WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

Docket No: 50-482
Facility Operating License No: NPF-42

ANNUAL SAFETY EVALUATION REPORT

Report No: 8

Reporting Period: January 1, 1992 through December 31, 1992

Prepared by: Kevin J. Moles

SUMMARY

This report provides a brief description of changes, test, and experiments performed at Wolf Creek Generating Station pursuant to 10 CFR 50.59(a)(1). This report includes summaries of the associated safety evaluations that were reviewed and found to be acceptable by the Plant Safety Review Committee for the period beginning January 1, 1992 and ending December 31, 1992. This report is submitted in accordance with the requirements of 10 CFR 50.59(b)(2).

On the basis of these evaluations of changes the following has been determined:

- There is no increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR)
- There is no possibility that as accident or malfunction of equipment important to safety of a different type than any evaluated previously in the USAR may be created.
- The margin of safety as defined in the basis for any Technical Specification is not reduced.

Therefore, all items reported herein were determined not to involve an unreviewed safety question.

A change to the safety evaluation tracking process occurred approximately mid 1991. This change involved sequential numbering of all safety evaluations upon initiation. Many evaluations reviewed and approved this year were generated prior to this numbering program and were not assigned a sequential number. Therefore this report is divided into two sections. Section I contains safety evaluations assigned a sequential number under the current process. Section II contains miscellaneous safety evaluations initiated prior to the numbering program.

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Section I

Safety Evaluation: 91-0014 Revision: 0

Generator Hydrogen Cooler Service Water Temperature Control Valve (EATV0007) Replacement

The function of the Generator Hydrogen Cooler Service Water Temperature Control Valve, EATV0007, is to control hydrogen temperatures in the Turbine Generator Hydrogen Coolers. This function is accomplished by controlling service water flow through the Hydrogen Coolers. Control valve EATV0007 functions adequately when lake temperature is high (95°) and the maximum/minimum service water flow rates are 3160 and 1042 gpm respectively. However, when lake water temperature is low (34°F), the valve is unable to control flow at the lower end of the valve's performance range. The valve serves no safety function and is located in a non-safety related system. Failure of the valve due to a Design Basis Accident (DBA) or design occurrence does not require Engineering to perform a single failure analysis, since the system in which the valve is located is not required to be designed for single failure criterion.

Replacement of control valve EATV0007 with a ball type control valve does not affect any of the accidents previously evaluated in the Updated Safety Analysis Report (USAR). However, the effect that the design change would have on USAR evaluated accidents, if applicable, would be a decrease in occurrence and consequences of an accident because the ball type control valve is better suited for system service. The ball type control valve function is the same as that for existing control valve EATV0007; however, it does have better rangeability and flow control characteristics. The design change does not affect any accidents previously evaluated in the USAR. Control valve EATV0007 or its ball type control valve replacement are non-safety related components and they are not equipment important to safety as evaluated in the USAR. The consequences of control valve EATV0007 malfunction is the shutdown of the turbine generator until the malfunction, is within previous safety evaluations. The design change requires that piping modifications be made for installation of the replacement valve; however, this design will be non-seismic. The new weight of the ball type control valve is 715 lbs. which is 430 lbs. more than the old valve weight. However, the pipe being replaced is 14" diameter compared to the 8" diameter new pipe size. The higher weight of valve is approximately compensated by lower pipe weight. As a result, piping stress and support loads are not adversely impacted.

Safety Evaluation: 91-0021 Revision: 0

Heat Trace as Built on Drain Line

Heat tracing and insulation were found not installed on a drain line in the Radwaste System as shown on drawings. This evaluation allows for use-as-is.

The proposed design change, to use-as-is and revise the documentation, only affects a non-safety system. This system has no impact on the function of safety-related equipment or accident initiation or mitigation. The system has functioned in the as built condition and has no effect on accidents evaluated in the Updated Safety Analysis Report (USAR).

Safety Evaluation: 91-0082 Revision: 0

Detergent Drain Tank Level Switch Replacement

The detergent wastes are collected in the Detergent Drain Tank until 11 1/2 inches of effluent are collected in the tank. At this time the normally open contact of the high level float switch closes and one pump is actuated. If the water level continues to rise to 16 1/2 inches, the normally open contact of the high high level float switch closes and the second pump is actuated. Both pumps continue to pump until the level drops to 5 1/2 inches and the low level switch contact opens. However, the soapy water has caused setpoint variations resulting in damage to the pumps.

This modification replaces the three existing float switches with a radio frequency (RF) capacitance three point level switch.

The new switch is an entirely electronic device. There are no moving parts to jam, clog or wearout resulting in minimum maintenance. The probe's pressure and temperature capability exceed the design on the Detergent Drain Tank. The replacement of the three level float switches with a tripoint RF capacitance level switch will enhance the function of the system and eliminate the problems that occurred previously due to failure of the float switches. This modification affects non-safety related equipment. Implementation of this modification has no affect on any safety-related equipment or systems.

Safety Evaluation: 91-0178 Revision: 0

Gravel Around Essential Service Water (ESW) Valve Pit

This modification removes 12 inches of existing soil and replaces it with the same depth of crushed rock (AB-3) surfacing in the yard area outside the powerblock (west of the Control Diesel Generator Buildings around the ESW valve pits). The removal of cohesive material and replacement with crushed rock (AB-3) is a change to the backfill described in the Updated Safety Analysis Report (USAR). The area will be excavated so that the finished elevation will be approximately the same as that of the surrounding grade. It will be sloped so that the run off is toward the road. Bumper posts will be placed around the ESW valve pits so that no excessive loads will be placed on the structural walls of the ESW valve pits.

This change in surface texture will not introduce any significant change to the storm drainage analysis in the USAR. The change in loads on safety-related structures caused by the different surface has been evaluated. Installation of the crushed rock (AB-3) will not affect operation of any equipment that is important to safety, required for plant shutdown or for required accident mitigation.

Safety Evaluation: 91-0192 Revision: 0

Fuel Repair Procedures for Wolf Creek Generating Station

This Westinghouse procedure provides a detailed guide to conduct fuel assembly inspection and repair. This procedure provides instructions for reconstitution of fuel assemblies, reassembly of fuel assemblies using a different skeleton and television visual inspection of removed fuel rods or complete assemblies. All reconstitution activities are performed in the cask loading pit using Westinghouse and plant equipment. The Multipurpose Fuel Repair System (MFRS) station is used to hold the spent fuel assembly and/or skeleton during reconstitution or skeleton replacement. The MFRS elevator is used to raise, lower, and invert the spent fuel assembly as required. The New Fuel Elevator, HKE10, will hold the Fuel Rod Storage Rack (FRSR) during reconstitution and is used to hold the removed failed rods and the replacement stainless steel dummy rods. The bottom nozzle platform is used to hold the removed bottom nozzle for inspection and preparation for reuse. The Fuel Rod Handling Tool (FRHT) is used to remove and insert rods into a fuel assembly or the FRSR.

Administrative controls provided in plant procedures and this vendor procedure will ensure that no criticality, personnel radiation exposure, or suspended load concerns are outside of previous analysis when spent fuel assemblies or fuel rod racks are moved into the cask loading pit for fuel reconstitution. Plant procedures also provide administrative controls on fuel movement and cask location during reconstitution activities to preclude fuel damage. The MFRS elevator and FRHT are designed with stops which are set by this procedure to prevent spent fuel from being raised too high. Failure of the new fuel elevator and the MFRS elevator during a seismic event were previously evaluated. The evaluation concluded that these failure will not result in a release of radioactivity in excess of that assumed in the design basis fuel handling accident.

Safety Evaluation: 91-0196 Revision: 0

Minimum Flow Orifice Plate

This modification installs a restricting orifice with an isolation valve in the Shop Building, in the downstream discharge piping of the raw water pumps at the Makeup Discharge Structure. This modification allows the continuous discharge of a minimum amount of water to prevent dead heading of the Filtered Water Transfer Pump. The modification establishes the original design intent.

There are no design basis accidents evaluated in the Updated Safety Analysis Report (USAR) that involve the non-safety related Makeup Demineralizer System. This change does not change the system function or failure modes.

Safety Evaluation: 91-0206 Revision: 1

Minimum Raceway Separation in Electrical Penetration Areas

Physical separation criteria is established for the independence of circuits and equipment comprising or associated with class IE systems. This physical separation is provided to maintain the physical and electrical independence of a sufficient number of circuits and equipment so that the protective functions required during and following any design basis event can be accomplished.

Updated Safety Analysis Report (USAR), Section 8.3.1.4, contains the minimum separation criteria based on IEEE-384 (1974) and Reg. Guide 1.75 which Wolf Creek is committed to. IEEE-384 (1974) allows a departure from established minimum separation distances by analysis. This analysis shall be based on tests performed to determine the flame retardant characteristics of the proposed cable installation. Wyle Laboratories conducted tests for Limerick Units 1 and 2 to justify separation which is less than the standard distance. They used similar cables of the same size (and larger) as the WCNOC cables in question. Their tests are based on the following failure mode assumptions:

1. The cable or equipment in the circuit develops a fault that is not cleared due to the failure of the primary overcurrent protective device.
2. The fault current level (660 amps) is just below the long-term trip setpoint of the next higher level overcurrent device.
3. The impedance of the fault adjusts itself automatically to maintain the fault current magnitude at a constant level as the resistance of the wire increases due to heating.
4. There are no other loads on the same circuit which would cause the next higher level overcurrent device to trip.
5. The overload wire can maintain the continuous overheated condition without an operator being aware of the condition.

Philadelphia Electric Company's Design Verification Test Report #48503 showed the heating effects due to wiring faults which caused sustained overcurrent conditions with the above assumptions had the greatest impact on adjacent wires. The results of their tests revealed the following:

1. Cable sized #4/0 AWG and smaller when energized with 660 amps and routed in an open cable tray, did not ignite. Cables were tested in both horizontal and vertical tray configurations and did not ignite in any case. Configurations with a 1 inch vertical separation between cable trays and zero separation between cable tray and enclosed raceway were tested successfully without damage.
2. No separation was required between an enclosed raceway and another enclosed raceway or cable tray when the enclosed raceway contains cables which are #4/0 AWG and smaller.
3. One inch separation between an enclosed raceway and another enclosed raceway or cable tray is required when the enclosed raceway contains cables larger than #4/0 AWG.

The Electrical Raceway Separation Verification Test Reports for Limerick Units 1 and 2 is applicable to WCNOC for the following reasons:

1. WCNOC uses similar cables made by the same manufactures as the cables used in the test. Cables used by WCNOC made by different manufactures than those used in the test use the same type of insulation material (cross-linked polyethylene or cross-linked ethylene propylene rubber) and jacket material (chlorosulfonated polyethylene or neoprene). In addition, all class IE cables are qualified to the same standards such as IEEE-323 and IEEE-383 and are manufactured to the same IPCEA standards.
2. The configurations described in the Work Requests are similar to the configurations used in the Electrical Raceway Separation Verification Tests.
3. The fault current used in the test is very conservative compared to the maximum credible fault current that could develop for the configuration of circuits described in the above Work Requests except for Work Requests 05867-91 and 05868-91.

Exposed Non-Class IE cables leaving tray 6A2A05 and 6A2B03 and Class IF conduit 4U2A2E can be used as is for the following reasons:

Class IE conduit 4U2A2E contain circuits which terminate at the Accumulator Tank A Vent Valve EPHV8950A. EPHV8950A is a solenoid-operated valve that is normally closed and fails closed. Solenoid-operated vent valves are provided to depressurize the accumulator tanks during emergency cold shutdown conditions. Accumulator tank A has only one vent valve (EPHV8950A), since redundancy in capability to prevent accumulator discharge to the RCS is provided by a motor-operated gate valve in the tank discharge line. This solenoid valve is connected to a different emergency bus than its associated motor-operated gate valve. During normal operation, in order for the vent valve to open, a hot short would have to occur in the cabling to EPHV8950A. If this unlikely event were to occur simultaneously with a design basis accident, adequate core cooling is provided by the other three accumulators.

Safety Evaluation: 91-0207 Revision: 0

Spent Fuel Pool Gate Seal Inflation Piping

This modification revises the air supply connections and provides local seal pressure monitoring with individual pressure gauges on each independent seal. It also provides quick connect fittings for the spent fuel pool gate seal to allow for the local pressure monitoring and a method to connect an alternate air supply. This modification establishes the original intent of the seal design to be completely independent from each other. The modification will not affect the ability to perform the intended design basis functions.

However, there are no design basis accidents associated with, or impacted by, any failure modes of the spent fuel pool gate seals. The design change failure modes do not change the system function or affect the system failure modes.

Safety Evaluation: 91-0215 Revision: 0

Implementation of the Configuration Change Package Program

This procedure revision incorporates the Configuration Change Package (CCP) process. The intent of the CCP is to implement definitions and concepts established in EPRI Document NP-6406 (NCIG-11), "Guidelines for the Technical Evaluation of Replacement Items in Nuclear Power Plants," and NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluations," to streamline the Design Change process.

NCIG-11 establishes that a design change does not occur if the equivalency of form, fit, function and interchangeabiltiy of a replacement item has been determined. The basis for this premise is that the plant, system or component design has not been altered.

NSAC-125, states that equivalent part changes are considered maintenance activities and thus, not considered a change in design. In addition, based on these guidance documents, the procedure defines Alternate Replacement Critical Characteristics for Design, Design Function, Design Change and Configuration Change.

Based on these concepts and definitions, the CCP process was created to allow configuration changes to be made while still maintaining design control.

Safety Evaluation: 91-0246 Revision: 0

Revision to Updated Safety Analysis Report (USAR) Figure 1.1-1, "Symbols and Legend for System Flow and Piping and Instrumentation Diagrams,"

This USAR change revises USAR Figure 1.1-1, "Symbols and Legend for System Flow and Piping and Instrumentation Diagrams" to reflect changes made by Plant Modification Requests to associated flow diagrams.

USAR Figure 1.1-1 provides the system designators used in associated flow diagrams and the number of sheets found in each. This is an editorial drawing change only and has no effect on plant equipment or operations.

Safety Evaluation: 91-0248 Revision: 0

Revised Licensed Operator Requalification Training Program Cycle

This Updated Safety Analysis Report (USAR) change alters the Wolf Creek Nuclear Operating Corporation Licensed Operator Requalification Training Program Cycle to better align the program with the National Exam Schedule. In order to change the cycle, a two month extension was required to the normal twenty four month program. The new program cycle begins on December 1 of even years. This change required an exemption from the requalification requirements of 10 CFR 55.59(a). This exemption was requested of and granted by the NRC. The requalification program extension or change allows operator examinations to be given in accordance with the National Exam Schedule promulgated in Generic Letter 89-03.

Safety Evaluation: 92-0001 Revision: 0

Containment Spray Additive Tank Relief Valve Setpoint Discrepancy

This modification revises the Updated Safety Analysis Report (USAR) to correct the high pressure relief valve setpoint for Containment Spray Additive Tank (CSAT) Relief Valve ENV0057 to reflect the design setpoint.

Changing the setpoint for relief valve ENV0057 is consistent with the design of the CSAT it protects. This change does not affect the function of or availability of the CSAT. The setpoint was changed from 5 psig to 10 psig. The higher relief valve setpoint does not exceed the design pressure or construction standards of the CSAT which it protects.

Revising the USAR for the relief valve setpoint to the CSAT meets the assumptions in the accident analysis.

Safety Evaluation: 92-0002 Revision: 0

Removal of Power for Safety Injection Pump A Upstream Isolation Valve

This modification removes power from the Residual Heat Removal Heat Exchanger A/Chemical Volume Control System to Safety Injection Pump A Upstream Isolation Valve EMHV8924 during Modes 1, Power Operation through Mode 3, Hot Standby. This modification will prevent spurious movement or movement due to operator error.

Valve EMHV8924 is a motor operated gate isolation valve located in the crossover line from the Centrifugal Charging Pump Suction to the Safety Injection Pump Suction. There are a total of three valves in the line leading from the suction side of the Safety Injection Pumps. There are two motor operated gate valves in parallel (EMHV8807A/B) which are normally closed. The third valve (EMHV8924), is in series with the first two and is normally open. The two valves in parallel meet the single active failure criteria since a failure of one valve to open will not prevent the opening of the crossover. Valve EMHV8924 was originally required for isolation of the crossover in the case of a passive failure of either of the other two valves. This is apparently the reason for listing EMHV8924 in the Updated Safety Analysis Report (USAR) Table 3.9(N)-11, "Active Valve List." The USAR Section 3.1.1.4 defines the various passive component failures which could be postulated for valves such as EMHV8807A/B. The only probable failure mode would be leakage due to a single valve stem packing leak.

