Nebraska Public Power District

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February 28, 1990

U.S. Nuclear Regulatory Commission
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 submits the Cooper Nuclear Station Annual Operating Report for the period of January 1, 1989, through December 31, 1989.

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,

G. A. Trevors
Division Manager
Nuclear Support

GAT/tja:cal 7717
Attachment

cc: NRC Regional Office
Region IV

NRC Resident Inspector
Cooper Nuclear Station

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COOPER NUCLEAR STATION
BROWNVILLE, NEBRASKA

ANNUAL OPERATING REPORT
JANUARY 1, 1989 THROUGH DECEMBER 31, 1989

USNRC DOCKET 50-298

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I. PERFORMANCE CHARACTERISTICS

FUEL PERFORMANCE

Cycle 12 operation was interrupted on January 25, 1989, as a result of a high neutron flux scram, caused by an MSIV stem-to-disc separation. (Reference LER 89-001). The unit was restarted on February 5, 1989, and continued operation through April 7, 1989. 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 CNS Technical Specifications. Comparisons of actual control rod densities to the control rod densities predicted by computer program calculations at various core average exposures, indicated no reactivity anomalies of 1% $\Delta k/k$ or greater.

During the period from April 8 through June 16, 1989, the reactor was shutdown and the reactor vessel disassembled for the scheduled refueling and maintenance outage. A core offload and reload was performed, which included replacement of 104 fuel assemblies. With the concurrence of General Electric, it was decided that sipping for leaking fuel assemblies was not warranted due to the low off-gas activity.

Cycle 13 operation commenced with initial reactor startup on June 16, 1989, and 100 percent thermal power was initially achieved on June 30, 1989. The startup physics test program was completed on July 26, 1989, with notification of test completion submitted to the NRC on August 15, 1989. Operation of the unit was interrupted by a main turbine trip, which resulted in a reactor scram on September 28, 1989 (Reference LER 89-025). The unit was restarted on September 30, 1989, and continued operation through November 25, 1989, when the unit scrammed due to an outboard Main Steam Isolation Valve (MSIV) closure as a result of depressurization of the Instrument Air (IA) System (Reference LER 89-026). The unit was restarted on December 6, 1989, and continued operation through December 31, 1989. Off-gas activity continued at essentially steady-state levels with the reactor coolant dose equivalent I-131 equilibrium values and off-gas release rates maintained well within the limits specified by the CNS Technical Specifications. Comparisons of actual control rod densities to the control rod densities predicted by computer program calculations at various core average exposures, indicated no reactivity anomalies of 1% $\Delta k/k$ or greater.

MSV AND MSRV FAILURE CHALLENGES
(Ref.: NUREG-0737, Action Item II.K.3.3)

There were two challenges to the relief valves during the November 11, 1989, scram. Both valve actuations were satisfactory.

There were no failures.

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

REPORTABLE SPECIAL PROCEDURES/SPECIAL TEST PROCEDURES

SMP 89-02

TITLE: Replacement of 480V "G" Transformer Bushings and Flexible Shunts

DESCRIPTION: The purpose of this Special Maintenance Procedure (SMP) was to provide authorization for the installation of temporary feeders and disconnect switches in the plant 480V distribution system for the purpose of providing temporary power to selected equipment during the replacement of bushings and flex shunts on the 4160/480 AC Station Service Transformer (SST) 1G. In order to repair the bushing leaks on SST "G", 480V Bus 1G (Division II) was de-energized for approximately three days. Temporary feeders and disconnect switches were installed permitting energization of several Division II MCCs and selected equipment from Division I sources.

SAFETY ANALYSIS: The installation of the temporary power feeders and disconnect switches during the bushing replacement on SST "1G" was performed while the reactor was in a shutdown, cooled down condition with no fuel movements in progress. During the power outage created by the repair to SST 1G, separation criteria and electrical separation were allowed to be compromised with the use of this detailed procedure and CNS Procedure 0.9, "Equipment Clearance and Release Orders." The requirements for these criteria are decreased, and the operational and accident concerns defined in the USAR and Technical Specifications are minimized with the reactor in a shutdown and cooled down condition and no fuel movements in progress. Therefore, the activities controlled by this Special Maintenance Procedure did not degrade personnel, equipment safety, nor was the capability to achieve and maintain cold shutdown as prescribed by 10CFR Part 50 compromised by the cross-connection of divisional power supplies. After completion of this SMP all temporary feeders and disconnects were removed and all system line-ups (Div. I/Div. II) were restored to normal prior to moving any fuel.

SMP 89-08

TITLE: 480 "G" Transformer High Voltage Bushing Repair

DESCRIPTION: The purpose of this Special Maintenance Procedure (SMP) was to provide authorization for the installation of temporary feeders and disconnect switches in the 480V distribution system for the purpose of providing temporary power to selected equipment during repair of a leaking high voltage bushing on the Station Service Transformer (SST) 1G. In order to repair the high voltage bushing leaks on SST "G", 480V Bus 1G (Division II) was de-energized for approximately three days. Temporary feeders and disconnect switches were installed permitting energization of several Division II MCCs and selected equipment from Division I sources.

SAFETY ANALYSIS: The installation of the temporary power feeders and disconnect switches during the repair of the leaking high voltage bushing on SST "1G" was performed while the reactor was in a shutdown, cooled down condition with no fuel movements in progress. During the power outage created by the repair to SST 1G bushings, separation criteria and electrical separation were allowed to be compromised with the use of this detailed procedure and CNS Procedure 0.9, "Equipment Clearance and Release Orders." The requirements for these criteria are decreased, and the operational and accident concerns defined in the USAR and Technical Specifications are minimized with the reactor in a shutdown and cooled down condition and no fuel movements in progress. Therefore, the activities controlled by this Special Maintenance Procedure did not degrade personnel, equipment safety, nor was the capability to achieve and maintain cold shutdown as prescribed by 10CFR Part 50 compromised by the cross-connection of divisional power supplies. After completion of this SMP all temporary feeders and disconnects were removed and all system line-ups (Div. I/Div. II) were restored to normal prior to moving any fuel.

STP 87-003

TITLE: Maximum Control Rod Drive (CRD) System Flow

DESCRIPTION: The purpose of this Special Test Procedure was to perform testing to determine the maximum obtainable flow rate and the optimum flow path through the Control Rod Drive system. This testing was performed to satisfy the requirements of INPO Finding TS.2-1, which specified testing of systems that provide high pressure make-up water to the reactor vessel. The Control Rod Drive system could be used in this case, as a backup to the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems during a loss of feedwater event. Also this test complies with the guidance of the Nuclear Regulatory Commission documented in NUREG 0619, that "...the applicable licensee will be required to demonstrate, by testing, concurrent two-CRD pump operation (if necessary to fulfill required flow capability), satisfactory CRD system operation, and required return flow capacity to the vessel."

