Nebraska Public Power District

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February 28, 1992

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, hereby, submits the Cooper Nuclear Station Annual Operating Report for the period of January 1, 1991, through December 31, 1991.

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,

Guy Horn

G. R. Horn
Nuclear Power
Group manager

GRH/tja:91an-rpt.ltr
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, 1991 THROUGH DECEMBER 31, 1991

USNRC DOCKET 50-298

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I. PERFORMANCE CHARACTERISTICS

FUEL PERFORMANCE

Cycle 14 operation continued from January 1, 1991 through March 22, 1991. Operation of the unit was interrupted on March 23, 1991, via a manual scram, for repair associated with a feedwater check valve. The unit was restarted on March 27, 1991 and continued operation until May 9, 1991. The unit was shutdown on May 9, 1991 for repair of a Core Spray Valve. Startup of the unit commenced on May 10, 1991 and operation in Cycle 14 continued until October 4, 1991.

Cycle 14 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% or greater.

During the period from October 4, 1991 through December 14, 1991, 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 164 fuel assemblies. With the concurrence of General Electric, it was decided that sipping for leaking fuel assemblies was not warranted due to extremely low off-gas activity.

Cycle 15 operation commenced with initial reactor startup on December 15, 1991 and 100% thermal power was initially achieved on December 26, 1991. The startup physics test program was completed on January 13, 1992, with notification of test completion to be submitted to the NRC under separate correspondence.

Cycle 15 operation continued through December 31, 1991. 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% or greater.

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

There were no operational failures or challenges to the Main Safety Valves or Main Steam Safety Relief Valves during the operational year of 1991.

The "challenges" made in accordance with Technical Specification 4.6.D.1, cyclic testing, were documented in LER 91-015.

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

REPORTABLE SPECIAL PROCEDURES / SPECIAL TEST PROCEDURES

SP 86-004

TITLE: PMIS Functional Check

DESCRIPTION: This Special Procedure (SP) provided specific instructions to operations personnel regarding verification checks during testing of the Plant Management Information System (PMIS) installation. This Special Procedure performed these checks to ensure that the PMIS/SPDS (Safety Parameter Display System) was completely functional for use by CNS Operations personnel.

SAFETY ANALYSIS: This SP was intended only to functionally check the PMIS/SPDS operation. No systems were directly affected by this SP. The PMIS computer system is part of the plant monitoring system only, and does not directly affect the performance or operation of any plant system. This SP did not create an accident or malfunction of a different type, nor decrease the margin of safety of CNS because it is only a plant monitoring system and does not directly affect the operation of any plant system including those important to safety.

SP 90-231

TITLE: In-Service Testing of REC-CV-16CV

DESCRIPTION: The purpose of this Special Procedure (SP) was to test the operation of REC-CV-16CV. ASME Section XI, requires all valves which are required to operate in order to perform their safety function be routinely tested. This SP simulated a depressurization of the non-critical portion of the Reactor Equipment Cooling (REC) System by opening two temporary installed valves upstream of the REC-CV-16CV and verifying closure of REC-CV-16Cv by observing a reduction in flow.

SAFETY ANALYSIS: This SP did not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. This SP did not shut down the critical REC loops or jeopardize the operability of the essential portions of the REC System. The SP was written to maintain the critical loops of REC operable during the testing of REC-CV-16CV. A partial REC System outage was required, but only shut down the non-critical, non-essential loops of the REC. The plant was in cold shutdown status (<212 °F) during the test with fuel pool cooling being the only non-critical load requiring support. Additionally, Spent Fuel Pool Cooling could have been provided by Residual Heat Removal System, if necessary.

SP 90-235

TITLE: Post Accident Sampling System (PASS) Line Backflush Procedure

DESCRIPTION: The purpose of this Special Procedure was to backflush the PASS to increase the flow rates by removing any potential foreign material or other obstructions that could possibly cause a partial flow restriction in this system.

SAFETY ANALYSIS: The PASS has no safety function. The only function of the PASS is to allow collection of reactor coolant, torus water, and containment atmosphere during accident conditions, to assist operations personnel in evaluating the severity of nuclear accidents. This procedure did not change the form, fit, or function of the PASS. In addition, all PASS sample lines were isolated from reactor systems during this procedure, and the pressure used to backflush was less than the rated capacity of the tubings and fittings in the PASS. Also, this procedure included steps to prevent discharged material from becoming airborne or discharged into the Reactor Building, and the flushing medium was non-toxic. Therefore, performing this Special Procedure did not create an unreviewed safety question.

STP 87-008

TITLE: Rod Worth Minimizer Operability

DESCRIPTION: This Special Test Procedure (STP) was designed to accomplish a thorough evaluation of the Science Applications International Corporation (SAIC) installed improvements to the Rod Worth Minimizer (RWM). The purpose of this STP was to establish that the corrections made to the RWM program eliminated previously identified program deficiencies, and that no new significant problems remained after these changes were installed.