It is believed that such a stem packing leak could be adequately isolated by closing the affected valve, EMHV8807A or B, which are flexible wedge type gate valves. Therefore, valve EMHV8924 is not required to function for this passive failure scenario. During development of the operating scenarios for EMHV8924 as part of the implementation of Generic Letter 89-10 evaluation, it was concluded that isolation of the Emergency Core Cooling System (ECCS) could be accomplished by closing EMHV8807A/B, EJHV8804A and EMHV8923A/B and these valves have been designed or are being appropriately modified for this purpose. This means that EMHV8924 would never have to operate under a differential pressure greater than zero.

Therefore, it has been concluded that EMHV8924 does not need to change position for any safety function and can thus be considered a passive valve. USAR Table 3.9(N)-11, (Sheet 7) and Figure 6.3-1 (Sheet 2) are affected. Even though EMHV8924 can be considered passive, there is a concern that the valve could misposition itself due to spurious movement referred to in Section 6.3.2.2 of the USAR or could activate due to operator error.

By physically removing power to EMHV8924, the likelihood of a mispositioning accident or malfunction are eliminated. The consequences of a misposition malfunction are also eliminated.

Safety Evaluation: 92-0003 Revision: 0

Blowdown and Sample Process Isolation Signal Temporary Disconnection

During normal operation, Condenser Air Removal System Radiation Detector GERE0092 has experienced some spiking activities. When spiking occurs, a Blowdown and Sample Process Isolation Signal is generated which caused the closure of effected Steam Generator Blowdown System valves. This temporary modification disconnects the isolation signal for an approximate two week period while the monitor's spiking activities are monitored. Radiation Detector GERE0092 remains operational and maintains its Control Room alarm function. While the signal is isolated, the system is monitored by other detectors that will initiate an isolation signal should an increase in secondary system contamination levels occur.

Radiation Detector GERE0092 does not serve a safety-related function. It only monitors the exhaust of the Condenser Air Removal System and is completely isolated during a design bases accident.

Safety Evaluation: 92-0004 Revision: 0

Pressure Test Procedure Change

This change to the Chemical and Volume Control System Charging Pump Common Discharge and Reactor Coolant System Charging Lines Pressure Testing Procedure temporarily installs a pressure gauge at Vent Valve BGV0446. It additionally provides the option of connecting a hydro pump to this vent valve. The intent of the change is to facilitate pressure testing of the alternate charging flowpath without incurring a transient during switchover from normal to alternate charging while at normal operating pressure and temperature with the plant in Mode 3, Hot Standby.

The temporary equipment is installed using flexible high pressure tubing and/or hose. Vent Valve BGV0446 remains isolated except when in use. Seismicity of the piping is not affected and the hydro pump is equipped with two relief valves to prevent overpressurization.

The normal charging flow path is in use during performance of this test and in the extremely unlikely event that line-up of the alternate charging flowpath is required, the pressure rating of the temporary equipment is equal to the system pressure rating. The ability of the charging system to perform its required design function is not degraded.

Safety Evaluation: 92-0005 Revision: 0

Temporary Removal of Positive Displacement Pump Relief Valve

This temporary modification facilitates repairs to the 1.75 inch by 2 inch relief valve off the Positive Displacement Pump (PDP) discharge line. The relief valve's discharge is hardpiped by a 2 inch line to a 3 inch common header which is connected to the Volume Control Tank (VCT) and is used by other relief valves. The 2 inch line is normally stagnant and contains flow only when the PDP relief valve lifts; no other relief valves use this line. To support maintenance work, this line has freeze seals applied per an approved plant procedure. A blind flange is installed on the line after the relief valve is removed to provide isolation when the freeze seals are not present. The freeze seals and temporary blind flange provide an isolation boundary in the line which connects to an operable portion of the Chemical & Volume Control System (CVCS). The PDP's suction and discharge piping are isolated by valving, controlled under the Clearance Order Procedure. After repairs are made, freeze seals are applied to remove the blind flange and reinstall the relief valve.

If the freeze seals fail, the consequences of this failure would not compromise the safety function of the CVCS nor increase the consequences of any previously evaluated CVCS equipment failure. Failure of the subject relief valve in the open position during normal plant operations with the PDP operating has been previously evaluated. The failure effects of this event have no effect on normal plant operation or the ability to bring the plant down to a safe shutdown condition. The event is resolved by removing the PDP from service and placing a Centrifugal Charging Pump in service. All radioactive fluids are contained.

Safety Evaluation: 92-0006 Revision: 0

Load Cell Settings of Refueling Machine Clarification

A Updated Safety Analysis Report (USAR) change was made to clarify setpoint values for the Refueling Machine load cell settings. The previous USAR wording specified a single value but did not specify a tolerance. This was not representative of the actual work practices and was misleading. The change addressed both the raising overload and master overload interlock values as well as the lowering underload interlock value. The tolerance that was clarified by this change was to state the setpoints were less than or equal to the single value previously given. This now gives a precise description of the actual work practices.

The USAR clarification to describe the tolerance used for the Refueling Machine load cell setpoints does not change the values used previously. Therefore, the clarification does not change the way in which the Refueling Machine is used in actual practice.

Safety Evaluation: 92-0007 Revision: 0

Updated Safety Analysis Report (USAR) Chapter 13 Organization Change

This is a change to Chapter 13 of the USAR which reflects change to the Wolf Creek Nuclear Operating Corporation organization. This change was initiated by a change in personnel.

Clear reporting lines and program responsibilities by qualified personnel have been maintained. This change does not cause any change in the overall operating philosophy or capabilities of Wolf Creek Generating Station.

This change does not make any changes to systems, components or methods of operation required to mitigate the consequences of an accident previously evaluated. This change does not effect equipment important to safety. This change does not affect systems, components or procedures required to operate the plant.

Safety Evaluation: 92-0008 Revision: 0

Replacement of Chlorinator in the Demineralized Water Makeup System

This modification replaces the existing chlorinator used to inject chlorine into the raw water in the Demineralized Water Makeup System. The existing chlorinator will not automatically control the chlorine addition to the raw water in the correct proportion to the flow. This modification installs a chlorinator designed to operate at approximately the mid point of the control valve's operating range to meet the chlorination requirements of the present raw water flow.

The Demineralized Water Makeup System chlorinates raw water by equipment capable of automatically providing chlorine to maintain the chlorine residual within the requirements of state regulations. The chlorinator dissolves chlorine gas into water and injects the chlorine solution into the inlet piping of the Chlorine Retention Tank. The chlorinated water then flows through the precipitators, sand and carbon filters to the Filtered Water Storage Tanks to be used to produce demineralized makeup water and potable water.

This modification establishes the original design intent to automatically inject chlorine into the raw water flow to maintain the proper chlorine residual in the raw water flow from the Chlorine Retention Tank. This modification replaces non-safety related equipment and does not alter the system functions or failure modes. Replacement of the chlorinator lessens the consequences of a line break that would discharge chlorine gas to the atmosphere.

Safety Evaluation: 92-0010 Revision: 0

Nuclear Instrumentation System Setpoint Changes and Time Delay Additions

This modification changes the setpoint for the Nuclear Instrumentation (NIS) Power Range Channel Deviation alarm, and adds time delays for the Quadrant Power Tilt Ratio alarm. These changes eliminate nuisance alarms which occur during full power operation as the result of a core flow anomaly.

The Power Range Channel Deviation alarm alerts the operators when the signal from one NIS channel differs from the average of the four channels by more than two percent. Westinghouse has provided justification as to why the short term fluctuations caused by this core flow anomaly do not represent an unreviewed safety question (WCAP-11528) and the Precautions, Limitations, and Setpoints document allows this alarm to be adjusted so that it is just outside the range of normal operating variations.

The Quadrant Power Tilt Ratio alarm is also subject to spurious actuations due to the core flow anomaly. This alarm is described in the Updated Safety Analysis Report (USAR) and the setpoint is specified in technical specifications. This change adds a time delay circuit which requires that the alarm signal be present for two seconds before an alarm is actuated and the information recorded by the plant computer. This delay prevents spurious actuations caused by the flow anomaly and still allows annunciation at the flux tilt conditions described in the USAR. No limits in the technical specification are changed.

Safety Evaluation: 92-0012 Revision: 0

Auxiliary Feedwater Pressure Test Using Motor Driven Auxiliary Feedwater Pumps (AFWPs)

Test procedure STS PE-053A, "Auxiliary Feedwater Pressure Test Using Motor Driven AFWPs," was revised to specify pressurizing the motor-driven auxiliary feedwater pump discharge piping through a drain valve connection with a temporarily installed hydro pump. This will allow testing during Mode 1 operation. Pressurization of the motor-driven auxiliary feedwater pump discharge piping will be performed on one train at a time, since the piping configuration during the test will render the affected train inoperable and Technical Specification LCO 3.7.1.2 will be entered.

The hydro pump assembly, hoses, fittings and test flange used at the drain line connections are all rated at 1500 psig minimum and will maintain the test pressure. The hoses will not rigidly couple the hydro pump load to the Auxiliary Feedwater System. The drain line valves will only be opened to pressurize the system and will then be restored to their normal position. To avoid discharging water into the Auxiliary Feedwater System, the applied test pressure will be maintained at least 50 psig below the pressure downstream of the boundary check valves. The test procedure also requires the static pressure downstream of the turbine-driven auxiliary feedwater pump be monitored periodically to check for possible backleakage through check valves in the turbine-driven pump discharge lines which tie into the test volume piping.

The revised test method does not alter the testing parameters previously used. Therefore, there is no change to the Auxiliary Feedwater System operation that may cause different failure modes than previously evaluated. The reliability, capability, and integrity of the Auxiliary Feedwater System in removing thermal energy from the Reactor Coolant System has not been degraded by this procedure change.

Safety Evaluation: 92-0016 Revision: 0

Essential Service Water Vault Construction

This modification installed concrete access vaults around sections of Essential Service Water (ESW) piping. These vaults are to provide access for lining the ESW piping.

A review of evaluated accidents discussed in the Updated Safety Analysis Report (USAR) indicated that construction of these access vaults had no effect on the initiators of any accidents. Administrative controls were established to protect the ESW piping during construction so that ESW system operation was not impaired. All pumps and equipment were protected by seismically designed shoring and the ESW piping was protected by administrative controls which prevented the potential for simultaneous failure of both ESW trains.

Assessment of the construction of the access vaults showed that the only safety-related equipment in the area was the ESW components and piping. Any malfunction of this equipment was already bound by previous evaluations.

Administrative controls were established to ensure that ESW operations was not degraded with respect to the bases for Technical Specification 3.7.4.

Reference also discussion associated to Safety Evaluation 92-0027.

Safety Evaluation: 92-0017 Revision: 1

Addition of MCC Distribution Panel Loads to Design Drawing

A revision to the Diesel Generator Loading List (Design Drawing E-11005 and Updated Safety Analysis Report (USAR) Figure 8.3-2) was made to incorporate as-built information regarding the Motor Control Center (MCC) distribution panels which were previously omitted. This drawing change adds MCC distribution panel loads of 61kW to the Diesel Generator Loading List. Other loads were evaluated, but were not added, as it is the intent of the loading list to show major loads supplied by the Class 1E dc system. Loads not added include the Essential Service Water (ESW) traveling screen, hydrogen analyzer and voltage regulating transformer which are not expected to operate continuously during Loss of Coolant Accident (LOCA) and/or station blackout. The newly identified MCC distribution panel load and the existing motor operated valve loads are conservative loads. These loads contain enough margin to address both the ESW traveling screen (2kW) and hydrogen analyzer (1kW) loads. The voltage regulating transformer is a backup source which will not be used unless the battery and battery charger fail and/or equipment failure occurs somewhere within the normal source path.

This change results in a worst case total diesel generator load of 5818kW during station blackout. The diesel generator has a continuous rating of 6201kW. Therefore, the ability of the diesel generator to perform its safety-related design function is not affected by the drawing changes.

Safety Evaluation: 92-0018 Revision: 0

Correction to Information Associated with the Emergency Diesel Engine Starting System (EDESS)

This Updated Safety Analysis Report (USAR) change deletes incorrect information associated to startup tests associated to the EDESS. The "no start cranking test" was performed on site rather than at the manufacturer's facilities. This information is discussed in Chapter 14 of the USAR.

This change is not associated with any design basis accidents. The location of the test does not affect the function of the Emergency Diesel Generator. The test was successfully performed on site.

Safety Evaluation: 92-0019 Revision: 0

Change to Updated Safety Analysis Report (USAR) to Incorporate Temporary Equipment as Permanent Equipment

This modification revised the required USAR drawings to show the as-built configuration of a temporary modification that has been made permanent. The temporary modification installed an additional breathing air compressor including all associated air, electrical, and water connections.

The drawing revisions associated with this change are being made to reflect the as-built condition resulting from the incorporation of a temporary modification that is being made permanent. No physical plant changes are being made as a result of these drawing revisions.

Safety Evaluation: 92-0020 Revision: 0

Pressurizer Spray Valve Temporary Shielding

Temporary lead shielding was required to radiologically protect personnel near the pressurizer spray valves. An evaluation was performed to determine the acceptability of additional loading of the associated piping by the shielding.

The installation of this temporary shielding has been evaluated and installation requirements have been established to ensure that no equipment important to safety would be compromised. The evaluation determined that the structural and seismic integrity of the affected piping was not degraded by the shielding when installed in accordance with the engineering disposition for this activity. The installation of temporary shielding does not introduce any new failure modes to this system.

Safety Evaluation: 92-0021 Revision: 0

Feedwater Flowrate Determination Test Temporary Procedure

This temporary procedure measures feedwater flow rate to each steam generator using a chemical tracer. Injection and sample skids will be connected to feedwater piping so that flow through the feedwater venturies can be measured and the condition of the venturies can be determined.

Performance of this test does not increase the probability of a loss of normal feedwater flow beyond the ANS Condition II frequency that it is currently evaluated at. The temporary connections are; flexible to maintain seismic integrity, pressure rated greater than the feedwater system, and connected upstream of the main feedwater isolation valves to prevent degradation of the Auxiliary Feedwater System.

The temporary test equipment did not interact with, or impair the operation of, any equipment important to safety. The only potential failure during the test is the loss of pressure boundary integrity which is bound by the existing loss of feedwater accident analysis. The only type of event that may be initiated by performance of this temporary procedure is a loss of normal feedwater flow. This event has already been analyzed. The temporary test equipment was installed upstream of the feedwater isolation valves, which ensures that auxiliary feedwater operation is not impaired.

Safety Evaluation: 92-0022 Revision: 0

Sample Recovery System Pump - Drainline

The Process Sampling System Sample Recovery Pump (PRM02) sends the sample water (after testing) to the Steam Generator Blow Down Demineralizer. The pump's placement in the sample panel provides little clearance for maintenance to drain the lubricating oil. The drain plug for the pump is located over the pump base without adequate room to place a drain pan between the drain and pump base. This change installs a drain line, with a valve, to permit easier maintenance of the pump. This change is a non-safety related change to a non-safety related component.

The safety design basis for the Process Sampling System, in Section 9.3.2 of the Updated Safety Analysis Report (USAR), concerns penetrations of containment. This change does not penetrate containment, nor does it affect any penetrations of containment.

This change will not affect the function, setpoints, or reliability of the Process Sampling System. The change does not alter any seismic or environmental qualification of any system or component.

Safety Evaluation: 92-0024 Revision: 0

Removal of Circulating Water Inlet Sample Line to Panel RM172

This change removes from service the circulating water sample line for the Process Sampling System sample panel RM172. The Process Sampling System and the Circulating Water System are affected by this change. The change is located in the Turbine Building. This is a non-safety related change. The existing influent sample point is not used, as sampling is done at the effluent verses the influent. If the circulating water influent ever needs to be tested, the sample will be taken at the inlet screenhouse. Based on this, the sample line will be cut downstream of the piping isolation valve and the line capped to make it available for future sampling, if needed. Also, this removes from service the circulating water inlet sample line to sample panel RM172, including its associated instrumentation.

The safety-related design basis for the Process Sampling System, in Updated Safety Analysis Report (USAR) Section 9.3.2, concerns penetrations of containment. The sample line for the Circulating Water System does not penetrate containment. There are no safety design basis for the Circulating Water System.

This change will not affect the function or integrity of the Circulating Water System because closing the 3/4 inch sample line will not affect the 120 inch circulating water line. This change does not alter the seismic, environmental or equipment qualification of any system, component or structure.

Safety Evaluation: 92-0025 Revision: 0

Installation of FTS-2000 Emergency Notification System (ENS) Phones

The NRC supplied ENS was replaced with a new "FTS-2000" system. The new ENS communication system was installed in parallel with the NRC supplied Tellabs system for a minimum one month trial basis and this modification was to provide that change. In addition, telephone outlets were added in the Power Block Computer Room.