SAFETY ANALYSIS: The Control Rod Drive Hydraulic system is designed to supply high pressure water to the Control Rod Drive mechanisms for cooling, positioning, and for providing charging water for the Hydraulic Control Unit (HCU) scram function. Performance of this Special Test Procedure did not affect the design capability of the CRD system to function in its prescribed manner during routine operation of the system. While this STP was being performed, however, the hydraulic system could not be in its normal configuration. It was therefore required that the reactor be scrammed and in a cold shutdown condition prior to performing the test to allow for flow through several flow paths through the CRD system to the vessel. Therefore, because the control rods were inserted prior to performance of this test (i.e., the reactor was

in a shutdown condition with the reactor scrammed and the scram was not reset), Technical Specifications were satisfied. Also, removal of the CRD return line and subsequent testing of the CRD system for vessel make-up flow is discussed in the USAR.

STP 87-012

TITLE: Stability Testing of Solidification Matrix

DESCRIPTION: The purpose of this Special Test Procedure (STP) was to ensure compliance with the radioactive waste burial site regulations regarding stability (10CFR61.56). This STP evaluated the stability of solidified, non-radioactive resin, and sludge lab samples prepared using a process control program based on the existing radioactive waste cement solidification system process. The criteria specified in the NRC Technical Position on Final Waste Form was used as a basis for evaluating the sample stability.

SAFETY ANALYSIS: All work performed under this Special Test Procedure was in accordance with procedures, practices, codes, and standards normally used and performed during routine operation of Cooper Nuclear Station. Therefore, this Special Test Procedure did not represent an unreviewed safety question.

STP 87-013

TITLE: Magna-Blast Circuit Breaker Closing Coil Minimum Operating Voltage Test

DESCRIPTION: The purpose of this Special Test Procedure was to directly measure and determine a reliable minimum operating voltage for the 125 VDC closing coil on the General Electric Magna-Blast 4160 volt circuit breakers. The STP was necessary to determine operability of the 4160 volt Magna-Blast circuit breaker closing coil during a worst case voltage drop situation in response to a concern identified during the Safety System Functional Inspection (SSFI).

SAFETY ANALYSIS: This STP was conducted on a spare 4160 volt circuit breaker in the CNS Electric Shop. No high voltages were utilized, because only the 125 VDC supply is required to operate the breaker. Electric Shop personnel are familiar with the operation of this breaker and have tested similar breakers by cycling them in the shop. Therefore, no unusual hazards or equipment alteration were conducted. Overall plant operation and safety were not changed by the conductance of this STP.

STP 87-015

TITLE: Kaman Power Supply Test

DESCRIPTION: This Special Test Procedure was performed to collect data on the Kaman HRH and HRN radiation monitor power supplies to identify and document those items requiring hardware and software modifications concerning the power supplies. This STP also performed testing on the power supplies by adding minimum load resistors and a signal diode between the +5 VDC line and the PFI/signal line.

SAFETY ANALYSIS: This Special Test Procedure did not in any way degrade the safety of Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. This test required the monitors to be off-line but the alternate G.E. sampling system was put into service to sample off-gas radiation levels at the stack. Technical Specifications allows for maintenance and testing of equipment and the monitors were returned to service within their specified limits.

STP 87-022

TITLE: Moisture Separator Performance Verification and Monitoring Test.

DESCRIPTION: The purpose of this Special Test Procedure was to verify that the moisture removal effectiveness of the moisture separators met the requirements of Contract 87-11A when the plant and Main Steam System are operating under specified plant conditions. Moisture removal effectiveness measurements were used to determine if the Contract 87-11A guarantee of at least 95 percent moisture removal had been met.

SAFETY ANALYSIS: This procedure involved portions of the Main Steam System which are non-essential. This Special Test Procedure did not in any way degrade the safety of Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. This test was conducted with the system and plant in their normal mode of operation. This STP did not require abnormal operation of any plant systems or procedures, and did not introduce any plant equipment alteration. Therefore, the affect on overall plant safety was not changed.

STP 88-014

TITLE: Special Procedure 6.4.5.2 Procedure Change Test

DESCRIPTION: The purpose of this Special Test Procedure was to allow testing of a proposed change to Procedure 6.4.5.2 "Fire Protection System Annual Inspection." During a fire/all-risk special visit to CNS conducted by American Nuclear Insurers (ANI), ANI determined that two features of the diesel engine fire pump 1D controller were not being tested sufficiently by Procedure 6.4.5.2. This test was performed to correct the two deficiencies identified by ANI,

and incorporate them into the procedure if the test was successful.

SAFETY ANALYSIS: This activity required diesel fire pump 1D to be out of service during the procedure, which is not a violation of the Technical Specifications. The operability of the two electric fire pumps were not in jeopardy throughout the test. Therefore, the requirements of Technical Specification 3.15 were not compromised. Appropriate safety practices were observed to ensure the safety of personnel and equipment while working on the low voltage diesel controller circuitry. The Fire Suppression System is classified as non-essential and this STP had no effect on nuclear safety and did not create an unreviewed safety question.

STP 88-015A

TITLE: Number 2 Feedwater Heaters Normal Liquid Level Determination

DESCRIPTION: The purpose of this Special Test Procedure (STP) was to determine the "optimum" nominal liquid level in Feedwater Heaters 1-A-2 and 1-B-2 following modification of their liquid level instrument loops by DC 88-015. The normal liquid level (NLL) of each heater was then set 1 inch above the "optimum" nominal liquid level determined for each heater. This higher normal operating liquid level will provide subcooling of the heater drains, thereby eliminating the flashing phenomena that is suspected of causing abnormal heater drain operation.

SAFETY ANALYSIS: This procedure involved portions of the Heater Drain System which are nonessential. The special test as described in this procedure did not in any way degrade the safety of Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. The test was conducted with the system and plant in their normal mode of operation. Health Physics was consulted prior to beginning testing to determine the extent of any radiological concerns regarding this Special Test Procedure.

STP 88-016

TITLE: Reactor Recirculation Motor Generator (RRMG) Slip Ring Brushes

DESCRIPTION: The purpose of this Special Test Procedure was to document the testing of new slip ring brushes in the RRMG sets. It also documented the testing of increased brush current density on RRMG "B" exciter and generator collector rings, and its effect on brush life. This test was a result of GE recommendations following their refurbishment of the RRMG sets.

SAFETY ANALYSIS: The Cooper Nuclear Station USAR discusses the safety function of the Reactor Recirculation system. The safety analysis performed for the recirculation system involves the loss of a recirculation pump. The modified current density on the brushes and/or new

brush type was not expected to change the operating characteristics of the system. If an MG set trip resulted, the following pump trip would be an analyzed event. Therefore, since the USAR discusses recirculation pump trips, this test did not represent an unreviewed safety question.