SAFETY ANALYSIS: Performance of this STP involved movement of control rods (limited to one notch) while in the 75% power region. A Rod Withdrawal Error Analysis of this sort has been analyzed in the USAR (Section XIV-5.1.5) for the continuous withdrawal of the fully-inserted maximum worth rod at maximum drive speed. This STP was bounded by the worst case condition in which the reactor is in its most reactive state with no xenon or samarium present in the core. Additionally, a limiting control rod pattern surrounding this maximum worth rod was imposed such that the withdrawal of the control rod would not breach thermal limits. This STP did not change the plant facility or procedures as described in the USAR or the Technical Specifications. All safety aspects were reviewed, and there was no possibility of an accident or malfunction of equipment important to safety as a result of performing this test procedure.

STP 87-021

TITLE: RRMG Set - Computer Point Installation/Data Collection

DESCRIPTION: This Special Test Procedure was performed to investigate certain minor undemanded speed changes previously observed in the Reactor Recirculation Motor Generator (RRMG) Control System. Additionally, this STP temporarily modified each RRMG Set's control system to gather data with the intent to monitor and troubleshoot the control system.

SAFETY ANALYSIS: The installations performed as a part of this STP only affected the speed control circuit of the recirculation pumps, and did not alter the effects of a recirculation pump trip, nor did this STP alter any of the logic initiating a pump trip. Therefore, by not affecting the results of recirculation pump trips, the probability or consequences of an accident analyzed in the USAR or Technical Specifications was not increased, or the possibility of an un-analyzed accident created.

STP 88-002

TITLE: Kaman Interrupt Removal

DESCRIPTION: The purpose of this Special Test Procedure was to collect data on the Elevated Release Point (ERP) Kaman high range high (HRH), and high range normal (HRN) radiation monitors during normal operating conditions. This STP also performed tests that removed jumpers for the unused interrupts on the system and CPU boards and required the monitors to be off-line.

SAFETY ANALYSIS: This Special Test Procedure did not degrade Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. This test required the ERP Kaman 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 this equipment, and the monitors were returned to service with-in their specified limits.

STP 88-189

TITLE: Trial Chemical Treatment Of Circulating Water System

DESCRIPTION: The purpose of this Special Test Procedure was to conduct a trial use of a chemical dispersant (non-ionic polymer antifoulant) to clean the turbine generator condenser tubes by intermittently adding this chemical to the Circulating Water (CW) System downstream of the traveling screens, and determine its effectiveness.

SAFETY ANALYSIS: The Circulating Water System is not safety related and is not credited with a safety function in any USAR accident or transient analysis. The only effect of this STP on plant operation was to potentially increase the efficiency of the turbine generator condenser by reducing fouling of the water side of the condenser tubes. The addition of the antifoulant will compensate for the reduced scouring action from river sand during the winter months. This STP did not effect station safety, operation or the function of any safety related equipment, and did not involve an unreviewed safety question.

STP 88-194

TITLE: PMIS Digital Point Overcurrent

DESCRIPTION: The purpose of this Special Test Procedure was to monitor the 28 volt power supplies that supply sensing voltage to the digital input points of the Plant Management Information System (PMIS) Multiplexer (MUX) Links. This provided data, to allow for troubleshooting and repair of intermittent short circuits that have been causing the power supply fuses to open.

SAFETY ANALYSIS: This Special Test Procedure was intended only to monitor the 28 volt power supplies to the PMIS-MUX-LINKS to provide data for troubleshooting. The affected PMIS multiplexers do not perform a safety function nor impact any Technical Specification. The PMIS computer system is part of the plant monitoring system only, and does not directly affect the performance or operation of any plant system including those important to safety. Therefore, performance of this STP did not involve an unreviewed safety question.

STP 88-201B

TITLE: Monitoring Of Steam Tunnel Cooling System Temperatures

DESCRIPTION: The purpose of this Special Test Procedure was to obtain ambient air, cooling water, and surface temperature associated with the cooling system and heat load contributors in the Steam Tunnel. Thermocouples were temporarily installed in the Steam Tunnel to achieve this purpose. The information obtained from this STP will be used in evaluating Steam Tunnel cooling system performance.

SAFETY ANALYSIS: The Steam Tunnel Cooling System which is part of the Turbine Equipment Cooling (TEC) System is a non-essential, non-safety related system. Thus, this STP did not affect any systems performing a safety function. The function of the Steam Tunnel cooling system is to provide normal operation cooling of the area. The only purpose of this STP was to monitor and obtain data for system performance evaluation. There was no changes in system components or system operating characteristics, therefore, the affect on overall plant safety was not changed.

STP 88-286

TITLE: Radiation Monitor Flow Switch (RMP-FS-332) Replacement

DESCRIPTION: The purpose of this Special Test Procedure was to provide for the temporary installation and performance testing of a replacement component for RMP-FS-332, Service Water effluent radiation monitor sample flow switch.

SAFETY ANALYSIS: The flow switch involved with this STP was non-essential, and is not required to perform any safety-related function. The replacement flow switch for the Service Water effluent is anticipated to improve overall monitor reliability. Performance of this STP was in conformance with plant Technical Specifications which require that process radiation monitoring equipment shall be operable; except for maintenance, and required tests, checks, and calibration. Therefore, this STP did not involve an unreviewed safety question.

STP 89-091

TITLE: Pall Condensate Filter Septum Performance Evaluation

DESCRIPTION: Special Test Procedure 89-091 provided for the installation and performance evaluation of Pall Corporation's "Profile" filter septa which was installed in the "C" and "D" condensate demineralizer vessels. This STP evaluated the septa for resin leakage, soluble and insoluble metal removal, pressure drop before and after precoat, total volume of water processed between precoats, and septa life span.