This change affects only non-safety related communications systems. The non-safety related telephone system is electrically separated from equipment important to safety. Therefore, no interaction with important to safety equipment can occur.

Safety Evaluation: 92-0026 Revision: 0

Replacement of Condensate Demineralizer Sight Window with a Steel Plate

This temporary modification replaced one of the sight windows on the "B" Condensate Demineralizer Tank with a steel plate until a new sight window glass was available. This temporary modification was made so that the water treatment plant could utilize this tank for resin storage and return the tank being used to normal service.

The condensate demineralizer tanks have no safety function and their failure would not affect any equipment important to safety.

The inability to observe the tanks' contents or its transfer will not affect the operation, indication, control or analysis of the system and normal operation will be maintained.

Safety Evaluation: 92-0027 Revision: 0

Essential Service Water System Access Vaults and Pipe Spools

This modification is for the purpose of facilitating a repair program for the underground portion of the Essential Service Water System (ESWS) piping. The modification consists of installing four new concrete access vaults, two each for train A and B, and installing bolted flanges, spool pieces, and drain and vent valves in the Ultimate Heat Sink (UHS) supply and return lines of both trains at the locations of the new vaults. The access provided by this modification will allow future pipe inspection/repair as needed.

The affects of cold weather, water intrusion, seismic events, fire, tornado missiles and aircraft hazards have been accounted for in the design and installation of the new concrete vaults and piping changes. Water level monitoring and water removing facilities have been provided on the outside of the vaults. Local temperature monitoring instrumentation has been provided on the outside of the vaults for monitoring the temperature inside the vaults without entering them.

The underground ESWS piping was originally designed as fully supported by the ground. The construction of the new vaults will render the pipes partially unsupported. The pipes have been evaluated for the unsupported length for both conditions (i.e., with and without the piping modifications) for all applicable design loads. The pipe stresses remain within the ASME Code allowables. No new supports are required for the pipes in the vaults. The modifications to the ESWS piping are also in accordance with ASME Code requirements.

This modification does not affect the ability of any safety-related system, component or structure to perform its safety-related function.

Reference also discussion associated to Safety Evaluation 92-0027.

Safety Evaluation: 92-0028 Revision: 0

Control Rod Drive Mechanism Temporary Shielding

Temporary lead shielding was required to radiologically protect personnel during repairs to the control rod drive mechanism canopy seal welds.

The installation of this temporary shielding has been evaluated and installation requirements have been established to ensure that no equipment important to safety would be compromised. The evaluation determined that, with the shielding adequately secured to prevent falling, a shielding load of up to 60 lbs/ft would be acceptable for Mode 5 and below. Therefore, the proposed shielding load not to exceed 40 lbs/ft with a Mode 4 restraint does not increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report (USAR). The installation of adequately secured temporary shielding does not introduce any new failure modes to this system.

Safety Evaluation: 92-0029 Revision: 0

Installation of Temporary Shielding in Reactor Cavity on Concrete Support Ledge and Nozzle Piping

Temporary lead shielding was required on the concrete support ledge surrounding the reactor and on the hot and cold leg nozzles to protect personnel working on the reactor cavity seal ring shielding.

The installation of this temporary shielding has been evaluated and installation requirements have been established to ensure that no equipment important to safety would be compromised. The evaluation determined that with the shielding adequately secured to prevent falling, a shielding load of up to 160 lbs/ft up to a maximum of 640 lbs could be placed on the hot and cold leg nozzles and up to 2560 lbs of shielding may be installed on the ledge. This installation was found acceptable for Modes 5 and below. The proposed shielding loads do not exceed those determined to be acceptable. Installation of temporary shielding has been evaluated to ensure pressure boundary integrity is maintained during a seismic event. The installation of adequately secured temporary shielding does not introduce any new failure modes to the affected systems.

Safety Evaluation: 92-0030 Revision: 0

Canopy Seal Weld Encapsulation

This modification installed a mechanical clamp to encapsulate the canopy seal weld area. The clamp assemblies were designed and fabricated in accordance with the requirements of ASME Section III, subsection NB.

The installation of the clamp assemblies was evaluated by Westinghouse and found to have an insignificant effect on the reactor vessel head and head adapter seismic analysis. The possible failure of the clamp assembly was evaluated and it was determined that an orderly shutdown of the reactor would not be prevented with potential leakage being negligible as compared to the Chemical Volume and Control System charging pump capacity. It was also determined that the amount of potential leakage was not sufficient for boric acid to penetrate the reactor vessel insulation and cause corrosion of the reactor vessel head area.

Safety Evaluation: 92-0031 Revision: 0

Reissue of Steam Dump Stroking Procedure

Procedure STN AB-001, "Steam Dump Stroking," which was previously deleted, was revised and reissued with steps to defeat the control interlocks for low low T_{avg} and condenser pressure. Steps to isolate each valve prior to testing are also included to prevent inadvertent dumping of steam.

Procedural steps to isolate the valves are included to prevent inadvertent steam dump and the low low T_{avg} and condenser pressure interlock defeats are installed in a manner which will not affect other system operation. Systems required to mitigate the consequences of evaluated accidents are not affected by this change.

This test is performed only in Mode 3, Hot Standby, through Mode 6, Refueling, and does not cause any unique or different challenges to equipment important to safety.

Safety Evaluation: 92-0032 Revision: 0

Changes to Correct Minor Errors and Drawing Inconsistencies Found During Electrical Distribution System Functional Analysis (EDSFA)

The EDSFA program identified minor drawing errors and inconsistencies that require document changes only. No field work is required as a result of these changes. The document change include correction of tag numbers associated with the 4160 volt Class 1E bus undervoltage protection; correct an error in the descriptive text of the Relay Setting Tabulation for the 4.16kV System (NB); corrects destination information of a conductor; corrects a labeling error with an annunciation logic continuation flag; corrects an error which identifies a cable tray with an incorrect raceway tag number; corrects a breaker coordination curve plot to incorporate the time dial setting; to reference the correct coordination curve sheet for other relays; and modifies the list of protective functions provided for each Diesel Generator to agree with design requirements and Updated Safety Analysis Report (USAR) statements.

The corrections associated with this change consists of miscellaneous minor errors and inconsistencies on several design documents that are being made to provide agreement with the as-built configuration. No physical plant or procedure changes are being made as a result of these design document changes.

Safety Evaluation: 92-0033 Revision: 0

Installation of Temporary Shielding in Reactor Cavity on Concrete Support Ledge and Nozzle Piping

Temporary lead shielding was required on the concrete support ledge surrounding the reactor and on the hot and cold leg nozzles to protect personnel working on the reactor cavity seal ring shielding.

See discussion associated to Safety Evaluation 92-0029.

Safety Evaluation: 92-0035 Revision: 0

Temporary Thermocouple Installation Below the Reactor Cavity Seal Ring

This temporary modification installs thermocouples below the reactor cavity seal ring to monitor temperatures near the neutron shield material. The thermocouples and cables will be securely fastened to prevent movement and routed to maintain proper cable separation.

The installation of temporary thermocouples below the reactor cavity seal ring has been evaluated and installation requirements have been established to ensure that no equipment important to safety would be compromised.

The installation of adequately secured and separated cables and thermocouples does not introduce any new failure modes to any system.

Safety Evaluation: 92-0036 Revision: 0

Change Organizational Responsibilities for Trending Program

This change involves reassignment of the administrative responsibility for trending of conditions adverse to quality and Performance Improvement Request (PIR) reviews. The previous responsibility was in the Quality Department/Quality Assurance and is now assigned to Plant Support. The Updated Safety Analysis Report (USAR) changes will reflect this reassignment of duties that is being done to satisfy an NRC commitment for a centralized trending program.

This change of administrative responsibilities for trending and PIR review does not affect any of the equipment associated with accident prevention/mitigation. The same administrative duties are being performed as before, but in a more centralized group.

Safety Evaluation: 92-0037 Revision: 0

Temporary Motion Detector Installation to Monitor Reactor Coolant System Movement during Heatup

This temporary modification installed five linear motion detectors to monitor Reactor Coolant System (RCS) movement during plant heatup. Four detectors were installed on the RCS crossover legs below the Reactor Coolant Pumps (RCP). These detectors were installed on temporary plates which were bolted across the base of an RCP support. The sensor cable from the detector was secured to the support attachment on the RCS piping. The fifth detector was mounted on a plate bolted across a support, for RCP lube oil drain piping. Output signals from the detectors were wired to the integrated leak rate test (ILRT) instrument cable termination boxes inside the bioshield. The detectors were connected to a datalogger outside containment via the ILRT cable termination points at the containment electrical penetration.

The temporary installation of five linear motion detectors has been evaluated and requirements have been established to ensure that no equipment important to safety would be compromised. All cabling will be installed so that proper separation is maintained and the mass of the cabling and detectors does not add any significant load to the piping. It is possible that if the detector mounted on the RCP lube oil drain piping were to become dislodged during a seismic event that the safety injection test line isolation valve actuators could have the air line or flex conduit severed. These valves fail closed under these conditions and would continue to perform their safety function.

By following the requirements established for installation of this temporary modification the operation of equipment important to safety will not be affected in any way that has not been previously evaluated.

Safety Evaluation: 92-0038 Revision: 0

Thermal Expansion Monitoring Procedure

This temporary procedure prescribes inspection methods and instrumentation to monitor component support thermal expansion in the containment during plant heatup.

This temporary procedure provides instructions for monitoring thermal expansion of Reactor Coolant System (RCS) piping, supports, and components during plant heatup. These inspections included visual observations, support/piping clearance measurements, snubber scale readings and/or distance measurements, out of plane displacement measurements for crossover piping and the "B" Reactor Coolant Pump (RCP) support column, loose parts monitoring system recordings, seismic monitoring system recordings, and RCP vibration monitors. All equipment used was evaluated when installed and determined not to affect the operation of any equipment important to safety.

All inspections performed in accordance with this procedure are passive and do not have any affect on the frequency, type, or consequences of any events evaluated in the Updated Safety Analysis Report (USAR).

Safety Evaluation: 92-0042 Revision: 0

Replacement of Differential Pressure Switches on the Essential Service Water Self-Cleaning Strainers

This modification to the Essential Service Water (ESW) System Self-Cleaning Strainer control circuitry eliminates the need for control cabling between the Control Room and the ESW pumphouse. The electrical losses in the control wiring for the strainers could result in the inoperability of the automatic backwash feature of the strainers under abnormal grid voltage conditions. The modification to the control circuitry installs new differential pressure switches in the control circuitry in close proximity to the strainers and does not alter the design function of the strainers.

Safety Evaluation: 92-0044 Revision: 0

Installation of Drip Pan Below the Reactor Vessel Head Vent Line

This temporary modification installs a drip pan below the reactor vessel head vent line. Very slight leakage of Reactor Coolant System water has been observed to be dripping from the vent line. This temporary modification is being installed to prevent boron from crystallizing on the vessel head seismic support structure. The drip pan will be placed on the support structure directly below the vent line and will not impair the operation of any equipment. The drip pan does not pose a seismic concern due to its low mass and should it become dislodged, the kickplate lip around the support structure is high enough to retain it during a seismic event.

Safety Evaluation: 92-0046 Revision: 0

Loose Parts Monitoring System Temporary Modification

The Loose Parts Monitoring System (SQ) is a non-safety related detection system designed to meet the recommendations of Regulatory Guide 1.133. The system consists of twelve channels which monitor the presence of metallic parts in the reactor and the four steam generators.

This temporary modification disconnected the twelve inputs to allow Westinghouse to monitor and record data from the twelve sensors at their discretion. The monitoring and recording equipment utilizes a Honeywell visicorder to process the signals from instrumentation inside containment. This equipment will not cause any detrimental effects to the conductors or containment electrical penetrations associated with the Loose Parts Monitoring System. The use of this non-safety related equipment in this manner will not cause any detrimental effects to any systems or components important to safety.

This temporary modification was removed prior to Mode 2, Startup. The system is only required in Modes 1, Power Operation, and Mode 2, Startup.

Safety Evaluation: 92-0047 Revision: 0

Revision to Heat Exchanger/Room Cooler Design Data

This modification revises WCNOC-62, "Heat Exchanger/Room Cooler Design Data," Updated Safety Analysis Report (USAR) Chapter 9 and System Descriptions M-00EA, M-10EF(Q) and M-10GL(Q). NRC Inspection Report No. 50-482/91-17 identified several concerns with WCNOC-62. A review of the design heat load calculation for the penetration room coolers resulted in the initiation of this modification to revise the affected design documents.

The penetration room coolers were analyzed for normal and post-LOCA modes of operation in calculation GL-02-W. The calculation establishes the design heat loads in penetration rooms 1409 and 1410 and available cooler capacity. The results of the calculation concluded that the penetration room coolers have sufficient capacity for cooling the rooms during normal conditions and post-LOCA conditions. There is adequate margin in the room coolers to allow for cooler tube plugging.

Safety Evaluation: 92-0048 Revision: 0

Pressure Transmitter Installation for Transient Pressure Analysis

This temporary modification installed four pressure transmitters to monitor the safety injection accumulator discharge piping during Reactor Coolant System (RCS) boundary valve leakage testing and subsequent plant heatup. One transmitter was installed on the test connection of each accumulator discharge line between the first and second off check valves using a tee fitting to facilitate connection of the hydro pump used for RCS boundary valve leak testing. Output from the transmitters was wired to the cables for RCS midloop level instruments and a datagraph acquisition system was installed at the balance of plant instrumentation panels.

The installation of the transmitters was performed in accordance with requirements defined by Nuclear Plant Engineering which assured that all equipment would be seismically qualified and proper cable separation would be maintained.

Safety Evaluation: 92-0049 Revision: 0

Change of Pipe Material for Essential Service Water (ESW) Piping to/from the Spent Fuel Pool Pump Room Cooler

This modification involves changing piping material from carbon steel to stainless steel. The subject piping is the Essential Service Water (ESW) Train "A" supply and return for the Spent Fuel Pool Pump Room Cooler and ESW emergency make-up supply to the Spent Fuel Pool. The reason for the change is to replace the carbon steel piping which has become clogged with scale. Stainless steel has better corrosion resistance than carbon steel. The required pipe wall thickness of the stainless steel piping is less for the same outside diameter, compared to carbon steel, therefore the internal diameter is larger. The larger internal diameter is desirable because of the reduction in pressure drop. The ESW System, Fuel Building Heating, Ventilation, and Air Conditioning (HVAC) System, and the Fuel Pool Cleanup and Cooling System are affected by this change.

The subject piping runs through the Auxiliary Building and into the Fuel Building. Both of these buildings are seismic category I structures.

The replacement piping and valve configuration is the same as the original design. The replacement valves have been evaluated and all their environmental, seismic and equipment qualification requirements are met. All design flows remain unchanged. Therefore, the capability to provide the required flow to the Spent Fuel Pool Pump Room Cooler and makeup flow to the Spent Fuel Pool is not adversely affected by this modification.

Reference Safety Evaluation 92-0067 for Train "B" evaluation.

Safety Evaluation: 92-0052 Revision: 0

Modification of Shims for the Reactor Coolant System Crossover Leg Whip Restraint Saddle Supports

A plant modification was performed to rework the crossover leg whip restraint saddle supports on the Reactor Coolant System (RCS) piping within the containment to provide a minimum 1/16 inch gap between the shim and saddle at an RCS temperature of 557° F. This modification allows machining of the shims to establish the original design configuration. This modification does not effect the consequences of previously evaluated accidents or effect operations of the plant.

Safety Evaluation 92-0054 Revision: 0

Class 1E Electrical Equipment Air Conditioning Units Document Corrections

This change updates Updated Safety Analysis Report (USAR) information to reflect the as-built design of the Class 1E Electrical Equipment Air Conditioning Units. This change is a documentation change only and does not cause any equipment changes to made in the field. The static pressure values and compressor power input values are being changed, but this will not affect the ability of the units to perform their safety-related function of cooling Class 1E electrical equipment in the Control Building as described in the USAR.

This documentation change to reflect as-built vendor data does not affect equipment performance as described in the USAR. Therefore, no safety-related system functions are affected by the change.

Safety Evaluation: 92-0056 Revision: 0

Essential Service Water Pump Horsepower Requirements

This modification revised the list of loads supplied by the Emergency Diesel Generator (EDGs) for the brake horsepower requirements for the Essential Service Water (ESW) pumps. The corresponding kW value was also modified to reflect the actual load.

The list of loads supplied by the EDGs indicated a value of 1630 for brake horsepower for the ESW pumps. During a design review, it was identified that the list of loads for the ESW pumps brake horsepower was not consistent with the power requirement as shown on the pump performance curves. The pump performance curves indicate that at a flow rate of 15,000 gpm the brake horsepower is 1700hp. The present kW load on the EDG is 5285 kW including the additional 70hp for the ESW pump. The impact of the additional 70hp load on the EDG is not significant with respect to precluding the EDG from successfully performing its intended design function.