STP 88-242

TITLE: Procedure for Using Equotip Model D Hardness Tester

DESCRIPTION: The purpose of this Special Test Procedure was to provide instructions and guidance for performing field hardness tests using the Equotip Model D instrument. The intent was to determine the tensile strength of certain components at CNS. Reference NRC Bulletin 88-05 "Nonconforming Materials Supplied by Piping Supplies Inc." The tensile strengths of the components were then compared with the minimum properties given in ASME or ASTM material specifications to determine acceptability.

SAFETY ANALYSIS: This procedure did not degrade Cooper Nuclear Station in any way with respect to personnel, equipment, and nuclear safety. The very small amount of material that was removed (for surface preparation) had a negligible impact on the strength of the component that was tested. The hardness test itself is a non-destructive type, and therefore, did not affect the performance of the component.

STP 88-261

TITLE: Core Spray Pump Minimum Flow Test

DESCRIPTION: The purpose of this Special Test Procedure was to conduct and document a minimum flow test of the two Core Spray Pumps. The test was needed to verify the adequacy of the pump(s) minimum flow capabilities as required by NRC Bulletin 88-04.

SAFETY ANALYSIS: This STP was performed while the plant was in a cold shutdown condition. This test consisted of operating the pumps per approved CNS procedures at off-design flow conditions including minimum flow and taking vibration readings with a multi-channel FM recorder and vibration transducers. Flow rates in the minimum flow line were obtained by using a portable UT flowmeter. The Core Spray Pumps were not operated in any manner which was not previously evaluated and SORC approved (CNS Procedure). The pumps were also continually monitored throughout the test to ensure the pumps did not exceed any vibration levels which could have been considered damaging. Therefore, no unreviewed safety question existed with the performance of this STP.

STP 88-283

TITLE: HPCI Pump Vibration Test

DESCRIPTION: The purpose of this Special Test Procedure was to conduct and document a vibration test of the High Pressure Coolant Injection (HPCI) pumps. The test was needed to obtain information necessary to determine the source of higher than normal vibration on the HPCI pumps.

SAFETY ANALYSIS: This STP was performed while the plant was in a cold shutdown condition. This test consisted of operating the HPCI pumps per approved CNS Procedure. The only physical interface with the system was to remove three shaft covers and attach magnetically mounted velocity transducers. The HPCI pumps were not operated in any manner which was not previously evaluated and SORC approved (CNS Procedure). The pumps were also continually monitored throughout the test to ensure the pumps did not experience any excessive vibration levels as defined by Inservice Testing Criteria. Therefore, no unreviewed safety question existed with the performance of this STP.

STP 88-293

TITLE: Primary Containment Purge and Vent Valve Quarterly Leak Rate Tests

DESCRIPTION: The purpose of this Special Test Procedure was to provide guidance for performing quarterly leak rate tests on the Primary Containment purge and vent isolation valves. The valves are normally leak tested once per each operating cycle per 10CFR50 Appendix J. However, NRC concerns have prompted increased frequency of the tests to verify the adequacy of the resilient seats.

SAFETY ANALYSIS: Since the tests described in this Special Test Procedure were performed with both the inboard and outboard valves of each purge and vent penetration in the closed position, the quarterly leak tests did in no way degrade the Primary Containment isolation capability of the subject valves. CNS Technical Specifications requires that these tests be performed routinely. Therefore, the margin of safety was not reduced.

STP 89-154

TITLE: Verification of Correct Fail Position on Loss of Instrument Air

DESCRIPTION: The purpose of this Special Test Procedure was to verify the correct failure position of valves and dampers on a loss of air. The District committed to perform these test in response to Generic Letter 88-14, "Instrument Air Supply Problems Affecting Safety-Related Equipment." Additionally, this Special Test Procedure verified the failure position of valves and dampers

due to loss of electrical power to the specific control device for each component.

SAFETY ANALYSIS: The performance of this STP did not affect the safety functions for any components tested in this procedure because failure position on loss of air and loss of electrical power had been previously evaluated and components are engineered to position properly on loss of air or electrical power. This test did not impact safety concerns at CNS due to the fact that all Technical Specification requirements for each component were met before testing. In addition, the plant was in a cold shutdown condition during testing of these components.

STP 89-173 and Revision 1

TITLE: Application of Abrasion Control Putty on SW-MOV-M089A and SW-MOV-M089B

DESCRIPTION: This Special Test Procedure authorized the use of Chesterton abrasion control putty and abrasive control liquid which was applied to the eroded portions of the valve bodies (downstream portions). This test also evaluated the erosion resistance suitability of this material in this application.

SAFETY ANALYSIS: The plant was in a cold shutdown condition during this STP. Service Water System function and operation were not altered by the application of abrasion control putty/liquid in the downstream side of the body of SW-MOV-M089A and SW-MOV-M089B valves. The wall thickness in the downstream portion of the valve bodies met minimum wall thickness requirements specified by code prior to the abrasion control putty/liquid application. The putty followed the original contours of the valves and had no effect on system and valve performance.

STP 89-176

TITLE: MSIV Closing Test with Instrument Air Valved Out

DESCRIPTION: The purpose of this Special Test Procedure was to measure MSIV closure time with instrument air to the accumulators valved out and compare it to the closure time with instrument air valved in. This STP was performed due to recommendations received in GE Service Information Letter SIL 482.

SAFETY ANALYSIS: This Special Test Procedure was performed during reactor cold shutdown in conjunction with CNS approved Procedure S.P.6.3.9.4 "Main Steam Isolation Valve Testing." This STP did not introduce any new or temporary design modifications of equipment or plant operation. The MSIV's were stroked with the supply air valve closed, using only air from the accumulator. MSIV valves are designed to close with no instrument air available, therefore, no unreviewed safety question existed.

STP 89-215

TITLE: Control Room Pressurization Test with Dampers AD-1021A(B) Locked Open

DESCRIPTION: The purpose of this Special Test Procedure was to determine the positive static pressure in the Control Room/Cable Spreading Room envelope relative to ambient and the surrounding areas with Supply Fan Dampers AD-1021A and AD-1021B open simultaneously in the Normal and Emergency Operating Modes. This information then could be used to justify changing dampers AD-1021A and AD-1021B from "fail closed" to "fail open".

SAFETY ANALYSIS: The plant was on-line and in the Run Mode during performance of this STP. System performance was slightly decreased in both the Normal and Emergency modes of operation resulting in a lower positive pressure maintained in the Control Room Envelope. Therefore, if the pressure reading at any location, during any phase of this test, gave any indication of being marginal with respect to maintaining a positive pressure, the test would have been immediately aborted.