SAFETY ANALYSIS: The "C" and "D" condensate filter septum replacement was necessary due to partial filter pluggage and was considered routine maintenance. The Pall septa were a direct retrofit of the existing Graver demineralizer. Therefore, no vessel modification was required prior to installation/replacement. The Filter septa were removed and replaced utilizing CNS Procedure 2.2.5 "Condensate Filter Demineralizer System", and filter performance data was collected utilizing CNS operating and chemistry procedures. Therefore, all margins of safety as defined by Technical Specification, USAR, and plant procedures were maintained. This STP did not involve an unreviewed safety question.

STP 89-212

TITLE: Measurement of Air Flow & Air Temperature of the Control Room

DESCRIPTION: The purpose of this Special Test Procedure was to measure and record air flows and temperature data in the Control Room and Cable Spreading Room under normal operating conditions. This data will than be used to calculate the Control Room Envelope internal heat load. This calculation will be compared to the original heat load calculation to determine if additional heat load has been added to the Control Room Envelope since initial plant start up.

SAFETY ANALYSIS: This STP took air flow and temperature measurements in the Control Room and Cable Spreading Room areas. The STP used a pitot tube, manometer, and thermometer to measure HVAC System air flows and temperature. The measurements were taken at selected locations in the Control/Cable Spreading Room (Control Room Envelope) during normal operating conditions. In addition, temperature in the rooms was monitored at 60 minute intervals during performance of the STP. Therefore, operability of all essential and non-essential components were not affected by the performance of this STP. This STP did not increase the probability or consequence of any accidents or malfunctions previously evaluated, nor create a possibility for an accident or malfunction of a different type than previously evaluated in the USAR. All applicable plant Technical Specifications pertaining to the Control/Cable Spreading Rooms and associated essential equipment, and room air flow/temperature were maintained at all times. Therefore, all margins of safety as defined in the Technical Specifications were maintained. This STP did not involve an unreviewed safety question.

STP 89-246

TITLE: Primary Cont. Purge & Vent Valves 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 operating cycle per 10CFR50 Appendix J. However, NRC concerns have prompted increased frequency of the tests to verify the adequacy of the valves' 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 not degrade the Primary Containment isolation capability of the subject valves. 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 90-174

TITLE: Loss of Gland Water to the Service Water Booster Pump Functional Test

DESCRIPTION: The Purpose of this Special Test Procedure was to determine what effect loss of gland cooling water would have on the Residual Heat Removal (RHR) Service Water Booster Pump (SWBP) mechanical seals.

SAFETY ANALYSIS: This STP was performed just prior to a scheduled maintenance inspection on a selected SWBP. The maintenance inspection and the performance of this STP required entering a Limiting Condition of Operation (LCO) while the SWBP was out of service. This STP did not adversely impact any of the design features of the Service Water System and the STP provided for sufficient monitoring of the pump to detect and prevent any damage. This STP was performed in compliance with the LCO as specified in the CNS Technical Specifications. Therefore, by complying with Technical Specification's operability requirements, the probability of occurrence or the consequences of an accident or malfunction previously evaluated was unchanged. The margin of safety as used in the basis for any Technical Specification was not reduced, therefore, an unreviewed safety question did not exist.

STP 90-269 Amendment 1

TITLE: Primary Cont. Purge & Vent Valves 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 operating cycle per 10CFR50 Appendix J. However, NRC concerns have prompted increased frequency of the tests to verify the adequacy of the resilient seats. In addition, STP 90-269 Amendment 1 replaced STP 90-269 "Primary Containment Purge and Vent Valve Quarterly Leak Rate Tests" in its entirety.

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 not degrade the Primary Containment isolation capability of the subject valves. 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.

REPORTABLE DESIGN CHANGES

DC's 87-015MA, MB, MD, and ME

TITLE: CNS Annunciator Upgrade Project

DESCRIPTION: These Design Changes were a continuation of the Detailed Control Room Design Review (DCRDR) Annunciation Upgrade Project. This Project replaced the existing non-essential (not safety related) Cooper Nuclear Station (CNS) Control Room Annunciator System with an upgraded non-essential Annunciator System which incorporates reflash and sequential events recording capabilities and is designed to meet Human Factors Engineering (HFE) guidelines. These Desig.. Changes accomplished the replacement of the existing Panalarm Control Room Windowboxes in the following Control Room Panels (A, E, H, K, M, P-1, P-2, Q, R and 9-3) with new windowboxes, Control Room Supervisor's CRT, Alarm printers, Panel CRT's, and the necessary connections.

SAFETY ANALYSIS: These Design Changes improved the annunciator system performance and reliability as well as resolved Human Factors Engineering deficiencies. These Design Changes did not affect the actual systems (alarm inputs) that the affected Panels monitor. The performance and reliability of the systems which provide input to the Annunciator system were not changed by these Design Changes, and the Plant was in a cold shutdown condition while the work was implemented. The Annunciator System is a non-essential system and its operation, although desirable, is not necessary to obtain safe shutdown of the plant. A failure of any portion, or the entire Annunciator System will not jeopardize the plant safe shutdown capabiliti·

Although the Annunciator System interfaces with safety systems, neither an Annunciator System failure or an interface failure will jeopardize the plant safety system capabilities. These Design Changes did not modify the function of any safety system. The upgraded Annunciator System only utilizes passive monitoring of existing instrumentation contacts. In addition, all applicable Technical Specifications were adhered to during implementation of these Design Changes. Therefore, these modifications did not change the existing accident analyzes for Cooper Nuclear Station, nor the probability or consequences of an accident as analyzed in the USAR. No reduction in the margin of safety was involved with implementation of these Design Changes.