Safety Evaluation: 92-0057 Revision: 0

Turbine EHC Hydraulic Power Unit Pump Discharge Piping Modification

This modification replaces the discharge piping for the main turbine EHC hydraulic power unit pump. The replacement of the discharge piping from 1 1/2-inch to 2-inch nominal pipe size is to prevent excessive vibration.

A General Electric Technical Information Letter identified the occurrences of weld failures in the hydraulic power unit pump discharge piping due to pump vibration. This piping is located between the discharge of the pumps and relief valves FV005 and FV006. Replacement of the discharge piping with 2-inch piping changes the stiffness and increases the weld area.

Safety Evaluation: 92-0058 Revision: 0

Temperature Correction for Outside Environment

This modification revises the Updated Safety Anslysis Report (USAR) and Mechanical/Nuclear Design Criteria to reflect site specific minimum outside winter temperatures. The previously identified values were based on the most conservative information of the five original SNUPPS sites.

This change makes the USAR and Mechanical/Nuclear Design Criteria consistent with site specific minimum winter temperatures. Because the new values are site specific and are bound by values previously accepted in the safety evaluation report there is no change in the ability of any equipment to perform its safety function.

Safety Evaluation: 92-0059 Revision: 0

Control of Locked Component Status Procedure Change

This procedure change deleted Fuel Oil Transfer Pump A Discharge Strainer Drain Valves JEV0063 and JEV0065 from ADM 02-102, "Control of Locked Component Status."

Valves JEV0063 and JEV0065 are drain valves located on Y-Strainers. These valves are not an active component in any system flow path. These valves are not required to be locked components. Removal of these valves from the locked component list does not degrade the ability of the Fuel Oil Transfer System to provide makeup to the Fuel Oil Day Tank.

Safety Evaluation: 92-0060 Revision: 0

Corrections to Updated Safety Analysis Report (USAR) Description of the Emergency Fuel Oil System

This USAR change makes corrections to the description of the Emergency Fuel Oil System to reflect actual system configuration. As a result of Revision 1 to calculation M-JE-321, USAR Section 9.5.4.2.2.C and Table 9.5.4-1 are being revised to correct the Emergency Fuel Oil Day Tank capacity and the fuel consumption rate to be consistent with ANSI N195-1976. Additionally, this change clarifies the capacity of the emergency fuel oil transfer pumps to be greater than or equal to 15 gpm. Emergency fuel oil transfer pump setpoints and alarm setpoint to maintain day tank fuel level were not changed, therefore, the current margin over the technical specification required 390 gallons of fuel oil in the day tank is unchanged.

Safety Evaluation: 92-0062 Revision: 0

Modification to Positive Displacement Pump Relief Valve Leakoff Line

This temporary modification crimps the leakoff line on the Positive Displacement Pump (PDP) relief valve to prevent the loss of an estimated .25 gpm of borated water from the Volume Control Tank (VCT). The relief valve bellows appeared to have failed allowing the VCT to drain into the valve bonnet and out the leakoff line to plant drains. Crimping the leakoff line decreased the loss of VCT inventory allowing stabilization of VCT level until the relief valve can be reworked.

The PDP relief valve is not required to bring the plant to a safe shutdown condition and its post accident position has no affect on plant safety. During a Loss of Coolant Accident, the Centrifugal Charging Pumps are aligned to the Refueling Water Storage Tank and the VCT outlet valves are closed.

Safety Evaluation: 92-0063 Revision: 0

New Procedure for Charging Individual Battery Cells

A new procedure was developed for charging individual cells of the Wolf Creek Generating Station Class 1E batteries while the battery bank is operable. The power supply provides a precisely regulated DC output voltage through the use of semiconductors and transistors to a selected or programmed value. Conditions are monitored during the charging process. The power supply is protected against the effects of overloads and internal shorts by automatic crossover and input circuit breaker features, respectively. Diodes in the power supply circuitry prevent current flow from the battery back into the power supply. The power supply is protected by a thermostatic control circuit to automatically shut the unit down in the event of thermal overload. Class 1E integrity of the battery bank being charged is maintained because a Class 1E fuse and Q cable are installed between them, which provides for an acceptable Class 1E isolation device.

The insulated alligator clips at the ends of the Q cable are not Class 1E qualified. Their failure, seismically induced or spontaneous, is unlikely because of their simplicity, current carrying capacity, very low mass and physical dimensions. The dimension restraint eliminates any possibility of short circuiting between conductors after assuming their failure. Tie wrapping the cable/clips at their terminations will restrict their failure to the termination area.

The power supply which will be located in the battery room during use, has been evaluated for seismic category 2 over 1 concern and seismic qualification of the existing battery banks. The distance between power supply and cell terminations is minimized and charging cables are not draped over any other conductors. The battery room fire protection features and ventilation features are not affected by the power supply and procedure actions.

The presence of the power supply across the low cell(s) of a Class 1E battery bank does not increase the malfunction consequences of the battery bank or its normal charger because ground fault, Class 1E integrity and normal charger monitoring and performance capabilities are maintained.

Safety Evaluation: 92-0064 Revision: 0

Change Elevation of Containment High Range Radiation Monitor

The containment digital high range radiation monitor (DHRRM) system detectors were listed in the Updated Safety Analysis Report (USAR) as located at El. 2052'-0". However, the as-built condition has one detector (GT-RE-60) at El. 2052'-0" and the other detector (GT-RE-59) is at El. 2073'-0". This condition requires a change to the USAR to reflect the two different elevations. These redundant instruments are used to detect, indicate and alarm high gamma radiation levels in containment. The location change meets the design, material, and construction standards applicable to the DHRRM system. This elevation change also complies with all applicable regulatory and design basis considerations for the system.

This change does not cause a physical plant change and the existing equipment continues to meet environmental and seismic qualification requirements for the DHRRM system. The higher elevation does not expose the detector to new hazards that have not been previously analyzed. Therefore, this change does not affect the ability of the DHRRM system to perform its intended function.

Safety Evaluation: 92-0065 Revision: 0

Radiological Emergency Response Plan (RERP) Revision

This revision to the RERP updated organizational titles, made Kansas Protective Action Guides (PAGS) consistent with State and County Emergency Plans and corrected typographical errors.

This change is to organizational titles only, the positions have not been deleted. The positions are now under different supervision only.

Safety Evaluation: 92-0067 Revision: 0

Change of Pipe Material For Essential Service Water (ESW) Piping to/from the Spent Fuel Pool Pump Room Cooler and ESW Make-up Supply to the Spent Fuel Pool

This modification involves changing pipe material from carbon steel to stainless steel. The subject piping is the Essential Service Water (ESW) Train "B" supply and return for the Spent Fuel Pool Pump Room Cooler and ESW emergency make-up supply to the Spent Fuel Pool. The reason for the change is to replace the carbon steel piping which has become clogged with scale. Stainless steel has better corrosion resistance than carbon steel. The required pipe wall thickness of the stainless steel piping is less for the same outside diameter, compared to carbon steel, therefore the internal diameter is larger. The larger internal diameter is desirable because of the reduction in pressure drop. The ESW System, Fuel Building Heating, Ventilation, and Air Conditioning (HVAC) System, and the Fuel Pool Cleanup and Cooling System are affected by this change.

The subject piping runs through the Auxiliary Building and into the Fuel Building. Both of these buildings are seismic category I structures.

The replacement piping and valve configuration is the same as the original design. The replacement valves have been evaluated and all their environmental, seismic and equipment qualification requirements are met. All design flows remain unchanged. Therefore, the capability to provide the required flow to the Spent Fuel Pool Pump Room Cooler and makeup flow to the Spent Fuel Pool is not adversely affected by this modification.

Reference Safety Evaluation 92-0049 for Train "A" evaluation.

Safety Evaluation: 92-0068 Revision: 0

Procedure Change to Add an Additional Check Valve to the ASME Section XI testing on the Safety Injection System

In accordance with Generic Letter 89-04 a procedure change was made to include an additional check valve in the suction line of the Safety Injection (SI) Pump from the Refueling Water Storage Tank (RWST) into the ASME Section XI testing program. To test this check valve, the motor operated valve (MOV) in the suction line upstream of the cross tie line to the redundant SI pump is isolated. This configuration changes that described in the Updated Safety Analysis Report (USAR). This change requires the redundant operable SI pump to take suction from the RWST via the alternate suction path (i.e., cross-connect line), if the diesel generator in the opposite train is assumed to fail under the single failure criteria should a postulated Loss of Coolant Accident (LOCA) occur during the testing. A calculation was performed confirming the adequacy of the Emergency Core Cooling System (ECCS) performance in this configuration.

In the event that an SI signal occurs during the testing, both pumps would have to take suction via the single suction line from the RWST. The operability of the pumps would not be adversely affected because the available net positive suction head for both pumps is still adequate in this configuration. Other system components such as piping, instrumentation, and actuation logic circuits are not affected by the procedure change.

The procedure change has also deleted the need to enter the action statement during testing because the discharge MOV of the SI pump in test is no longer allowed to be closed when the SI pumps are operable. This change eliminates the potential to damage the SI pump in test if a valid SI signal were to occur during testing. The SI pump discharge valve is allowed by the procedure to be closed when the SI pumps are no longer required to be operable (i.e., Modes 4, 5, and 6). To prevent inadvertent injection into the Reactor Coolant System (RCS) in Mode 3 when the SI pumps are operable, the procedure is restricted from use unless the RCS pressure is equal to or above 1,800 psig.

The procedure change does not diminish nor compromise the SI system in fulfilling its ECCS safety function. In addition the change does not cause any unanalyzed condition from those previously evaluated to occur.

Safety Evaluation: 92-0069 Revision: 0

Relocation of Sample Point for Condenser Air Discharge Monitor

A plant modification was made to relocate the sample point for the Condenser Air Discharge Monitor. The sample point for this radiation monitor was located downstream of the Demineralized Water Degassifier vacuum pump exhaust to the Turbine Building Heating, Ventilation and Air Conditioning (HVAC) exhaust. The oil mist eliminators on these vacuum pumps produced oil in the Turbine Building HVAC exhaust which clogged the pre-filters and coated the beta detector windows on the radiation monitor. In addition, the previous location provided a poor sample since the upstream air was diluted with Turbine Building air. The new location is upstream of both of these lines as well as a line which exhausts air from the Miscellaneous Condensate Drain Tank.

The Condenser Air Discharge Monitor provides: (1) a back up to the steam generator liquid and the steam generator blowdown processing radiation monitors for detection of primary-to-secondary leaks in the steam generator, and (2) a signal to close the steam generator blowdown isolation valves on high radiation to prevent discharge of radioactive fluid and to limit radioactive contamination of the blowdown demineralizers. In order to detect these types of radiation leaks, the monitor need only sample exhaust air from the Condenser Air Removal (CAR) System. The new sample location would continue to provide this function since the CAR System exhaust to the Turbine Building Ventilation (TBV) System remains upstream.

The two systems exhaust to the TBV System that are no longer sampled due to the relocation of the Condenser Air Discharge Monitor have been evaluated for the need to be monitored. Only the Oily Waste System could potentially contain trace amounts of radioactive contaminates. Any radioactive contaminates from this Oily Waste System exhaust would be detected by the Unit Vent Radioactive Monitor located downstream.

The new location, with a different air flow from the original position, no longer provides isokinetic sampling. Isokinetic sampling affects particulate sampling only. Currently, the Unit Vent is used for particulate sampling so the ability for this sampling has not been lost by the relocation of the Condenser Air Discharge Monitor.

Therefore, the relocation of the non-safety Condenser Air Discharge Monitor sample point will not affect its intended function to detect radiation.

Safety Evaluation: 92-0070 Revision: 0

Installation of Corrosion Product Samplers

This temporary modification installs corrosion product sample stations at connections to various systems to monitor corrosion product transport. The formation, transport and deposition of corrosion products have led to a rarity of problems in nuclear steam cycles. Application of long term integrated sampling techniques for preconcentration of corrosion products is the optimum way to obtain accurate estimates of corrosion product transport.

The sample stations are all located in the Turbine Building which does not contain any equipment important to safety. The sample streams are all brought to the sampler by stainless steel tubing. The cooling water is brought to the sampler's cooler by a hose. Failure or spray effects from the sampler, tubing or hose would not increase the probability or consequences of any equipment important to safety to fail because of the absence of this equipment in the Turbine Building. The sample stream is at a higher pressure than the cooling water system; therefore, the potential assuming sampler cooler failure for raw lake water degrading or adversely affecting sensitive components or their performance in the secondary cycle is not created. The sample stream and cooling water, after passing through the sampler unit, are discharged to the Turbine Building sump. This pathway is the normal collection point for any leakages from the various systems associated with the modification. The modification has not created a new or different release path to the environment from those previously evaluated.

Safety Evaluation: 92-0071 Revision: 0

Emergency Diesel Generator Governor Access Platforms

This modification installs two small platforms in the Diesel Generator Building to provide access to the emergency diesel generator governors. Installation of the access platforms prevents personnel from stepping on process lines during periodic inspection of the emergency diesel generator governors. The platforms have been seismically designed to ensure the integrity of any safety-related equipment in the vicinity of the platform. The platform grating is immediately below a flame detector and suppression system heat detectors located above the platform. The grating has sufficient openings so that if a fire were to develop below the platform, its detection would be permitted by the flame detectors while the function of the heat detectors would be unaffected.

Safety Evaluation: 92-0072 Revision: 0

Addition of Radwaste Processing Equipment and Increase Storage Area

See Safety Evaluation 92-0096

Safety Evaluation: 92-0074 Revision: 0

Plant Change to the Computer Room HEATING, VENTILATION AND AIR CONDITIONING (HVAC) Units

An earlier plant modification removed all the computers from the Computer Room and turned this room into office space. This modification was done to compensate for the computer removal (heat load) from the room. This modification replaces the existing sheave with an adjustable one to reduce the air flow and noise level and throttle the chilled water flow to reduce the chilled water through the unit. Also, modification of the control circuits was performed to eliminate the dehumidification aspect of the system for this room. Without the computers this feature was no longer needed.

Reduction in the air flow and chilled water flow will not change the original function of the Heating, Ventilation and Air Conditioning (HVAC) units by maintaining the Computer Room temperature at 72^{O}F without operating the units out of their design limits. These units are non-safety related and there is no safety-related equipment in the Computer Room.

Safety Evaluation: 92-0075 Revision: 0

Procedure Change to use Hydrogen Purge Subsystem to Reduce Containment Building Atmospheric Pressure

A procedure revision has been made to allow use of the Hydrogen Purge Subsystem to reduce Containment Building atmospheric pressure within technical specification limits when the normal Containment Building Purge Systems are out of service. The penetration used by the Hydrogen Purge Subsystem is normally isolated, but is equipped with isolation valves which receive an automatic isolation signal. The procedure change allows opening the penetration path and an analysis was performed assuming a loss of coolant accident (LOCA) with a loss of offsite power. The single limiting failure of one protection train is also assumed. The subsequent result of this postulated event is that the penetration remains open for up to 25 seconds before closure, during which time about 136 cubic feet of Containment Building atmosphere is exhausted via the Emergency Exhaust System. The analysis results indicate that the calculated LOCA thyroid dose in the Exclusion Area would be increased by about 14 rem because of the delay in isolating the penetration. However, the resulting dose consequences from the hydrogen purge time delay, combined with the calculated dose contributions from Containment Building and Emergency Core Cooling System recirculation leakages, remain within the guideline values of 10 CFR Part 100.

The failure of both valves to close need not be postulated because this condition would require double failure. Therefore, the use of the Hydrogen Purge Subsystem to reduce Containment Building pressure is within previously analyzed conditions for plant operations and accident analysis.

Safety Evaluation: 92-0076 Revision: 0

Drawing Corrections for the Diesel Generator Intercooler System

The flow restriction orifice arrangement for the intercooler system on the diesel generators was reviewed on plant drawings against the vendor piping drawings and the as-built field configuration. The discrepancies noted are that plant piping and instrumentation drawings (P&ID's) show two flow orifices that are not shown on vendor prints nor installed in the field and the air cooler cooling water orifices are shown on the air cooler inlets on the P&ID, but are shown on the outlets in vendor piping assembly drawings and verified on the outlets in the field. Both of these differences were evaluated with vendor input and confirmed that the as-built configuration is the proper arrangement so the P&ID's were revised.

In addition to the above, an illegible drawing was enhanced for readability as part of the above changes. No plant physical changes were made as a result of the drawing changes.

