition described above. This change did not degrade the operation of any system or equipment used to mitigate the consequences of an accident, nor was equipment malfunction increased as a result of this change. All previous accident analyses as documented in the USAR remain bounding, and no unreviewed safety question was created.

LCR 94-0104

TITLE: Description of Plant Operation in Cold Shutdown

DESCRIPTION: The purpose of this License Change Request (LCR) was to document a change in the USAR discussing the mode of operation during refueling outages concerning the Service Water pumps. The Service Water (SW) pumps are used to remove the shutdown cooling heat load via the RHR heat exchanger. With the Reactor in the Refueling mode the RHR Service Water Booster Pumps (RHRSWBPs) are shutdown and the SW pumps are used to drive the cooling water through the RHR heat exchanger directly. Since the water also passes through the RHRSWBPs it is also known as windmilling the RHRSWBPs.

SAFETY ANALYSIS: The design analysis performed by Burns & Roe in response to FSAR question 10.20 and subsequent NPPD testing demonstrated that windmilling of the RHRSWBPs has no detrimental effect on any Service Water equipment. This change does not increase the probability of any accident or transient which could occur during the refueling mode. This change did not introduce any new failure modes that could affect safety-related components. The probability of occurrence or consequences of an accident important to safety previously evaluated in the USAR was not increased because no safety-related equipment was impacted. This activity did not create a possibility for an accident or malfunction of a different type than any previously evaluated in the USAR.

LCR 94-0110

TITLE: USAR Change Clarifying Reactor Building to Suppression Chamber Vacuum Relief System

DESCRIPTION: The purpose of this License Change Request was to revise the description in the USAR to clarify the Reactor Building to Suppression Chamber Vacuum Relief System. The change is clarifying the fail open requirement of these valves to accurately describe the operation of these valves.

SAFETY ANALYSIS: This USAR change did not involve any hardware modification to the plant since the as-installed design of the vacuum breakers is acceptable. This USAR change was solely a paperwork work change to update the equipment description to accurately reflect equipment operation. As such, there are no new failure modes, probability of accidents, and the consequences of an accident are not increased. The requirement for vacuum relief and primary containment continue to be met. Implementation of this USAR change does not present an unreviewed safety question or require a change to the Technical Specifications as defined in 10CFR50.59. Therefore, this USAR revision did not change the existing safety analysis for Cooper Nuclear Station, nor increase the probability or consequences of an accident as analyzed in the USAR.

LCR 94-0117

TITLE: Reorganization/Realignment of Reporting Channels in the Nuclear Power Group

DESCRIPTION: The purpose of this License Change Request (LCR) was to revise the USAR documenting organizational changes to the Nuclear Power Group. The changes reflect the new reporting alignments of the Instrument and Control Department. The Instrument and Control Department will report to the Maintenance Manager instead of the Operations Manager.

SAFETY ANALYSIS: The changes in the Nuclear Power group management organization and alignment do not affect the design or operation of any system, structure or component. The changes are simply in management reporting requirements. This change will continue to provide for qualified and responsible individuals in the management positions. The change is considered to be an administrative change to the NPG organization which does not affect the performance of the organization or the station to effectively respond to plant transients or emergencies. This change revises reporting alignments only.

LCR 94-0152

TITLE: USAR Change to Clearly Define the Reactor Building Design Basis

DESCRIPTION: The purpose of this License Change Request (LCR) was to document a USAR change to clarify that the Reactor Building design basis description is not a safety design basis of the Secondary Containment system. This change also revises the Reactor Building internal pressure specification to be consistent with the as-built plant.

SAFETY ANALYSIS: The USAR accident analysis does not rely upon the reactor building maintaining a positive pressure to mitigate the consequences of an accident. The as-built positive pressure structural specification of the reactor building is not being revised but rather clarified in psig format. No equipment important to safety relies upon the structural specification in order to perform their safety-related functions. Since no system design parameters were changed, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR was not created. All margins of safety as defined by Technical Specifications, USAR, and plant procedures continue to be maintained.

Procedure Change Notice (PCN) 0.3 (Revision 15)

TITLE: Station Operations Review Committee

DESCRIPTION: This Procedure Change Notice (PCN) revised the Station Operations Review Committee (SORC) membership to disciplines as stated in the Technical Specifications as opposed to specific position titles. This PCN also established a matrix which will be utilized to identify primary and alternate members of SORC.

