Attachment to WM 89-0077

WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

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

ANNUAL SUMMARY OF SAFETY EVALUATIONS

Report No. 4

Reporting Period: January 1, 1988 through December 31, 1988

Prepared by

Merlin G. Williams

March 29, 1989

IE47
1/1

EXECUTIVE SUMMARY

This report provides a brief description of changes, tests and experiments and a summary of the associated safety evaluations which were reviewed and found to be acceptable by the Plant Safety Review Committee between January 1, 1988, and December 31, 1988. This report is being submitted pursuant to 10 CFR 50.59(b)(2).

Three major categories of safety evaluations are included in this report. Section I, Safety Evaluations, generally includes items such as temporary plant modifications, procedure revisions and use of temporary procedures. Section III, Plant Modification Requests, primarily involves permanent plant modifications and design drawing revisions. Section III addresses Updated Safety Analysis Report (USAR) revisions and proposed changes to the plant's Technical Specifications.

SECTION I

SAFETY EVALUATION 87-SE-017 Revision: 1 and 2

Title: Instrument Air System Temporary Modification

Description: This temporary modification removes the internals from check valve KAV214 in the Compressed Air System to allow the air being supplied by the temporary compressor to flow into the air dryers/filters and be used as Instrument Air. This modification also provides for the replacement of the portable rented Atlas Copco Compressor with a Sullair Compressor.

Safety Evaluation: The check valve performs no safety related function. This check valve prevents backflow from the Service Air System into the Instrument Air subsystem during normal plant system alignment. The removal of the check valve's function is temporary and will be restored when the system is improved with a larger more reliable air system. The quality, quantity, and pressure of Sullair Compressor Air is adequate to supply and meet the requirements of the Instrument Air System with no degradation to supplied system components.

SAFETY EVALUATION 87-SE-017 Revision: 3

Title: Instrument Air System Temporary Modification

Description: The subject temporary modification revision is being made to isolate the service air header from the instrument air subsystem by closing valve KAV004 while check valve KAV214 internals are removed and the Sullair Compressor is installed per Revision 0 of this modification. Valve KAV004 will be maintained closed by this modification revision and only opened in case of an emergency, i.e., an air compressor trouble alarm or a compressor air pressure low alarm, and as determined necessary by Operations. The check valve internals will remain removed under this alignment configuration to allow, in case of an emergency, the Sullair Compressor air to flow back into the air dryers/filter train of the Compressed Air System (CAS). The air would then be processed into instrument air for subsequent use.

Safety Evaluation: The check valve, KAV214, is an ANSI B31.1 valve which performs no safety related function. With the Servise Air System isolated from the Instrument Air subsystem by the actions of this modification revision (KAV004 closed), the function of the check valve is inconsequential. The quality, quantity, and pressure of Sullair Compressor air, if it were to be used during an emergency, is adequate to supply the requirements of the Instrument Air System with no degradation to supplied system components. The air from the Sullair Compressor is essentially oil free (completely eliminates all lubricant aerosols of .01 micron in size). The Sullair Compressor, if needed in an emergency, can adequately provide the instrument air loads. Use of the Sullair Compressor for instrument air via valve KAV214 will not degrade system components, create a different type of malfunction or accident, nor increase the probability or possibility of these events taking place.

SAFETY EVALUATION 87-SE-019 Revision: 1

Title: Gasket Replacement With Non-Quality Material

Description: This temporary modification was issued because the gasket on AB PV-3, Steam Generator "C" Atmospheric Relief Valve, required replacement following disassembly of the valve to repair seat leakage. The only available gasket material was supplied "non-quality" but is the same part number, style, dimension and pressure rating as the gasket recommended by the valve manufacturer. These are the critical characteristics. As described in Wolf Creek Generating Station Procurement Evaluation Request (PER) No. 097, Revision 0, the gasket filler material shall be certified by the manufacturer to have a leachable chloride content of 220 ppm maximum. The replacement gasket does not have the certification; however, the valve manufacturer's control valve quotation sheet found in Manual 10466-J-601B-0062-03 has no such requirement and in Flexitallic Bulletin 171 a note states that an asbestos paper (this gasket has asbestos filler material) meets the 220 ppm requirement for total soluble chlorides. If chloride induced stress corrosion cracking were to occur in carbon steel, it is reasonable to believe any stress corrosion cracking will open to the gasket seating surface (because of stress, temperature and proximity to chloride leaching). Accordingly, the gasket will be replaced at the next available outage.

Safety Evaluation: The critical characteristics of style, dimension and pressure rating remain the same because the part number is the same and visually verified. Operation of the valve will be tested prior to return to service. Function of the gasket is to provide a pressure barrier against leakage. This is analyzed in USAR Chapter 15.1.5. If leakage should develop to the extent that the steam line becomes inoperable, then three loop operation is not permissible and reactor shutdown is required. Technical Specification bases are not affected. Failure of this valve is analyzed and shown in USAR Table 10.3-3. Two of the four valves are adequate to meet shutdown requirements and overpressure protection is provided by the Safety Valves. Main Steam Line Break is analyzed in USAR Chapter 15.1.5 and steam supply loss to Auxiliary Feed Pump Turbine (Steam Loop #3) will be recovered by Motor Driven Auxiliary Feed Pump for 100% feedwater requirements.

SAFETY EVALUATION 87-SE-043 Revision: 0

Title: Defeat Of Automatic Closure Of Reactor Coolant Pump Cooling Water Return Valve

Description: Off-normal procedure OFN 00-005 provides actions for operators in the event of a Reactor Coolant Pump (RCP) malfunction or loss of support function which could lead to RCP failure. The changes being made to revision 5 of this procedure are to: 1) make the procedure flow in a more logical manner, 2) add additional information such as a caution note, computer temperature trend points, actions to defeat T and T ave for the affected loop, and 3) ensure flow to the thermal barrier when the #1 seal high leakoff flow is isolated.

Safety Evaluation: The changes addressed in Items 1 and 2 are viewed as procedure enhancements and do not involve safety concerns. The third item of adding the step to defeat automatic closure of the Component Cooling Water return thermal barrier isolation valve by opening its breaker on the affected pump, is being added as a preventative action to minimize the amount of seal and/or bearing damage. This action is only undertaken when all other procedure actions have failed and the #1 seal is outside the safe operating range and considered inoperative, with a primary pressure drop occurring across the #2 seal. This action is recommended by the pumps' manufacturer in their instruction manual under Emergency Operations.

SAFETY EVALUATION 87-SE-071 Revision: 1

Title: Excessive Casting Thickness Of Positive Displacement Pump Discharge Valve

Description: The replacement valve body for Chemical and Volume Control System (BG) PRV-8818, Positive Displacement Pump (PDP) discharge relief to the Volume Control Tank (VCT), has casting thickness such that heavy hex nuts cannot be used on the inlet or high pressure side flange. This flange is 4 holed and uses 1" bolting. PRV-8818 is installed to protect the PDP (PBG04) discharge piping. It is desired to complete installation of this valve using the heavy hex nuts machined down to standard hex nut size, to accommodate other outage work involving filling of the VCT. Bechtel Specification MS-2 requires the use of heavy hex nuts on Class BCB (2500 lbs., Stainless Steel, ASME Section III Class 2) piping. ANSI B16.5, applicable to Class 2 piping, allows the use of standard or heavy hex nuts. This will be resolved by Nuclear Plant Engineering (NPE) prior to operation of the PDP. The PDP will be tagged out for the duration of this Temporary Modification under a clearance order until NPE resolution.

Safety Evaluation: This modification does not affect the operation of the relief valve (PRV-8118) to protect the lines and components that might be pressurized above design pressure (2800 psig) by improper operation or component malfunction. (Reference USAR page 9.3-55).

SAFETY EVALUATION 87-SE-076 Revision: 1

Title: Extend Service Air Piping To Exciter Bearing Seal

Description: The intent of this Temporary Modification Order (TMO) is to hard pipe service air to main generator exciter bearings 11 and 12 to prevent bearing oil inleakage into the exciter housing by reversing the pressure gradient that currently exists. The added carbon steel piping, supports, valves and regulators maintain service air pressure integrity of 125 psig. The added piping is located under the turbine building operating deck floor on the 2065' elevation. It is approximately 40' from the point of tie-in until it terminates at the exciter bearings. The Compressed Air System (CAS) provides a reliable, continuous supply of filtered, dry, and essentially oil free air for pneumatic instrument operation and control of pneumatic valves. Safety related pneumatically operated valves are listed in USAR, Section 9.3, Table 9.3-2. All listed valves fail in their safe position upon loss of air throughout the plant. Instrument and service air are both supplied from the three plant compressors with only instrument air being processed through a dryer/filter train before utilization. To relieve the three plant air compressors and free them for instrument air loads, all of the service air loads are currently being supplied by the Sullair skid mounted compressor, which is temporarily installed under TMO #87-035-KA (Safety Evaluation 87-SE-017.)

Safety Evaluation: The portion of the CAS with a safety related function will not be compromised by adding to the turbine building service air piping. The safety related backup compressed gas (N_2) supply for the Auxiliary Feedwater Control and Main Steam Atmospheric Relief Valves will remain unaffected by this added piping. Additionally, the reliable backup supply of compressed gas for the Main Feedwater Control Valves will not be affected either. The 8-hours of reliable compressed gas in the accumulators for these valves will not be comprised by the added service air piping to the exciter in the turbine building. The quality of instrument air has not been degraded by the added piping to the Service Air Subsystem since the added piping does not alter or degrade the operability of the filter/dryers in the instrument air supply lines. The added service air piping will not degrade system components, create a different type of malfunction or accident, nor increase the probability or possibility of these events taking place.

SAFETY EVALUATION 87-SE-107 Revision: 0

Title: Increased Combustible Loading Of Auxiliary Building

Description: This procedure change to Administrative Procedure ADM 13-102, Revision (Rev.) 6, "Control Of Combustible Materials", is made to reflect the additional amounts of combustible materials allowed to be stored in the plant. Since this loading is different than that described in the current USAR revision, this 10 CFR 50.59 safety evaluation is required. Recent incorporation of Plant Modification Request (PMR) 2159, Rev. 0, and 2206, Rev. 1, into the plant's design have provided additional fire protection features to accommodate these additional combustible loads. PMR 2159 incorporated the permanently approved protective clothing storage areas for the Auxiliary Building in Fire Areas A-1, 8 and 19. This action increases the amount of fixed combustibles (Class A) in these areas from their current Fire Hazards Analysis in the USAR. PMR 2206 Rev. 1, incorporates an increase to the fixed combustible loading (Class A and B) in Fire Area A-1, due to the existence of the hot tool crib, waste compactor and maintenance/test equipment storage on the 1074' level of the Auxiliary Building. Revision 1 of this PMR also incorporated additional fire protection features, such as a thermo-lag barrier and additional fire detectors. Revision 2 of this PMR, which includes dampers in the HVAC duct between floor levels 1974 and 1989, has not been incorporated as of this procedure change evaluation date.

Safety Evaluation: The changes made to the procedure are allowable provided compliance with Administrative Procedure ADM 13-100, "Fire Protection Manual", is maintained until the dampers are installed. The addition of the protective clothing in the fire areas did not affect the Fire Hazard Analysis for these areas since the permanent storage areas are in remote locations within these areas and the clothing is stored on metal shelves in an orderly fashion. These storage locations do not provide a fire source or base which could affect both trains of a safety related system. The procedure change did not reduce the separation distance between safety or non-safety related equipment. Therefore, space and configuration have been maintained. The number and location of portable extinguishers and hose stations remains unaffected and they are fully capable of fighting fires in the storage locations. Until the additional fire dampers are incorporated, the appropriate fire impairment controls, i.e., firewatch/patrol are to be implemented per ADM 13-100. The additional loadings allowed by the procedure change, with the added fire protection features and required impairment controls, ensure that a fire in this area (A-1) will be confined and adequately retarded from spreading to adjacent areas (A-33).

SAFETY EVALUATION 88-SE-001 Revision: 0

Title: Containment Building Area Radiation Monitor Wiring Disconnected

Description: This temporary modification pertains to the area radiation monitor detector assembly SDRE41 and its local indicator RIA41 which are located on the manipulator bridge crane in the Reactor Building. The detector is used to protect crane operators during refueling operations. The wiring feeding the manipulator crane was disconnected during the Refuel II Outage to allow setting the reactor head. This wiring will not be connected until the next refueling outage. An Engineering Evaluation Request is being prepared by Maintenance Engineering to address relocation of the wiring to prevent interference with head movement in the future. The present configuration is causing control board annunciator windows A-62 (Hi-Hi Rad), B-62 (Hi Rad), and C-62 (Failure) to be in continuous alarm status.

Safety Evaluation: The Area Radiation Monitoring System (ARMS) performs no function related to safe shutdown of the plant or to mitigate the consequences of accidents that could result in off site exposures (Reference USAR 12.3.4.10). The system functions continuously to alert plant personnel and the control room of the occurrence and approximate location of increasing radiation. The jumpering of the switch outputs of the out-of-service monitor will restore control room awareness of the remaining active detectors. The jumpers do not affect any Class 1E equipment or wiring.

SAFETY EVALUATION 88-SE-002 Revision: 0

Title: Radwaste Building Temporary Shielding

Description: This temporary shielding request is for temporary shielding to be placed on Valve HC V7300 (primary spent resin header to primary spent resin storage tank isolation), HC V7357 (primary spent resin header to primary spent resin storage tank check valve), density element HC DE-35 and line HC-036-HCD-3 located in the Radwaste Building in order to reduce exposure to personnel working in the area.

Safety Evaluation: The primary spent resin header line HC-036-HCD-3 is a non-seismic, non-safety related, "D" augmented line which accommodates transfer of resins from reactor purification systems. The Solid Radwaste System performs no function related to the safe shutdown of the plant, and its failure does not adversely affect any safety related system or component, and therefore has no safety design basis.

SAFETY EVALUATION 88-SE-003 Revision: 0

Title: Isolation Of Essential Service Water Warming Line

Description: This Temporary Modification Order (TMO) is to facilitate weld repair of the 30" Essential Service Water (ESW) "A" train warming line downstream of EF V264, (ESW traveling water screen 1A warm water header downstream isolation). This valve and 6' of connected upstream piping between EF V264 and EF V262 (ESW traveling water screen 1A warm water header upstream isolation) are being removed. Valve EF V262 has been closed and tagged "do not operate" to isolate the warming line supply. To provide additional personnel safety a temporary blind is being bolted onto the mating flange connection on EF V262 via the subject TMO. The 30" warming line provides freeze protection of A train ESW during its operation, reference USAR 9.2.1.2.2.3. Two trains of ESW are required to be operable in Modes 1 through 4 per Technical Specification 3.7.4.

Safety Evaluation: It has been determined that the warming line need not be operable when the cooling lake temperature is above 36°F since the ice hazard is not present. Therefore, the isolation of the warming line above 36°F does not render "A" train ESW inoperable or require entry into the Limiting Condition for Operation (LCO). No reduction in the margin of safety defined in the bases of Technical Specification 3.7.4 will result and should lake temperature fall to 36°F, compliance with the LCO shall be met. With only one train of ESW operable, the Technical Specification LCO allows normal plant operations for up to 72 hours.

SAFETY EVALUATION 88-SE-004 Revision: 0

Title: Jumper Installation Around Switch Controlling Air Conditioning Unit

Description: Switch 155 as shown on drawing M-622.1 023-W08 is the Start-Reset-Stop switch to the Control Room Air Conditioning Unit SGK04B. This switch has failed. A like replacement is not available at this time. For operability of this unit per Technical Specification 3/ .7.6, this temporary modification will install a physical jumper across the poles at switch 155 to allow SGK04B to be energized.