Safety Evaluation: 92-0077 Revision: 0

Modifications to the Demineralized Water Storage Tank Overflow Line

A modification was made to the Demineralized Water Storage Tank overflow line to remove a check valve and relocate the power feed point of the freeze protection from the top of the line to the bottom. The change to remove the check valve was made to prevent possible freezing of the check valve, which would prevent overflow protection. The check valve had been added in the past to keep cold air out of the inside of the pipe from giving false indication of freeze protection failure due to cold temperatures. The revision of the power feed point on the freeze protection was made to help maintain a higher temperature inside the overflow line and had successfully been used at another power plant.

The Demineralized Water System is non-safety related and the changes will not impact safety-related systems.

Safety Evaluation: 92-0078 Revision: 0

Procedure Revision to Reflect Operations Organization Changes and Correct Titles

An administrative procedure change was made to reflect organizational changes to show that the Shift Supervisors report to the Supervisor Operations. Also, a procedure change was made to show that the title Fire Protection Coordinator should be Fire Protection Specialist.

These administrative procedure changes do not affect the physical plant arrangement nor the way equipment is operated.

Safety Evaluation: 92-0079 Revision: 0

Addition of Chemical Injection/Effluent Piping to Service Water/Circulating Water System

Chemical injection lines and chemical effluent piping were added from vendor skids in and about the chlorine cylinder storage facility via a below-grade trench. The lines were routed to injection quills and to Service Water/Circulating Water System connections inside the Circulating Water Screenhouse. All equipment and interface systems are classified non-nuclear safety related and therefore will not affect safety-related components.
The on-site storage and use of the chemicals proposed to be used with this equipment is addressed by a separate evaluation.

SAFETY EVALUATION: 92-0080 Revision: 0

Updated Safety Analysis Report (USAR) Change to Clarify Job Descriptions

A change was made to Section 13.2.2.8.1 of the USAR to more clearly describe which positions require job descriptions. The criteria used was that of American Nuclear Standards Institute (ANSI) 3.1-1978, Section 4. This administrative change to the USAR does not affect physical plant configurations or operation. Therefore, no impact on nuclear safety-related equipment will result from this USAR clarifying change.

Safety Evaluation: 92-0083 Revision: 0

Addition of a Computer Point for Reactor Coolant Drain Tank (RCDT) Level Indication

The RCDT level indication previously was available in the Radwaste Control Room only. A plant technical specification surveillance procedure requires monitoring of the RCDT level for determining reactor coolant leak rates which was difficult since the location of the readout was outside the Control Room. Therefore, the existing signal was re-routed to a panel to provide computer input for Control Room readout. The RCDT level signal is non-nuclear safety related as is the computer point. The additional conduit for re-routing the signal to the panel is supported on the safety-related Auxiliary Building lls. These supports were evaluated and found acceptable.

Therefore, the plant modification to provide the RCDT level signal to a computer point does not adversely impact any nuclear safety-related equipment.

Safety Evaluation: 92-0084 Revision: 0

Failed Fuel Monitor Setpoint Changes

The non-nuclear safety related Failed Fuel Monitor setpoints were described as fixed values in the Updated Safety Analysis Report (USAR). The vendor drawings allow the alarm points to be varied as necessary to eliminate alarms that are determined not to be failed fuel related. The Alert Alarm and High Alarm setpoints were revised to allow variation tied to changing background levels. The Alert Alarm has been reset to 0.5 uCi/ml plus background, which will prevent alarms on the average unplanned crudburst while remaining low enough to alarm on iodine spikes/failed fuel occurrences. The High Alarm has been reset to 5.0 uCi/ml, which will prevent alarms on most unplanned non-clad failure activity changes while remaining low enough to alarm on possible failed fuel. The background levels will be checked periodically to reset the alarms as appropriate.

The above setpoint changes will still provide the intended alarm indications while eliminating the previous unnecessary alarms caused by non-fuel failure activity changes. This non-nuclear safety related setpoint change does not affect any safety-related hardware or the operation of safety-related equipment.

Safety Evaluation: 92-0085 Revision: 0

Revise the Allowable Solution Strengths of Certain Chemicals

The Condensate Hydrazine Supply Tanks, Condensate Ammonia Supply Tanks, and the Feedwater Hydrazine and Ammonia Mixture Solution Tank were limited to 5% concentration limits during plant operation and 17% concentration limits during plant shutdown per the Updated Safety Analysis Report (USAR). The lower value allowed during plant operation was identified to protect personnel from the effects of the chemicals. However, allowing the concentration to be maintained at the 17% value during all phases of operation will enhance personnel safety since the frequency of adding concentrated chemicals to the tanks would be reduced. Solution strengths of 5% or less are at times impractical during power operation because of system demand. Allowance of strengths up to 17% is desirable during these high demand periods.

This non-nuclear safety related operation change will not adversely affect safety-related systems. Also, the personnel hazard associated with handling these chemicals will be reduced by the less frequent handling of the concentrated chemicals being added to the tanks.

Safety Evaluation: 92-0086 Revision: 0

Addition of Nitrogen Supply to Lower Condensate Dissolved Oxygen

A plant change to add piping, tubing, valves and flow meters to allow the supply of nitrogen to the condenser hotwell was made to reduce the dissolved oxygen content of the condensate. Control of the dissolved oxygen reduces the formation of corrosion products. The addition of a nitrogen blanket in the condenser hotwell allows the plant to meet established oxygen level goals and Institute of Nuclear Power Operations guidelines.

The condenser and nitrogen system are non-nuclear safety related and the changes will not interact with safety-related systems.

Safety Evaluation: 92-0087 Revision: 0

Updated Safety Analysis Report (USAR) Clarification on Chlorine Alarms

Sections of the USAR were revised to describe procedural controls that are in place if a chlorine alarm is received in the Control Room. The procedural action taken when a chlorine alarm is received is to close any open doors leading to the Control Room, which was not stated in the previous USAR wording. The revised wording will clarify that actions are taken to meet Regulatory Guide 1.95 for Control Room access door closure if an accidental chlorine release occurs.

This USAR change clarifies personnel actions and does not adversely affect nuclear safety-related equipment. The procedural controls described will enhance personnel safety in the event of a chlorine release.

Safety Evaluation: 92-0088 Revision: 0

Replacement of Main Steam Drain Piping

Pipewall thinning occurred immediately downstream of a certain Main Steam System piping isolation valve. The cause of this thinning was two-phase erosion/corrosion. A plant modification was made to replace all piping downstream of the affected valve and other valves that have similar configurations and up to the first elbow after the common drain header. The piping change replaced the previous carbon steel piping and fittings with low-alloy steel (2 1/4 Cr-1 Moly) pipe and fittings. This specific low-alloy steel has essentially the same mechanical properties as the original carbon steel, yet is more resistant to erosion/corrosion wear, at the operating temperature of interest.

The piping changes are on a non-nuclear safety related system and are located within the Turbine Building. No high energy line break, environmental qualification, flood, seismic or piping stress values are adversely affected by the plant change.

Safety Evaluation: 92-0089 Revision: 0

Miscellaneous Changes to the Health Physics Facility Locations and Program

This Updated Safety Analysis Report (USAR) change revised Section 12.0 of the USAR to reflect changes to the Health Physics office area, name and title changes, upgrade equipment, present utilization of mens' and womens' locker rooms, instruments contained in and utilization of the Instrument Storage Room, location of the Tool and Equipment Decontamination Room and protective clothing storage area locations.

Drawings and wording in the USAR have been revised and/or deleted to reflect the relocation of Health Physics facilities from the Shop Building to the Technical Support Building. The physical relocation of the Health Physics facilities were initiated to provide better organization and utilization of office and work areas. Storage locations for protective clothing and modesty garments have been designated in a manner to best facilitate access to and from contaminated or potentially contaminated areas. This relocation will not affect the ability of the Health Physics organization to provide daily plant support or respond to emergency conditions.

Electronic Dosimeters are being procured to enhance or expand the protection of personnel. This type of dosimeter has an exposure alarm that alarms on dose rate or accumulated dose exposure. This gives the carrier an active alarm versus the present Pocket Ion Chamber meter readout. Personnel protection is in no way compromised by this dosimeter and the margin of safety to other equipment is not diminished.

Safety Evaluation: 92-0091 Revision: 0

Design Criteria and Methodology for Snubber Reduction

This modification provides a revision to the Updated Safety Analysis Report (USAR) for the utilization of ASME Code Case N-411, "Alternate Damping Values for Seismic Analysis of Class 1, 2, and 3 Piping ASME Section III, Division 1" when performing snubber reduction stress analysis. Document C-1, "Design Criteria and Methodology for Snubber Reduction, WCGS" has been issued to establish the technical design criteria and associated methodology which shall be used on the snubber reduction program. The design criteria is consistent with the original criteria as specified in Specification 10466-M-200 and applicable section of the USAR. The snubber reduction methodology reflects current analytical techniques and methodology applied to reduce snubber effectively. The methodology in Document C-1 utilized the alternate damping values identified in ASME Code Case N-411 as endorsed in Regulatory Guide 1.84 and results in acceptable accurate seismic analysis of piping. This new methodology, will not invalidate the techniques and methods developed and used in the original procedures and calculations.

Safety Evaluation: 92-0092 Revision: 0

Reactor Vessel Head Vent System (RVHVS) Piping Rerouting

The original structural qualification analysis for the RVHVS did not conservatively bound all actual plant specific operating loads. As a result, pipe stresses and support loads exceed industry standard allowable values. Interim operation was evaluated and found acceptable, but long term operation required rerouting of the subject piping to ensure the stress levels are brought within ASME Code allowable. The effect of the modification to the RVHVS on the head lift rig legs, the head vent nozzle, and Control Rod Drive Mechanism cooling shroud were evaluated and determined to bring the system stress levels within the ASME Code allowable.

The modification to the RVHVS has no adverse effect on the capability for venting noncondensible gases, which could impede natural circulation cooling through the reactor in the event of beyond-design-basis events.

Safety Evaluation: 92-0093 Revision: 0

Replacement of 3rd Stage Extraction Pipelines Due to Wear Related Pipewall Thinning

Pipewall thinning occurred in the 12" pipe sections of the 3rd stage extraction pipelines leading to the high pressure heaters. The cause of this abnormal pipewall thinning is either two-phase flow-acceleration corrosion, commonly called erosion/corrosion or impingement erosion by high velocity liquid. A plant modification was made to replace the affected piping. The piping change replaced the previous carbon steel piping with low-alloy steel (2 1/4 Cr-1 Moly) pipe. This specific low-alloy steel has essentially the same mechanical properties as the original carbon steel, yet is more resistant to wear.

The piping changes are on a non-nuclear safety related system and are located within the Turbine Building. No high energy line break, environmental qualification, flood, seismic or piping stress values are adversely affected by the plant change.

Safety Evaluation: 92-0094 Revision: 0

Replacement of 5th Stage Extraction Pipelines Due to Wear Related Pipewall Thinning

Pipewall thinning occurred in the 16" pipe sections of the 5th stage extraction pipelines leading to the high pressure heaters. The cause of this abnormal pipewall thinning is either two-phase flow-acceleration corrosion, commonly called erosion/corrosion or impingement erosion by high velocity liquid. A plant modification was made to replace the affected piping. The piping change replaced the previous carbon steel piping with low-alloy steel (2 1/4 Cr-1 Moly) pipe. This specific low-alloy steel has essentially the same mechanical properties as the original carbon steel, yet is more resistant to wear.

The piping changes are on a non-nuclear safety related system and are located within the Turbine Building. No high energy line break, environmental qualification, flood, seismic or piping stress values are adversely affected by the plant change.

Safety Evaluation: 92-0096 Revision: 0

Addition of Radwaste Processing Equipment and Increase Storage Area

This plant modification adds equipment for solid and liquid radwaste processing to replace obsolete equipment and to modify the Radwaste Building and Waste Bale Drumming Area to allow storage of three to five years of low level radioactive waste. The major design features include:

Location of the Liquid Radwaste Demineralizer Skid in room 7215, and the Solid Radwaste Disposal Station in room 7218, and routing all necessary piping to these rooms.

Increasing the capacity of the Solid Radwaste Bridge Crane from 7.5 tons to 9.33 tons, and modifying the crane to allow handling of radwaste shipping containers (HICs and LSA boxes).

Installation of a shield wall on the south end of the Truck Bay to provide radiation protection for radwaste operations in the Truck Bay, and an electric roll-up door and an emergency exit door on the west side of the Truck Bay to allow easier truck access.

Modification of room 7228 to provide rooms for storage of compacted waste in drums.

The systems modified or affected by this plant change include the Liquid Radwaste System, Solid Radwaste System, Secondary Liquid Radwaste System, Compressed Air System, and Floor and Equipment Drain System. The plant changes do not affect any safety-related components, meets seismic requirements per Regulatory Guide 1.143, and has no impact on the environmental program since there is no change to the liquid, gaseous, or solid wastes released to the environment. Tornado missile and dispersion concerns have been evaluated as acceptable for the new arrangement which provides a different type of protection than the original design. Storage of waste in the Waste Bale Drumming Area has been evaluated and the arrangement will ensure that a radioactive release is within existing limits during design basis conditions.

Safety Evaluation: 92-0097 Revision: 0

Installation of a Resin Trap Backwash Piping

The resin trap in the Mix and Storage Tank vent pipe fouls with resin fines or sludge resulting in down time of the regeneration system. A resin trap backwash connection was provided by supplying an ANSI B31.1 piping system with isolation valves.

The equipment is located in the Turbine Building and is non-nuclear safety related. No high energy line break, environmental qualification, flood, seismic or piping stress values are adversely affected by the plant change.

Safety Evaluation: 92-0098 Revision: 0

Non-Q Drawing Discrepancies

Various non-nuclear safety related drawings were revised to correct typographical and editorial errors. These drawing changes did not modify plant equipment or affect the design function of any safety related equipment. The changes include:

- Correcting an incorrect reference to another drawing.

- Correcting line designators to show the correct line number going to an 'A' and 'B' tank as the designators were reversed.

- Correcting the designation of which pump was the 'north' pump.

- Correcting the representation that there is a separate flow control valve and a bypass valve even though the vendor gave both valves a single number.

- Correcting the abbreviated designation of the Closed Cooling Water System as CLCW versus the incorrect CCW designation.

- Adding a sectional control valve on the drawing that is the correct as-built configuration of the fire protection piping.

Safety Evaluation: 92-0100 Revision: 0

Uprating of the Main Step-up Transformers

The uprating of the power output of Wolf Creek Generating Station, planned for mid-cycle 7 operation, will impose additional heat loads onto the main step-up transformers. A modification will be made during the refueling outage to replace the oil coolers, replace the cooling oil pumps, add piping and fittings to the new coolers, and provide miscellaneous interface equipment. The modifications will increase the MVA rating from 415 MVA to 448.2 MVA.

The new coolers are non-nuclear safety related and do not affect any safety-related equipment. The equipment is located outside of the Turbine Building.

Safety Evaluation: 92-0102 Revision: 0

Application of Temporary Lead Shielding to Volume Control Tank Piping

Temporary lead shielding was added to the piping from the Volume Control Tank (VCT) to the Charging Pump suction piping and to the piping from the Seal Water Heat Exchanger to VCT outlet piping in support of application to decontamination coatings in Room 1318. An evaluation was performed for installation of this shielding in Mode 1 while the system remains in service. In order to maintain system operability and pressure boundary integrity during a seismic event, certain conditions were imposed. These conditions include:

1. Installation and removal of shielding was controlled by administrative procedure.

2. All lead shielding was removed after application of decontamination coatings.

3. Maximum allowable shielding weight was no more than 10 pounds per foot of piping.

4. Sections of piping were designated to avoid the addition of lead shielding where necessary.

The installation of temporary shielding has been evaluated and installation requirements have been established to ensure that no equipment important to safety will be compromised.

Safety Evaluation: 92-0103 Revision: 0

Temporary Installation of Water Treatment Equipment in the Circulating Water Screenhouse

Temporary installation of water treatment equipment was provided in the Circulating Water Screenhouse due to work being done on the system by another plant modification. This temporary installation was used to provide circulating and service water chlorination capability. A vendor skid was connected to the Service Water System by removal of a spool piece and adding piping and 2" petroleum hose. This change was performed in the Circulating Water Screenhouse.

The equipment affected by this temporary change is non-nuclear safety related and non-seismic. The chemical used poses no more of a health hazard than chemicals already in use or stored at the plant.

Safety Evaluation: 92-0104 Revision: 0

Temporary Installation of Water Treatment Equipment in the Circulating Water Screenhouse

Temporary installation of water treatment equipment was provided in the Circulating Water Screenhouse due to work being done on the system by another plant modification. This temporary installation was used to provide service water anti-scale chemical feed capability. Additional tubing was routed from the anti-scale facility to the Service Water System. This change was performed in the Circulating Water Screenhouse.