SAFETY ANALYSIS: There are no changes to SORC as a review and approval committee; the change is in how membership is defined, from position title to discipline. These changes do not affect the design or operation of any plant system, structure, or component described in the USAR accident analysis. The changes in SORC membership definition does not impact the effectiveness of SORC or its ability to perform their required responsibilities. This change is considered to be an administrative change to the NPG organization which does not affect the performance of the organization or the station to effectively respond to plant transients or emergencies.

Procedure Change Notice (PCN) 2.1.10 (Revision 26)

TITLE: Station Power Changes

DESCRIPTION: This Procedure Change Notice (PCN) added statements to the procedure to decrease the probability of an instability event. The changes are administrative and were based on the BWROG guidelines for stability interim corrective actions as discussed in GE Service Information Letter (SIL) 551.

SAFETY ANALYSIS: The Procedure Change Notice added conservatism and awareness to decrease the probability of an instability event. No equipment was changed by this PCN, this change was strictly an editorial change that added clarifications to the procedure to alert operators of a possible instability event and the proper steps that need to be taken. These changes were based on BWROG guidelines which were implemented to address the concerns of SIL No. 551, regarding interim corrective actions for stability. This PCN involved no technical or operational aspects that directly affect normal station operation. This procedure change did not require abnormal operation of any plant systems or procedures, and did not introduce any plant equipment alteration.

Procedure Change Notice (PCN) 2.4.1.6. (Revision 4)

TITLE: Abnormal Neutron Flux Oscillations, or Operation in the Instability Region

DESCRIPTION: This Procedure Change Notice (PCN) added statements to the procedure to further clarify administratively that operation in the high power/low flow region of the power to flow map is prohibited. The changes are administrative and were based on the BWROG guidelines for stability interim corrective actions as discussed in GE Service Information Letter (SIL) 551.

SAFETY ANALYSIS: The Procedure Change Notice (PCN) added conservatism and awareness to alert operators, including or requiring a scram in some instances to decrease the possibility of entering the high power/low flow region of the power to flow map or the instability region. No equipment was changed by this PCN, this change was strictly an editorial change that added clarifications restricting further the high power/low flow region of the power to flow map, and the proper steps that need to be taken by the operators. These changes were based on BWROG guidelines which were implemented to address the concerns of SIL No. 551, regarding interim corrective actions for stability. This PCN involved no technical or operational aspects that directly affect normal station operation. This procedure change did not require abnormal operation of any plant systems or procedures, and did not introduce any plant equipment alteration.

Procedure Change Notice (PCN) 2.4.8.5 (Revision 9)

TITLE: Toxic Gas in the Control Room

DESCRIPTION: This Procedure Change Notice (PCN) added statements and steps to the procedure to isolate the Main Control Room and Cable Spreading Room from outside makeup air in response to a toxic gas event. This change secures the emergency booster fan in the event of a outside toxic gas event.

SAFETY ANALYSIS: The only event this abnormal procedure 2.4.8.5 is applicable to, is a toxic gas release incident. Revising the procedure to ensure that the emergency bypass filter system is not providing outside makeup air, and that normal ventilation systems are also not drawing air from the outside, decreases and\or eliminates the consequences of a toxic gas event. By providing steps to ensure that the Control Room and Cable Spreading Room HVAC continues to provide recirculation and cooling, personnel are protected from serious degradation of the control room environment. All accident analyses remain the same, and no new accidents or malfunctions were created.

Procedure Change Notice (PCN) 6.3.17.18 (Revision 3)

TITLE: Control Room Envelope Pressurization Test

DESCRIPTION: This Procedure Change Notice (PCN) added statements and steps to specify that the performance of this test shall be limited to conditions when wind speeds are between 0-8 MPH. This requirement will allow performance of this test under the conditions identical to those where pressurization will be required.

SAFETY ANALYSIS: Control Room pressurization, with respect to atmospheric pressure, is not required for wind speeds greater than 8 MPH as determined by ERIN Report No. TR 122-93-01-01. The Control Room envelope remains positive with respect to adjacent buildings. There were no hardware or operating procedure changes. This change simply defined the proper conditions that are allowed for performance of this test. Therefore, no possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this procedure change.

Procedure Change Notice (PCN) 6.4.5.2.12 (Revision 5)

TITLE: Fire Door Examination

DESCRIPTION: This Procedure Change Notice (PCN) revised the dimensional requirements against which fire door gaps are measured and verified. These new criteria are only acceptable for certain doors that will be specifically identified in the inspection procedure. The remaining doors must meet the original requirements and will have different procedure acceptance criteria from those doors described above.