Safety Evaluation: The equipment used to jumper this switch will be sufficiently rated to handle the required conditions at the switch (i.e. current, voltage, heat load). This jumper as described will not affect any of the safety features of the unit as it does not bypass any of the protective circuitry to SGK04B. A switch may be installed in series with the jumper to allow for the operation at this switch by Operations personnel provided the switch is also sufficiently rated to handle the required conditions. The mass at this jumper is so small that it does not present any seismic concern.

SAFETY EVALUATION 88-SE-005 Revision: 0

Title: Installation Of Temporary Differential Pressure Gauge On Safety Injection Pump Discharge

Description: This safety evaluation is applicable to proposed changes to the procedure governing Reactor Coolant System Isolation Check Valve leak testing. The changes to the surveillance procedure STS PE-019E, Revision 0 are mainly enhancements or editorial changes/corrections and, as such, do not affect or change the USAR analysis or Technical Specifications. The only change which does create a change from its description (Figure 6.3-1 Sheet. 2) in the USAR is the incorporation of the temporary differential pressure (D.P.) cell at flow element EM FI-928 (Safety Injection System Pump to Reactor Coolant System hot leg 4 test flow indicator). The optional installation of the additional flow indicator provides for redundancy, greater accuracy, and more sensitive readings during testing.

Safety Evaluation: The test line and D.P. cell are utilized only for leakage testing and do not provide for plant power operation or safe shutdown of the reactor. These components provide for measurement of leakage during surveillance testing. In this capacity they do not interface with plant equipment important to safety.

SAFETY EVALUATION 88-SE-006 Revision: 0

Title: Simulation Of Normal Steam Generator Levels While The Unit Is Shut Down

Description: The temporary change being made is to emulate normal Steam Generator (SG) levels on 3 of the 4 channels of each loop. This will be accomplished by placing a resistor at each instrument's input test jack in the 7300 process protection cabinets which will give the voltage value to emulate normal SG levels. One instrument channel will not be emulated in order to provide actual SG level indication to the Control Room operators. This change is limited to plant Modes 4, 5, and 6 since Reactor Trip and Auxiliary Feedwater Actuation Signals are defeated. In Modes 4, 5, and 6, this Engineered Safety Features Actuation System (ESFAS) function is not required since the reactor is tripped and the plant is being cooled down to Tave<350°F. Feedwater requirements are minimal and not required for emergency needs in these plant modes.

Safety Evaluation: This temporary change shall not be implemented until the plant is in Mode 4, entering Refuel III. This temporary change is being installed to prevent unnecessary challenges upon the system. This modification is required to be returned to normal operation prior to Mode 3 entry following Refueling III. The basic function of the Lo-Lo Steam Generator water level is to preserve the steam generator heat sink for removal of residual heat by acting to cause a Reactor Trip before the steam generators are dry. This feature also acts to start Auxiliary Feedwater system, refer to USAR 7.2.2.3.5. A coincident 2 out of 4 channels in the Lo-Lo condition initiates these ESFAS functions. The postulated accident in this case is loss of normal feedwater flow and is analyzed in USAR 15.2.7. This analysis sets initial reactor conditions at 102% power. Technical Specification Basis (2.2.1) for limiting safety system settings states the protective function and agrees with the USAR as stated above. Technical Specification Table 3.3-3 requires a minimum of 3 operable channels per operating per steam generator while in Modes 1, 2, and 3. Therefore, defeating the Reactor Trip and Auxiliary Feedwater Actuation Signals during plant Modes 4, 5, and 6 is not a reduction in the margin of safety since the safety feature is not required to be present by the Technical Specification.

SAFETY EVALUATION 88-SE-007 Revision: 0

Title: Installation Of Additional Gai-Tronics Station

Description: This Temporary Modification Order (TMO) will tie into Gai-Tronics station 103 in the Reactor Building and provide for an extension of this station into the refueling pool to support Reactor Vessel head removal work activities.

Safety Evaluation: The communication system is described in Section 9.5.2 of the USAR. Design bases 9.5.2.1.1 states that there is no safety design basis for the communication system. Figure 9.5.2-2 of the USAR and Bechtel drawing E-1L1595 shows station 103. The Gai-Tronics system is not Class 1E. The function of the Gai-Tronics stations to allow communications throughout the site is not changed. The function and design of the communications system is maintained.

SAFETY EVALUATION 88-SE-008 Revision: 0

Title: Installation Of Temporary Shielding Above Reactor Vessel

Description: Temporary lead shielding is to be placed between the reactor vessel flange and upper control rod internals to provide personnel protection during decontamination of the flange. The shield assembly consists of 9 lead sheets, each sheet 1/8 of an inch thick, 6 foot width by 6 foot height sandwiched between 2 steel plates 6 foot width and 6 foot height, with a thickness of 3/4 inches each. Two tag lines secured to the assembly provide additional means of control. Assembly weight is approximately 4500 pounds. The lead/steel combination results in approximately a tenth thickness of attenuating medium, reducing anticipated gamma fields of 1.5 R/hr to 300 - 400 mR/hr.

Safety Evaluation: USAR Section 9.4.1.3 addresses postulated polar crane failure during a reactor vessel head removal or reassembly. If this unlikely event should occur, various consequences would prevail depending on the position of the vessel head assembly in relation to the vessel during the crane failure. The vessel head assembly impacting the vessel flange following a 4 foot fall through air and subsequent 24 foot fall through water has been analyzed as being the limiting accident case in terms of maximum impact velocity. Buckling of control rod drive rods on head impact has been analyzed as the only major force experienced by the fuel assemblies, and found to not damage fuel assemblies, thus, maintaining fuel cladding integrity. The weight of the shield assembly places any fall of the assembly onto the flange, control rod drive assemblies, or upper internals within the bounds of the head drop analysis. The shield assembly shall be wrapped in visquine, herculite, etc., to prevent dropping of foreign objects into the reactor vessel during use. In Mode 6 with the vessel head removed, the vessel does not serve as a pressure boundary. With the upper internals in place, the shielding assembly would not come into contact with fuel upon crane or cable failure. Integrity of the fuel cladding would be maintained. Reactivity control would not be challenged.

SAFETY EVALUATION 88-SE-009 Revision: 0

Title: Installation Of Temporary Shielding For Reactor Vessel Head

Description: This Temporary Modification Order (TMO) will place temporary lead blanket shielding around the reactor vessel head while it rests in the reactor vessel head decontamination and storage area. The methodology of support and attachment of shielding has previously been evaluated. Installation, surveillance and removal of the temporary blanketing will be administratively controlled. This evaluation was performed for the same shielding applications during the Refuel I Outage and the Refuel II Outage.

Safety Evaluation: The possibility or probability of the occurrence of a malfunction of equipment important to safety has not been created or increased since in the highly unlikely event of shielding shroud framework failure, the collapse of the lead shielding blankets from about 4 feet above the floor to the floor would not be on to any equipment important to safety or otherwise. For Mode 6 operations, temporary lead blanket shielding of the vessel head at the storage area will not affect safety related functions of the head. The shielding will be removed prior to entry into Mode 4.

SAFETY EVALUATION 88-SE-010 Revision: 0

Title: Installation Of A Jumper In Class 1E 125 Volt DC System

Description: This Safety Evaluation analyzes the installation of a jumper in the 125 volt DC system (Class 1E) between battery charger NK25, a spare, and Bus NK04. The normal battery charger for Bus NK04 is being taken out of service for maintenance.

Safety Evaluation: This use of battery charger NK25 is according to its design intent. The jumper will be connected to Bus NK04 through breaker 89NK0409 using 2/0 conductor cable. The cable conductor size was chosen to be large enough to carry the normal full load of Bus NK04. This load is approximately 59 amps. Breakers 52NG0203 and 89NK0402 shall be tagged OPEN and "do not operate" to prevent cross connection of load centers NG01 and NG02. All fire barriers breached by installation of the jumper will be controlled by the fire impairment procedure. This evaluation is valid in Modes 5 and 6 only.

SAFETY EVALUATION 88-SE-011 Revision: 0

Title: Installation Of Replacement Lug On Class 1E Battery Charger

Description: A lug or wire terminal on the control circuit board of Battery Charger NK24 has been replaced with another similar lug. The lug lands the 14 AWG wire feed to the board terminal. The NK Battery Charger was bought and procured as Class 1E IEEE 323, 283, hence its Quality status. The original lug, per conversation with the Charger's manufacturer, is a commercial grade component which is not uniquely qualified or tested for use in the nuclear industry. The manufacturer recommends replacement with a 1/4", fully insulated blue push-on-tab made by Hoffmann or a 1/4" Hollingsworth or other commercially available alternate. The replacement lug being installed is a McMasters 1/4", UL listed insulated, tin plated copper lug which may be crimped with any standard crimping tool.

Safety Evaluation: This replacement lug meets the design and is an equivalent replacement in like and kind, therefore the operability of the battery charger will be maintained as designed. The probability, possibility or consequences of previously evaluated accidents, malfunctions, or consequences thereof has not been increased since original specification requirements have all been satisfied.

SAFETY EVALUATION 88-SE-012 Revision: 0

Title: Blocked Open Door Between Auxiliary And Turbine Buildings

Description: Missile Door 41017 is being blocked open to provide a vent in the Auxiliary Building to support operability of the Control Room Emergency Ventilation System (CREVS) concurrent with the operability of the Emergency Exhaust Systems (EES). Technical Specification 3/4.7.6 requires two independent CREVSs to be operable in all modes. Technical Specification 3/4.7.7 for Loss of Coolant Accident considerations requires two independent EESs to be operable in Modes 1 through 4 and for refueling considerations, Technical Specification 3.9.13 requires two EESs whenever irradiated fuel is in the spent fuel pool. The subject missile door is in the north barrier wall of the Auxiliary Building which separates the Auxiliary and Turbine Buildings. This door is on the 2047'6" level of the Auxiliary Building and opens out into a Turbine Building stairwell. When the door is opened, appropriate security measures and a firewatch shall be established to prevent any decrease in security or increase in fire hazards from occurring. Missile Door 41017 provides protection against the four general sources from which missiles are postulated. These four sources are rotating component failure, pressurized component failure, tornados, and missiles associated with activities in the proximity of the site, reference USAR Section 3.5.

Safety Evaluation: In the case of the rotating and pressurized component failure postulated missile sources (from Turbine into Auxiliary Building considered since Auxiliary to Turbine is no safety concern), it is deemed that no significant increase in the probability of occurrence of this type of missile hazard to safety related equipment has occurred. This conclusion was reached due to the very low probability to begin with, the small opening created, 3'4" X 6'8" (22.2 square feet), the layout and configuration barriers (reinforced concrete masonry wall of the stairwell), the 2065' Turbine Building operating floor, the absence of safety related equipment in the target area exposed, trajectory, and the angles and rebound energies required. In the case of a tornado, the door will be closed when a warning is required by Natural Emergency Procedure OFN 00-003 and the Control Room will instruct the continuous firewatch to remove the block and close the door. This action will restore the protection provided to guard against tornado missiles. Therefore, no increase in the probability, possibility, or consequences of tornado missiles from previous analyses will be incurred. The consequences of flooding need not be considered since this door is not watertight and is above the maximum evaluated flood level of 1995.2' per USAR Table 3.4-1. Fire consequences have been prevented from any increase due to the presence of a firewatch. Should a fire occur in the Turbine Building which could spread into the stairwell and threaten the Auxiliary Building, the firewatch shall be instructed to close the missile door to provide the three hour rating shield between Auxiliary and Turbine Building inherent to design. This action will prevent fire spread from Turbine to Auxiliary Buildings or between fire areas, thereby maintaining the plant within previous fire hazards analysis. The temperature effects due to the door being open are viewed as inconsequential since direct exposure to outside temperature/elements has not been introduced. Additionally, the area directly affected on the 2047'6" level of the Auxiliary Building is not a temperature controlled environment required by Technical Specifications. Due to the small opening, temperature transmission mediums/buffers (enclosed stairwell), and similar environments between the interface of the Auxiliary and Turbine Buildings, no significant changes in Auxiliary Building temperatures are foreseen. However, this condition will be monitored by the Auxiliary Building watch and firewatch.

SAFETY EVALUATION 88-SE-013 Revision: 0

Title: Blocked Closed Tendon Gallery Damper

Description: The actuator of damper GF D036, Tendon Gallery to Auxiliary Building HVAC downstream isolation, is being removed and the damper is physically being blocked closed in its safeguards position. The damper's actuator is going to be put into service in the Control Building HVAC (GK) System on Damper GKD081, Control Room air conditioning unit 4A Control Room Discharge Isolation Damper. The damper actuators of these two dampers were procured and supplied under Specification M627-A and are equivalent.

Safety Evaluation: Damper GFD036 closes on a Safety Injection (SI) signal and minimizes the potential release of radioactivity from the Tendon Gallery to the Auxiliary Building. With the damper blocked in the closed position this function is already fulfilled. Damper GFD036 is the outer most damper of a double isolation damper combination on the outlet HVAC duct from the Tendon Gallery. Its tandem damper GFD035 is also blocked closed since its actuator has also been removed; see TMO #87-119-GK and Safety Evaluation #87-SE-092. The tornado damper GFD046 or fire damper GFD074 employed between the double isolation dampers and the Tendon Gallery are not affected. The Tendon Gallery is below the base slab of the reactor building and provides for installation and inspection of the vertical post tensioning system. Heated or cooled air is supplied to the Tendon Gallery area where it goes through the Auxiliary/Fuel Building normal exhaust filter adsorber unit prior to exhausting via the unit vent. Blocking the return side damper GFD036 closed from the Tendon Gallery does not increase the probability, possibility or consequences of previously evaluated accidents. Radioactive materials will not be released or spread to areas not monitored or radiologically controlled. All equipment important to safety will function as required.

SAFETY EVALUATION 88-SE-014 Revision: 0

Title: Blocked Open Fire Dampers

Description: The following 3-hour-rated fire dampers are being physically blocked open in order to provide heat to the Auxiliary Building/Main Steam Enclosure. The four listed dampers are fusible link fire dampers in the supply/exhaust duct from the Main Steam enclosure building supply air unit SGF01. This air unit is part of the Miscellaneous Building HVAC System and is not safety related. Fire Damper GFD002 is in the floor between the 2047' and 2026' levels of the Auxiliary Building. This damper provides the 3-hour-rated fire barrier between Fire Areas A-19 and A-18. Fire Damper GFD051 is in the floor between the 2000' and 1974' levels of the Auxiliary Building. This damper provides the 3-hour-rated fire barrier between Fire Areas A-24 and A-1. Fire Damper GFD070 provides the same fire protection function as GFD002; i.e., separates Fire Areas A-19 and A-18. Fire Damper GFD078 is in the wall at column line A-3 and provides for a 3-hour barrier between Fire Areas A-23 and A-18 at elevation 2035, floor level 2026.

Safety Evaluation: These fire dampers are normally open but have closed due to the hot air supplied by the air unit SGF01. While these dampers are blocked open, appropriate fire impairments are to be issued in accordance with Administrative Procedure ADM 13-103. The actions required by ADM 13-103 prevent the possibility or consequences of a fire spreading between fire areas from increasing as previously analyzed. All of the affected fire areas have either an ionization or an infrared flame detection system within the area. These detection systems alarm locally and in the control room. These detection systems have not been impaired or degraded by the action of this modification. The Auxiliary Feedwater Pump Rooms A, B, and C temperature limit as defined by Table 3.7-4 of Technical Specification 3.7.12 will be complied with, i.e. the maximum temperature limit is 119^{o}F. The operability of the Auxiliary Feedwater Pump Room Coolers has not been affected by this modification since these coolers are powered and supplied cooling water from vital or essential sources. The operability of the back draft, tornado, or balance dampers in the duct work has not been affected by blocking the fire dampers open. The seismicity of the duct work has not been degraded by physically blocking the dampers open.