This Special Test Procedure did not affect the safety functions for any components tested in the procedure, it only operated filtering equipment in their Normal and Emergency modes of operation. The Emergency Bypass System was still able to supply the required CFM and perform its essential safety related function throughout the test duration. This test was to determine if the Control Room envelope could maintain a positive pressure should loss of instrument air occur and the Control Room Normal HVAC supply fan dampers failed open.

STP 89-218

TITLE: 4160V AC Critical Switchgear Room Ventilation Distribution and Entrainment

DESCRIPTION: This Special Test Procedure collected air flow velocity data in 4160 volt AC Critical Switchgear Rooms 1F and 1G during normal ventilation operation and during operation of the backup Portable Ventilation System to verify that sufficient distribution and entrainment are occurring to preclude the buildup of localized temperatures (hot spots).

SAFETY ANALYSIS: Operability of the 4160V AC Critical Switchgear was not affected by this STP. This STP only took air flow velocity and temperature measurements in the Critical Switchgear Rooms 1F and 1G under two approved means of ventilation (normal and backup). All applicable Plant Technical Specifications pertaining to the 4160V AC Critical Switchgear equipment and room air temperatures were maintained at all times. Therefore, the affect on overall plant safety was not changed.

STP 89-223

TITLE: Testing of a Gortex Lube Oil Filter Gasket

DESCRIPTION: The purpose of this Special Test Procedure was to install and evaluate the performance of gortex self-adhering joint sealant for use in the Diesel Generator Lube Oil System filter. This teflon type material was used as a gasket between the lid and barrel of the lube oil system filter.

SAFETY ANALYSIS: There were no changes in the operation, design, or control of the Diesel Generator System with the performance of this test. This STP only installed and evaluated the use of gortex self-adhesive tape for gasket material in the DG lube oil filter.

REPORTABLE DESIGN CHANGES

DC 83-023

TITLE: High Pressure Coolant Injection (HPCI) Exhaust Line Vacuum Breaker Installation

DESCRIPTION: The purpose of this Design Change was to provide for the installation of an additional vacuum breaker in the HPCI exhaust line. This added vacuum breaker allows vacuum to be broken quickly in the HPCI exhaust line to alleviate the HPCI turbine exhaust line reflood problem.

SAFETY ANALYSIS: This Design Change was performed with the plant in a cold shutdown condition. This change decreased the possibility of HPCI exhaust line reflood and, therefore, improved the reliability of the HPCI system. This Design Change did not alter any operating characteristics of the HPCI system nor did it affect HPCI system operation as defined in the CNS Technical Specifications. This Design Change resulted in HPCI's operational safety being increased, while having no detrimental effect on nuclear safety.

DC 85-005

TITLE: Replacement of 24 Volt DC Batteries

DESCRIPTION: The purpose of this Design Change was to replace the 24 volt DC Batteries 1A1, 1A2, 1B1, and 1B2, and the battery racks. These 24 volt batteries supply power to control room for the area radiation monitors and startup range neutron monitors upon loss of A-C power supply to the battery chargers.

SAFETY ANALYSIS: The existing safety function of the 24 volt DC power system remains unchanged. No additional safety concerns beyond those already evaluated in the USAR result from this Design Change. The capacity of the new batteries envelope their design basis load profile. The margin of safety associated with the 24 volt DC power system was maintained. System performance was improved, the original batteries were nearing the end of their design life and showed signs of physical deterioration, which indicated the need for replacement. The new batteries have larger ampere hour rating, as compared to the original batteries. The battery replacement did not change the function or operation of the 24 volt DC system or the equipment to which it provides power.

DC 85-015F

TITLE: Control Room Design Review (CRDR) Panel 9-3 Modification

DESCRIPTION: This Design Change is a continuation of the Control Room Design Review (CRDR) program. The primary purpose of this Design Change was to modify the existing Cooper Nuclear Station (CNS) Control Room Panel 9-3 in order that the displays and controls are

arranged more in accord with established Human Factors Engineering (HFE) guidelines. Application of these HFE guidelines modify the existing control and display positional relationships and labeling practices. This Design Change also included the mounting provisions for displays added and substituted per Design Change 85-106, "Differential Pressure & Pressure Transmitter Replacement", it also included the removal and panel repair for a control and display, which is being decommissioned by DC 86-78, "Removal of the Reactor Vessel Head Spray Line."

SAFETY ANALYSIS: The modifications to Panel 9-3 were done during a planned outage period. Work did not commence until the plant was off-line, and the reactor was in a stable, cold shutdown mode of operation. The work was performed in two phases, Division I modifications, and Division II modifications. All work completed in the first phase was completed, checked out, and was fully operational before work was allowed to commence on the second phase. No functional modifications were made to any of the affected systems and all design criteria remained in accordance with governing codes, standards, and practices. Therefore, Nuclear Station safety and equipment safety were not impaired either during or following the modification.

DC 85-045

TITLE: Emergency Feeder to the Emergency Operating Facility (EOF)

DESCRIPTION: The purpose of this Design Change was to provide an emergency 480V power feeder to the Emergency Operations Facility (EOF) that would be available under accident conditions. The EOF is designed to provide a safe environment at CNS for the overall management of an emergency. The NRC provided NPPD guidance in Generic Letter 81-10, NUREG-06961, and NUREG-0737, Supplement No. 1.

SAFETY ANALYSIS: THe modifications outlined by this Design Change did not in any way degrade the safety of Cooper Nuclear Station with respect to equipment or nuclear safety. This change did not change the function or operation of any system or component related to safe shutdown. By providing emergency power to the EOF the District will be able to provide overall management for CNS in case of a loss of off-site power while the EOF is activated.

DC 85-106

TITLE: Pressure Transmitter Replacement

DESCRIPTION: The purpose of this Design Change was to replace four transmitters with environmentally and seismically qualified transmitters. The replacement of the transmitters was necessary to bring CNS into compliance with the requirements of NUREG-0737, Supplement 1 in accordance with the guidance given in Reg. Guide 1.97.

SAFETY ANALYSIS: This Design Change involved work on safety related components. This modification did not degrade the safety of Cooper Nuclear Station with respect to plant safety and nuclear safety. This change did not modify any operational function of systems or components. These changes will increase the reliability of the affected instrumentation during and after an abnormal event or accident because they are now environmentally and seismically qualified.

DC 85-108

TITLE: Torus Temperature Recorded and Resistance Temperature Detector Replacement and Residual Heat Removal Heat Exchanger Outlet Thermocouple Replacement

DESCRIPTION: The purpose of this Design Change was to replace 16 Resistance Temperature Recorders (RTD's) which measure the suppression pool water temperature, replace two thermocouples which measure the Residual Heat Removal Heat Exchanger Outlet temperature, and to replace two suppression pool water temperature recorders with seismically and environmentally qualified equipment to meet Reg. Guide 1.97 requirements.