DC 87-023

TITLE: RAD Chemical Laboratory Modifications

DESCRIPTION: This Design Change performed environmental and instrumentation modifications to the Radiological Chemical Laboratory in the Radwaste Building at Cooper Nuclear Station. In addition, this Design Change provided for improved monitoring for the Condensate Filter Demineralizer Conductivity Sampling System as well as improved environmental and space conditions for the RAD Chemical Laboratory personnel.

SAFETY ANALYSIS: This Design Change did not affect any safety related systems, nor did it affect the safe operation or shutdown of any essential system, and was classified as non-essential. Since none of the changes associated with Design Change 87-023 affected any safe shutdown systems or components and the quality of materials were equal to or greater than those specified in the original construction, implementation of this design change did not increase the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the USAR. In addition, this design change did not require any changes or additions to the Technical Specifications and involved no decrease in the margin of safety. Therefore, no unreviewed safety question existed with the implementation of this design change.

DC 87-023 Amendment 1

TITLE: RAD Chemical Laboratory Modifications - Amendment 1

DESCRIPTION: The purpose of this Amendment to Design Change 87-023 was to provide for the further expansion of the Radiological Chemical Laboratory Office located in the Radwaste Building at Cooper Nuclear Station.

SAFETY ANALYSIS: This Amendment did not involve modifications to safety equipment and was, therefore, considered non-essential. The implementation of this amendment did not degrade plant personnel safety, equipment safety or nuclear safety during or following the modifications. This Amendment did not increase the possibility of any accident occurrence, nor did it decrease the safety margin as defined by the basis for any Technical Specification, therefore, it did not involve an unreviewed safety question.

DC 88-053B

TITLE: Essential Control Building Ventilation System

DESCRIPTION: This Design Change (DC) installed an essential (safety related) ventilation system to the Critical Switchgear Rooms and to the Control Building (903'6" elevation) to provide

cooling under abnormal and accident conditions. This change utilized a network of essential supply and exhaust fans to remove the heat generated by both the essential electrical equipment and normal equipment in the control building. This system was classified as a ventilation system because no chilled water or mechanical refrigeration were involved. This DC fulfilled the District's commitment made in response to NRC Inspection Report 50-298/89-13, dated April 10, 1989, to install the essential ventilation system for cooling the Critical AC and DC electrical equipment rooms under abnormal and accident conditions.

SAFETY ANALYSIS: The new Essential Ventilation System was designed for redundancy, single failure criteria, separation criteria, and other rules applicable for a nuclear safety-related system. This new ventilation system will provide a safety function in that it will maintain acceptable room temperatures to further ensure that the necessary electrical equipment is available to safely shutdown the reactor. No new safety concerns were created with the implementation of the Design Change, in addition, CNS plant safety was enhanced by the installation of the Essential Control Building Ventilation System. 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 Design Change.

DC 88-201B

TITLE: Steam Tunnel Cooling System Upgrade

DESCRIPTION: This Design Change modified the Steam Tunnel Cooling System. The modifications included the installation of replacement fan coil units (FCU), and the addition of insulation to the Main Steam Line penetration guard sleeves in the Steam Tunnel to reduce the Steam Tunnel heat load. The Steam Tunnel FCU's replacement and insulation installation were performed to decrease area temperature during normal operation.

SAFETY ANALYSIS: The implementation of this Design Change did not degrade plant personnel safety, equipment safety, or nuclear safety. The Steam Tunnel Cooling system is non-essential and is not required for safe shutdown of the plant, or mitigating the consequences of an accident. The ability of the Steam Tunnel Cooling system to perform its function was increased by this modification. The performance of the cooling system was enhanced by an increase in system heat load removal capabilities and by the reduction in heat input from the Main Steam Line penetration guard sleeves. The design function and operation of the Steam Tunnel Cooling system remained unchanged. Therefore, this Design Change did not create an unreviewed safety question or have an adverse effect on nuclear safety.

DC 89-002

TITLE: West Warehouse Utilities Connection

DESCRIPTION: The purpose of this Design Change was to connect the new West Warehouse at CNS to the existing fire potable H_2O_2 protection lines, Caitronics intercommunications system, and the 12.5kV ring bus power line. In addition it also provided for the installation of the sprinkler system in the building and the fire alarm annunciator panel in the Central Alarm Station.

SAFETY ANALYSIS: The modifications outlined by this Design Change did not 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 of the plant. This DC did not create an unreviewed safety question, nor did it reduce the margin of safety defined in the Technical Specifications.