SAFETY EVALUATION 88-SE-015 Revision: 0

Title: Discharge Of Heated Service Water To The Ultimate Heat Sink

Description: This operational change evaluation covers the environmental concerns with diverting a portion of the Service Water System (SWS) flows through the Essential Service Water (ESW) system and then discharging it to the Ultimate Heat Sink (UHS) during winter conditions. Normally discharged with the Circulating Water System (CWS), this diverted flow was intended to supply warming lines that prevent ice formation at the ESW intake and to provide chlorinated water to the ESW components to reduce microbiologically induced corrosion. This environmental evaluation is valid only for winter operations given the conditions discussed below. In the event that discharges to the UHS are needed during other seasons, especially summer, variables are expected to be different enough to warrant a separate evaluation. In completing this evaluation, actual monitored conditions combined with the best estimates available were used. Of primary importance were monitored discharge temperature increases (delta T), total residual chlorine (TRC) concentrations from the ESW, and estimated SWS flows to the UHS. With WCGS operating at 100% power, the SWS discharge to the UHS was heated an average of 35°F over ambient lake temperatures. This delta T average was computed from temperatures taken from January 10 through January 21, 1988. When Wolf Creek Generating Station was at 0% power, temperatures monitored from January 22 through February 11, 1988 varied more, however the delta T averaged only 5°F. Measurements of TRC from January 10 until winter discharges were discontinued on March 12, 1988, ranged from 0.01 to 0.44 mg/l and averaged 0.19 mg/l. Actual flow rates of this discharge were estimated to be between 3500-4000 gpm. Postulated operational modes, flow rates, and TRC levels initially intended for evaluation purposes were not used because they either did not reflect actual conditions or conditions changed during the course of this evaluation.

Safety Evaluation: It is concluded that discharging heated and chlorinated SWS effluents to the UHS will not significantly increase the exposure of the Wolf Creek Cooling Lake (WCCL) fishery to cold shock, increase area waterfowl disease or depredation events, nor increase chlorine effects on the fishery. Based on these, no increases of previously evaluated environmental impacts will occur. This operational change does constitute a change in a WCGS effluent route, however, significant changes in the overall National Pollution Discharge Elimination System (NPDES) regulated effluent volume and composition to WCCL as a whole will not result. Similarly, routine discharge to the UHS was not previously evaluated by the NRC, however, given the conditions stated above, this practice will not result in significant adverse environmental impacts.

SAFETY EVALUATION 88-SE-016 Revision: 0

Title: Jumpering Of Essential Service Water Valve Torque Switch

Description: This Temporary Modification Order (TMO) will jumper the open torque switch and allow the valve to operate with only limit switch protection. Essential Service Water valve EF HV-49 (Essential Service Water from containment air coolers) is located in the north pipe penetration room of the Auxiliary Building 2000' level. This normally closed valve is on the 14" Essential Service Water (ESW) outlet from containment coolers SGN01A and SGN01C. In the event of a Safety Injection Signal (SIS), EF HV-49 is designed to open 44% of full travel (open limit switch set at 44% based on Start-up flow balance test). It has been determined that the open direction torque switch for EF HV-49 is operating while the valve is coming off its seat and, thereby de-energizing the operator and preventing the valve from assuming its normal SIS position.

Safety Evaluation: Manual operation of the valve is possible and, therefore, it has been concluded that the opening torque experienced by the operator is not excessive but is above the present torque switch setting. Jumper the open torque switch and allow the valve to operate with only limit switch protection. Plant Modification Request (PMR) 01988 has been approved to allow bypassing this torque switch up to 40% of the valve travel in the open direction (to allow the operator to overcome opening torque), but has not been implemented. When implemented, the torque switch will effectively protect the valve for 4% before the open limit switch operates at 44%. In the event the open limit switch fails to operate at 44%, the valve would continue to operate until the stop nuts associated with the operator bottom out on the housing. The motor would then fail due to locked rotor with the butterfly disk in its full open position. This is the same position the valve would be in if the open limit failed with the torque switch active. Failure of the motor is not considered to be outside of previous analyses since the motor overloads are jumpered by design. This forces the motor to perform its safety function to failure. Upon implementation of PMR 01988, high torques experienced in the 0 to 40% open range which resulted in locked rotor would produce the same effect. In the event the valve motor fails and it is necessary to close the valve, it could be closed manually.

SAFETY EVALUATION 88-SE-017 Revision: 0

Title: Administrative Procedure Changes

Description: This safety evaluation addresses an administrative procedure change applicable to ADM 01-006, Revision 4. The title changes of Superintendent to Manager of Technical Support and I&C Supervisor to Manager of I&C are administrative changes made in the procedure and do not involve an unreviewed safety question. These nomenclature changes do not reduce the margin of safety defined in the bases of the Technical Specification. The procedure change of adding to the scope of the Manager of Technical Support, the Health Physics and Emergency Plan organizations due to the new plant organization structure are viewed as administrative in nature. The Health Physics and Emergency Plan organizations previously report to the Superintendent of Plant Support. The changes made to the subject procedure now place the Health Physics and Emergency Plan under the responsibility of the Manager of Technical Support. The Manager of Technical Support position remains reportable to and directly responsible to the Plant Manager as previously stated. The Health Physics and Emergency Plan Programs have not been diminished by this administrative change. The qualifications of the Manager (previously Superintendent) of Technical Support are no different than the Manager (previously Superintendent) of Plant Support, reference USAR Table 13.1-1 and ANSI/ANS 3.1-1978.

Safety Evaluation: These changes in the administrative structure of the plant's organization do not increase the possibility, probability, or consequences of previously evaluated accidents, nor cause a reduction in the margin of safety.

SAFETY EVALUATION 88-SE-018 Revision: 0

Title: Temporary Replacement Of Sanitary Lift Station Sump With Holding Tank

Description: This Temporary Modification Order (TMO) temporarily replaces a sanitary liftstation pump with a holding tank to allow application of a protective coating to the sump floor. The sanitary waste lift station for the power block (LA System) is located at the 1974' level of the communications corridor. It collects all power block sanitary wastes (restrooms, showers, mop sinks, etc.) for transfer to the sewage treatment plant. The waste is pumped by sump pumps PLA01A and PLA01B. The lift station sump is constructed of concrete and it has become necessary to remove the sump from service in order to apply a protective coating to the sump floor and walls. The temporary modification involves installation of a 540 gallon (approximate) holding tank near the existing sump and then routing the normal influent into the holding tank. This will be accomplished by removing 3 existing cleanout plugs near the 1974' level, installing inflatable dams, and then routing the waste from the cleanouts to the tank using flexible hose. Pump PAL01A will be removed from the sump and installed into the holding tank to automatically pump collected wastes to the existing sump discharge through a temporary connection. The pump will be powered by its normal power source. The temporary discharge line will connect into the normal discharge line downstream of the existing check valve.

Safety Evaluation: The sanitary waste drainage system is part of the floor and equipment drainage system as described in USAR Section 9.3.3. It performs no safety function and, therefore, has no safety design basis. The holding tank will not be in the vicinity of any equipment important to safety and affects no existing safety analysis involving design basis accidents, seismic events, fires, pipe whip events, or tornados that are provided in the USAR. In the event that the holding tank ruptured and making a conservative assumption that the tank was full (present intent is to set up the controls to maintain a low volume in the tank), up to 540 gallons of waste could be deposited directly on the floor. No appreciable water level would develop since the 1600 gallon sump will be open during this period and, therefore, liquid waste will flow back into it and into the open oily waste drains in the area. The Control Building oily waste sump is also located in this area and it could easily contain the entire amount of water received through the drains mentioned above. Some seepage could be expected into the Essential Service Water (ESW) pipe space (Room 3101) on the 1974' level of the Control Building through door 31011, but this small amount of water would have no affect on any structures, systems or components important to safety. Present intent is not to vent the holding tank. This is not considered necessary since the entire system is vented through the power block and, therefore, should prevent accumulation of sewer gas. In the very unlikely event that gases did collect in the tank and were somehow ignited causing the tank to burst apart, no safety related components or systems would be affected since the Control Building missile resistant walls and pressure door 31011 are considered adequate to resist any flying debris from this type of rupture. High pressure gas bottles are in the area (O_2, argon, acetylene, N_2, argon-methane mixture). A tank burst that damaged bottle connections containing combustible gases could initiate a fire in this area. Since this area contains no safe shutdown equipment and is separated from adjoining safe shutdown areas by a 3-hour rated barrier, any resultant fire cannot prevent safe shutdown.

SAFETY EVALUATION 88-SE-019 Revision: 0

Title: Procedural Change To Allow Operator Action To Prevent Spurious MSIV Or MFIV Closure

Description: The change made to procedure GEN 00-004, Revision 10 is to add actions to stop a spurious or inadvertent slow closure of a Main Steam Isolation Valve (MSIV) or Main Feedwater Isolation Valve (MFIV) at power and recover it to full open position. This change was prompted by events which occurred at a similar plant, namely inadvertent operator action and card failure. The Main Steam and Feedwater Isolation Actuation System (MSFIS) provides outputs to energize or de-energize control solenoids which operate or test the plant's MSIVs and MFIVs. The MSFIS is divided into two actuation channels, one active and one in standby, which are independent. The control signals are generated from a rotary selector switch which is "make before break". The procedure steps added provide direction for the operator to go to the MSFIS Manual Test Panel in back of the Control Room and take the active actuation channel, either red or yellow, of the associated valve to the bypass position. This action involves entry into Technical Specification 3.3.2 Table 3.3-3 Action 21 (MSIV) or Action 27 (FWIV) Limiting Condition for Operation. This action statement states that with the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE. With the operating signals defeated, that is in bypass, the associated valve's active side closed position limit switch is depressed by an operator which emulates the closed mode. The MSFIS Test Panel switch is then turned to the operate position, and the Main Control Board switch is then actuated to open the valve. These procedural steps defeat the valve's active operating controls, since they are placed in bypass, to simulate closed position so that the valve can be opened when the controls are switched back to active. The MSFIS Panel is labeled and located in a well lit area. The closed position limit switch for each valve is clearly identifiable by its electrical separation group color code (red or yellow) and is readily accessible.

Safety Evaluation: The probability or possibility of the occurrence of an equipment important to safety malfunction has not been increased since the procedure change is explicit and the equipment clearly identified. A chart reflecting the active or standby actuation channels and associated color for each MSIV and MFIV is provided as part of the procedure change. Inadvertent closure of a MSIV and a FWIV malfunction or closure during power operation has been evaluated. The analysis and consequences of these events are detailed in USAR Section 15.2.4 and 15.2.7. These events are considered American Nuclear Society Condition II events and are not as severe as the turbine trip event. These procedure actions do not increase the probability or consequences of these previously evaluated accidents. The procedural actions when undertaken shall require compliance with the Technical Specification. Compliance with the Limiting Condition for Operation will retain the plant within previous analysis and not reduce the margin of safety. Each MSIV remains operable in accordance with Technical Specification 3.7.1.5 since the standby channel is fully operable and capable of a 5-second closure when the active channel is in bypass.

SAFETY EVALUATION 88-SE-021 Revision: 0

Title: Procedural Installation Of Cables From Spare Charger To Battery Bank

Description: The subject procedure MPE E051Q-03, Revision 0 provides the means to maintain power on a 125 Volt DC System NK bus during its charger maintenance or during a Low Medium Voltage 4160 Volts Class 1E NB switchgear outage when the plant is in Mode 5 or 6. The procedure in the case of charger maintenance runs temporary 2/0 cables from the spare charger NK25, to the battery bank with the charger out of service. In the case where the normal AC feed bus is out of service the procedure allows for connecting the DC busses of the outage associated AC bus to the DC buses of the energized AC bus by 2/0 cables. The resultant configuration parallels the DC busses with the energized charger maintaining each busses' battery.

Safety Evaluation: Steps in the procedure verify that systems are de-energized and isolated, polarity is checked, and fire impairments issued on all fire doors breached by the cable runs. These procedural requirements and controls maintain the probability, possibility and consequences of previously evaluated accidents within previous analyses. The procedure controls prevent an increase in the possibility or probability of a short circuit, ground fault, electrical fire, over current degradation, or backfeed to the outage bus from occurring which could cause a challenge to or compromise plant safety features. The spare charger has a rated output of 300 amps and is identical to the normal battery chargers, all of which are Class 1E. Each battery charger has sufficient capacity to restore the battery from the design minimum charge (one duty cycle) to its fully charged state while supplying the largest combined demand of the steady-state loads. The full load rating of the battery chargers is 59 amps. USAR Section 8.3.2 states that the spare battery charger is centrally located and is used in the event of failure of a charger or inverter by connecting it to the affected system. Therefore, by design, the spare charger is suited for this use; however, the electrical connections between the spare charger and the affected DC system are not provided by design. The 2/0 copper conductor cables used between the spare charger and the DC bus or between DC busses are rated to at least 189 amps (reference NEC Handbook 3rd Ed.). The temporary cables are capable of carrying these currents without any degradation to themselves or the electrical sources or distribution system to which they are attached. Group separation is lost by the actions of the subject procedure but is not required since the plant is in Mode 5 or 6 before the procedure can be used. The Shift Supervisor is cognizant of the installation, existence and restoration of the affected DC system through the Work Request procedure. The mode restraint is also tracked by the Work Request program.

SAFETY EVALUATION 88-SE-022 Revision: 0

Title: Temporary Removal Of Battery Room Drain And Acid Neutralizer Tank For Maintenance

Description: This Temporary Modification Order (TMO) allows the PJ System (non-Class 1E 250 Volt DC) Battery Room Acid Neutralizer Tank located on the 2000' level of the Turbine Building to be removed to facilitate inspection and repair activities. The PJ Battery Room is on the 2033' level of the Turbine Building. The drain in this room is also being plugged for personnel safety while the neutralization tank is removed.

Safety Evaluation: This tank and associated drain are part of the oily waste collection system. The USAR describes how each battery room floor drain network is provided with an acid neutralization tank designed to neutralize the amount of acid contained within approximately 25% of the battery cells in the event of a break in the batteries. Temporarily removing the PJ battery room neutralizer tank and plugging the only drain/drain line that drains to the tank does not increase the probability of occurrence of previously evaluated accidents. These accident probabilities are not a function of the existence or nonexistence of the drain or tank since they do not affect the flood analyses. The probability of occurrence of a malfunction of equipment important to safety, or of being of a different type than previously evaluated has not been introduced since the changes made lie entirely in the Turbine Building. The Turbine Building provides no equipment vital to safe operation or shutdown of the reactor. The consequences of this temporary change do not affect the Control Building positive pressurization capability since the drain and associated piping lie entirely in the Turbine Building and do not communicate to the Control Building. The consequences of a gross PJ Battery Bank break, however unlikely, with the room drain plugged will not have a deleterious effect on any equipment important to safety since all leakage will be confined to the battery room with possible leakage under the room doors and onto the 2033' level of the Turbine Building. The potentially noxious and corrosive vapors will be vented from the room by the normal ventilation system in the same manner as designed. No increase or different release pathway of these vapors has been created by this temporary change. The Control Room habitability has not been compromised since all HVAC systems which provide a function important to safety have not been affected or been introduced to an increase in malfunction probability.

SAFETY EVALUATION 88-SE-023 Revision: 0

Title: Defeat Of Cask Handling Crane Interlocks

Description: This Temporary Modification Order (TMO) is to cover removing the limit switches and mechanical stops from the cask handling crane to allow it to rollover to the cask loading pool just 3' south of the spent fuel pool to facilitate access to the burnable poison tool in the cask loading pool. The cask handling crane and accessories are used to handle spent fuel shipping casks between the railroad cars or trucks, the loading pool, and the wash down pit. The cask handling crane is equipped with a monorail and hoist which is used to transfer new fuel from the new fuel storage vault to the new fuel elevator. Under normal use, limit switches and mechanical stops are located to prevent any crane (other than the spent fuel pool bridge crane) from traveling over the spent fuel pool. During scheduled maintenance periods, the cask handling crane is used to provide access over the spent fuel pool for servicing light bulbs and fire detectors. During these periods, the rail stops are removed to allow crane travel. These rail stops, which are not heavy loads, are hinged such that they can be rotated out of the path of the cask handling crane. The stops do not require lifting to clear the cask handling crane, but are permanently attached to the crane rail support girder to preclude a drop. In addition, geared type upper and lower limit switches are used in the control circuit in each hoist system of the cask handling crane (per USAR Section 9.1.4.2.2).