SAFETY ANALYSIS: The plant was in a cold shutdown condition during implementation of this Design Change. This modification will not degrade the safety of Cooper Nuclear Station (CNS) with respect to personnel, equipment, or nuclear safety. The temperature monitoring equipment in the RHR Heat Exchanger areas, the torus areas, and the Control Room is intended to provide post accident monitoring capability and will not degrade the function of any safety related system. These changes will increase the reliability of the systems to maintain monitoring operations during and after an abnormal event or accident.

DC 85-112

TITLE: Critical AC Bus Breaker and Fuse Coordination Modifications

DESCRIPTION: The purpose of DC 85-112 was to modify various CRITICAL AC circuits as recommended in Cooper Nuclear Station Critical Bus Coordination Study performed by the Protection and Control Engineering Department, T&D Project Division of Nebraska Public Power District., One circuit breaker was replaced with a fused disconnect switch, fuses were upgraded to a larger fuse size, amptector relay settings and 4160V relay settings were modified, and amptectors were replaced.

SAFETY ANALYSIS: The plant was in a cold shutdown condition during implementation of this Design Change. During the implementation of this DC the affected systems were inoperable. The modifications performed by this DC provided for adequate circuit coordination margins on all critical AC bus coordinate pairs at CNS. With the new

coordination margins, a failure of any piece of equipment or cable will isolate the failure to a minimum area, thereby decreasing the probability that the failure of a piece of equipment or cable will not affect additional equipment.

ESC 86-024

TITLE: Turbine Equipment Cooling (TEC) Heat Exchanger Tube Replacement

DESCRIPTION: The purpose of this Equipment Specification Change (ESC) was to replace the original tube material (arsenical admiralty) with titanium tubing per ASME SB-338, Grade 2. The titanium tubing material is strongly resistant to both chemical corrosion and erosion.

SAFETY ANALYSIS: This Equipment Specification Change did not interface with any safety shutdown systems and was considered non-essential. This change did not affect the operating characteristics of the heat exchangers, the only change was in the tube material used in the heat exchangers to a more corrosion-erosion resistive type. Therefore, the change in tube material did not create an unreviewed safety question or affect nuclear safety.

DC 86-024
DC 87-061
DC 89-186

TITLE: Diesel Generator Control Cabinet Upgrades and DG Starting Air Tube Replacement

DESCRIPTION: The purpose of these Design Changes was to increase the availability and improve the reliability of Diesel Generators No. 1 and 2 by upgrading the local control systems, minimizing control air, lube oil, fuel oil, and jacket water cooling tubing failures from vibration, fatigue, and tubing interferences. To reduce these failures, the existing copper tubing was replaced with stronger stainless steel tubing. Any stainless steel tubing which showed signs of wear was replaced and all clamped tubing was provided with vibration isolation rubber cushions and thereby provides a less severe operating environment. To reduce tube failures caused by interferences, the final installation was inspected for tube-to-tube contact and relative motion during diesel generator operation.

SAFETY ANALYSIS: With the implementation of this Design Change the overall safety of the control system of the diesel generators is substantially improved by the reduction of vibration induced damage and thereby enhances system performance. The Diesel Generators retained their designed safety shutdown features and their emergency operation/function remains as specified in the CNS USAR. The implementation of this Design Change did not create a possibility for an accident more severe than the Total Loss of Off-Site Power during normal operation as already analyzed in the USAR.

DC 86-053

TITLE: Main Steam High Flow Setpoint

DESCRIPTION: This Design Change provided for a nonconservative change to the Technical Specification setpoint limit of Main Steam Line high flow from 140 percent of rated steam flow to 150 percent of rated steam flow. The Design Change required changes to station documents only. Documents changed were: Technical Specifications, USAR, Station Procedures, Training Manual, Set Point Log, and Vendor Manuals. The actual instrument setpoint of 112 psid did not change, therefore, no changes to station equipment were required.

SAFETY ANALYSIS: DC 86-53 did not present an unreviewed safety question. Although changing the setting limit to 150 percent results in an increase of approximately 10 percent in the dose calculated for a setting of 140 percent, either dose value is an extremely small fraction of the 10CFR100 allowable or the bounding design basis accident (a guillotine break of the steam line). Indirectly the change provided a positive effect by providing margin needed to allow closure of one Main Steam Line isolation valve at rated power. Both effects are in keeping with the intent of the basis for the USAR. For additional information refer to Cooper Nuclear Station License Amendment 96 dated March 17, 1986.

DC 86-078

TITLE: Removal of Reactor Head Spray Line

DESCRIPTION: The purpose of this Design Change was to permanently remove the Head Spray Subsystem of the Residual Heat Removal System and associated equipment.

SAFETY ANALYSIS: The Head Spray Subsystem has no safety function and no credit has been taken for its use in current safety evaluations or emergency operating procedures. Removal of the head spray line was beneficial in that it removed a potential location for a primary system pipe break and reduced ALARA concerns. Removal of the head spray neither increased nor decreased RHR System reliability since it has no impact on the other operating modes of the system. Therefore, head spray removal has no significant impact on plant safety. Also, applicable CNS Technical Specifications were revised in Amendment 103 of Cooper Nuclear Station operating license.

DC 86-154

TITLE: SGT Charcoal Adsorber Bank Capacity Upgrade

DESCRIPTION: The purpose of this Design Change was to upgrade the Standby Gas Treatment (SGT) system by increasing the capabilities of the

charcoal adsorbent in capturing and retaining radioiodine. This was accomplished by adding two adsorber traps to the existing bank of four in each SGT train.

SAFETY ANALYSIS: The modifications as described by this Design Change in no way degraded the safety of Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. The modifications increased the reliability and safety margin of the operation of the SGT system by further mitigating the effects of a DBA LOCA on the safety of the public. The addition of the two charcoal absorber traps to each SGT train increased the safety margin of the system by decreasing the specific iodine loading in each train.

DC 87-024 and Revision 1

TITLE: Office Building Construction

DESCRIPTION: The purpose of this Design Change was to describe and authorize the renovation and expansion of the existing Office Building at Cooper Nuclear Station.

SAFETY ANALYSIS: The implementation of this Design Change in no way degraded plant personnel safety, equipment safety, or nuclear safety during or following the modification. This Design Change involved no systems which are credited to function in the mitigation of an accident nor were safety related systems involved with this change.

DC 87-122 and Revision 1

TITLE: IRM/SRM Negative Voltage Sensing Relay and Fuse Replacement

DESCRIPTION: The purpose of this Design Change was to provide instructions for the addition of Negative Voltage Sensing Relays to each IRM and SRM Channel and to increase the amp rating of the IRM chassis fuses from 3/4 amp to 1½ amp. This change increased channel availability and will give a more reliable indication of operable channels. This Design Change was written in response to RIC SIL 007, SER 33-86 and SIL 445.