DC 89-180

TITLE: Testable Check Valve Actuators

DESCRIPTION: The purpose of this Design Change was to improve the testing which ensures the operability of the Core Spray (CS) and Residual Heat Removal (RHR) systems testable check valves. This was accomplished by removing the air operated actuators and the motor operated bypass valves (CS system only) associated with these testable check valves. Modifications to the RHR and CS testable check valves are consistent with the NRC recommendations per AEOD/C502, "Overpressurization of Emergency Core Cooling Systems for Boiling Water Reactors". The motor operated bypass valves (CS system only) were no longer required once the air operators were removed, therefore, they were also removed. In addition, the disc position indicators were modified to improve the valve position indication reliability.

SAFETY ANALYSIS: The implementation of this DC was performed while the plant was in a cold shutdown condition. This DC did not change the original design basis of the testable check valves or affect the safety function of the affected systems. However the changes did improve the reliability of the CS and RHR testable check valves by modifying the reed switches for more reliable position indication. In addition, the CS system bypass valves piping and instrument air drywell penetrations were cut, capped, and hydrostatically (leak) tested. This Design Change did not alter the capabilities of the CS or RHR testable check 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 89-207

TITLE: H_2O_2 Analyzer Particle Filters - Isolation Valve Modifications

DESCRIPTION: The purpose of this Design Change was to install four particle filters in the sample lines to the H_2O_2 Analyzer Units to improve Analyzer operation. Four additional isolation valves were installed in the sample lines to provide Operations a simpler way to isolate the drywell and torus from the H_2O_2 Analyzer Units. Two shutoff valves were installed in the return lines from the H_2O_2 analyzers to the torus to provide a simpler way for Operations to isolate the Analyzer Units. Finally this DC installed four sample point valves in the sample lines to provide a easier way to perform grab samples when the H_2O_2 Analyzer Units are inoperable.

SAFETY ANALYSIS: The Containment Atmosphere Monitoring System operation did not change with the implementation of this Design Change. The installation of the particle filters provide additional assurance that abrasive particles will not enter the analyzer units. The implementation of this DC increased system reliability, efficiency, and operator performance. Installation of the filters and valves did not affect the H_2O_2 monitors, in that the system will retain its safety features and that operation of the system during a design basis event remains as specified. This DC was implemented with the Reactor in a cold shutdown condition. This DC did not constitute an unreviewed safety question, nor did it reduce the margin of safety defined in the Technical Specifications.

DC 89-256

TITLE: Reactor Water Cleanup (RWCU) Pipe Replacement

DESCRIPTION: The purpose of this Design Change (DC) was to replace portions of the Reactor Water Cleanup (RWCU) piping system with material that is resistant to intergranular stress corrosion cracking (IGSCC). In addition, this DC implemented several minor modifications (upgrade of the RWCU pumps, addition of a subcooling line, bypass lines, and replacement of excess flow elbow taps with an Annubar flow element) to improve system performance and reliability.

SAFETY ANALYSIS: This Design Change enhanced the existing plant design by upgrading RWCU pump performance, replacing portions of the RWCU piping and adding bypass lines, subcooling lines, annubar flow elements etc.. These changes resulted in several benefits including simplification of operation, reduced maintenance, and reduced radiation exposure. These changes were implemented while the plant was in a cold shutdown condition. No safety design basis or functional requirements of the systems were affected. Therefore, this modification did not change the existing safety analysis for Cooper Nuclear Station, nor the probability or consequence of an accident as analyzed in the CNS USAR.

DC 89-272

TITLE: Combustible Gas Control - Standby Nitrogen Injection (SBNI)

DESCRIPTION: This Design Change provided a Standby Nitrogen Injection (SBNI) System for injecting nitrogen gas into the Primary Containment. The purpose of this nitrogen injection is to provide an emergency backup system to the normal Primary Containment nitrogen supply system. These systems are to dilute the combustible gases produced by radiolytic decomposition of reactor coolant and metal-water chemical reactions which occur following a Loss of Coolant Accident (LOCA). The SBNI system will provide the necessary supply of nitrogen for a minimum of 48 hours by which time an off-site long-term source of nitrogen will be available. The conceptual design, system classification, and other details were given in a Letter from NPPD to NRC dated September 28, 1989, "Post Accident Combustible Gas Control".

SAFETY ANALYSIS: The SBNI system is provided with two independent connection points, and two independent and redundant injection paths into both the drywell and wetwell (suppression chamber) portions of the CNS Primary Containment. The existing purge and vent system was the primary means of nitrogen inerting and was therefore subjected to gaseous nitrogen under normal conditions. The replacement of the existing ACAD system with the new SBNI system will enhance the ability to inject dilution gas into the primary containment during accident conditions. The SBNI system is provided with two complete and redundant pathways and nitrogen sources ensuring against a single component or pathway failure which would prevent the overall system from performing its function. This DC did not constitute an unreviewed safety question, nor did it reduce the margin of safety defined in the Technical Specifications.

DC 89-286A

TITLE: Performance/Reliability Monitoring Instrumentation

DESCRIPTION: This Design Change installed instruments to monitor the performance and reliability of safety related components. The safety related components included the Diesel Generators, the Residual Heat Removal System, and the Diesel Generator Service Water System. These instruments were included in the Performance/Reliability Monitoring Program implemented at CNS. The installation of these instruments will allow for the collection of data required for in-depth analysis of system components and their performance. This analysis will further ensure the reliability of the safety systems by providing valuable trending data.