Safety Evaluation: The use of the cask handling crane shall be limited to the work area required over the cask loading pool. At no time shall the cask handling crane or associated 150 ton hook be allowed to travel over the spent fuel pool. Defeating the interlocks will allow the cask handling crane to travel over to the work area such that the 5 ton monorail hoist can be utilized. The use of the 5 ton monorail hoist will be over the cask loading pool and will not be over the spent fuel pool at any time. The Technical Specification actually allows loads up to 2250 pounds to travel over fuel assemblies (reference Technical Specification 3.9.7), but no loads over the fuel assemblies in the spent fuel pool are desired during use of the crane to lift the burnable poison tool. In restricting cask handling crane movements from over the spent fuel pool, it is concluded that all postulated accident scenarios fall within previous analysis since previous analysis includes ruptured fuel assemblies due to load drops (Reference USAR 15.7.4). Good rigging practices will be used and the load of the burnable poison tool shall be lifted in a vertical position, plumb to the hoist hook.

SAFETY EVALUATION 88-SE-025 Revision: 0

Title: Addition Of A Sampling Point To Obtain Condensate Storage Tank Samples

Description: This Temporary Modification Order (TMO) allows addition of a sampling point on a tubing supply line to allow sampling of the Condensate Storage Tank. The Condensate Storage and Transfer System (CSTS) consists of one 450,000-gallon, 42 feet high and 43 feet diameter, Condensate Storage Tank (CST) and associated valves and piping. The CST serves as a reservoir to supply or receive condensate, as required by the condenser hotwell level control system. The tank is also a nonseismically designed source of water to the Auxiliary Feedwater System. To verify acceptable water quality of the CST, Chemistry is required to sample CST contents. A sample point has been incorporated into the 3/8" stainless steel tubing supply line to level transmitter AP LT04 by way of a Swagelok tee, 3/8" stainless steel tubing and an isolation valve. These added components maintain system pressure rating. The effects of opening and closing the sample isolation valve on the level transmitter have been tested. The test which was conducted when the tank was at its normal level of 90% has shown that with the 3/8" sample point line valve fully open the level indication reads about 9% low. Due to the false reading in CST level the subject change can cause when a sample is being taken, Chemistry shall inform the Control Room that AP LI04A will temporarily read low when a sample is being taken.

Safety Evaluation: These effects on the transmitter present no increase in the malfunction probability or possibility of equipment important to safety since the transmitter does not align Auxiliary Feedwater suction from the CST to the Essential Service Water System upon a low suction pressure signal. The Auxiliary Feedwater suction header pressure transmitters and associated setpoints are not affected or degraded by the added sample point. The Technical Specification requirement of an assured CST volume of 281,000 gallons (60%, 26') is not degraded or reduced as a result of incorporating the sample point from CST level instrument AP LT04. The consequences of the false indication on the level transmitter when a sample is taken are not compromising or challenging to any of the plant's safety features. By Chemistry technicians informing the Control Room of the desire to take a CST sample, the operators will be cognizant of the false level indication and potential low level alarm.

SAFETY EVALUATION 88-SE-026 Revision: 0

Title: Installation Of A Demineralized Water Supply Line To A Temporary Tank

Description: New procedure CKL PE-162, Revision 0, was written to support the efforts of sludge lancing the Steam Generators during an outage. This activity requires a supply of demineralized water. A convenient source of demineralized water is being provided by running a 1-1/2" hose from the Demineralized Water Tank (TAN01) at AN V003 (Demineralized Water Storage Tank Drain Valve) to fill a tank in the sludge lancing trailer.

Safety Evaluation: USAR 9.2.3 describes the demineralized water system, including tank TAN01. The tank is vented, insulated, and heated. The system is non-safety related and the accident analyses in Chapter 15 of the USAR take no credit for its existence. Therefore, it is concluded that the possibility or probability of accident or malfunction occurrence or consequences thereof increasing from that which has been previously evaluated has not been incurred by this procedure. The connection from AN V003 to the trailer shall maintain a configuration to maintain system separation to prevent back-up of unanalyzed water to the Demineralized Water Storage Tank. Back flow protection on the trailer tank shall be verified prior to connection. The hose shall be disconnected at valve AN V003 when no flow conditions exist. The hose shall be physically disconnected to allow water drain off to prevent freezing and damage to valve AN V003.

SAFETY EVALUATION 88-SE-027 Revision: 0

Title: Temporary Removal Of Control Building Oily Waste Sump From Service For Maintenance

Description: This Temporary Modification Order (TMO) temporarily removes the Control Building Oily Waste Sump from service to allow application of a protective coating to the sump floor and walls. The Control Building oily waste sump is located at the 1974' level of the Communications Corridor. It collects non-radioactive oily waste via equipment drains and floor drains from the Communications Corridor, the Control Building, and the Auxiliary Building. The waste is pumped by sump pumps PLE07A and PLE07B to the site oil/water separator for disposal. The sump is constructed of concrete and it has become necessary to remove the sump from service in order to apply a protective coating to the sump floor and walls. The temporary modification involves installation of a 540 gallon (approximate) holding tank near the existing sump and then routing the normal influent into the holding tank. This will be accomplished by removing three existing cleanout plugs near the 1974' level, installing inflatable dams, and then routing the waste from the cleanouts to the tank using flexible hose. Pump PLE07A will be removed from the sump and installed into the holding tank to automatically pump collected wastes to the existing sump discharge through a temporary connection. The pump will be powered by its normal power source. The temporary discharge line will connect into the normal discharge downstream of the existing check valve. Four floor drains on the 1974' level and three floor drains on the 2000' level of the Communications Corridor will be plugged since they feed into the sump downstream of the cleanouts. Two equipment drains on the 1974' level and one equipment drain near the domestic hot water package (KD System) on the 2000' level of the Communications Corridor will require diversion of influent to temporary barrels or directly to the sanitary waste lift station (LA System) since these drains also feed into the sump downstream of existing cleanouts.

Safety Evaluation: The Control Building oily waste sump is part of the floor and equipment drainage system as described in USAR Section 9.3.3. It performs no safety function and, therefore, has no safety design basis. The holding tank will not be in the vicinity of any equipment important to safety and affects no existing safety analyses involving design basis accidents, or seismic, fire, pipe whip or tornado events that are provided in the USAR. In the event that the holding tank ruptured and making a conservative assumption that the tank was full (present intent is to set up the controls to maintain a low volume in the tank), up to 540 gallons of waste could be deposited directly on the floor. No appreciable water level would develop since the 1100 gallon sump will be open during this period and liquid waste will flow back into it and into the communications corridor elevator pit in the area. Some seepage could be expected into the Essential Service Water (ESW) pipe space (Room 3101) on the 1974' level of the Control Building through door 31011, but this small amount of water would have no affect on any structures, systems or components important to safety. The entire 540 gallons deposited in the pipe space would produce a water level of less than 1/4". The three plugged floor drains at the 2000' level of the communications corridor near the domestic hot water package will not pose any threat to structures housing safety related equipment. In the event of flooding in this area, there are still other oily waste drains at this level which will remain active. Doors in the area could also be opened to divert water to the outside.

SAFETY EVALUATION 88-SE-029 Revision: 0 And 1

Title: Removal Of Interlock Between Condenser Vacuum Pumps And Discharge Dampers

Description: This Temporary procedure change to surveillance STN PE-027 will secure the Condenser Air Removal System in support of Auxiliary and Control Building pressure tests. The condenser mechanical vacuum pumps are interlocked with the condenser filtration system discharge dampers such that when the dampers are closed (as desired) the vacuum pumps are deenergized. The subject temporary procedure change will defeat the interlock such that the filtration system can be secured, i.e., dampers closed, while the vacuum pumps remain operable to sustain condenser vacuum. With the dampers closed the noncondensable gases removed by the vacuum pumps will be vented via the local vent to the Turbine Building upstream of condenser radiation monitor GE RE-92 and filter.

Safety Evaluation: Releasing the condenser air via this route does not compromise the plant's safety features since per USAR 9.4.4.1.1 safety design bases, the Turbine Building HVAC Systems serve no safety function except for those dampers and ductwork in the condenser air removal filtration system which are required to provide isolation of the Auxiliary Building. These dampers and associated ductwork are not affected by removing the interlock between them and the vacuum pumps. The dampers will be closed in their safeguards position during the test since the filtration system is secured. The automatic closure signal of these dampers upon a Safety Injection Signal (SIS) has also not been affected by removing the damper vacuum pump interlocks. The alternate condenser gas removal discharge path vents the gases prior to passing by radiation monitor GE RE-92. This monitor closes the blowdown and sample process valves and alarms in the Control Room. There is no Technical Specification requirement for this monitor. Steam Generator blowdown liquid monitor BM RE-25, Steam Generator liquid radioactivity monitor SJ RE-02, blowdown discharge line monitor BM RE-52 all provide indication of a primary to secondary leak and isolation of the blowdown and sample isolation lines. The operability of these monitors and their automatic isolation functions have not been impaired by the subject change. The ability to isolate the blowdown and sample lines and provide an indication of a primary to secondary leak has not been lost nor has the Technical Specification margin of safety defined by Limiting Conditions for Operation 3.3.11 or 3.11.2.3 been reduced.

SAFETY EVALUATION 88-SE-031 Revision: 0

Title: Defeat Of Fire Protection Alarm And Deluge Functions For The Unit Auxiliary Transformer

Description: This Temporary Modification Order (TMO) allows defeating the fire protection alarm and deluge functions for the Unit Auxiliary Transformer. The Unit Auxiliary Transformer is provided with a fire detection and extinguishing system which actuates the automatic suppression system installed in the zone. This system is powered by a non-Class 1E source. An intermittent ground between the power supply non-Class 1E source feed and the detectors above the Unit Auxiliary Transformer is causing spurious alarm flashes. Investigation as to where the ground fault is occurring is not desired while the plant is at power due to the presence of the high voltage. This work is being scheduled for the next opportune time. To clear the spurious alarms the subject fuse in the local control panel is being pulled. This action, in addition to clearing the alarm window, will also defeat the automatic deluge at the transformer. With the detector and automatic deluge defeated, compensatory actions as required by Fire Impairment 88-163, which requires increased fire watch surveillance, provide adequate fire protection compensatory measures. This action prevents the possibility or probability of fire spread. Manual deluge of the zone remains available.

Safety Evaluation: The site fire hazards analysis has evaluated, as required by Appendix R of 10 CFR 50, the potential fire hazards which could damage equipment outside the power block or equipment required for safe shutdown in USAR Appendix 9.5.B. The Unit Auxiliary Transformer is in fire zone 11 and is listed as a non-safety related structure in USAR Table 9.5 B-5. The transformer, as stated in USAR 9.5.B.1, is a structure which is remotely located from safety related structures. A postulated fire within this area does not pose a hazard to structures and systems required for safe shutdown. The zone 11 sensor detects specific ambient temperatures which, upon reaching the alarm setpoint, alarms in the Control Room and automatically activates the deluge valve of the sprinkler system. Configuration of space and fire barriers has not been changed. Defeating the alarm and deluge of zone 11, therefore, does not increase the probability or possibility of previously evaluated accidents or equipment important to safety malfunction occurrence or consequences of these events.

SAFETY EVALUATION 88-SE-032 Revision: 0

Title: Injection Of Chemical For Scale Control In Circulating Water System

Description: This Temporary Modification Order (TMO) allows injection of Betz Powerline WCN01 into the WCGS circulating water system to control condenser scaling and improve plant performance. The chemical will be stored and injected using the acid feed (AX) system which was initially designed for sulfuric acid injection. Powerline WCN01 will be tried for a 6-month period after which its performance will be evaluated. During this time, it will be injected continuously at a rate of 70 gallons per day. During this spring, summer, and fall period, 3 circulating water pumps will be in operation which will provide flows of approximately 500,000 gpm.

Safety Evaluation: Betz Powerline WCN01 is an organic phosphonate liquid which inhibits scale formation by covering the periphery of $CaCO_3$ (Calcium Carbonate) crystals and prevents their expansion. Its pH is 13.4 due to the NaOH (Sodium Hydroxide) added to enhance its storage capabilities. This high pH presents no environmental problems due to the small feed rates and high circulating water flow. This will result in no noticeable circulating water pH changes. The two ingredients which inhibit scaling are the organic phosphonate (1-hydroxyethylidine 1,1-diphosphonic acid, abbreviated HEDP) and modified hydroxylated copolymer (MHC). Scale formation is inhibited by HEDP and MHC is a dispersant which discourages particle attachment to piping. The concentration of chemical will be approximately 1/10,000 of the amount evaluated to cause no mortality in test organisms. Consequently, no adverse environmental impact on Wolf Creek Cooling Lake organisms is anticipated.

SAFETY EVALUATION 88-SE-033 Revision: 0

Title: Temporary Procedure To Allow Operation Of The Letdown Chiller In The Recirculation Mode

Description: This temporary procedure TP-OP-71 allows the operation of the letdown chiller portion of the Boron Thermal Regeneration System (BTRS) to operate without the mode selector switch in borate or dilute. A jumper will be placed in the letdown chiller heat exchanger shell side outlet temperature control valve BG TCV386 control circuit to allow the valve to be opened without the mode selector switch (BG HIS27) being in borate or dilute. This abnormal mode of operation is desired to enable chemistry to obtain a more representative sample of the chiller water without having the BTRS subsystem in the borate or dilute mode. The temporary procedure provides for recirculating the chiller water by running a chiller pump and jumpering open the temperature control valve BG TCV386 to the letdown chiller heat exchanger when the system is in standby. The jumper on BG TCV386 does not cause other interlocked valves within the BTRS subsystem to actuate. This recirculating mode of operation of chiller coolant will not place a load on the letdown chiller heat exchanger as no Reactor Coolant System letdown flow through the heat exchanger will be present since the BTRS is not placed into operation.

Safety Evaluation: The BTRS is a subsystem of the Chemical and Volume Control System (CVCS) which consists of several other subsystems, namely: the Charging, Letdown, and Seal Water System; the Reactor Coolant Purification and Chemistry Control System; and the Reactor Makeup Control System. The BTRS is not part of the CVCS associated with emergency boration charging for the Emergency Core Cooling System, reactor coolant pressure boundary isolation or containment isolation. These portions are safety related and required to function following a Design Basis Accident and to achieve and maintain the plant in a safe shutdown condition. Recirculating the chiller water without BTRS processing does not increase the possibility or probability of equipment important to safety malfunction occurrence since none of this equipment is affected by the use of the procedure. The BTRS has no safety design basis. The BTRS is designed for power generation to allow load follow operations as required by the design load cycle. The BTRS portion of the CVCS is designed and fabricated in accordance with quality group D (augmented) codes and standards. The BTRS is capable of controlling the changes in the reactor coolant boron concentration to compensate for the Xenon transients during load follow operations, without adding makeup for either boration or dilution. A boron dilution accident through the reactor makeup portion of the CVCS has been evaluated in USAR Section 15.4.6. The probability of increasing the occurrence of this accident has not been increased since the temporary procedure does not introduce a BTRS dilution process mode of operation. The consequences of a dilution event has not been increased since the CVCS dilution isolation function and associated isolation valves have not been affected by the temporary procedure.