SAFETY ANALYSIS: This Design Change provided for the addition of negative voltage sensing relays to the IRM and SRM. This will now increase monitoring by giving an alarm and indication of inoperable channel with the loss of negative voltage. The loss of positive voltage already gives such an indication. The remaining safety functions and alarm circuitry were unaffected. Therefore, the original designed safeguards remain the same. The objective of this DC was to increase startup range neutron monitoring dependability by increasing the dependability of the SRM and IRM, and minimizing the frequency of inoperative SRM/IRM. Therefore, the implementation of this Design Change did not constitute an

unreviewed safety question. Refer to Cooper Nuclear Station (CNS) Licensing Amendment 115 dated February 11, 1988, and CNS Licensing Amendment 128 dated February 21, 1989.

DC 87-189

TITLE: Non-Essential Control and Turbine Building HVAC System Modification

DESCRIPTION: This Design Change installed a 3-section chilled water/glycol solution cooling coil in the Control Building HVAC System Heating and Ventilating Unit 1-HV-C-1A to add cooling capability to the Unit. In addition, a new air handling unit was installed in the Turbine Building electrical shop and a new roof-mounted packaged air conditioning unit was installed in the Turbine Building non-critical switchgear room.

SAFETY ANALYSIS: The implementation of this Design Change did not degrade plant personnel safety, equipment safety, or nuclear safety during or following the modification. Addition of the chilled water cooling coil to the Control Building H&V Unit 1-HV-C-1A, and of the new air handling unit in the Turbine Building electric shop, and the addition of the packaged air conditioner for the non-critical switchgear room, enhanced the operation of the existing HVAC systems for these areas. Overall system performance was improved by this change because the existing HVAC system will be able to maintain the more desirable temperatures for optimum equipment performance in the areas served.

DC 87-191

TITLE: Removal of Auto Open Logic HPCI/RCIC

DESCRIPTION: The purpose of this Design Change was to remove the auto open logic from certain HPCI/RCIC motor operated valves upon HPCI/RCIC system initiation respectively, move the high water level HPCI trip signal from HPCI-MOV-M015 to HPCI-MOV-M014, and install annunciation in the Control Room for RCIC-MOV-M015, M016 not fully open.

SAFETY ANALYSIS: The startup and operational configuration of the HPCI and RCIC systems remained the same. The changes documented in this DC will enhance the performance of the HPCI/RCIC system by removing the possibility of water hammer events occurring without the HPCI/RCIC steam lines being preheated. This Design Change did not alter the isolation capabilities of the HPCI/RCIC steam isolation valves, nor did it change any functions of the affected components during operation. The margin of safety was not reduced nor was the possibility of an accident or malfunction created or increased by the implementation of this Design Change.

DC 87-195

TITLE: Alternate Power Supply to Foxboro Cabinets and Drywell Rad Monitors

DESCRIPTION: The purpose of this Design Change was to install new essential power distribution modules in each Foxboro Cabinet to meet the District's commitments for Regulation Guide 1.97 for torus temperature. This DC also changed power sources for the High Range Area Radiation Monitors for the Drywell to an essential power source.

SAFETY ANALYSIS: These modifications did not make any functional changes in systems or equipment. The operation of the equipment in the Foxboro panels and operation of the Drywell Radiation Monitors did not change. By supplying the Drywell Radiation Monitors and torus temperature measurement equipment with essential power sources, system operability and reliability were increased as compared to original design. These systems are now qualified to Reg. Guide 1.97 requirements which provides assurance that operation will be maintained throughout a seismic event or design basis accident.

DC 87-198 and Amendment 1

TITLE: Reg. Guide 1.97 Essential 120 VAC Power

DESCRIPTION: The purpose of this Design Change was to replace Critical 240/120 VAC Instrumentation and Control Distribution Transformers with equivalent essential qualified transformers to comply with Reg. Guide 1.97 commitments. The amendment to the Design Change provided for the installation of two Line Conditioners on the alternate power supply line to both divisions of the Reactor Protection System.

SAFETY ANALYSIS: There were no nuclear or equipment safety concerns during or after implementation of this Design Change. The modifications were performed with the Reactor in cold shutdown and work was limited to one division at a time. It was the intent of this DC to upgrade plant safety and reliability by installing qualified seismic equipment that will ensure that the 120 VAC post-accident monitoring systems instrumentation is powered by an essential source in accordance with Reg. Guide 1.97. Also the Line Conditioners will maintain the specified voltage (Technical Specification 4.9.A.2) at the RPS panelboards, when powered from the alternate feeder, during high and low grid voltage conditions

DC 88-036

TITLE: Feed Pump Turbine Control Upgrade

DESCRIPTION: This Design Change provided for the replacement of the existing DeLaval turbine speed control system with a new speed control

system manufactured by the Lovejoy Controls Corporation (LCC). The LCC system is an electronic/pneumatic unit which performs Reactor Feed Pump Turbine (RFPT) speed control and monitoring functions. The system normally receives an input speed demand signal from the GE reactor level manual/automatic control station. The system output consists of a pneumatic signal which positions a new Foxboro actuator which, in turn, positions the DeLaval turbine steam control valve servomotor. In addition, the existing RFPT zero-speed monitor was replaced with a LCC three-proximity-probe zero-speed monitor.

SAFETY ANALYSIS: The Design Change (installation) was performed when the reactor was in a Cold Shutdown Condition, the affected mechanical systems were isolated and electrical circuits were de-energized. Following completion of the installation, the feed pump turbine individual hydraulic, pneumatic and electrical control, protection and surveillance circuits were tested in a predetermined and deliberate sequence. To assure personnel and equipment safety, individual systems were returned to service only after that system was inspected, calibrated and tested. The feed pump turbine was operated only after the individual systems were returned to service and in accordance with approved plant procedures. This Design Change affected electrical, electronic, mechanical, hydraulic and pneumatic control devices, components, tubing and wiring which are not relied on to perform nuclear safety functions, but do control a plant process which has an impact on plant normal operation. The Design Change increased the availability and improved the reliability of the feed pump turbines by minimizing failures of the control system. This modification did not create any new mode of plant operation or affect any operating limits or setpoints. The Feedwater Controller failure transient as described in the USAR remains bounding.