SAFETY ANALYSIS: No personnel, equipment, or nuclear safety concerns existed with the implementation of this DC. The components installed per this DC are passive components (local indicators only) and do not provide any controlling functions. The addition of

these instruments does not affect the operational characteristics of any of the safety related or non-essential components/systems. Since there was no change in any system components or operating characteristics, the effect on overall plant safety was not changed.

DC 89-289

TITLE: Instrument Air (IA) Post Filter "B" Replacement

DESCRIPTION: The purpose of this Design Change involved the replacement of the Instrument Air Dryer B Post Filter, and the replacement of the existing filter cartridges on both Trains A and B with high temperature cartridges. In addition, piping was rerouted, a filter blowdown valve was added and a connection for a temperature indicator was installed.

SAFETY ANALYSIS: This Design Change did not create the possibility of an accident or malfunction of a different type than previously identified. The portions of the affected system that provide engineered safeguards and reactor protection functions were not altered by this DC. The Instrument Air system design, operation, capacity and, capability were not degraded by this DC. The Instrument Air system operates in the same manner as it did prior to the modifications. Therefore, this DC did not constitute an unreviewed safety question, nor reduce the margin of safety as defined in the Technical Specifications.

DC 90-004

TITLE: Non-Critical AC Bus Coordination

DESCRIPTION: Design Change DC 90-004 involved modifying various non-critical 4160V relays and 480V circuit breakers by changing the setpoints, refurbishing solid-state trip devices and changing their setpoints, or replacing the circuit breakers with fused disconnects. The changes were based on the analysis of the Non-Critical AC Buses at Cooper Nuclear Station by Nebraska Public Power District. These changes provide improved breaker coordination and will reduce the possibility of false trips during fault conditions.

SAFETY ANALYSIS: Equipment reliability was enhanced by the implementation of this Design Change. The modifications performed provided for adequate circuit coordination margins on non-critical AC buses which supply power to non-safety related loads at CNS. With the new coordination margins, a failure of any equipment, component, or cable will isolate the failure to a minimum area, thereby, decreasing the probability that the failure of the equipment or cable will affect additional equipment. All work done by this DC was done in accordance with established plant procedures for setting and/or testing breakers. No unreviewed safety questions were created, nor was any Technical Specification margin of safety reduced.

DC 90-021

TITLE: PMIS Augmentation, Phase II

DESCRIPTION: The purpose of this Design Change was to provide additional plant parameter inputs to the Plant Management Information System (PMIS), to correct human factors deficiencies observed during the Detailed Control Room Design Review (DCRDR), and to comply with NUREG-1342 requirements for implementation of Safety Parameter Display System (SPDS). In addition, this DC implemented additional PMIS points to improve monitoring of system operation.

SAFETY ANALYSIS: Although the PMIS interfaces with safety systems, PMIS failure will not jeopardize the plant safety system capabilities. All interfaces with safety systems utilize essential components. This Design Change did not modify the function of any safety system. The upgraded PMIS only utilizes passive monitoring of existing instrumentation contacts. Therefore, this modification did not change the existing accident analyses for Cooper Nuclear Station, nor the probability or consequences of an accident as analyzed in the USAR. No reduction in the margin of safety was involved with implementation of this Design Change.

DC 90-181

TITLE: Modification Of DC Westinghouse DB Breakers

DESCRIPTION: The purpose of this Design Change was to convert the existing Westinghouse DB-series circuit breakers into fused disconnect switches. The conversion and sizing of the fuses in the modified breakers were changed for DC breaker coordination purposes. In addition, fuse status indicating lights were installed (with annunciation provided) in the door of the DC Switchgear cubicles to indicate blown fuses.

SAFETY ANALYSIS: The work involved in this Design Change did not result in any personnel, equipment, or nuclear safety problems, nor did it result in any operational changes for the systems affected. The use of fuses for electrical coordination in DC electrical systems is superior to circuit breakers for protection against switchgear faults therefore, system reliability was increased. The work for this DC was performed in the plant's Electrical shop, where the breakers were modified and acceptance tested prior to reinstallation. The safety function of the DC system was enhanced and the ability of the DC batteries to power safety-related components during a Design Basis Event was not changed. Therefore, the affect on overall plant safety was not changed.

DC 90-218

TITLE: Residual Heat Removal (RHR) Pressure Maintenance Test Gauge

DESCRIPTION: The purpose of the this Design Change was to provide a positive means for pressure monitoring of Check Valves RHR-CV-24CV and RHR-CV-25CV in the RHR "A" Loop and Check Valves RHR-CV-18CV and RHR-CV-19CV in the RHR "B" Loop as required for ASME Section XI In-Service Testing. This was accomplished by installing tubing and a pressure gauge to allow for testing of the valves.

SAFETY ANALYSIS: The affected portion of the system where the modifications were involved was the non-essential portions of the system. This DC installed tubing and a new pressure gauge that is isolated by a normally closed root valve. These components are in service only during periods of Inservice Testing and are isolated during normal plant operation. The installation of the pressure gauge provides for positive monitoring of the check valve performance during the required testing. The overall reliability and the safety design function of the check valves has not been altered. The implementation of this DC did not affect the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor did it create an accident or malfunction of a different type as defined by the USAR and Technical Specifications.