SAFETY EVALUATION 88-SE-034 Revision: 0

Title: Installation Of A Site Fabricated Coupling In Class 1E Air Conditioning Unit

Description: This Temporary Modification Order (TMO) allows installation of a Site Fabricated coupling in a Class 1E air conditioning (A/C) unit. The filter dryer assembly of Class 1E electrical equipment air conditioning unit SGK05B is on the liquid side of the water cooled condenser. There is a screw at the bottom of the filter housing that applies tension to the 2-filter stack inside the housing. A brass encapsulation consisting of a cap, nipple and coupling is used to cover this tensioning screw and provide a pressure boundary to prevent loss of freon. The coupling has been cracked, thereby allowing loss of freon. The temporary modification consists of fabrication of another coupling using commercial material. Vendor drawing D-9386 specifies a 1 1/2" X 2 1/16" long brass coupling with no other special attributes (grade, schedule, etc.). The coupling will be fabricated in the machine shop from available stock. The fabricated coupling will have at least the same wall thickness as the original and, therefore, will retain the fit, form and function of the original.

Safety Evaluation: The fabricated brass cap will not degrade the operability of the A/C Unit. The A/C Unit will be capable of keeping the safety related Class 1E electrical equipment from being exposed to excessive temperature. Therefore, the margin of safety defined in the bases of Technical Specification 3/4.7.12 is not reduced by this subject change.

SAFETY EVALUATION 88-SE-035 Revision: 0

Title: Use Of Waste Evaporator Condensate Pump Without Certified Material Test Report

Description: The Temporary Modification allows installation of a new replacement Waste Evaporator Concentrate Pump SHB09. This replacement pump was received without its associated Certified Material Test Report (CMTR). This pump is supplied from the same manufacturer as the original pump and is a direct replacement, conforming to fit, form and function. This pump was issued from the warehouse per Conditional Release CR 293. Non-Conformance Report M-1129 has been issued concerning this pump's CMTR. The pump is classified as Special Scope since it serves to process radioactive fluids from the evaporator body of the Waste Evaporator Package.

Safety Evaluation: As a result of this modification, the worst possible accident would be that of a pump failure and a leakage path from the Waste Evaporator Package into the Radwaste Building. This accident is within the bounds of the accident described in USAR 15.7.2 which evaluates for both the failure of the Boron Recycle Holdup Tank and the Primary Evaporator Bottoms Tank. Both of these accidents result in doses well within the guidelines of 10 CFR 100. Therefore, any failure of the replacement pump SHB09 would also fall well within the bounds of USAR 15.7.2.

SAFETY EVALUATION 88-SE-036 Revision: 0

Title: Installation Of MIC Test Coupons In Essential Service Water System

Description: This Temporary Modification Order (TMO) allows the insertion of test coupons in the Essential Service Water (ESW) 30" main supply and return lines in support of the Microbiological Induced Corrosion (MIC) program. The test coupons will be inserted into the 1" drain lines using a dip rod assembly with the drain line closed off by a thermocouple type connector. The ESW piping is Seismic Category I and is rated to 150 pounds. The commercial pipe fittings/connectors employed in the closure of the drain lines retain this rating. The added weight of 2 pounds, due to the test coupon dip rod assembly off of the drain line, does not significantly alter the seismicity of the ESW line. Failure of the 1' Schedule 160 (ID .815") drain line or thermocouple type connector cap off of the 30" (ID 29.25") main header would result in an insignificant flow loss. The small test coupon, about 3/8" X 2" X 1/8", will not diminish the 30" header flow.

Safety Evaluation: In the unlikely event the coupon became detached from the dip rod, it would represent a small piece of debris which would not be capable of producing any adverse mechanical effects. The two coupons (one in each header) in the return line, should they get free, would probably get swept out to the Ultimate Heat Sink (UHS) and not affect any plant equipment. The coupons in the supply header are downstream of the ESW pumps and system strainers and could be swept into the ESW distribution piping and associated heat exchangers in the power block. The test coupon is not large enough to block a cooling line to any of the ESW supplied heat exchangers, coolers or condensers. The line sizes to most of the safety related coolers are 4" or greater and the smallest ESW supply line is 2" which is to the Fuel Pool Cooling Pump Room Cooler. The operability of the Containment Cooler and Component Cooling Water valves, which are manipulated during a Safety Injection Signal, would not be impaired by the small debris piece the test coupon would represent. The tube sheets of the ESW supplied heat exchangers would not be introduced to a significant blockage should the coupon break free. If the coupon become lodged in a tube and caused fretting wear and subsequent breach, the resultant effects would not degrade the heat exchangers performance. Heat exchanger tube failure or insignificant blockage is not a new or different type of failure or malfunction occurrence considered from previous analysis. The possibility of the coupon degrading the ESW pump discharge pressure or strainer differential pressure instrumentation and associated alarms is not possible since the coupon would have to go against the flow and then through the strainer. The 1/2" instrument line to the ESW Radiation Monitors would only be slightly blocked should the coupon lodge in this line. The monitor would still receive a representative stream of the ESW water. This monitor provides system contamination indication only and is not Technical Specification related. It is concluded in view of previous analysis that the insertion of the test coupons in the ESW main supply and return headers does not increase the probability or possibility of the occurrence of equipment malfunction or failure. The subsequent consequences of coupon debris would not be consequential nor would it introduce a new or different type of malfunction than those which have been evaluated.

SAFETY EVALUATION 88-SE-037 Revision: 0

Title: Evaluation Of Organization Change

Description: The WCNOC Organization Chart, USAR Figure 13.1-2 and Technical Specification Figure 6.2-1, reflects that the Manager Facilities and Modifications reports to the Vice President Engineering and Technical Services. Recent organizational and administrative changes have now designated the position of Manager Facilities and Modifications to report to the newly created position of Manager Maintenance and Modifications who reports to the Plant Manager who reports to the Vice President Nuclear Operations. Step 5.1 of procedure KP2-500 Rev. 4 has been changed to incorporate these recent organizational and administrative changes. This revised organizational structure is different than that described in the USAR; hence, the reason for this safety evaluation. The other changes made in the procedure do not differ from the USAR analysis or description. These changes bring the procedure into compliance with the USAR since USAR Section 13.2.2.8.1 states that job specifications will be prepared for middle and upper level professional positions. The other changes made to the procedure accomplish this requirement.

Safety Evaluation: Changing the line of responsibility of the Facilities and Modifications Group is an administrative change and does not diminish the quality of work performed or qualification of the professionals in the group. The Manager Facilities and Modifications and the Manager Maintenance and Modifications are technically qualified to fulfill these positions. The change to the subject procedure in Step 5.1 does not diminish the level of expertise at WCNOC. The Shift Crew composition, qualifications and training have not been affected by the changes. The procedure changes are administrative in nature and do not involve any unreviewed safety questions.

SAFETY EVALUATION 88-SE-038 Revision: 0

Title: Evaluation Of Organizational Changes

Description: The WCNOC Organization Chart, USAR Figure 13.1-2 and Technical Specification Figure 6.2-1 have title positions of Superintendent shown. These position titles of Superintendent have been changed recently to Manager. Additionally, the organizational charts reflect that the Manager Facilities and Modifications reports to the Vice President Engineering and Technical Services. The recent organizational and administrative changes have now designated the position of Manager Facilities and Modifications to report to the newly created position of Manager Maintenance and Modifications who reports to the Plant Manager who reports to the Vice President Nuclear Operations. The charts also reflect the positions of Superintendent of Plant Support and of Regulatory, Quality, and Administrative Services. These two positions have now been combined under a single position called Manager of Plant Support, reporting to the Plant Manager. Procedure ADM 07-100 establishes the process for preparation, review, approval and distribution of procedures for Wolf Creek Generating Station (WCGS). The changes made to this procedure incorporated these recent organizational, administrative and title changes. These changes in the organizational structure and position titles are different than that described in the USAR.

Safety Evaluation: Changing the line of responsibility of the Facilities and Modifications Group, creating the Manager of Maintenance and Modifications position, and changing other titles from Superintendent to Manager are administrative changes and do not reduce the quality of work performed or level of expertise of company personnel. The administrative controls in Section 6.0 of the Technical Specifications are not reduced by the subject changes. The qualification and training of the Unit Staff, Plant Safety Review Committee function, procedure and program controls addressed in this section of the Technical Specifications remain intact and within compliance. The Shift Crew composition, qualifications and training have not been affected by the subject changes. In conclusion, the subject procedure changes are administrative in nature and do not involve any unreviewed safety questions.

SAFETY EVALUATION 88-SE-039 Revision: 0

Title: Procedure Change To Allow Use Of Temporary DP Cell In Safety Injection System Test Line

Description: The changes to the surveillance procedure STS PE-019D, Revision 0 are mainly enhancements or editorial changes/corrections and, as such, do not affect or change the USAR analysis or Technical Specifications. The only change which does create a change from its description (Figure 6.3-1 Sheet 2) in the USAR is when the procedure allows for the incorporation of the temporary differential pressure (DP) cell at flow element EM FI-928 (Safety Injection System pump to Reactor Coolant System (RCS) hot leg 4 test line). The optional installation of the additional flow indicator provides for redundancy, greater accuracy, and more sensitive readings during testing. The option of using the DP cell across flow element EM FI-928 to record leakage from the RCS pressure isolation valves in support of surveillance testing (per Technical Specification 4.4.6.2.2) provides for equivalent test parameter indication in measuring leakage flows. By measuring the DP across element taps, a flow correlation can be achieved. The accuracy of the DP cell meets or is better than the accuracy of the permanently installed element.

Safety Evaluation: During surveillance testing, the 3/4" Safety Injection (SI) test line is in use as designed and the potential for an SI exists since the plant is in Mode 3. This occurrence would not have any affect on the SI function or performance since the DP cell does not interface with any SI equipment. The DP cell and associated instrument connections maintain system pressure rating. The use of the DP cell will not increase the consequences of previously evaluated accidents since accuracy and system tubing pressure rating are maintained.

SAFETY EVALUATION 88-SE-040 Revision: 0

Title: Relocation Of Chlorine Monitor Air Sample Pumps

Description: This Temporary Modification Order (TMO) will allow relocation of the Chlorine Monitor air sample pumps. Redundant chlorine monitors are located in the Control Building supply system ductwork, downstream of the control room filter adsorber. These chlorine monitors, upon detection of chlorine, initiate isolation of the Control Building Normal Supply and Exhaust Systems. These monitors have been causing spurious Control Room Ventilation Isolation Signals (CRVIS). As a result of an engineering review of these events, it has been concluded that the cause of these spurious occurrences is due, in part, to the vibrations of the WISA (air) pump inside the detector's wall mounted cabinet. The temporary modification will idle these pumps and mount identical pumps outside the cabinets in the immediate area.

Safety Evaluation: The new outside pump mounting has been approved and seismically evaluated. The outside auxiliary pump support maintains the seismic design of the system. The existing monitor cabinet is a NEMA type 12 enclosure which provides a degree of protection against dust, falling dirt and dripping noncorrosive liquids. The liquid tight seal at the cabinet and outside pump provide a better seal than that of a NEMA 12 enclosure. The qualified spare WISA pump will be used as the auxiliary pump for Chlorine Monitor GK AITS-2. The existing WISA pump for Chlorine Monitor GK AITS-3 will become the auxiliary pump for this unit. A non-quality (identical) WISA pump will be put into the GK AITS-3 cabinet to maintain cabinet seismic qualification. The auxiliary WISA pump outside of the NEMA 12 enclosure will not result in a reduction of physical protection to the electrical or mechanical components in the pump since the pump components are located inside a metal enclosure which provides equivalent protection. The auxiliary WISA pump is located within 5' of the existing cabinet; therefore, the same environmental conditions are experienced by both. The qualified tygon air sampling tubing and Swagelok fittings from the auxiliary pump maintain system design since tygon tubing is used by normal design to provide sample air to the cabinet. An assessment of this modification has concluded that system design is exceeded or is maintained and the system malfunction occurrence has not been increased and, in fact, has been reduced by the changes.

SAFETY EVALUATION 88-SE-041 Revision: 0

Title: Operation With One Pressurizer Spray Valve Closed

Description: The subject temporary procedure change for Procedure GEN 00-004, Revision 10, is to support continued plant operations with one pressurizer (PZR) spray valve BB PCV-455B and its associated manual throttle valve BB V082 closed. USAR 5.4.10.3.4 discusses the pressurizer spray function and states that two separate, automatically controlled spray valves with remote manual overrides are used to initiate pressurizer spray. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow through both spray lines to reduce thermal stresses and thermal shock when the spray valves open and to help maintain uniform water chemistry and temperature in the pressurizer. Spray flow is modulated by air operated valves automatically or manually controlled from the Control Room. The spray valves of the PZR Control System help maintain or restore the PZR pressure to the design pressure +-35 psi (within reactor trip and relief and safety valve actuation set point limits) following normal operation transients. The power operated relief valves (PORV's) limit system pressure for large positive pressure transients. These PORV's provide a safety function to limit system pressure.

Safety Evaluation: The closing of one redundant spray valve and its associated manual throttle valve does not reduce the capability of the Pressurizer Control System to maintain and control PZR pressure nor does it compromise the plant's nuclear safety features. The PZR Pressure Control System, which include the spray valves, is part of the plant control systems not required for safety. USAR Section 7.7 discusses the design objective, description, analysis and failures of plant control systems not required for safety. In the analysis section of the Plant Control Systems (USAR 7.7.2) it states that the plant control systems will prevent an undesirable condition in the operation of the plant that, if reached, is protected by reactor trip. Worst case failure modes of the plant control systems are postulated in the analysis of off design operational transients and accidents covered in USAR Chapter 15.0. These analyses show that a reactor trip setpoint is reached in time to protect the health and safety of the public under those postulated incidents. The closure of one spray valve and associated manual throttle valve does not create a different type of malfunction or accident than previously evaluated since the USAR analysis has evaluated the complete loss of the PZR spray function, the worst case failure. Isolation of the one spray line does not increase the effects of thermal stress and thermal shock to the Reactor Coolant System piping. The remaining bypass line provides adequate warming of the spray lines to prevent thermal shock to the line following spray actuation.

SAFETY EVALUATION 88-SE-043 Revision: 0

Title: Addition Of Microbiological Induced Corrosion Test Coupon

Description: This Temporary Modification Order (TMO), Revision 1, is to add an additional test coupon in support of the Microbiological Induced Corrosion (MIC) program. The added test coupon will be inserted into the 1" drain line off of the 6" discharge line from Chiller Pump B of the Boron Thermal Regeneration System (BTRS) using a dip rod assembly. The Chiller coolant is a closed coolant system which takes heat from the Reactor Coolant System Letdown flow, during BTRS operation via the letdown Chiller Heat Exchanger, and rejects it to the Service Water System via the Chiller Unit. Revision 0 of this temporary modification was addressed by Safety Evaluation 88-SE-036.

Safety Evaluation: The BTRS is a subsystem of the Chemical and Volume Control System (CVCS). It is not a part of the CVCS associated with emergency boration charging for Emergency Core Cooling System, reactor coolant pressure boundary isolation or containment isolation. These portions are safety related and are required to function following a Design Basis Accident to achieve and maintain the plant in a safe shutdown condition. The BTRS has no safety design basis. The BTRS is designed for power generation to allow load follow operations as required by the design load cycle. The BTRS portion of the CVCS is designed and fabricated in accordance with Quality Group D (augmented) codes and standards. The BTRS Chiller pump discharge line is not Seismic Category I nor II/I qualified. It is an ANSI B31.1 line rated to 150 lbs. The commercial test coupon pipe fittings/connectors employed in the closure of the drain line after insertion retain this rating. The added weight of 2 lbs., due to the dip rod assembly off of the drain line, does not significantly alter the mechanical integrity of the line. The small test coupon, about 3/8" X 2" X 1/8", will not diminish the Chiller coolant flow. In the unlikely event the coupon became detached from the dip rod, it would represent a small piece of debris which would not be capable of producing adverse affects on any safety related equipment. The BTRS Chiller coolant is under low pressure, approximately 90 psig, and low temperature, approximately 46^{O}F. The fretting wear this small piece of debris could cause to the shell side of the Letdown Chiller Heat Exchanger, where the flow velocity is very low and the pressure and temperature environment is not severe, is deemed to be insignificant and minimal. Heat exchanger tube failure as a result of this debris piece is deemed not credible; however, should it occur it would not be considered a different type of malfunction or occurrence. Heat exchanger tube failure is within the analysis of system design.