DC 88-156

TITLE: Replacement of 250 VDC Station Batteries, Battery Racks, and Battery Chargers

DESCRIPTION: The purpose of this Design Change was to replace the 250V Batteries 1A and 1B, the battery racks, and the 250V Battery Chargers 1A and 1B. This Design Change also added a third charger, designated 1C, which can operate in parallel with either Charger 1A or 1B. Manual Transfer Switches and fusible disconnects were installed so that Charger 1C can be used in parallel with either Charger 1A or 1B. Charger 1C, under normal operating conditions, will be in a standby condition. The batteries in the 250V system were replaced by lead-calcium batteries which have a higher ampere-hour rating. The battery chargers for the 250V system were also replaced, but retained their amp rating. Due to the increased size of the replacement battery cells, it was necessary to replace and rearrange the battery racks. The seismic design basis of the battery cells and the racks was not changed. The existing DB-25 breakers, which

are associated with the DC output of Chargers 1A and 1B, had their coil assemblies changed from 300 ampere coils to 400 ampere coils. The 400 ampere coils provide better coordination with the feeder breakers. The existing DB 50 breakers, which are the 250V Battery feeders to the 250V Switchgear Buses 1A and 1B, had their coil assemblies changed from 800 ampere coils to 1600 ampere coils. The new 1600 ampere coils provide better coordination with the system feeder breakers. Revision to CNS Technical Specification pages were approved in Cooper Nuclear Station License Amendment No. 130 dated May 24, 1989.

SAFETY ANALYSIS: The existing safety function of the 250V DC Power System remains unchanged. No additional safety concerns beyond those already evaluated in the USAR result from this Design Change. The capacity of the new batteries envelope their design basis load profile. The margin of safety associated with the 250V DC Power System was maintained. Refer to Cooper Nuclear Station License Amendment No. 130 dated May 24, 1989. System performance was improved. The original batteries were nearing the end of their design life and showed signs of physical deterioration, which indicated the need for replacement. The new batteries have a rating of 1800 ampere hours, as compared to the original 250V batteries' rating of 1368 ampere hours. Also, with the addition of the third battery Charger 1C, reliability is also increased because it will enable operators to transfer to a live bus in the event one train of AC power is disabled.

DC 88-201

TITLE: Drywell Cooling System and Steam Tunnel Cooling System Upgrade

DESCRIPTION: The purpose of this Design Change was to install new cooling coils, coil housings, mixing dampers, and to modify the ductwork for the Drywell cooling system to increase the cooling capacity of the system. Also the Steam Tunnel coolers were modified to increase the cooling capability by increasing the air flow with larger motors and adding ductwork to the fan-coil units.

SAFETY ANALYSIS: The Drywell and Steam Tunnel cooling systems are non-essential systems and are not required for safe shutdown of the plant or mitigating the consequences of an accident. The interface between the Drywell cooling system and the Essential portion of the REC system has been reviewed and it was determined that the Drywell fan-coil units are tripped off in an accident condition, so no thermal increase on the Essential operation of the REC system occurs. Therefore, there is no affect on the Essential requirements of the REC system. The implementation of this Design Change did not degrade plant personnel safety, equipment safety, or nuclear safety. Replacement of the drywell cooling coils and dampers and modifying the ductwork to change the flows improves the ability of the drywell cooling system to perform its design function. Adding ductwork to the Steam Tunnel also improves the ability of the system to perform its design function. This

modification did not affect the design basis of the affected systems as described in the USAR. Therefore, no accident or malfunction of a different type than any previously evaluated were created.

DC 88-201A

TITLE: Reactor Recirculation Motor Generator (RRMG) Set Ventilation Upgrade

DESCRIPTION: The purpose of this Design Change was to remove the MG set exhaust louver and replace it with a steel framed exhaust stack. The new exhaust stack was installed to discharge MG set exhaust air vertically up and away from the adjacent Reactor Building HV and RRMG Set HV air intake louvers. The new exhaust will provide effective separation between the discharge and intake pathways.

SAFETY ANALYSIS: The portion of the RRMG Set ventilation system involved with this Design Change does not provide a safety function required for safe shutdown of the reactor or mitigate an off-site release. However, with the implementation of this DC improves system performance and reliability of the RRMG Set ventilation system and HV system during normal operating conditions was increased. The margin of safety was not reduced nor was the possibility of an accident or malfunction created or increased with the implementation of this Design Change.

DC 88-222A

TITLE: DCRDR Modifications to Control Room Panel A

DESCRIPTION: Design Change 88-222A modified Control Room Panel A to comply with recommendations made per the CNS Detailed Control Room Design Review (DCRDR) Summary Report issued February 4, 1985. It was the primary purpose of this Design Change to modify the existing Cooper Nuclear Station (CNS) Control Room Panel A in order that the displays and controls are arranged more in accord with established Human Factors Engineering (HFE) guidelines. Application of these HFE guidelines modifies the existing control and display positional relationships and labeling practices.

SAFETY ANALYSIS: There were no major modifications made to any of the systems that would change the operability of the system as a result of this DC. The majority of the modifications were to panel arrangements and were solely to the positional and labeling relationships of the operator controls within the Control Room. This modification did not make any functional changes in system operation, components or equipment. No possibility for an accident or malfunction of a different type than previously evaluated in the USAR exists.

DC 88-263 and Amendment 1

TITLE: Second Level and 69 kV Undervoltage Relay and DG Auto-Start Modification

DESCRIPTION: The purpose of this Design Change was to modify the existing second level undervoltage relaying system, add second level undervoltage relays to the Emergency Transformers (ET), and remove the unnecessary auto-start signals from the Diesel-Generator (DG) control circuitry. The second level undervoltage relaying scheme was modified to comply with NRC Branch Technical Position PSB-1. The second level undervoltage relays were added to the Emergency Transformers to prevent the transfer of the Critical Buses 1F and 1G to a degraded source. The auto-start signals were removed from the DG control circuitry to reduce the number of unnecessary starts of the DGs.

SAFETY ANALYSIS: The implementation of this DC was performed while the plant was in cold shutdown. Disabling the second level undervoltage system for one of the critical 4160 Volt buses caused the bus to be inoperative. Therefore, the work was coordinated with the annual inspection of the diesel generators to prevent having both critical 4160 Volt buses inoperative. The critical bus undervoltage protection and diesel generator auto-start circuitry were operational for the operable diesel generator. There were no nuclear or equipment safety concerns during or after implementation of this Design Change since the modifications were performed with the Reactor in cold shutdown and limited to work on one division at a time. The Control Room was notified when the second level undervoltage protection for a critical 4160 Volt bus was out-of-service so that the bus voltage could be monitored regularly. This precluded operating any equipment at a degraded voltage.

The implementation of the changes to the second level undervoltage protection for Buses 1F and 1G did not degrade the safety function of the undervoltage protection system. The non-essential loads and motor loads will be shed from the critical buses during a sustained degraded voltage condition prior to transferring the critical buses to the Emergency Transformer. This will increase the ability of the Emergency Transformer to perform its function if safety-related motors have to be sequenced back on to the critical buses. The addition of second level undervoltage relays on the Emergency Transformer will prevent transferring Buses 1F and 1G to degraded power source. All sustained degraded voltage conditions will be annunciated in the Control Room.