DC 90-302

TITLE: ASCO Transfer Switch Modification

DESCRIPTION: This design Change was required to close out Temporary Design Change (TDC) 90-300 which allowed Motor Control Center (MCC) MCC T to be the normal feeder for MCC X with MCC L as the emergency feeder thereby, providing a more appropriately distributed load on the diesel generators. In addition, this DC also removed the unused automatic transfer logic on some ASCO model 935-307 transfer switches. The removal of the unused circuitry will allow for more reliable operation since the new design will contain fewer components with less maintenance.

SAFETY ANALYSIS: This DC reduced the ECCS loading on DG #1, and increased the ECCS loading on DG #2, which was not as heavily loaded during postulated accident scenarios. This DC ensured an adequate source of electrical power to operate the essential equipment by reducing the peak loading of DG #1. The additional loading on DG #2 was well within its rated output. The manual operation of the ASCO transfer switches did not change, only the unused components of the automatic transfer switches were removed. This DC did not affect the normal operational or maintenance characteristics of the ASCO transfer switches. Therefore, no margin of safety was increased, and no unreviewed safety question existed.

ESC 91-045

TITLE: Non-Critical AC Fuse Replacement

DESCRIPTION: The purpose of this Equipment Specification Change was to replace the existing fuses which did not conform with the Non-Critical AC Bus Coordination Study, Nuclear Engineering Department Calculation (NEDC) NEDC 86-105F. These fuses were changed out to match the coordination study and provide standardization on fuse brand and types used at CNS.

SAFETY ANALYSIS: The new fuses that were installed have the same physical dimensions, are capable of handling the full load and inrush currents of the circuits, will provide short-circuit protection, and will provide the proper fuse coordination. 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 the equipment or cable will affect additional equipment. Therefore, since the design and operation of the systems involved were not changed by this ESC, no unreviewed safety questions were created.

REPORTABLE ACTIVITIES

Setpoint Change Request 89-26

TITLE: ARI/RPT ATWS High Reactor Pressure Trip Setpoint Change

DESCRIPTION: The purpose of this setpoint change to the high pressure Recirculation Pump Trip (RPT) and Alternate Rod Insertion (ARI) was to ensure that ARI/RPT will occur at Cooper Nuclear Station to provide Anticipated Transient Without Scram (ATWS) protection when initiated at lower power levels, but will minimize the potential for inadvertent occurrence. This setpoint change (89-26) raised the setpoint of Pressure Switches NBI-PS-102A, B, C and D from 1060 psig to 1074 psig. The selection of the reactor high pressure setpoint (1074) for the ARI/RPT logic of the Cooper Nuclear Station is described in General Electric Company Document EAS-16-0389, "Evaluation of ARI/RPT High Pressure Setpoint for Cooper Nuclear Station".

SAFETY ANALYSIS: This activity involved only a setpoint change and did not involve any modifications to the Reactor Recirculation System or its control system. Therefore, this activity did not change the safety function of the Reactor Recirculation System or its performance and reliability. The modification did not involve any changes to ARI/RPT system hardware.

The GE Analysis, EAS-16-0389, utilized normal input parameters (e.g., service water temperature = 75°F) rather than conservative bounding conditions. ATWS involves multiple failures, therefore, more realistic evaluations are acceptable than in the case of design basis accidents discussed in the USAR, which analyzes single failure with subsequent mitigating backup and redundant components. Accordingly, it was acceptable to NRC that the analysis input parameters be less conservative than USAR values. The new setpoint for high pressure ARI/RPT is below the Technical Specifications limit of 1120 psig for Cooper Nuclear Station RPT. This setpoint change did not therefore, present an unreviewed safety question and did not require a change to the CNS Technical Specifications.

Procedure Change Notice (PCN) 0.3 (Revision 11)

TITLE: Station Operations Review Committee

DESCRIPTION: The purpose of this Procedure Change Notice was to document the addition of the Senior Manager of Staff Support position to the CNS Station Operations Review Committee (SORC) membership. This change provided for additional expertise in the diverse areas associated with nuclear plant operation. In addition, this change increased the SORC Quorum requirements from five to six.

SAFETY ANALYSIS: This procedure change did not in any way degrade the safety of Cooper Nuclear Station with respect to personnel, equipment, or nuclear safety. The change was administrative in nature and involved no technical or operational aspects that directly affect station operation. This procedure change did not require abnormal operation of any plant systems or procedures, and did not introduce any plant equipment alteration. Therefore, the effect on overall plant safety was not changed.

<u>Procedure Change Notice (PCN) 2.2.3 (Revision 40)</u>

TITLE: Circulating Water System

DESCRIPTION: The purpose of this procedure change notice was to add administrative limits in the Circulating Water System procedure restricting the backwashing of both condensers while the mode switch is in RUN. During power operations, backwashing both condensers at the same time under certain conditions could result in a significant loss of condenser vacuum and a possible plant shutdown.

SAFETY ANALYSIS: The Circulating Water System is not relied upon to mitigate the affects of any analyzed transients or accidents, and system failure would not result in an unanalyzed accident. However, by adding administrative limits restricting certain backwashing activities, plant safety was enhanced by removing a potential plant transient, resulting from a loss of condenser vacuum. There was no change in system components or system operating characteristics, therefore, no unreviewed safety questions were created.