The equipment affected by this temporary change is non-nuclear safety related and non-seismic. The chemical used poses no more of a health hazard than chemicals already in use or stored at the plant.

Safety Evaluation: 92-0105 Revision: 0

Organizational Changes at Wolf Creek Generating Station

Several organizational changes were made at Wolf Creek Generating Station (WCGS) that included title changes, changes in reporting relationships, personnel changes, etc. These changes improve the operating philosophy of WCGS and do not adversely affect the safe and reliable operation of the plant. The specific changes were:

1. Assignment of the Director Quality and Director Nuclear Plant Engineering to the Performance Enhancement Program Team and elimination of the position Director Nuclear Plant Engineering.

2. Appointment of an Acting Director Quality.

3. Change of title from Vice President Engineering & Technical Services to Vice President Engineering.

4. Reassignment of Nuclear Safety Analysis to the Vice President Engineering and definition of the reporting chain for managers previously reporting to the Director Nuclear Plant Engineering.

5. Creation of the Nuclear Assurance Department, including assignment of a Vice President and definition of this organization. Eliminated the position,Updated Safety Analysis Report (USAR) Director Nuclear Services.

6. Eliminated the position, Director of Administrative Services and definition of the Human Resources reporting chain.

7. Licensing and Compliance functions have been combined under Regulatory Services.

8. Emergency Planning has been consolidated under Technical Services.

Safety Evaluation: 92-0106 Revision: 0

Domestic Potable Water and Makeup Water Demineralizer System Modifications

This modification installed the necessary instrumentation and equipment to the Potable Water System and Makeup Water Demineralizer System to comply with the 1993 Surface Water Treatment Rule and Disinfection By-Product Rule. These changes to add instrumentation, equipment and piping do not affect the ability of any basic component to perform its intended design basis function.

The systems affected are non-nuclear safety related and non-seismic. The equipment is located in the Shop Building.

Safety Evaluation: 92-0107 Revision: 0

Temporary Procedure for Reactor Coolant Pump Component Cooling Water Motor Operated Valve Differential Pressure Testing

This temporary procedure provides instruction for differential pressure testing of the Reactor Coolant Pump Thermal Barrier Component Cooling Water (CCW) return isolation valves in response to Generic Letter 89-10. This test requires these motor-operated valves (MOV) to be stroked open and closed against normal CCW pressure and flow, and is restricted to plant modes 5, 6, or no mode.

The temporary presser gauges used for this test will be connected to permanent plant indicator manifold connections using flexible tubing and fittings rated at 200 psig. These connections are capable of retaining CCW pressure and do not affect the system during a seismic event because the flexible connections impose no additional loading on system piping.

During Modes 5, 6, and No Mode, the only significant essential CCW load is Spent Fuel Pool Cooling. If CCW flow to the Spent Fuel Pool Heat Exchanger is adjusted during the test the Spent Fuel Pool Temperature will be monitored. All CCW loads will be isolatable during the test and the isolation valves at the test connections will be closed except when taking pressure readings. Since the test will briefly interrupt Reactor Coolant Pump (RCP) Thermal Barrier Cooling the associated RCP will either be secured or supported with seal injection as specified in the Updated Safety Analysis Report (USAR).

Safety Evaluation: 92-0109 Revision: 0

Add Seismic Qualification Report on 4.16 KV Switchgear to Vendor Documentation

Additional information was added to the vendor documentation on the 4.16 kv switchgear to address seismic qualification. The switchgear was evaluated for additional configurations from that originally evaluated and found to be acceptable for operation with the breaker in any of three different positions. The breaker was also evaluated for rolling, tipping, etc.

The switchgear cubicles were not modified, the additional documentation provided justification for operation in an orientation different from that previously analyzed.

Safety Evaluation: 92-0110 Revision: 0

Procedure to Differential Pressure Test Residual Heat Removal (RHR) Motor Operated Valves

This test procedure was developed to determine the ability of motor operated valves for the RHR inlet and Reactor Coolant System (RCS) hot leg to RHR pump suction to perform their design safety function. The test can only be performed in Mode 5, with the reactor coolant loops filled and a bubble in the pressurizer. The RHR train not being tested will be in operation. Reactor coolant pump 'D' will be in operation to assure pressurizer spray. The pressurizer spray and heaters will control RCS pressure. Letdown and cleanup through the letdown heat exchanger will be unavailable for a short duration during this test. Excess letdown will be available, if required during the test. The RCS pressure boundary will be extended by the use of temporary pressure gauges. The gauges will be valved out unless they are in active use.

Upon completion of the test, the systems will be restored to a normal lineup and condition as required by the existing plant conditions. The systems/components will not be subjected to conditions beyond their design capability and will be unaffected except for the increased service due to manipulation.

Safety Evaluation: 92-0111 Revision: 0

Performance of Fuel Assembly Inspection and Partial Disassembly

A fuel assembly inspection was performed as part of the root cause fuel rod failure investigation. The inspection plan was developed to provide a detailed guide for fuel assembly partial disassembly and inspection. Inspection activities included the following:

1. TV examination of selected fuel assemblies using the XYZ visual inspection fixture which incorporates a high magnification camera.

2. Remove and reinstall the fuel assembly bottom nozzle using the Multi-purpose Fuel Repair System (MFRS) for inspection of the bottom grid using a TV camera.

3. Cut off and remove the fuel assembly' top nozzle in the MFRS for inspection of thimble tubes and measurement of selected fuel rod breakaway forces using the Vantage 5 Motorized Fuel Rod Handling Tool (V5 MFRHT).

4. Transfer selected rods from certain fuel assemblies to the failed rod basket in the Spent Fuel Pool using the V5 MFRHT.

5. Remove rods from the failed rod basket using the motorized fuel handling tool (FRHT) for profile measurement and visual inspection using the profilometer and rod visual fixture.

6. Install a new top nozzle and perform bulge technique on thimble tubes to secure the top nozzle to the affected fuel assemblies.

7. Remove the bottom nozzle and perform fiberscope inspection and pin gauge measurements on all locations where rods were removed.

8. Insert stainless steel pins in selected empty positions and then install bottom nozzle.

9. If necessary, install submerged Foreign Object Search and Retrieval (FOSAR) system for retrieval of dropped or missing parts.

An evaluation of the seismic considerations, heavy loads concern, criticality concern, and cooling was performed to ensure safety.

Safety Evaluation: 92-0112 Revision: 0

Procedure to Test RCP Seal Water Return and Excess Letdown Isolation Motor Operated Valves

This test procedure will be used to determine the ability of the Reactor Coolant Pump (RCP) seal water return and excess letdown isolation valves to perform their design safety function. This test can be performed only in Modes 5, 6 or when fuel is not in the reactor vessel. A test boundary will be established around these valves and flow/pressure is established in the normal flow/pressure direction using water from the reactor make-up water system. The water will be rejected to Auxiliary Building drains. The subject valves will then be individually operated in both the open and closed direction while flow, temperature, and pressure as well as actuator performance are monitored using temporary instrumentation. The affected systems will remain capable of performing their safety functions.

Upon completion of the test, the system will be restored to a normal lineup and condition as required by the existing plant conditions. The system/components are not subjected to conditions beyond their design capability and will be unaffected except for the increased service due to manipulation.

Safety Evaluation: 92-0113 Revision: 0

Administrative Procedure Change to the Exempt Temporary Equipment List

The administrative procedure on "Control of Temporary Equipment" was revised to add two items to the Exempt Temporary Equipment List. The two items added are the use of Emergency Eye Wash Stations and the use of plastic/Tygon tubing to direct seal or packing leakoff water to associated drains.

The Emergency Eye Wash Stations addressed are units that have capacities of 5, 10 and 37 gallons (holding 4, 7.5 and 30 gallons of water, respectively). The two smaller stations, 5 and 10 gallon units, have a low center of gravity and can be favorably compared to equipment already on the exempt list (i.e., portable air samplers, small portable test equipment, and trash cans when full). In order for the 37 gallon unit to be exempt, the procedure requires the Station to be secured to the cart it is transported on and the cart to be supplied with a tie off chain or strap. The tie offs will be used to secure the Station to a non-nuclear safety component/structure capable of keeping the Station from falling or getting away if a seismic event were to occur.

The plastic/Tygon tubing will be used to direct packing or seal leakoff water from skid units to the nearest drain. The tubing will be attached to an existing drain and will not be routed over active safety related components. The leakoff water is typically condensate or demineralizer water at atmospheric pressure. Without the use of the proposed drain tube the water leaks down over the equipment creating house keeping and corrosion problems.

The procedure changes to the Exempt Temporary Equipment List will not pose a safety hazard to surrounding equipment, when used as described by the procedure.

Safety Evaluation: 92-0116 Revision: 0

Test Procedure for the Main Steam Dump Valves

A test procedure was developed to perform the steam dump in service valve test. The test is described in the Updated Safety Analysis Report (USAR) with the exception of the method used to actuate the valve by electrically jumpering the actuating solenoids rather than using the controls in the Control Room. Any affect on reactor control is eliminated by closing the downstream isolation valve as described in the USAR thereby preventing any flow from occurring.

The main steam dump valves have no safety function, are non-nuclear safety related, and no credit is taken for their operation in the maintenance of nuclear safety. Adequate protection against loss of heat removal capability is maintained by the atmospheric relief valves and steam generator safety valves.

Safety Evaluation: 92-0119 Revision: 0

Changes Due to Implementation of Offsite Dose Calculation Manual

Changes were made to the Updated Safety Analysis Report (USAR) based upon replacement of the Radiological Effluent Technical Specifications with the Offsite Dose Calculation Manual (ODCM). Certain USAR sections were revised to properly reflect the relocation of information that was in the technical specifications to the ODCM. The high alarm and alert alarm setpoints for certain monitors were reversed in a USAR table and this was corrected and other clarifications were made.

The USAR updating to reflect approval of the ODCM, correcting setpoint values and administrative discrepancies for radiation monitors will not change the original design function or cause systems to be operated outside their design limits.

Safety Evaluation: 92-0120 Revision: 0

Revise Drawings on Various Security Fences and Gates to Reflect As-built Information

An Updated Safety Analysis Report (USAR) figure and various drawings were revised to accurately reflect the as-built configuration of the plant and provide consistency in the numbering of gates. The changes are related to the security fences and gates only. This revision is a document change only and does not cause physical changes to hardware in the field.

Safety Evaluation: 92-0121 Revision: 0

Auxiliary Feedwater Pump Turbine Vibration Probe Removal

This modification removes Auxiliary Feedwater (AFW) pump turbine vibration probes which were supplied with the original equipment, but never used, and replaces them with steel plugs.

The intended function of the vibration monitors being replaced is performed by portable equipment. Removing these unused monitors will improve accessibility for inspections and maintenance. The location, and weight of the replacement plugs is such that there is no effect on the seismic characteristics and the material used is compatible with the lubricating oil used in the AFW turbine.

Safety Evaluation: 92-0123 Revision: 0

Centrifugal Charging Pump Miniflow, Motor-Operated Valve Differential Pressure Test

This temporary procedure governs the differential pressure testing of the Centrifugal Charging Pump (CCP) miniflow, motor-operated valve to satisfy Generic Letter 89-10 requirements.

Performance of this temporary procedure will increase the pressure in the Chemical Volume Control System (CVCS) and cause lower flow through the CCP due to the isolation at the recirculation line. The associated pressure increase for the CVCS is within the design limits of the system and the CCP is designed to operate at this low flow for a period not to exceed 30 minutes. A standing order is in place to prevent exceeding the 30 minute time limit and each train will be tested independently with the appropriate Limiting Condition of Operation entered for one train inoperable during the test.

Safety Evaluation: 92-0124 Revision: 0

Differential Pressure Testing of the Charging Line Isolation Motor Operated Valves

This temporary procedure governs the differential pressure testing of the charging line isolation motor operated valves to satisfy Generic Letter 89-10 requirements.

Performance of this test is limited to mode 6 or when there is no fuel in the reactor vessel. If the test is performed at a time when containment closure is required, individuals will be stationed at the appropriate valves and maintain communications with the control room. This test will cause a pressure increase in the system which is limited by the shutoff head of the Centrifugal Charging Pump (CCP). The CCP will operate at less than minimum flow for a limited time during this test. There is a standing order in place which provides instructions for low flow operations.

All temporary instruments will be connected using flexible tubing and isolated, except when measurements are being taken, to maintain the seismic integrity of the system.

Safety Evaluation: 92-0125 Revision: 0

Temporary Installation of Pressure Gauge Downstream of Sample Connection for Reactor Coolant Drain Tank Vent Line

This temporary modification installed a temporary pressure gauge downstream of the sample connection for the Reactor Coolant Drain Tank (RCDT) vent line. This instrumentation was needed to allow monitoring of the RCDT pressure while an inaccuracy problem with the permanent RCDT pressure transmitter was being corrected. The pressure gauge is part of the process local sample line. The line is intended for providing Hydrogen samples from the RCDT that may be vented to the gaseous radwaste system.

The weight of the gauge is not significant and will not impact the seismic qualification of the associated piping. The level controller is not affected. Pressure rating, accuracy, interfaces, compatibility and weight were all considered in the selection of the temporary gauge. Operations will verify pressure once per shift to ensure no drastic changes in pressure occur. Failure of the pressure gauge has no impact on the probability of Radioactive Liquid Waste System Failure as discussed in the Updated Safety Analysis Report (USAR) 15.7.2. Failure of the gauge is not anticipated. However, mechanical or structural failure will not interfere with the mitigation of any analyzed accidents. This temporary change will have no effect on containment isolation valves. This gauge will not introduce any new failure scenarios that are not bounded by the existing analyses. The pressure gauge does not interfere physically or functionally, with any equipment important to safety.

Safety Evaluation: 92-0126 Revision: 0

Essential Service Water Flow Verification for Component Cooling Water Heat Exchanger

This temporary procedure aligns the Essential Service Water (ESW) System to simulate emergency flow conditions so that component cooling heat exchanger flow can be verified.

The alignment of the ESW system will be accomplished in accordance with existing, approved, plant procedures. Flow through the heat exchangers will be greater than normal during the test which will prevent technical specification containment temperature limits from being exceeded. During the test, personnel will be stationed to restore the system to its normal lineup if directed to do so by the Control Room. ESW Pump parameters will be maintained within high and low flow design conditions throughout the test which assures that flow and pressure conditions remain within design limits.

Safety Evaluation: 92-0128 Revision: 0

Updated Safety Analysis Report Changes to Emergency Diesel Generator Description

Changes were made to the Updated Safety Analysis Report (USAR) to correct a typographical error and correct the functional description of the Emergency Diesel Generator (EDG) jacket water coolant temperature switches. The functional description change describes the four temperature setpoints as one is at a higher setpoint and the other two at the highest temperature setting. The previous description stated that all four were at increasing setpoints. Also, the engine shutdown caused by the temperature switches now states that any two (one of which must be at the highest temperature setpoint) will cause this to occur. The previous description stated that any two would cause engine shutdown.

This change to the USAR for the EDG jacket water coolant temperature switches reflect the existing design and configuration. The basic function is not changed, there are still four jacket water temperature switches and two out of four will trip the engine.

Safety Evaluation: 92-0130 Revision: 0

Temporary Procedure to Attach a Pressure Transmitter to Equipment Vent and Drain System

Two pressure transmitters were temporarily added to the Equipment Vent and Drain System to trend pressure data. A temporary procedure was developed to add these transmitters to the non-nuclear safety related system. This temporary change was done to monitor equipment operating parameters as part of a hardware failure analysis.

No impact was made on safety related equipment due to the remote location of the transmitters on the non-nuclear safety related system.

Safety Evaluation: 92-0131 Revision: 0

Biofouling/Erosion/Corrosion Monitoring Equipment Changes

This modification replaces obsolete sample pumps; and reroutes piping and wiring to support the operation of corrosion, deposit, and biofoul monitors used on the Service Water and Circulating Water Systems. This modification allows monitoring of these systems in accordance with the requirements of Generic Letter 89-13. The Updated Safety Analysis Report is being revised to show new piping, wiring and equipment locations.

This modification affects non-safety related equipment which is located in the turbine building. The added piping is small-bore piping containing low energy cooling water. Therefore, any pipe break would not be a personnel safety hazard nor contribute significantly to the turbine building flooding previously evaluated. There are no technical specifications related to non-safety related piping.

Safety Evaluation: 92-0139 Revision: 0

Licensed Operator Requalification Examination Process Changes

This change to sections 13.2.1.2.8 and 13.2.1.2.9 of the Updated Safety Analysis Report (USAR) modifies the licensed operator requalification examination process to match draft NUREG-1021, Rev. 7, "NRC Examiner Standards." This change reflects the current regulatory guidance for operator examinations.