Removal of the diesel-generator auto-start signals will reduce the number of unnecessary starts of the diesels without degrading the safety function of the diesel generators. Loss of voltage to the critical 4160 Volt Buses 1F and 1G and a LOCA will still cause the diesel generators to autostart. The signals that are being removed are signals of conditions that could ultimately result in a loss of voltage on Buses 1F and 1G. Since this

condition still causes an auto-start of the diesel generators, the safety function of the diesel generators is not being degraded. General Electric has performed a Safety Analysis of the removal of the auto-start signals. The conclusion of the Safety Analysis by G.E. is that the removal of the auto-start signals does not degrade the system and will not have any adverse affect on the safety and reliability of the plant.

DC 89-011

TITLE: Recycle Elimination for Augmented Off Gas Recombiner Train A

DESCRIPTION: The purpose of this Design Change was to eliminate the recycle loop of the Augmented Off Gas (AOG) Recombiner Train A and to replace the dilution source with a new steam line taken directly from the main steam supply line. This retrofit of the AOG system will reduce the possibility of internal hydrogen rapid burns caused by catalyst migration, will make "A" train more reliable, will reduce airborne radioactivity, and will increase the reliability of the hydrogen analyzers.

SAFETY ANALYSIS: This modification did not interface with any safe shutdown systems and was considered non-essential. This modification enhanced the overall effectiveness and functionality of the AOG Recombiner Train A by providing better controls in respect to the treatment and release of radioactive gases. This DC changed the autoisolation capabilities of the AOG system, thereby increasing the reliability of the system against an uncontrolled release of contaminated materials. This modification in no way affected the current system for monitoring the levels of activity or the quantities of contaminated gases. This modification enhanced the overall safety of the plant by reducing system maintenance and associated personnel radiation exposure, and improved system performance by reducing radioactive gaseous release, and the likelihood of off gas ignitions.

ESC 89-126

TITLE: Main Steam Line LRM Replacement

DESCRIPTION: The purpose of the Equipment Specification Change was to replace the four Main Steam Line (MSL) logarithmic radiation monitors (LRM) with new microprocessor-based log radiation monitors.

SAFETY ANALYSIS: This activity was performed while the plant was in cold shutdown when the scram function for the instrument is not needed per Table 3.1.1 of the CNS Technical Specification. The replacement log radiation monitors are microprocessor-based and are significantly less susceptible to noise, due to a built-in software feature that discriminates between valid and noise input signals based on their expected rate of change. These monitors are equipped with an auto-calibration feature, a significant

operational improvement. Input power requirements and output detector and alarm circuitry were unchanged. This activity did not impact any other plant systems nor did it alter the design basis of any associated equipment or components. Therefore, the margin of safety was not reduced nor was the possibility of an accident or malfunction created or increased with the implementation of this ESC.

DC 89-184

TITLE: CNS Cycle 13, Reload 12 Design and Safety Analysis

DESCRIPTION: The purpose of this Design Change was to document the loading of 104 new fuel assemblies into the core for Cycle 13 operation. The fuel bundle design was the same as previous reloads. This change also reviewed the results of the safety analysis performed by General Electric for Cooper Nuclear Station Cycle 13 core reload design.

SAFETY ANALYSIS: The Cycle 13 reload design was reviewed and analyzed using methodologies described in NEDE-24011-P-A (latest approved versions). The analysis for the specified abnormal operational transients and design basis accidents of Section XIV of the CNS USAR remains bounding for the Cycle 13 reload. 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. No physical changes to plant safety systems were implemented by this Design Change and all safety systems will continue to be operated in their normal (as designed) configuration.

ESC 89-228

TITLE: Barton Wire Replacement

DESCRIPTION: The purpose of this Equipment Specification Change was to replace the local wires from 30 Barton Switches to the local terminal boxed located in the Reactor Building to ensure compliance with 10CFR50.49.

SAFETY ANALYSIS: This Equipment Specification Change was necessary to ensure qualification of DPIS's. The subject wire replacement was approximately 15 ft. in length. There was no effect on operation because this ESC was performed while the plant was in a cold shutdown condition. There was no impact on other plant systems nor did this change alter the design basis of any associated equipment or components. This change improved plant reliability because the new wire that was installed is qualified to survive the postulated environment in the installed area as described in 10CFR50.49.

III. PERSONNEL AND MAN-REM EXPOSURE BY WORK AND JOB FUNCTION

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 & SURV.						
Maintenance Personnel	5	0	9	0.284	0.000	0.960
Operating Personnel	46	0	0	16.343	0.000	0.000
Health Physics Personnel	24	0	24	5.668	0.000	5.158
Supervisory Personnel	7	0	1	2.127	0.000	0.152
Engineering Personnel	13	3	13	4.314	0.021	2.838
ROUTINE MAINTENANCE						
Maintenance Personnel	74	0	353	49.999	0.000	163.089
Operating Personnel	1	0	0	0.001	0.000	0.000
Health Physics Personnel	21	0	25	13.947	0.000	7.722
Supervisory Personnel	5	0	2	0.776	0.000	0.541
Engineering Personnel	4	25	28	0.129	10.454	19.799
SPECIAL MAINTENANCE						
Maintenance Personnel	5	0	1	0.374	0.000	0.009
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	5	0	1	0.318	0.000	0.017
Supervisory Personnel	1	0	0	0.037	0.000	0.000
Engineering Personnel	3	0	12	0.074	0.000	1.039
WASTE PROCESSING						
Maintenance Personnel	2	0	6	0.017	0.000	0.219
Operating Personnel	5	0	0	2.725	0.000	0.000
Health Physics Personnel	7	0	1	0.395	0.000	0.004
Supervisory Personnel	1	0	0	0.049	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
REFUELING						
Maintenance Personnel	0	0	4	0.000	0.000	0.576
Operating Personnel	19	0	0	1.261	0.000	0.000
Health Physics Personnel	0	0	4	0.000	0.000	0.720
Supervisory Personnel	1	0	0	0.083	0.000	0.000
Engineering Personnel	2	0	0	0.146	0.000	0.000
INSERVICE INSPECTION						
Maintenance Personnel	0	0	21	0.000	0.000	13.575
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	1	0	0	0.002	0.000	0.000
Engineering Personnel	0	0	4	0.000	0.000	0.778
TOTALS						
Maintenance Personnel	74	0	373	50.674	0.000	178.428
Operating Personnel	48	0	0	20.330	0.000	0.000
Health Physics Personnel	24	0	25	20.328	0.000	13.621
Supervisory Personnel	7	0	2	3.074	0.000	0.693
Engineering Personnel	13	25	40	4.663	10.475	24.454
GRAND TOTALS	166	25	441	99.069	10.475	217.196