SAFETY EVALUATION 88-SE-044 Revision: 0

Title: Disabling Radwaste Building Ventilation Wide Range Gas Monitor Sample Isolation

Description: This Temporary Modification Order (TMO) prevents the Radwaste Building Ventilation System wide range gas monitors from isolating their particulate and iodine sample paths upon receipt of a high activity signal. The flow path provides sampling of ventilation exhaust for all parts of the building structure and components within the building and the discharge path for the Waste Gas Decay Tank release line.

Safety Evaluation: The Radwaste Building Ventilation System functions to provide a suitable atmosphere for equipment and personnel. This system serves no safety function.

SAFETY EVALUATION 88-SE-046 Revision: 0

Title: Installation Of Temporary Demineralizer System During System Maintenance

Description: This Temporary Modification Order (TMO) allows installation of a Temporary Demineralizer System during Demineralizer System maintenance. The Demineralized Water Storage and Transfer System (DWSTS) stores water for use upon demand for makeup within the plant. The DWSTS receives filtered and demineralized water from the Demineralized Water Makeup System (DWMS). Due to the numerous maintenance work activities, the DWMS needs to be taken out of service. While the DWMS is out of service, temporary Ecolochem portable water treatment equipment will be utilized to provide demineralized water. Ecolochem equipment will be tied into the DWMS in the Shop Building east of the Turbine Building. The existing caustic and acid day tanks and associated pumps will be removed and the remaining parts of the system restored with Nuclear Plant Engineering design approval. Ecolochem water treatment equipment does not defeat the capability to adequately treat the raw water of the DWMS in meeting chemical specifications for demineralized water as given in USAR Table 9.2-16.

Safety Evaluation: The DWMS does not interfere with any equipment important to safety and this system is not needed to shut down the reactor. The DWMS is not Class 1E qualified or powered, serves no safety function and contains no equipment important to safety. All previously evaluated accidents evaluated in USAR Section 15.0 do not take any credit for operability of the DWMS. The DWSTS is located in the Shop Building outside of the power block. The quality of demineralized water has not been affected by the subject change and the cleanliness requirements of plant systems which use demineralized water will be retained. The probability and possibility of previously evaluated accidents and malfunctions of equipment important to safety has not been increased by this temporary change made to the Water Treatment System in the Shop Building.

SAFETY EVALUATION 88-SE-047 Revision: 0 and 1

Title: Lack Of Support Brackets On Diesel Generator Turbocharger Piping

Description: This Temporary Modification Order (TMO) allows the installation of field fabricated brackets on the Diesel Generator (D/G) turbocharger piping. Two brackets on the turbocharger water piping for each have never been installed. The Engineering Report for the seismic documentation package for the D/Gs supplied by Colt Industries/Fairbanks Morse reflects that since this piping is of short lengths and is directly attached to the massive diesel engine, a rigid connection is made and no dynamic magnification of vibratory input is expected and no seismic analysis is considered necessary. Additionally, the Seismic Calculations Skid Mounted Piping Report states that pipe runs with a nominal outside diameter greater than 3" are analyzed in detail. The analysis for the small pipes shows that small pipes will not resonate and that the stresses will not exceed allowable values. The missing brackets are on 1" water piping which spans approximately 36" and weighs less than 10 lbs. The function of the brackets is to limit vibration during normal operation.

Safety Evaluation: The revision to this modification and subsequent safety evaluation is to allow the bracket on the east side of each D/G to be customized as symmetry is not present. The east side bracket will still be made from qualified 1/4" angle iron but to the design configuration needed for the application. The overall length of the stiffener needs to be trimmed to suit field location of pipe. This bracket will restore the integrity of the water piping to design and maintain the reliability of the D/G; therefore, no increase in the probability of malfunction occurrence will be incurred. The installation of the bracket on each D/G will restore or establish compliance with the spare parts drawing, which is the only drawing showing these brackets. These brackets are considered for vibrational support only and are not considered for seismic analysis in view of the referenced reports. The field fabricated brackets are deemed by engineering judgement to satisfactorily secure the small diameter and span of water piping of the D/Gs, as would the original brackets supplied by the vendor. This cooling water line is not a high temperature or pressure line and experiences no significant operating design stresses. The field fabricated brackets are not expected to introduce a different type of malfunction occurrence or increase the consequences of previously evaluated malfunction occurrence since they provide the same securing ability as vendor supplied brackets.

SAFETY EVALUATION 88-SE-048 Revision: 0

Title: Temporary Patch On Steam Generator Feed Pump Bypass Line

Description: This Temporary Modification Order (TMO) allows installation of a temporary patch on the Steam Generator Feed Pump (SGFP) "B" minimum flow line to the Condenser. This line has developed a leak at one of the elbows downstream of AE FV1B. This flow path is used to allow a minimum flow of water to pass through the SGFP so not to allow the pumps to overheat during low discharge flow rates as seen during startup. With valve AE FV1B leaking by, there is flow through a small hole in the elbow. Application of a patch will stop the leakage.

Safety Evaluation: The piping is non-safety related, non-seismic and will not affect any equipment important to safety. The temporary soft patch will stop the leakage and reduce the degradation of the elbow caused by the dynamic forces of the fluid traveling through the leak. The addition of this patch will not affect any of the functions of this system. Section 10.3.6.1 of the USAR lists the materials and standards applicable to this system. The line is a carbon steel Schedule 100, ANSI B31.3 Critical Service, non-seismic line. The Critical Service designation of this line means that the line has been examined beyond the ANSI B31.1 code requirement. This soft patch will not restore the system to these standards, but will stop the leakage from the elbow and provide personnel safety. Loss of feedwater and failure of one Steam Generator Feed Pump have been evaluated in the USAR.

SAFETY EVALUATION 88-SE-049 Revision:

Title: Temporary Procedure To Filter Diesel Generator Stored Fuel Oil

Description: This temporary procedure was developed to filter the particulates from the fuel oil of the emergency fuel oil storage tank of the Emergency Diesel Engine Fuel Oil Storage and Transfer System (EDEFSTS) of each diesel engine. The procedure isolates the storage tank from the day tank and connects the storage-to-day-tank transfer header to the oil fill line of the storage tank by a temporary hose with a filter in it. The filter used will be compatible for a fuel oil medium. The control circuitry of the transfer pump is jumpered to defeat the high level trip of the transfer pump and allow the transfer pump to be energized to pump fuel oil through the filtered recirculation path. The high and low level alarms of the day tank are also defeated, however level indication of the day tank remains.

Safety Evaluation: Assessment of the procedure actions reflects that system chemistry is maintained, i.e., Class C cleanliness maintained, correct valve alignment achieved, and that plant equipment and systems restored. The actions of the procedure are localized and isolated to the EDEFSTS of each diesel engine. Existing system strainers and filters are outside of the created recirculation path; therefore, their function is not affected by the procedure. The probability or possibility of increasing the malfunction occurrence of the diesel generator has not been incurred by the subject procedure actions and actually may be decreased. The fire protection features of the Diesel Generator Building have not been degraded. The temporary recirculation hose is employed outside the Diesel Generator Building attached to non-seismic connections. The failure of this hose would not introduce a different type accident or increase the consequences of previously evaluated accidents since normal tanker truck off loading hazards at the storage tank location are foreseen as enveloping the otential hazard consequences of the temporary recirculation hose. Upon completion of the filtering evolution, the verification of restoration of the system alignment ensures system reliability. A fuel oil compatible filter will be used by the procedure and its utilization is not to be considered or construed as a qualification process of the filtered oil. This evaluation has evaluated the procedure as written in accordance with the 10 CFR 50.59 rule and has found it to have no adverse affect on the filtered fuel oil. There is no increase in the malfunction probability of equipment important to safety, no creation of a different type of malfunction or accident, and no increase in the consequences thereof.

SAFETY EVALUATION 88-SE-050 Revision: 0

Title: Setpoint Changes Of Process Radiation Monitors

Description: This USAR Change Request will change the setpoint values of ten Airborne Process Radiation Monitors listed in USAR Table 11.5-3 to agree with the revised Technical Specifications setpoints. The controlling isotope has been changed from Kr-85 to Xe-133.

Safety Evaluation: The controlling isotope listed for these monitors in USAR Table 11.5-3 has been changed from Kr-85 to Xe-133 as Xe-133 comprises 85% of total noble gas activity expected. (See Offsite Dose Calculation Manual (ODCM), Section 3.1.2). This change is in agreement with the methodology for the setpoint calculation of these monitors as specified in the ODCM and Power Generation Design Basis One in USAR Section 11.5.1.2. The changes made to USAR Tables 11.5-3 and 4 and 12.3-3 do not increase the consequences of previous USAR accidents since the setpoint changes are in compliance with Technical Specifications and the ODCM. The changes made to these USAR tables will align this document with the ODCM and Technical Specifications.

SAFETY EVALUATION 88-SE-051 Revision: 0

Title: Removal Of Power To Safety Injection Accumulator Level Transmitter

Description: This modification will lift the leads of electrical scheme 6EPI06FA at terminal block A points 10 and 11. This will remove power to EP LT-955, Safety Injection Accumulator "C" level transmitter. This is due to the fact that the transmitter is causing an errant high level indication in the alarm condition. The modification will remove the errant alarm from the control room while the instrument loop is reworked.

Safety Evaluation: The accumulator has a redundant level indicator, EP LT-954, which also alarms in the common alarm window. The alarm ability and level indication provided by EP LT-954 is not affected by this modification. Compliance with Technical Specification 3/4.5.1 which applies to accumulator operability can be verified using EP LT-954. This modification will not affect any other instruments or equipment.

SAFETY EVALUATION 88-SE-052 Revision: 0

Title: Installation Of Cables Through Containment Penetrations

Description: This evaluation applies to a checklist procedure which addresses the routing of cables through and foaming of Containment penetrations P-34 and P-65, and Auxiliary Building penetrations OP131W1296 and OP131W1297 in support of outage Inservice Inspection (ISI) related work activities. The cables will then exit the Auxiliary Building through OP131W1296 and/or OP131W1297 (which are currently grouted) and terminate at the vendor's trailer. These penetrations will also be foamed when in use. Dow Corning 3-6548 Silicone RTV foam will be used. The foam expands and dries following application, and will provide an adhesive seal and a fire barrier. This procedure is used when the plant is in Mode 5 or 6.

Safety Evaluation: The foam has a 3-hour fire rating. Fire impairment controls in accordance with the WCGS Fire Protection Program are to be implemented when the subject barriers are removed. The foam and cables will be removed following testing and the penetrations restored. Flanges will be installed on P-34 and P-65, and then local leak rate tested per 10 CFR 50 Appendix J to verify operability. The grout in OP131W1296 and OP131W1297 will be replaced. Core alterations and fuel movement in Containment will not be permitted while P-34 or P-65 is open between the Containment and Auxiliary Building to comply with Technical Specification 3/4.9.4. Containment closure (required when the plant is in Mode 5 or 6) is maintained by the procedure.

SAFETY EVALUATION 88-SE-053 Revision: 0

Title: Revision Of Offsite Dose Calculation Manual to Reflect Correct Plant Instruments

Description: The changes made to the Offsite Dose Calculation Manual (ODCM) by this revision are corrections to paragraph 3.1.4 to reflect actual plant design and system features. Paragraph 3.1.4 of the ODCM is titled "Alert Alarm Setpoint Calculations." The first sentence in this paragraph of the ODCM states:

> USAR Table 11.5-4 states that the alert alarm for the Plant Unit Vent Monitor (GT RE-21), Radwaste Building Exhaust Monitor (GH RE-10), Auxiliary Ventilation Exhaust Monitor (GL RE-60), and Access Control Area Ventilation Monitor (GK RE-41) is set to alert operators to that average concentration which if maintained for a full year would result in the 10 CFR 50 Appendix I annual dose guidelines being reached.

This statement is only partially correct since the Auxiliary Ventilation Exhaust Monitor, GL RE-60, and the Access Control Area Ventilation Exhaust Monitor GK RE-41, are not Airborne Effluent Radioactivity Monitors (AERM's) and need to be deleted from the ODCM.

Safety Evaluation: This discrepancy is being corrected by USAR Change Request (Safety Evaluation 88-SE-050). Changing the ODCM into a more valid and accurate document will improve plant operation. The margin of safety has not been reduced by this ODCM paragraph correction since no monitor setpoints have been changed as a result of this ODCM change.

SAFETY EVALUATION 88-SE-054 Revision: 0

Title: Removal Of One Air Trap From Emergency Diesel Generator Starting Air System

Description: This temporary modification removes one air trap (liquid drainer) from the non-safety related portion of the Emergency Diesel Generator Starting Air System.

Safety Evaluation: Capping the 3/4" line to air trap KJAT4A on the non-safety related portion of the Emergency Diesel Generator Starting Air System (EDGSAS) has no effect on the quality of air or capacity of air to start the emergency diesel generator. The spare parts to restore the trap to normal service are not available. The 3/4" tie-in line to the trap will be cut and capped by the actions on this modification. This action will prevent continuous blowdown of air through the trap to the drain until it is returned to service. The function of the trap (liquid drainer as described in the technical manual) is to automatically drain moisture to the floor drain upon accumulation in the trap to the float ball trip setpoint. Its design does not provide or facilitate a continuous blowdown function of starting air supply to the air dryers piping header, moisture accumulation level or alarm indication. Its failure during normal system operation would go undetected until discovery. This failure is foreseen as encompassing any and all potential consequences which may be possible by this modification. Trap KJAT2A upstream of KJAT4A remains operable; therefore, the ability to drain moisture out of the starting air system before it enters the air dryers still exists. As a precautionary measure in compensating for the loss of function of the trap, the isolation valve KJV886B shall be opened once every day to blowdown any moisture which accumulates at the filter. This action will lessen any additional loading to the air dryers which may be caused by the loss of the one trap.

SAFETY EVALUATION 88-SE-055 Revision: 0

Title: Temporary Installation Of Equivalent Gasket On Pressure Regulator For Auxiliary Feedwater Control Valve

Description: This Temporary Modification Order (TMO) was originated to allow the temporary installation of a safety related asbestos paper gasket in place of a metallic gasket on the Nitrogen Pressure Regulator for one Auxiliary Feedwater Control Valve.

Safety Evaluation: The paper gasket is functionally equivalent to the metallic gasket. The gasket will be leak tested following installation. Use of a paper gasket for a period of 60 days or less will not degrade the operation of the valve. This time period will allow the correct gasket to be procured. The valve will not be affected by high energy line break in the auxiliary feedwater pipe chase since the only postulated break in this area is a 3/4" nozzle break on the accumulator which causes extreme humidity or temperature environments (USAR Table 3.6-4 Sheet 25).

SAFETY EVALUATION 88-SE-056 Revision: 0

Title: Installation Of Temporary Pressure Gauge For Pressurizer Relief Tank

Description: This temporary modification installs a pressure gauge on the non-safety related gaseous Radwaste header outside of Containment to compensate for the loss of the Pressurizer Relief Tank (PRT) pressure transmitter signal.

Safety Evaluation: This pressure instrument is not controlled or addressed by any Technical Specification. This pressure instrument provides for Control Room PRT pressure indication and an alarm at 6 psig. Normal system pressure is 3 psig. To compensate for the loss of Pressure Transmitter BBPT469, an alternate pressure gauge has been installed on the PRT to gaseous radwaste header, on the 3/4" test connection on valve BBV243. The alternate gauge is installed on the non-safety related portion of the line outside containment. PRT pressure is read from this alternate local gauge every 4 hours by opening the Containment isolation valves between the PRT and the alternate gauge. The alternate gauge and monitoring action provides a commensurate response until BBPT469 can be restored. PRT temperature and level indication and alarms are still available to the Control Room operators. Overpressure protection of the PRT remains. Operability of the PRT is maintained and is unaffected by this temporary change.