This change does not effect any plant equipment or the knowledge requirements of the licensed operators being examined. The revised testing failure criteria described in draft NUREG-1021, Rev. 7 continues to prove the ability and knowledge of the licensed operators being examined.

Safety Evaluation: 92-0141 Revision: 0

Revision to "As Low As Reasonably Achievable" (ALARA) Committee Chairman Requirements

This change to section 12.1 of the Updated Safety Analysis Report (USAR) allows the chairman of the "As Low As Reasonably Achievable" (ALARA) committee to be any member of management rather than specifying that the Manger Radiological Services will hold the position.

This change resulted from organization changes which had no effect on the operating philosophy of Wolf Creek Generating Station and does not represent any decrease in concern for the health and safety of the public. This change had no effect on any systems, components, or procedures used to operate the plant and therefore has no effect on accidents which have been evaluated in the USAR.

Safety Evaluation: 92-0144 Revision: 0

Update Electrical Schematic and P&ID to Resolve Drawing Discrepancies

These drawing modifications delete references to two temperature switch interlocks which were inadvertently left on the drawing after an earlier design change and drawing revision. It also adds a reference to the associated status panel drawings which had been inadvertently deleted.

This modification makes no changes to any plant equipment or procedures. It is a drawing change only to an electrical schematic and P&ID to reflect the as-designed and as-built co..figuration of the Control Room Pressurization System.

Safety Evaluation: 92-0145 Revision: 0

Thermal Expansion Monitoring Procedure

This procedure provides for monitoring Reactor Coolant System (RCS) crossover leg pipe movement and reactor coolant pump tie rod and crossover leg vertical drop restraint tie rod gap clearances. Safety Evaluation 92-0037 concerning the installation of temporary motion detectors and evaluation 92-0038 concerning a temporary procedure for monitoring component support thermal expansion in the Containment Building during plant heatup are associated with the subject safety evaluation.

The procedure provides for the installation of temporary measuring equipment for monitoring RCS movement during plant heatup. The installation of this equipment is installed in Modes 3 and below and must be removed prior to entry into Mode 2. Installation of this equipment will not meet the minimum separation criteria of E-11013 and Regulatory Guide 1.75. The amount of heat that could result from the combustion of these low voltage instrument cables is inconsequential to any safety-related components or conduits and cables.

Safety Evaluation: 92-0147 Revision: 0

New Component Numbers for Reactor Coolant Pump Motor Vibration Monitors

Reactor Coolant Pump Vibration Monitors were originally configured as a single instrument. However, each monitor is actually two individual monitors. This modification provides configuration identification to show both a shaft monitor and a frame monitor.

There have been no hardware changes associated to this modification. This is a document change only. The vibration monitoring system for the Reactor Coolant Pumps is non-nuclear safety related.

Safety Evaluation: 92-0150 Revision: 0

Addition of a Hose from the Recycle Hold Up Tanks to the Spent Fuel Pool

A hose was added to the end of a pipe from the Recycle Hold Up Tanks in the Boron Recycle System so that borated water can be transferred to the Spent Fuel Pool (SFP). The hose will remain in place due to radiological considerations with removal and replacement. Removal may occur whenever the SFP Bridge Crane is required to operate in the area of the hose. The hose is secured in at least two places independently by rope to restrain and restrict the open end of the hose feeding the pool.

An evaluation of the hose has been performed to adequately address concerns with heavy loads, deborating the SFP, cooling capability, siphoning, and seismic concerns.

Safety Evaluation: 92-0150 Revision: 0

Addition of a Hose from the Recycle Hold Up Tanks to the Spent Fuel Pool

A hose was added to the end of a pipe from the Recycle Hold Up Tanks in the Boron Recycle System so that borated water can be transferred to the Spent Fuel Pool (SFP). The hose will remain in place due to radiological considerations with removal and replacement. Removal may occur whenever the SFP Bridge Crane is required to operate in the area of the hose. The hose is secured in at least two places independently by rope to restrain and restrict the open end of the hose feeding the pool.

An evaluation of the hose has been performed to adequately address concerns with heavy loads, deborating the SFP, cooling capability, siphoning, and seismic concerns.

Safety Evaluation: 92-0155 Revision: 0

Drawing Discrepancies on Valve Configurations and Trouble Alarm Circuit Configurations

Six valves were incorrectly identified as locked closed on Piping and Instrument Diagrams (P&ID's). The drawings were revised to remove the locked closed designation. The valves are located in the Lube Oil Storage, Transfer and Purification System. The valves basic function is to isolate alternative ways of supplying lube oil to different equipment in case of failure of the primary system. Status indication is available to determine the correct pressure and flow of lube oil to the respective equipment.

The Rod Drive M-G Set trouble alarm circuitry was shown as having different configurations on separate drawings. The Rod Drive M-G Sets supply power to the reactor control rod drives. The incorrect configuration was revised on two drawings to show the proper configuration. This change will have no impact on the equipment in the field which is correctly configured.

Both of these items involve non-nuclear safety related system drawing changes only. No physical plant modifications resulted from these changes.

Safety Evaluation: 92-0156 Revision: 0

Temporary Procedure to Determine Water Level in the Positive Displacement Pump Suction Pulsation Dampener

In order to assess the potential for gas binding a temporary procedure was developed to determine the water level in the Positive Displacement Pump suction pulsation dampener. The procedure will take the measurements as unobtrusively as is reasonable. While the valves are open, an individual will be available to restore the system to design condition should conditions make it necessary. The only equipment important to safety involved in this test is the piping pressure boundary. The operated valves are open only long enough to obtain the data and complete the venting.

Safety Evaluation: 92-0157 Revision: 0

Temporary Procedure to Verify Internal Temperatures on the Control Room Pressurization Filter Adsorber Units

A temporary procedure was written to support troubleshooting of the heaters for the Control Room Pressurization Filter Units. During the test, the heaters will be powered by a non-Class 1E power source and declared inoperable. Heater failure would cause the filter unit to be less efficient due to high humidity which degrades the carbon bed. Upon completion of the test the units would be reconnected to the Class 1E power source and verified operable.

The performance of this test will not affect any safety related component that is associated with accident initiation. The Control Room Ventilation System is used to mitigate accident consequences.

Safety Evaluation: 92-0164 Revision: 0

Installation of Freeze Seal for Repair of Valve ECV0018

This temporary modification provides for the installation of a freeze seal between valve ECV0018 and the spent fuel pool for the repair of valve ECV0018. The Fuel Pool Cooling System is designed to maintain the spent fuel pool water temperature below prescribed limits by removing decay heat generated by stored spent fuel assemblies. The Fuel Pool Cooling System is comprised of two redundant cooling trains, each of which have the capacity to provide 100 percent residual heat removal. It is acceptable to isolate one train by freeze sealing while the other fuel pool cooling train is in operation.

Normal water level of the spent fuel pool is Elevation 2046'-0:. The top of the spent assemblies in the pool is Elevation 2020"-8.3". The spent fuel assemblies will be kept under a specific depth of water to avoid possible overheating and damaging of the cladding. The spent fuel pool water is normally supplied by the Reactor Water Makeup System. An alternate source of makeup water to the spent fuel pool is the Refueling Water Storage Tank. The failure of the freeze seal on one train of cooling water will not compromise the safety function of the safety function of the redundant train of fuel pool cooling water. Additionally, an antisiphon hole is located in each return line, near the surface of the spent fuel pool. This antisiphon hole will prevent any possible draining of the spent fuel pool due to freeze seal failure. Therefore, freeze seal failure will have no affect on the normal plant operation or safe shutdown of the plant.

Safety Evaluation: 92-0166 Revision: 0

Clarification of Time to Perform Chlorine Analysis

This Updated Safety Analysis Report (USAR) change request involves a clarification to commitments associated to chloride analysis as required by NUREG-0737, "Clarification of TMI Action Plant Requirements" and Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

USAR Section 18.2.3 on post-accident sampling capability provides the NUREG-0737 position and clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13 and October 30, 1979 clarification letters. Item (5) states that "The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is not seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days.

Since Wolf Creek Generating Station (WCGS) coolant water is not seawater or brackish water, the chloride analysis can be accomplished in 4 days instead of 24 hours. Correcting the time for a chloride analysis to be performed will not change the original design function of system or cause systems to be operated outside their design limits. This USAR change will not modify, degrade, or prevent actions described or assumed in the accident analysis; nor does it alter any assumptions previously made in evaluating the radiological consequences of an accident.

Safety Evaluation: 92-0168 Revision: 0

Revision of Ultimate Heat Sink Volume of Sediment for Consistency

The Updated Safety Analysis Report (USAR) was revised to provide consistency between sections and to reflect the current analysis on the volume of sediment within the Ultimate Heat Sink (UHS). The current analysis is based upon 155 acre-feet. Two sections of the USAR had different values for the sediment buildup in the UHS and this value will be used to resolve the discrepancy. In addition, minor revisions of heat rejection loads and integrated totals were made to be consistent with the data presented in plant calculations for the current 100% power licensing basis.

These changes are documentation changes only and do not cause changes to the actual plant equipment.

Safety Evaluation: 92-0169 Revision: 0

Temporary Procedure to Differential Pressure Test Motor Operated Valve

A new temporary test procedure was developed to perform a differential pressure test on the Residual Heat Removal (RHR) pump hot leg injection motor operated valve per NRC Generic Letter 89-10. Performance of this test will isolate cold leg injection for one train of the RHR System. This will isolate for a short period of time all discharge flow paths of the train selected for the test. The RHR pump will then automatically enter recirculation mode. The test then opens and closes hot leg injection to the Reactor Coolant System. No abnormal pressure or flow transients will exist and the RHR System will remain within design limits during the performance of this test.

Performance of this test is restricted to plant conditions that allow one train of the RHR System to be out of service. This test will only affect the train of the RHR System that is out of service.

Safety Evaluation: 92-0170 Revision: 0

Replacement of Auxiliary Steam Feedwater Pumps (PFB03A/B) with New Model

The Auxiliary Steam Feedwater Pumps are being replaced due to obsolescence and spare parts procurement problems. Mounting and piping configurations must be adapted to the new pumps. The original motors will remain, only the pumps are affected. This modification also adds temporary instrumentation to monitor the pumps during normal operation and removes orifices to accommodate new pump minimum flow requirements.

The Auxiliary Steam System serves no safety function. Failure of this system will not compromise a safe shutdown of the reactor. The design function of the system is to provide steam required for plant heating and operation of the recycle evaporator system and waste evaporators. The new equipment satisfies the original design basis thermal and hydraulic performance parameters. The design basis is not being changed. The Auxiliary Steam System and Auxiliary Steam Feedwater Pumps are not involved with the initiation or mitigation of any accidents. The system and pumps are non-safety related. The pumps do not interact with any equipment important to safety during normal or accident plant operations. The Auxiliary Steam System and its components are not used in defining the margin of safety for equipment referenced in the Technical Specifications.

Safety Evaluation: 92-0171 Revision: 0

Temporary Procedure to Differential Pressure Test Motor Operated Valve

A new temporary test procedure was developed to perform a differential pressure test on the Reactor Coolant System (RCS) cold leg injection motor operated valve per NRC Generic Letter 89-10. The valve will be differential pressure tested in the open and closed positions against "A" Residual Heat Removal (RHR) pump pressure and flow. "A" train RHR will be declared inoperable during this test. No abnormal pressure or flow transients will exist and the RHR System will remain within design limits during the performance of this test. The temporary gauges installed upstream and downstream of the valve being tested will not be left unattended when they are valved in, therefore failure of connections/flex hose will not go undetected.

Performance of this test is restricted to plant conditions that allow one train of the RHR System to be out of service. This test will only affect the train of the RHR System that is out of service.

Safety Evaluation: 92-0172 Revision: 0

Revise Miscellaneous Drawings to Reflect As-Built Conditions and Correct Deficiencies

Several drawings were revised to correctly reflect as-built conditions or to correct discrepancies with other drawings that did not reflect as-built conditions. In particular, the drawing revisions included:

- Revise drawing E-13RJ09 to depict breaker PG12KCF1 as the alternate feed for PQ003.

- Revise drawing M-6[illegible]00142 to reflect as-built pre-action sprinkler system above the battery room on elevation 2033'-0" of the turbine building.

- Revise drawing M-12AC02 to show the output of ACPT0116 going to AC119 as shown on the schematic.

- Revise drawing M-13BB02 to show the Residual Heat Removal suction relief valve, EJ8708B, line drain valve as BBV0297 instead of BBV0279.

The drawing changes reflect the actual configuration in the plant and do not cause physical hardware changes. Therefore, no adverse impact on nuclear safety related equipment will occur.

Safety Evaluation: 92-0173 Revision: 0

Replacement of a Backflow Preventer in the Domestic Water System

A new replacement backflow preventer was added to the Domestic Water System because the existing unit was obsolete and could not be maintained. The new unit meets the standards of the existing unit, but certain piping modifications were made due to different size, weight and drain capabilities. The drain line was modified to add a funnel, union, and pipe support.

These changes maintain the capability of the previous backflow preventer on this non-nuclear safety related system.

Safety Evaluation: 92-0176 Revision: 0

Replacement of Auxiliary Steam Feedwater Pumps

See discussion associated with Safety Evaluation 92-0170.

Safety Evaluation: 92-0177 Revision: 0

Incorporation of As-Built Information Associated with the Waste Water Treatment Facility

This general modification provides as-built information for drawing incorporation associated with the Waste Water Treatment Facility. These drawing changes reflect the installation of piping and instrumentation or the deletion of instrumentation and valves not required or needed.

There are no design basis accidents evaluated in the Updated Safety Analysis Report (USAR) for the non-safety related Waste Water Treatment System. No system function or failure modes are affected by the design changes.

Safety Evaluation: 92-0178 Revision: 0

Clarification of Existing Emergency Diesel Generator Loads on Load List

The Emergency Diesel Generator load list currently depicts only load totals for load group 1. This format is based on the fact that load group 1 is the worst case or highest load group and therefore, envelopes load group 2. However, as loads are added to a single load group over time, configuration control may become confusing with the current format. Therefore, showing applicable load totals for both load groups concurrently on the load list would simplify configuration control. This modification is a document change only. There is no increase or decrease to the connected loads to either Emergency Diesel Generator.

Safety Evaluation: 92-0179 Revision: 0

Operational Design Basis Flow Through the Chemical Volume Control System (CVCS) Letdown Line and the Postulated Radiological Consequences from CVCS Letdown Line Rupture

The letdown flow is normally maintained at 120 gpm during plant power operation. This flow is conservatively calculated to increase to 141 gpm if a Chemical Volume Control System (CVCS) letdown line rupture is assumed to occur on the down stream side of the orifices. Updated Safety Analysis Report (USAR) Section 15.6.2 is being revised to reflect the operational design basis flow through the CVCS letdown line and the postulated radiological consequences that result from the assumed worst case CVCS letdown line rupture.

This USAR change has no effect on mechanisms postulated in the USAR to cause design basis events. The postulated radiological consequences resulting from an assumed worst case CVCS letdown line rupture with a calculated maximum break flow of 141 gpm remain well within 10 CFR 100 limits. The evaluation provides the basis for the conclusion that the impact of increased break flow on jet impingement, internal flooding, and subcompartment pressurization is insignificant. Temperatures are shown to increase, however, all the components affected are qualified to higher accident temperatures. Therefore, conclusions drawn from previous equipment qualification evaluations remain valid. The proposed change does not change assumptions addressing the malfunction of equipment important to safety. There are no physical modifications or changes in methods of operation associated with this change.

Safety Evaluation: 92-0182 Revision: 0

Temporary Jumper to Defeat High Discharge Air Temperature Trip on Compressed Air System

Due to spurious trips of the air compressor, a temporary modification was made to install a jumper on the automatic trip due to high discharge air temperature for the compressed air system. This modification will improve the reliability of this non-nuclear safety related system and the remaining trip functions and indications will ensure equipment protection is maintained.

This temporary modification will not adversely affect any nuclear safety related equipment or accident analysis initiators.

Safety Evaluation: 92-0184 Revision: 0

Procedure Change to Reflect Organizational Title Changes and Form Changes

The procedure titled "Conditional Release" was revised to reflect organizational title changes and to allow warehouse nonconformances to be documented and resolved on a Commodity Discrepancy Report, which has the same controls as a Nonconformance Report. These procedure changes will not affect installed hardware in the plant.

These administrative controls and title changes will not adversely affect any nuclear safety related equipment.

Safety Evaluation: 92-0186 Revision: 0

Updated Safety Analysis Report (USAR) Changes to Reflect Organizational Changes in the Quality Department

The position of Director Quality was deleted from the Wolf Creek Generating Station organization. This caused duties and responsibilities for this position to be assigned to the applicable Quality organization. The Updated Safety Analysis Report (USAR) was revised to reflect these changes. Also, the words "Quality Department" were changed to either "applicable Quality Organization" or to a specific manager.