<u>Procedure Change Notice (PCN) 5.2.5 (Revision 20)</u>

TITLE: Loss of Normal AC Power - Use of Emergency AC Power

DESCRIPTION: This procedure change notice was a direct result from actions taken by NPPD in response to NRC Information Notice (IN) 86-70 "Potential Failure of All Emergency Diesel Generators". This IN discussed design deficiencies that could disable both diesel generators (DG) by placing unanalyzed loads on the DG powered buses.

The CNS review used GE Specification 22A1259, "Standby AC Power System" to determine the ECCS equipment operational requirements in addition to the electrical drawings to determine which loads automatically shed during transfer to the emergency transformer or diesel generators (DG's). This PCN accentuated the loading restrictions on the DG's, to prevent overloading during the initial two (2) hours when the maximum ECCS load is present, and to clarify the load shed details of the BUS 1F and 1G transfer from normal power to the emergency transformer. All essential loads were analyzed for adequate voltage and current when the non-essential MCC's are powered from the emergency transformer for both high and low line voltage and load conditions.

SAFETY ANALYSIS: This PCN ensured that the 4160V and 480V auxiliary power distribution system will, under all transient and accident conditions, distribute AC power required to safely shutdown the reactor, maintain the safe shutdown condition, and operate all auxiliaries necessary for station safety by placing loading restrictions on certain non-essential loads. This PCN also provided administrative guidance to ensure that the DG's are not overloaded during the initial two (2) hours in an accident scenario when the maximum ECCS load occurs, and clarifies the load shed details of the BUS 1F and 1G transfer from normal power to the emergency transformer. In addition, this PCN ensures that an adequate source of electrical power to operate the essential equipment is available by placing loading restrictions on certain non-essential loads which increases the margin of safety for the diesel generator's loading capability. Therefore, the diesel generator's reliability, and capability to perform its intended safety function was not jeopardized.

Other Activities

TITLE: Evaluation of Dropped Control Blade in Spent Fuel Storage Pool

DESCRIPTION: This activity involved the movement of Control Rod Blades into and out of the Spent Fuel Storage Pool during a refueling outage along restricted paths in the vicinity of irradiated fuel without the benefit of Secondary Containment. An analysis was performed with respect to resulting fuel damage in the event a Control Rod Blade was dropped near irradiated fuel and subsequently impacted the fuel to determine if Secondary Containment was required for this activity.

SAFETY ANALYSIS: The analysis (GE proposal 295-1CB6L-HP1-91) showed that the energy from a falling control blade was insufficient to fail any fuel rods in the struck bundle when the blades were restricted to certain areas. Consequently, there would be no release of radioactivity from fuel bundles hit by a control blade that dropped vertically to the fuel rack and then fell over onto the stored fuel. Based on the results of this evaluation (and controlled movement of the refueling machine), it was concluded that if the arrangement of irradiated fuel bundles in the spent fuel storage pool are positioned so that a control blade cannot be directly dropped on the fuel during the blade movement through the pool, then this specific refueling operation would not require Secondary Containment integrity to be maintained during this activity.

III. PERSON.   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	7	0	0	0.337	0.000	0.000
Operating Personnel	53	0	0	18.272	0.000	0.000
Health Physics Personnel	25	0	37	8.284	0.000	11.753
Supervisory Personnel	4	0	0	0.913	0.000	0.000
Engineering Personnel	27	1	11	6.238	0.060	1.893
ROUTINE MAINTENANCE						
Maintenance Personnel	90	0	335	66.264	0.000	184.946
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	31	0	37	21.135	0.000	11.693
Supervisory Personnel	0	1	1	0.000	0.340	0.203
Engineering Personnel	8	33	33	0.216	18.177	8.065
SPECIAL MAINTENANCE						
Maintenance Personnel	1	0	20	0.018	0.000	13.463
Operating Personnel	6	0	0	0.039	0.000	0.000
Health Physics Personnel	2	0	8	0.184	0.000	1.592
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	3	0	1	0.124	0.000	0.170
WASTE PROCESSING						
Maintenance Personnel	1	0	0	0.004	0.000	0.000
Operating Personnel	4	0	0	1.625	0.000	0.000
Health Physics Personnel	5	0	1	1.103	0.000	0.017
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	0	0	0	0.000	0.000	0.000
REFUELING						
Maintenance Personnel	0	0	0	0.000	0.000	0.000
Operating Personnel	31	0	0	1.010	0.000	0.000
Health Physics Personnel	2	0	2	0.233	0.000	0.237
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	2	0	0	0.089	0.000	0.000
INSERVICE INSPECTION						
Maintenance Personnel	1	0	21	0.002	0.000	11.755
Operating Personnel	0	0	0	0.000	0.000	0.000
Health Physics Personnel	0	0	2	0.000	0.000	0.095
Supervisory Personnel	0	0	0	0.000	0.000	0.000
Engineering Personnel	1	0	0	0.001	0.000	0.000
TOTAL						
Maintenance Personnel	90	0	375	66.625	0.000	210.164
Operating Personnel	55	0	0	20.946	0.000	0.000
Health Physics Personnel	33	0	39	30.939	0.000	25.387
Supervisory Personnel	4	1	1	0.913	0.340	0.203
Engineering Personnel	27	33	40	6.668	18.237	10.128
GRAND TOTALS	209	34	455	126.091	18.577	245.882