SAFETY EVALUATION 88-SE-057 Revision: 0

Title: Procedure Change Involving Testing of Fire Protection Hose Stations

Description: Procedure STN FP-431 describes the method of performing valve operability checks for fire hose stations accessible during plant operation and hydrostatic tests for fire hose station hoses. This test should be performed at least once every three (3) years. This test was performed beginning November 11, 1984, and was repeated in May, 1988 except for the 13 containment fire hose stations. The temporary change to this procedure addresses deleting the valve operability check and water flow for the 13 hose stations inside containment until the containment fire protection header can be filled. The next scheduled available time these conditions will exist will be during Refueling III Outage scheduled to begin October 6, 1988.

Safety Evaluation: The capability to fight a containment fire remains readily available and no decrease in fire protection capability has resulted. The fire protection hose station locations are such that all accessible areas of the containment are adequately covered by at least one hose station. A fire at any hose station may be extinguished by using an adjacent hose station. The USAR states that an extra length of hose can be added to the adjacent station if required. These system design features are not affected by this temporary procedure change. It should be noted that the hoses for the containment stations are not allowed inside containment when the plant is at power. The change also does not relate to or affect the operability of the containment's fire detection and alarm system, fixed manually charged sprinkler system over the cable trays, portable fire extinguishers, or the oil collection system for the Reactor Coolant Pumps.

SAFETY EVALUATION 88-SE-057 Revision: 1

Title: Procedure Change Involving Testing Of Fire Protection Hose Stations
Description: This temporary procedure change allows delaying the conduct of the 3 year operability test, for the 13 fire hose stations inside of containment, until the containment fire protection header is filled during Refueling Outage III. Revision 1 allows extending the time past September 15, 1988.

Safety Evaluation: The other 56 hose stations have been successfully tested, so these 13 are not expected to fail. The 3 year periodicity was arbitrarily chosen; it is not a code requirement. The extension results in being less than 11 months past the 3 year period. Other containment fire protection equipment is not affected.

SAFETY EVALUATION 88-SE-059 Revision: 0

Title: Change To Containment Integrated Leak Rate Test Procedure

Description: This change to the containment Integrated Leak Rate Test Procedure includes placing 24 Resistance Temperature Detector (RTD) elements on the containment spray headers in containment. This procedure change activity introduces a II/I concern which has not been evaluated, hence the reason for this written safety evaluation.

Safety Evaluation: Each RTD weighs less than one pound. The evaluation determines that these small objects do not present a significant missile hazard should they become dislodged and fall. The reactor vessel head will not be removed during the time the RTD's are in place. The presence of the RTD elements on the spray piping when the Reactor Coolant System is depressurized and the plant is in Mode 5 or 6 does not introduce any significant II/I hazard concerns.

SAFETY EVALUATION 88-SE-060 Revision: 0

Title: Installation Of A Temporary Breathing Air Compressor Skid

Description: A redundant breathing air compressor skid, CKB201B, is being temporarily installed into the existing Breathing Air System to provide additional backup system capability. The added compressor skid will be located in the Turbine Building communications corridor on the 2000' elevation level next to the existing breathing air compressor skid "A". Air compressor skid "B" will need supply and return water from the Central Chilled Water System.

Safety Evaluation: This system serves no safety function and failure of the system does not affect the safe shutdown of the plant, (USAR 9.4.10.1.1). Thus, the increased loading on the Chilled Water System will have no deleterious effect on any safety related component or system. The air compressor skid also needs a supply of domestic water. The Domestic Water System serves no safety function and has no safety design basis, (USAR 9.2.4). Supply water from the Domestic Water System will not affect any safety related component or system. The tie-in to the Central Chilled and Domestic Water Systems will be from the lines which provide these waters to compressor skid "A". Power to compressor skid "B" will be supplied from the non-vital, non-safety related, non-class 1E load center. The added piping and power feed to the compressor does not introduce a II/I consideration since the runs are confined to the Turbine Building. The Turbine Building does not contain any safety related systems or components. The added air compressor skid will be located near the high pressure N_2 supply bottles which supply N_2 to the Main Steam Isolation Valves (MSIV) and the Feedwater Isolation Valves (FWIV). The valve actuators on the MSIVs and FWIVs have two independent N_2 accumulators, each capable of closing the valve in the event of an isolation signal. Each accumulator is pre-charged and the N_2 supply valve, KH-V130, is closed during normal plant operation. The accumulators and their isolation valves are safety related and are in the Auxiliary Building, but the N_2 supply piping and valve KH-V130 are not safety-related and are in the Turbine Building. Failure of the non-safety related portion of the N_2 supply piping in the Turbine Building has been evaluated and this failure does not affect the operability of the MSIVs or FWIVs. The added compressor is protected from overpressurization by relief valves. The added compressor skid does not represent a potential source of missiles which need to be postulated by design (refer to USAR 3.5.1.1, internal missile selection conditions) since the design pressure of the compressor is less than 275 psig and 120% overspeed occurrence failure is highly unlikely since the motor is powered off the utility grid which is maintained at 60 Hz. The Auxiliary Building provides protection against missiles from striking essential components and its integrity has not been affected by the added compressor.

SAFETY EVALUATION 88-SE-061 Revision: 0

Title: Installation Of Temporary Shielding
Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 1 Resistance Temperature Detector (RTD) Isolation valves and piping.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-062 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 1 RTD hot leg and cold leg manifold pipes.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-063 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 1 to the Reactor Coolant Drain Tank Valves BB V008, BB V009 and line BB-20-BCA-2.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-064 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 1 Steam Generator Primary Side Drain Valve BB V265 and line 230-BCB-3/8.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-065 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 2 RTD Isolation Valves and piping.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-066 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 2 RTD hot leg and cold leg manifold pipes.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 5, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-067 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Valve BB V266, Loop 2 Steam Generator Primary Side Drain and pipe 231-BCB-3/8.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-068 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Valves BB V028 and BB V029, Loop 2 Valves to the Reactor Coolant Drain Tank, and line BB-37-BCA-2.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-069 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 3 RTD Isolation Valves and piping.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-070 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 3 RTD hot leg and cold leg manifold pipes.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-071 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Valve BB V267 Loop 3 Steam Generator Primary Side Drain, and line 232-BCB-3/8.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-072 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Line BB-54-BCA-3.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-073 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 4 RTD Isolation Valves and piping.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-074 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Loop 4 RTD hot leg and cold leg manifold pipes.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-075 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Valve BB V268, Loop 4 Steam Generator Primary Side Drain, and line 233-BCB-3/8.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-076 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on Reactor Coolant System Valve BB V065, Loop 4 Excess Letdown Heat Exchanger Isolation, and line BB-74-BCA-2.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-077 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed to shield hot spots on the regenerative heat exchanger and associated piping.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-078 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed around the reactor vessel head while it is in its storage area.

Safety Evaluation: Analysis of the temporary shielding shows that, during Modes 5 and 6, it will not affect the integrity of the system's pressure boundary or seismic rating.

SAFETY EVALUATION 88-SE-079 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed slightly above the reactor vessel flange mating surface during stud hole repair if required.

Safety Evaluation: The shielding would be supported by the polar crane. Drop of the shielding would be less severe than dropping the vessel head.

SAFETY EVALUATION 88-SE-080 Revision: 0

Title: Installation Of Temporary Shielding

Description: Temporary shielding will be placed on the pressurizer spray valves and associated piping.

Safety Evaluation: The shielding is sized so as to not affect structural integrity of the system. A concurrent seismic event was not considered. However, loss of structural integrity of the pressurizer spray piping is acceptable in Modes 5 and 6.

SAFETY EVALUATION 88-SE-081 Revision: 0

Title: Installation Of Temporary Alternate Fire Pump

Description: This Temporary Modification Order (TMO) allows installation of a temporary alternate fire pump. The Fire Protection System (FPS) is described in USAR Section 9.5. The major components of the water supplied fire protection system are: two fire pumps (one electric motor-driven, the other diesel engine driven), one jockey pump for system pressure maintenance, distribution mains, fire hydrants with hose houses, stand pipes and sprinklers. Currently the electric motor-driven pump is out of service and an alternate pump of equal capability is being hard piped into the testing header at the Circulating Water Screen House (CWSH). The Fire Protection Manual requires 2 fire pumps to remain operable when the plant is in Modes 1 through 4. With one pump out of service, the manual requires the out of service pump be returned to service within 7 days or an alternate pump of equal or greater capacity and pressure shall be provided. This modification provides for the alternate pump and piping to support the fire protection needs of the plant in accordance with the Fire Protection Manual. The alternate pump and piping incorporated by this modification maintain system design and function. The alternate pump is a diesel-powered skid mounted unit which is rated at 3500 gpm at 350 feet (151 psia) of head. The pump skid will be located next to the CWSH to take suction from the lake. The discharge will be hard piped with 8" Schedule 40 or better carbon steel piping installed and supported per ANSI B31.1.

Safety Evaluation: After installation, the temporary alternate fire pump and associated piping shall be tested to verify acceptance with the systems surveillance requirements. The major limitation of this change is that the temporary pump must be manually started rather than the automatic start on low pressure. This loss of the automatic function is acceptable, however, as the alternate temporary pump can be manually started within 5 minutes if the need arises. This time period is of such a short duration and would only be necessary in case of normal diesel driven fire pump failure that the loss of automatic system function is viewed as inconsequential. The intent of the Fire Protection Manual to provide an alternate pump equal or better to the existing pump has been met by this temporary modification. Equipment important to safety has not been subject to an increase in malfunction probability occurrence since, in the event of a fire, system physical separation distance between redundant or diverse equipment important to safety is maintained.

SAFETY EVALUATION 88-SE-082 Revision: 0

Title: Temporary Use Of Nitrogen Bottles To Supply Containment Penetration Seals

Description: Temporary Procedure TP-OP-90 allows the Nitrogen (N_2) supply header to be isolated so maintenance can be performed on valve BB PCV-8034, Pressurizer Relief Tank Nitrogen Supply. This action will remove supply N_2 to the electrical penetrations. The N_2 gas provides an inert seal for the electrical penetrations. While the normal N_2 supply line is valved out, N_2 bottles will be temporarily connected to the N_2 piping downstream of the isolated portion of the N_2 service gas system. This action will introduce high pressure N_2 bottles in the north and south electrical penetration rooms and in the 2000' Auxiliary Building Corridor.

Safety Evaluation: USAR Section 9.3.5 describes the Service Gas System of which the N_2 piping/supply is a part. The USAR states that this system has no safety design basis and serves no safety function. The rupture of the N_2 piping will not cause unacceptable impairment of a safety related system, structure, or component from blast forces, missile impacts, or pipe whipping. The N_2 from the N_2 bottles will be supplied to the electrical penetrations from a vent valve connection in the N_2 distribution piping in each penetration room and in the Auxiliary Building Corridor. The N_2 will be routed by stainless steel tubing or high pressure flex hose. The probability of a missile generated from pressurized component failure impacting a safety related component has not been increased, but remains within previous analysis since corrective actions are taken to limit the consequences of catastrophic N_2 bottle failure. Stanchion enclosure will limit catastrophic failure of the N_2 bottle to the enclosure with no increase in missile induced malfunction occurrence to Class 1E electrical equipment in the rooms. Fire in the penetration rooms and corridor have been evaluated. A halon system is provided for fire suppression in the penetration rooms. The presence of the N_2 bottle, which is enclosed to contain any potential missile, is not foreseen to increase the consequences of this previously evaluated accident.

SAFETY EVALUATION 88-SE-083 Revision: 0

Title: Simulation Of Normal Steam Generator Levels While The Unit Is Shut Down

Description: The temporary change being made is to emulate normal Steam Generator (SG) levels on 3 of the 4 channels of each loop. This will be accomplished by placing a resistor at each instrument's input test jack in the 7300 process protection cabinets which will give the voltage value to emulate normal SG levels. One instrument channel will not be emulated in order to provide actual SG level indication to the Control Room operators. This change is limited to plant Modes 4, 5, and 6 since Reactor Trip and Auxiliary Feedwater Actuation Signals are defeated. In Modes 4, 5, and 6, this Engineered Safety Features Actuation System (ESFAS) function is not required since the reactor is tripped and the plant is being cooled down to Tave<350°F. Feedwater requirements are minimal and not required for emergency needs in these plant modes.

Safety Evaluation: This temporary change shall not be implemented until the plant is in Mode 4, entering Refuel III. This temporary change is being installed to prevent unnecessary challenges upon the system. This modification is required to be returned to normal operation prior to Mode 3 entry following Refueling III. The basic function of the Lo-Lo Steam Generator water level is to preserve the steam generator heat sink for removal of residual heat by acting to cause a Reactor Trip before the steam generators are dry. This feature also acts to start Auxiliary Feedwater system, refer to USAR 7.2.2.3.5. A coincident 2 out of 4 channels in the Lo-Lo condition initiates these ESFAS functions. The postulated accident in this case is loss of normal feedwater flow and is analyzed in USAR 15.2.7. This analysis sets initial reactor conditions at 102% power. Technical Specification Basis (2.2.1) for limiting safety system settings states the protective function and agrees with the USAR as stated above. Technical Specification Table 3.3-3 requires a minimum of 3 operable channels per operating steam generator while in Modes 1, 2, and 3. Therefore, defeating the Reactor Trip and Auxiliary Feedwater Actuation Signals during plant Modes 4, 5, and 6 is not a reduction in the margin of safety since the safety feature is not required to be present by the Technical Specification.

SAFETY EVALUATION 88-SE-084 Revision: 0

Title: Evaluation Of Anti-Corrosion Chemicals Effect On Plant Discharge

Description: The Wolf Creek Generating Station (WCGS) Environmental Protection Plan states that plant discharges to surface water (Wolf Creek Cooling Lake (WCCL)) are governed by the WCGS National Pollutant Discharge Elimination System (NPDES) Permit. The WCGS NPDES permit requires discharges to WCCL to be within the pH range of 6.0-9.0. Anticorrosion chemicals added to the Component Cooling Water (CCW) system typically elevate the water's pH above 9.0. By requiring that CCW temporarily being used to cool the air compressor skids used for the Integrated Leak Rate Test (ILRT) be below a 9.0 pH, this temporary procedure will preclude high pH discharges which could violate the NPDES permit.

Safety Evaluation: This procedure complies with the WCGS NPDES permit.

SAFETY EVALUATION 88-SE-085 Revision: 0

Title: High Pressure Safety Injection System Pressure Test

Description: Procedure STS PE-044A involves performing a system functional pressure test on the High Pressure Coolant Injection System (Safety Injection (SI) System) piping as required by Technical Specifications and the ASME Section XI Inservice Inspection (ISI) Program. The change made to the procedure is to incorporate a method of testing the hot leg injection portion of piping without having to align the Safety Injection pumps discharges to both the cold and hot leg injection points simultaneously. This requires entry into Technical Specification 3.5.2a Limiting Condition for Operation. The alternate method to be incorporated by the change avoids this alignment, and will replace it. Normally the hot leg injection piping is isolated from the pump's discharge and is unisolated by operator action to allow hot leg recirculation during post LOCA or shutdown conditions. The alternate method uses the 3/4 inch test line of the injection system to communicate the discharge pressure of the Safety Injection pumps, which is approximately 1500 psia, from the cold leg to the hot leg injection portion of piping without having to unisolate the hot leg injection headers. Three valves in the 3/4 inch test line are opened to communicate the system pressure to the hot leg injection piping. The test line, by design, maintains injection piping pressure rating of 2500 pounds and has the inherent design capability to automatically isolate itself from the main injection headers. In the unlikely event that a SI signal occurs during the test, the injection flow to the cold legs will be maintained due to the automatic Containment Isolation Signal (CIS) "A" closure function on the test line valves. The test is not expected to last for more than thirty minutes. The failure mode of the test line valves is closed. Two of these test line valves would have to fail to close during the presence of an SI before some of the cold leg injection flow would be directed out of the 4 inch (ID 3.438") cold leg injection header via the 3/4 inch (ID 0.612") test line to the hot leg injection points. This event is considered not credible and involves system double active failure.