SAFETY ANALYSIS: The USAR states that fire doors are listed by a nationally recognized testing laboratory as having 1½ hour or 3 hour fire rating. This PCN allows certain doors to exceed the gap requirements associated with this listing. Additional testing by Warnock Hersey International Inc. has shown that these doors are still qualified as 1½ hour or 3 hour fire doors with larger gaps. Because these doors maintain their required fire rating this PCN did not change the existing safety analysis for Cooper Nuclear Station, nor increase the probability or consequences of an accident as analyzed in the CNS USAR.

Other Activities

TITLE: HPCI Vacuum Breaker

DESCRIPTION: During a review of SIL 30, it was discovered that an open commitment exists in report number 50-298-83-14 (LER 83-14). The LER was caused by a 10" vacuum breaker on the HPCI turbine exhaust line that stuck open during surveillance testing. In the LER, it was stated that the damaged vacuum breaker was removed and replaced by a blank flange, restoring the HPCI system to its original configuration. In addition, the LER stated that a suitable vacuum breaker would be installed during a future outage. Analysis completed after the LER shows that the 10" vacuum breaker did not need to be replaced, but that the original design was satisfactory with an increase in preventive maintenance frequency. This decision is based on further analysis that determined that the original design for the exhaust line was satisfactory. This decision was supported by testing before the system was declared operational. However, no communication could be found where the District informed the NRC of this decision. This evaluation showed that original configuration in the HPCI turbine exhaust line is acceptable.

SAFETY ANALYSIS: The performance and reliability of the HPCI System or the Primary Containment was not affected by this change. This is due to the fact that the piping line modification of a blank flange does not impact any accident analyses and cannot increase the probability of malfunction of equipment. No possibility of an accident or malfunction of a different type than previously evaluated in the USAR or Technical Specifications was created as a result of this change. The functionality, reliability, and integrity of the HPCI system and Primary Containment system nor any of their design performance margins, were reduced by this evaluation or change. Therefore, all accident analyses remain the same, and no new accidents or malfunctions were created.

III. PERSONNEL AND MAN-REM EXPOSURE

PERSONNEL AND MAN-REM BY WORK AND JOB FUNCTION

Work and Job Function	Number of Personnel (> 100 mRem)			Total Man-Rem		
	Station Employees	Utility Employees	Contractor & Others	Station Employees	Utility Employees	Contractor & Others
REACTOR OPERATIONS & SUPV.						
Maintenance Personnel	70	1	6	1.861	0.043	0.225
Operating Personnel	37	0	0	8.707	0.000	0.000
Health Physics Personnel	30	0	4	7.618	0.000	1.222
Supervisory Personnel	2	2	1	0.220	0.052	0.152
Engineering Personnel	7	13	7	1.187	1.168	0.471
ROUTINE MAINTENANCE						
Maintenance Personnel	76	3	17	23.949	0.867	3.276
Operating Personnel	32	0	0	1.299	0.000	0.000
Health Physics Personnel	31	0	4	13.681	0.000	0.567
Supervisory Personnel	2	2	2	0.161	1.400	0.167
Engineering Personnel	7	13	6	0.769	4.099	0.700
SPECIAL MAINTENANCE						
Maintenance Personnel	0	0	0	0.000	0.000	0.000
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	0	0	0	0.000	0.000	0.000
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
WASTE PROCESSING						
Maintenance Personnel	17	1	2	0.101	0.002	0.018
Operating Personnel	8	0	0	0.691	0.000	0.000
Health Physics Personnel	18	0	2	0.913	0.000	0.020
Supervisory Personnel	0	0	1	0.000	0.000	0.001
Engineering Personnel	0	2	0	0.000	0.005	0.000
REFUELING						
Maintenance Personnel	0	0	0	0.000	0.000	0.000
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	0	0	0	0.000	0.000	0.000
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
INSERVICE INSPECTION						
Maintenance Personnel	0	0	1	0.000	0.000	0.001
Operating Personnel	1	0	0	0.005	0.000	0.000
Health Physics Personnel	0	0	0	0.000	0.000	0.000
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
TOTAL						
Maintenance Personnel	76	3	17	25.911	0.912	3.520
Operating Personnel	37	0	0	10.702	0.000	0.000
Health Physics Personnel	31	0	4	22.212	0.000	1.809
Supervisory Personnel	2	2	2	0.381	1.452	0.320
Engineering Personnel	7	13	7	1.956	5.272	1.171
GRAND TOTALS	153	18	30	61.162	7.636	6.820