These changes do not reduce the QA Program implementation from that which is approved and does not reduce any commitments, since all the activities are still being performed and no program changes were made.

Safety Evaluation: 92-0188 **Revision:** 0

Revision of Flood Levels in Rooms 1411 and 1412

An Updated Safety Analysis Report (USAR) change was made to correctly reflect the flood level in Rooms 1411 and 1412. These rooms are located in the Auxiliary Building and the level was increased from 1'-4" to 1'-9". The drain piping from area five of the Auxiliary Building provides a means for removing the water. The revised flood level is still well below the design level and does not affect any safety related equipment.

The original design basis flood level has not been exceeded by this change, so no adverse impact on the accident analysis will occur.

Safety Evaluation: 92-0190 Revision: 0

Update Functional Diagrams to Resolve Drawing Discrepancies

A revision was made to the functional diagram for the pressurizer trip signals to depict the PORV block valves as closing on decreasing pressurizer pressure. Also, labeling for three circuits for the Circulating Water Pump discharge valve motor heaters and compartment heaters was added. These changes were made to resolve drawing discrepancies.

The changes made do not affect actual plant equipment and therefore do not adversely affect nuclear safety related equipment.

Safety Evaluation: 92-0191 Revision: 0

Revise Updated Safety Analysis Report to Reflect Organizational Change

The Manager Maintenance and Modifications was changed from Mr. R. Holloway to Mr. C. Fowler. This change required updating the resume' in the Updated Safety Analysis Report (USAR). This personnel change maintains the previous scope and commitment to the affected position.

The personnel change does not affect the overall operating philosophy or capabilities of the Wolf Creek Generating Station organization.

Safety Evaluation: 92-0193 Revision: 0

Procedure Change to Jumper Alternator for the Potable Water Pumps

This procedure change to SYS WD-120, "Shop Building Domestic Water System" allows the alternator to be jumpered on the potable water pumps when out of service. This change is necessary to prevent air binding when only one potable water pump is in service. The modification does not affect the system function. Failure modes have not changed. This is a non-safety system.

Safety Evaluation: 92-0195 Revision: 0

Torque Switch Setting Adjustment

This modification changes the torque switch settings for the Residual Heat Removal System cross connect motor operated valves. The Updated Safety Analysis Report (USAR) is also revised to reflect the actual design basis. These changes are being made in accordance with the requirements of Generic Letter 89-10.

This modification incorporates new design requirements developed in response to Generic Letter 89-10. Both the modification and the corresponding USAR Revision are in accordance with the design basis specification and do not affect the design basis function of the components. The performance of these components remarks consistent with all assumptions made previous accident analysis.

Safety Evaluation: 92-0198 Revision: 0

Updated Safety Evaluation Report (USAR) Table 3.2-1, Section 6.2 and Chapter 9.5 Revision

This modification revises the Updated Safety Analysis Report (USAR) to reflect the appropriate classification of emergency diesel components. This classification was originally provided and evaluated in the safety evaluation report (NUREG-0830) but was not incorporated into the USAR at the time.

This USAR change is a document change only. The change reflects information which has been previously evaluated and approved and has no effect on any equipment operation or procedures.

Safety Evaluation: 92-0202 Revision: 0

Circulating Water Valve Position Determination

This temporary procedure determines the valve throttling requirements for the operation of two circulating water pumps (CWP). This determines the valve position required to avoid cavitation, maintain waterbox level, avoid condenser sediment buildup, and operate the CWPs within the manufacturers requirements.

This procedure will throttle the circulating water valves to reduce system flowrate and prevent pump runout. This is a non-safety related system and has no interface with any safety-related equipment. Performance of this procedure will cause a reduction in condenser vacuum. The procedure provides for monitoring and compensatory actions to prevent a turbine trip on a loss of condenser vacuum. A turbine trip is a moderate frequency event which has been evaluated in the Updated Safety Analysis Report (USAR).

Safety Evaluation: 92-0213 Revision: 0

Installation of Temporary Drain Hose and Pressure Gage

This temporary modification installs tubing to drain leakage past an Auxiliary Feedwater System valve to a floor drain and installs a pressure gage to monitor the pressure downstream of the leak to give any indication of any pressure buildup which may get past a check valve and contaminate the secondary side water with essential service water. (lake water)

The leaking valve is required to open on command to provide emergency Essential Service Water (ESW) to the Auxiliary Feedwater Pumps in the event the Condensate Storage Tank is no longer available and to act as a pressure boundary for the ESW system. The current leak rate of this valve when added to other unmitigated system leaks remains well within limits required for Ultimate Heat Sink inventory calculations for safe shutdown and the valve can continue to perform its safety functions with the temporary modifications in place.

Safety Evaluation: 92-0215 Revision: 0

Isolation of Auxiliary Feedwater Valve with Freeze Seal

This temporary modification installed a freeze seal in one of two Essential Service Water (ESW) supply lines to the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) so that a valve could be repaired.

The installation of the freeze seal and the valve repair are accomplished within the time allowed by technical specifications. Emergency measures were put in place as a contingency for freeze seal failure. Throughout the procedure the second ESW supply line to the TDAFWP and both motor driven auxiliary feedwater pumps remain available.

Section II

Plant Modification Request: 02780 Revision: 3

Surgeline Stratification Analysis

Thermal stratification in the pressurizer surge line has been identified as a concern which can affect its structural integrity. Stratification can result from the difference in densities between the hot leg water and generally hotter pressurizer water. Stratification with large temperature differences can produce very high stresses, and this can lead to integrity concerns. Study of the surge line behavior has concluded that the largest temperature differences occur during certain modes of plant heatup and cooldown.

A pressurizer surge line thermal stratification reanalysis was performed. In the reanalysis, plant-specific transients and seismic loads were used in the fatigue usage factor calculation. The results of the reanalysis indicate that the requirements of ASME Code allowable are satisfied. Also a revised pipe break location was performed, the Code equations did not exceed the 2.4 Sm limit nor did the fatigue usage factor exceed the 0.1 limit. Therefore no intermediate break point is required to be postulated utilizing criteria from NRC "Branch Technical Position MEB 3-1 Rev. 2 - June 1987."

Configuration and operating characteristics of the system are not affected in any manner by this reanalysis. Pipe routing, pipe supports and whip restraints remain the same. Pipe stresses have been change due to the affect of thermal stratification, however the stresses are found to be within the allowable code limits for both the OBE and SSE conditions.

Plant Modification Request: 03372 Revision: 0 and 1

Addition of Gate Valve in the Diesel Generator Sprinkler Fire Protection System

This modification adds a 6 inch gate valve in each of the diesel generator sprinkler Fire Protection Systems at the recommendation of American Nuclear Insurerers (ANI). The valves have been added for sprinkler system isolation to facilitate maintenance. The valves are to be locked open to avoid inadvertent isolation of the sprinkler system supply water. They have been analyzed to have no adverse impact to seismic loads. The proposed change does not affect the operation of the fire protection system. The fire protection system is not relied upon for accidents evaluated in the Updated Safety Analysis Report (USAR).

Plant Modification Request: 03518 Revision: 0

Chemical Additions to the Essential Service Water System (ESW)

This modification replaces existing carbon steel piping with stainless steel and alloy piping. It also provides two chemical addition pumps and a storage tank to add a non-oxidizing biocide into the ESW system. This modification was generated to correct degradation of the ESW system warming lines caused by excessive corrosion resulting from chlorination of stagnant and low flow lines. This modification also removes the chlorine equipment located in the ESW Chlorination Building along with detectors and Control Room chlorine annuciators.

The addition of biocide improves the cleanliness of pipe and components served by the ESW thus reducing their changes of failing. Elimination of the gaseous chlorine at the ESW pumphouse reduces the probability of a chlorine release. The physical chemical form of the biocide when stored or added produces safer accident scenarios. The chemical is shipped in dedicated shipping tanks and handled in a closed system. The chemical is located such that it is prevented from coming in contact with other chemicals that could produce a reaction and create a hazard.

The piping modifications have been reviewed for impact on supports. Two new supports were added. No other physical change to existing supports were required.

Plant Modification Request: 03604 Revision: 0

Revised Fire Pump Start Settings

Pinhole leaks developed on two relief valves in the Fire Protection System from erosion corrosion. In addition excessive pressure surges were also experienced in the Fire Protection System piping upon pump start. To remedy this condition, relief valve pressure settings were adjusted to accommodate flows, start pressures of each fire pump were raised, the jockey pump was replaced and a flow orifice was replaced. The results of this modification will reduce the amount of flow through the claval while maintaining maximum system pressure and not reduce system flow or pressure requirements. The modification also reduces the "delta P" experienced during the starting of the pumps thus preventing water hammer.

This modification does not change the function of the fire protection system. The change to pressure setpoints remain within the original design values. All equipment on the fire protection system is designed to withstand static pressure well above those settings implemented by this modification. The relief valve pressure setting modification will increase the reliability of the valves. The fire protection system has redundant supply lines and pumps to provide fire protection to safety equipment. The supply of water for the intended function of the system has not been changed to reduce the present level of safety.

Plant Modification Request: 03763 Revision: 0

Centrifugal Charging Pumps - Drip Pocket Drain Lines

This modification adds drain lines to the Centrifugal Charging Pumps (CCPs) so that leakage from the mechanical seals will run directly to floor drains rather than mixing with oil on the skid base plates. This is to prevent large quantities of oil from entering the radwaste system when the skid base plates drain.

The safety-related CCPs are part of the Chemical and Volume Control System (BG). As described in Updated Safety Analysis Report (USAR) Section 9.3.4, the BG system has a number of safety design bases. Specifically, this modification does not affect the ability of the CCPs to fulfill the requirements as stated in Safety Design Basis Two (i.e. the pressure boundary of the CVCS will remain intact after a Safe Shutdown Earthquake (SSE), the CCP will remain functional after an SSE, and the CVCS will still perform its intended function following postulated hazards, internal missiles, or pipe breaks). This modification is in accordance with the remaining design bases, as well.

The equipment drains are part of the Floor and Equipment Drains System (LF). As described in the USAR, Section 9.3.3, this system has a number of safety design bases. Since the drip pocket drain lines are routed to the same equipment drain as the base plate drain lines, this modification is in accordance with these bases.

The CCPs are located in separate rooms on the 1974' - 0" level of the Auxiliary Building, in radiation zone E. By eliminating the major source of leakage from leaking onto the base plate, the modification will reduce the radiation dose to workers. This is because there are no other major sources of leakage that would require manually pumping out the contaminated pump skid base plates.

With the major source of leakage eliminated, the possibility of the skid base plate overflowing (due to the closed drain line valves) is practically eliminated. Even if the skid base plates were to overflow, the leakage would be contained by the floor drains in the respective pump rooms. These floor drains are routed to the same location as the equipment drains utilized by the drain lines. The Fire Hazards Analysis has already considered the quantity of oil contained within the pump skid, as well as an additional transient 55 gallons. Collection of any oil on the skid base plate (with the drain valve now normally closed) will not add to the quantity of oil within the room.

As stated in the USAR, Section 9.3.4.2.2, the CCP seals are provided with leakoffs to collect the leakage and therefore prevent leakage to the atmosphere. This modification routes the leakage directly to an equipment drain, preventing it from collecting on the CCP skid base plates, further reducing possible leakage to the atmosphere.

The drain lines are 3/4 inch, field routed, stainless steel tubing. Design documents ensure that the tubing hangers are installed such that no II/I concerns will result. The II/I Report has been updated to reflect this modification.

The seismic qualification of the CCPs is not affected by the addition of the 3/4 inch tubing to the 3/4 inch NPT drip pocket drain holes. The modification does not affect any seismic, environmental, or equipment qualifications of any system, component, or structure.

Plant Modification Request: 03796 Revision: 0

Approved Use of AWS D1.1-90 for Welding of Structures

This modification allows the use of the latest edition welding code (AWS D1.1-90). This welding code includes performing welding procedure, welder performance qualification , prequalified joint or weld details and approved base metals.

Structural Welding Code-Steel AWS D1.1 covers welding requirements applicable to welded structures, It is to be used for the design and construction of carbon and low alloy steel structures. It is not intended to be used for pressure vessels or pressure piping. Welding procedures which conform to the AWS welding requirement, are deemed as prequalified and are therefore approved for use without performing procedure qualification tests.

AWS D1.1-90 contains additional welding joint details. The current edition does not effect the allowable stresses, the load carrying capacity of the welded joint, approved base materials, or inspection criteria.

A comparison of the welding code currently approved for use and the latest welding code edition, has determined that the latest edition is more detailed and restrictive and conservative. Special emphasis was placed on the areas described above as well as material stress allowables.

Plant Modification Request: 04148 Revision: 0

Closure Restrictions for Boron Injection Tank (BIT) Discharge to Reactor Coolant System and Charging Pump Discharge to BIT Valves

This modification revises operating procedures for closure of Boron Injection Tank (BIT) Discharge to Reactor Coolant System and Charging Pump Discharge to BIT valves EMHV8801A and B or EMHV8803A and B. Closure of valves will be restricted to times when differential pressure across the valves is less than 1380 psi or when the Centrifugal Charging (CC) and Positive Displace (PD) Pumps are secured. This restriction is required to ensure the valves will not be damaged during closure under high differential pressure conditions which could occur in some scenarios beyond the original valve design.

Valves EMHV8801A/B and EMHV8803A/B are required to close in various Emergency Operating Procedures (EMGs). The reason for closing the valves are:

1. The Safety Injection Signal (SIS) has been terminated, and the normal charging lineup is to be restored.

2. The accident which caused the SIS has been brought under control, and the normal charging lineup is to be restored.

3. The accident requires charging flow to be terminated.

Since the PMR only requires the CC/PD Pumps to be secured when the dP across the valves is >1380 psi, or when RCS pressure is less than 1300 psig, this evaluation will only address this case. if the dP across the valves is <1380 psi, or if the RCS pressure is greater than or equal to 1300 psig, no pump restriction applies. For reasons 1. and 2., securing the CC/PD Pumps produces no adverse condition, since either Safety Injection (SI) has been terminated or the accident has been brought under control. For reason 3., the accidents are Steam Generator Tube Rupture (SGTR) and Pressurized Thermal Shock (PTS).

In these two accidents, charging flow must be stopped to prevent the accident from becoming worse. Currently, operating procedures isolate charging flow by closing the subject valves, however, securing the CC/PD Pumps is just as effective. Once the valves are closed, the CC/PD Pumps may be restarted as required.

The other purpose of the CC/PD Pumps is to provide seal injection to the Reactor Coolant Pumps (RCPs). If CCW flow to the seals is not available during the time the CC/PD Pumps are secured, analysis has shown that no damage would occur to the RCP seals if seal injection is restored within 10 minutes. Since the subject valves stroke in <15 sections, there is ample time to restart a CC/PD Pump to restore seal injection flow. If a CC/PD Pump cannot be restarted for some reason, the RCPs can be stopped to preclude any damage to the seals.

Valve closure is required only as a result of an accident having already occurred. Isolation of charging flow is a normal or desired function follow ng an accident. The method of isolation can be either valve closure or pump stoppage. The purpose of the change is to assure the valves will not malfunction. Securing of the CCP(s) with subsequent CCP restart also does not increase the probability of occurrence of CCP malfunction, because this is a

normal control function and the CCPs are additionally protected from the effects of single failures. No other equipment will be adversely affected by the change. By the time the CC or PD pumps might be secured, the pumps and valves have either already performed their function as defined in the technical specifications, or securing the pumps will accomplish the same function as closing the valves.

Temporary Modification: 92-53-EC Revision: 0

Installation of Temporary Shielding on Spent Fuel Pool Cooling Pump

This temporary modification installed temporary shielding on Spent Fuel Pool Cooling pump PEC01A frame to deflect leaking water down onto pump base. The Spent Fuel Pool Cooling pump PEC01B was out of service. During this time it was necessary to use PEC01A. However, pump A had developed a leak spraying coolant around the pump seal. To contain the contaminated coolant within the pump's drain area, a temporary wrap was installed. The weight of the wrap was negligible therefore having no seismic impact on the equipment. If the wrap should become loose or fall, no damage would occur to equipment or piping in the area. The wrap will be localized and will not cause any ventilation concerns for area equipment.

USAR Change Request: 92-149 Revision: 0

Responsibility Transfer for Distribution of Plant Procedures

This change in program responsibilities transferred the distribution of plant procedures from the Plant Safety Review Committee to Document Services to consolidate control and distribution of documents. No program requirements were deleted, nor were Quality Program requirements deleted. The change is administrative in nature and does not affect plant equipment or operation.