Safety Evaluation: The portion of the test line used to support the pressure test lies entirely in containment. It is branched off of the injection piping upstream of two in series check valves before the Reactor Coolant System (RCS) pressure boundary. Evaluation of RCS loop branch line breaks has been completed in USAR Section 3B.4.4. The portion of the test line piping beyond the test line isolation valves is ANSI B31.1 piping designed for critical service, which is also rated at 2500 lbs. and is not Seismic Category I. If an earthquake occurs during the short time of the test and the nonseismic portion of the test line was ruptured (passive failure), the consequences would be water spray and minor flooding of containment until an operator isolated the test line. The utilization of the line by the subject procedure change will not increase the probability, possibility or consequences of the previously evaluated piping hazards analysis since the piping is not exposed to any different stresses or increased temperature or pressure variables than that considered within its current design.

SAFETY EVALUATION 88-SE-086 Revision: 0

Title: Procedure Change To Defeat The Automatic Isolation Feature Of The MFIV In The Hot Standby To Cold Shutdown Procedure

Description: This permanent procedure change made to the Hot Standby to Cold Shutdown procedure GEN 00-006 incorporates the actions to disable the Feedwater Isolation Signal on Lo average temperature (Tave) and P-4 (Reactor Trip Breakers Open) during plant shutdown, i.e., Modes 4, 5 and 6. This Engineered Safety Features Actuation System (ESFAS) function is disabled by the subject procedure change when the plant is in Mode 4. This function is restored prior to Mode 3 entry upon returning the plant to power. The Equipment Out Of Service Log and plant procedures provide for restoration, retest and subsequent verification prior to Mode 3 entry.

Safety Evaluation: Defeating the automatic isolation feature of the Main Feedwater Isolation Valves (MFIVs) is being done to avoid inadvertent actuation while in Mode 4, 5 and 6 and is in effect only when the plant is in Modes 4, 5 or 6. The purpose of the MFIV is to prevent uncontrolled blowdown from more than one steam generator in the event a pipe rupture in the Turbine Building and to separate safety and non-safety related systems. The MFIV also limits the quantity of fluid that enters Containment in the event of a downstream rupture from it and provides a pressure boundary for the controlled addition of auxiliary feedwater (refer USAR 10.4.7.2.2). With the plant in Modes 4, 5 or 6, the previous accident analysis assumes initial conditions that are more restrictive, i.e., Tave greater than 350^{o} F. Technical Specification Table 3.3-3 requires automatic isolation capability in Modes 1, 2 or 3. Defeating this function in Modes 4, 5 and 6 does not reduce the margin of safety as defined in the Technical Specification since the plant is in a more conservative condition than when the safety feature is required to be present.

SAFETY EVALUATION 88-SE-087 Revision: 0

Title: Procedure Change To Remove Water From Steam Generator U-Tubes Using Nitrogen During Draindown

Description: The subject procedure SYS BB-211, Revision 0, provides a method to displace the water from the Steam Generator (SG) tubes during Reactor Coolant System (RCS) drain down. Nitrogen (N_2) is introduced into the RCS at the loop flow instrumentation tap directly below the SG outlet in the crossover leg of each loop. The nitrogen from this injection point rises to displace the water in the high point, the SG of the system, the SG U-tubes to provide for a smooth drain down evolution. The normal drain down evolutions in the past have seen rapid fluctuations in RCS loop level due to random U-tube drainage. This procedure will be utilized during Mode 5 drain down and has been developed from experience from within the nuclear field.

Safety Evaluation: The plant is in cold shutdown, i.e., average temperature (Tave) less than or equal to 200^{o}F, when the procedure is used. Most previously evaluated accident occurrences in the USAR consider much more severe temperature and pressure conditions. Therefore, their occurrence probabilities have not been increased since their accident initial conditions are not present. The probability of a fuel handling accident in containment has not been increased and cannot occur since the head is still in place. Residual Heat Removal (RHR) suction nozzle vortexing and subsequent pump cavitation leading to possible malfunction is not created by using the N_2. RHR system operation has not been affected by this procedure's actions. The high pressure N_2 bottle will be adequately secured such that regulator failure would not produce an unconfined missile which could damage any equipment important to safety in the Reactor Building.

SAFETY EVALUATION 88-SE-088 Revision: 0

Title: Reactor Head Lift Or Set Without Refueling Pool Level Follow And With Studs, Nuts And Washers In Place

Description: The procedure change to FHP 02-007, Revision 5, allows for lifting or setting the reactor head without refueling pool level follow, and with the option of lifting the head with the studs, nuts, washers, and hold collars with it. These changes are incorporated into the head lift procedure as options in addition to the head lift with refueling pool level follow. The USAR describes the head lift with pool follow and the head set with pool follow until about 14' above the flange at which time the pool is drained prior to completely lowering the head. These evolutions are also described without the associated accessories with the head. In USAR 9.1.4.3, the effects on the reactor vessel nozzles, due to a postulated head drop, were evaluated by conservatively assuming the head drop through 28' of air. In an effort to reduce the probability of vessel support damage due to high impact loads, in the same section of the USAR, the accident evaluation input parameters were revised in the evaluation on the vessel supports to more closely reflect actual WCGS plant conditions during refueling. In the reactor vessel support evaluation where actual lift and set parameters were used, the limiting case was the head drop through 14' of air. This procedure change requests the head to be lifted 1' above the guide pins, or 18' 8" above the vessel, prior to lateral movement to the storage stand area without any water in the pool. The procedure change also requests setting the head from a height of 18' 8" above the vessel without the refueling pool flooded.

Safety Evaluation: Westinghouse/Site Nuclear Plant Engineering has evaluated and recalculated the effects of a head drop with the closure studs, nuts, washers and hold collars in place and with no water in the refueling pool. The results provided in PMR 01843, Revisions 0 and 1 remain within the previously evaluated design envelope for the consequences of a head drop. The evaluation reflects that there will be no consequential damage to the structural integrity of the reactor vessel, its nozzles or supports, core cooling capability or fuel cladding to the extent which will reduce the functioning of these important safety features to an unacceptable safety level. The calculated resultant vertical deformation of the reactor vessel and supports of 0.72 inches remains well within prescribed limits. The change does not affect the traverse path of the head lift evolution; therefore, the targets of the lift remain the same as those previously evaluated. The appropriate health physics measures as deemed necessary by the Site Health Physicist and ALARA program will be undertaken to maintain compliance with regulatory dose requirements.

SAFETY EVALUATION 88-SE-089 Revision: 0

Title: Two Control Building Doors Blocked Open

Description: During Refuel III, when the plant is in modes 5 and 6, doors 32011 & 32013 will be blocked open 1" to allow communication and computer cables to pass through. Door 32011 is a locked 3 hour rated fire door in the south stairwell of the Control Building which provides access to the 1984 elevation where the showers and dressing rooms for the Radiological Controlled Area entry are located. Door 32013 is a locked security and pressure resistant boundary door in the west wall of the Control Building at grade level (2000') which provides entry into the buildings south stairwell from the outside.

Safety Evaluation: The appropriate fire impairment and security measures will be implemented. Site Nuclear Plant Engineering has determined that potential infiltration of a radiological or chlorine accident is within their respective regulatory limits.

SAFETY EVALUATION 88-SE-090 Revision: 0

Title: Install Headset Capable Amplifier

Description: This temporary modification is to install a headset capable amplifier to provide for a means of communication during fuel off-load and reload for Reactor Engineering Personnel in the Control Room. The headset capable amplifier is being wired into the refueling channel at the Gai-tronics test and distribution panel (#QF076), which is in the southeast corner of the control room. The headset capable amplifier will enhance fuel movement evolutions over the installed handset station by providing more convenience to the Reactor Engineers.

Safety Evaluation: The added amplifier is the headset compatible type manufactured by Gai-tronics which has been and still is being installed within the plant (reference PMR 01235). The added amplifier retains Gai-tronics design and does not degrade system performance or represent an additional significant load to the Gai-tronics power supply. The communication system is described in USAR 9.5.2. In the design bases of the system's description, it states that there is no safety design basis for the communication system. Figure 9.5.2-2 of the USAR depicts the system's layout. All accident consequences and probabilities have not been increased from their previous evaluations since no credit has been taken for the existence of the Gai-tronics system within these previous evaluations. The approximately 60 feet of cable run between panel and headset compatible amplifier does not degrade the control room cabinets which are under it. The cable weighs approximately 1/2 pound per foot and is not rigid, but flexible, and will be secured to prevent displacement.

SAFETY EVALUATION 88-SE-091 Revision: 0

Title: Defeating The Horizontal Limit Switch On The Trolley Of The Polar Crane

Description: This temporary modification is to defeat the horizontal limit switch on the trolley of the Polar Crane to facilitate placing the reactor head in its storage and decontamination area.

Safety Evaluation: This is normal practice and currently addressed in the head lift procedure for initial and final head maintenance activities. This does not compromise its' seismic integrity or place the crane in a restricted zone of operation.

SAFETY EVALUATION 88-SE-092 Revision: 0

Title: Installation Of Additional Gai-tronics Communication Station

Description: This temporary modification provides for an additional Gai-tronics Communication Station installed in Containment directly inside the personnel hatch to enhance refueling activities.

Safety Evaluation: The communication system is described in USAR 9.5.2. In the design bases of the system's description, it states that there is no safety design basis for the communication system. Figure 9.5.2-2 of the USAR depicts the system's layout. All accident consequences and probabilities have not been increased from their previous evaluations since no credit has been taken for the existence of the Gai-tronics system within these previous evaluations.

SAFETY EVALUATION 88-SE-093 Revision: 0

Title: Temporarily Removing Nitrogen Supply Header From Service And Installing Nitrogen Bottle In Support Of Integrated Leak Rate Test

Description: This evaluation addresses temporary procedure TP-OP-102. Currently, the plant is in Mode 5 at the beginning of Refuel III Outage. The low pressure Nitrogen (N_2) supply header needs to be isolated in support of the Integrated Leak Rate Test (ILRT). This action will remove N_2 supply capability to the Volume Control Tank (VCT). The N_2 gas provides an inert cover gas during plant shutdown conditions. While the normal N_2 supply line is valved out, N_2 from a N_2 bottle will be temporarily connected to the VCT from a vent valve in the normal N_2 supply piping to the VCT. The N_2 will be routed by stainless steel (SS) tubing or high pressure SS flex hose to the vent valve. This action will introduce a high pressure N_2 bottle into the 2000' Auxiliary Building Corridor.

Safety Evaluation: The high pressure N_2 bottle shall be secured to the 4 inch X 6 inch tube steel support for the non-vital separation group 5 pressure transmitter, BG PT-115. This tube steel support is located beside the corridor door in the labyrinth entrance to the VCT valve gallery. Securing the bottle in the labyrinth entrance to the steel support with a chain or a wire rope will prevent the bottle from becoming an unlikely missile hazard.

A pressure regulator on the high pressure N_2 bottle will reduce the pressure to the pressure necessary for system operation as specified by the Control Room. The relief valve of the VCT maintains its overpressurization function.

Spontaneous explosion of the high pressure N_2 bottle is deemed not credible in view of industry experience and integral internal construction. The more likely hazard the N_2 bottle presents is as a missile. Since no safety related targets exist above the bottle a catastrophic regulator failure ejection missile does not threaten any equipment important to safety. The N_2 bottle in the entrance to the VCT valve gallery is not in the immediate area of any safety related equipment. The high pressure N_2 bottles which are to supply the Central Chilled Water Expansion Tank do not introduce a missile hazard to equipment important to safety since they are outside the buildings which house this equipment and also provide a safety barrier against these missiles. By securing the N_2 bottle and having evaluated an unlikely regulator ejection, it is concluded that no increase in the probability, possibility or consequences of an internal pressurized generated missile impacting a safety a safety related component has been incurred. The N_2 bottles shall be removed after the normal N_2 supply is restored. The expected duration of this work activity is less than 5 days.

SAFETY EVALUATION 88-SE-093 Revision: 1

Title: Temporarily Removing Nitrogen Supply Header From Service And Installing Nitrogen Bottle In Support Of Integrated Leak Rate Test

Description: This evaluation addresses temporary procedure TP-OP-102. Currently, the plant is in Mode 5 at the beginning of Refuel III. The low pressure Nitrogen (N_2) supply header needs to be isolated in support of the Integrated Leak Rate Test (ILRT). This action will remove supply N_2 capability to the Volume Control Tank (VCT). The N_2 gas provides an inert cover gas during plant shutdown conditions. While the normal N_2 supply line is valved out, N_2 from a N_2 bottle will be temporarily connected to the VCT from a vent valve in the normal N_2 supply piping to the VCT. The N_2 will be routed by stainless steel (SS) tubing or high pressure SS flex hose to the vent valve. This action will introduce a high pressure N_2 bottle in the 2000' level of the Auxiliary Building Corridor. The high pressure N_2 bottle shall be secured to the 4 inch X 6 inch tube steel support for the non-vital, separation group 5, pressure transmitter, BG PT-115. This tube steel support is located directly off the corridor door in the labyrinth entrance to the VCT valve gallery. Securing the bottle in the labyrinth entrance to the steel support with a chain or a wire rope will prevent the bottle from becoming an unlikely missile hazard. Subject revision is for this evaluation to remain valid in Mode 5 or 6. The N_2 bottle may be used to supply N_2 in these modes using the temporary procedure TP-OP-102/0 for reasons other than ILRT to support plant operation during N_2 supply header isolation.

Safety Evaluation: A pressure regulator on the high pressure N_2 bottle will reduce the pressure to the pressure necessary for system operation as specified by the Control Room. The tank to regulator connection inside diameter is 0.250 inch and the orifice diameter of the 3 inch X 4 inch relief valve on the VCT is 1.906 inch. Thus, the flow is limited at the outlet of the N_2 bottle within the relief capacity of the VCT's relief valve should the bottle's regulator fail. The relief valve of the VCT maintains its over pressurization function. Spontaneous explosion of the high pressure N_2 bottle is deemed not credible in view of industry experience and integral internal construction and the more likely hazard the N_2 bottle presents as a missile. Since no safety related targets exist above the bottle a catastrophic regulator failure ejection missile does not threaten any equipment important to safety. The N_2 bottle in the entrance to the VCT valve gallery is not in the immediate area of any safety related equipment. The high pressure N_2 bottles which are to supply the Central Chilled Water Expansion Tank do not introduce a missile hazard to equipment important to safety since they are outside the buildings which house this equipment and also provide a safety barrier against these missiles. By securing the N_2 bottle and having evaluated an unlikely regulator ejection it is concluded that no increase in the probability, possibility or consequences of an internal pressure generated missile impacting a safety related component has been incurred. The N_2 bottles shall be removed after the normal N_2 supply is restored.

SAFETY EVALUATION 88-SE-094 Revision: 0

Title: Evaluation Of Temporary Shielding

Description: Temporary shielding is to be placed in the Letdown Heat Exchanger Valve Room due to numerous refueling work activities in this high radiation area.

Safety Evaluation: The shielding will not affect the system's pressure boundary or introduce seismic concern.

SAFETY EVALUATION 88-SE-095 Revision: 0

Title: Local Control Of Temperature Control Valve

Description: This evaluation addresses a temporary modification order which will locally control valve BG TCV-129 with instrument air supplied by a regulator. This is desired because during the PA01 bus (13.8 Kv) outage, valve BG TCV-129 (Letdown Heat Exchanger Outlet to Volume Control Tank (VCT)) failed open. In normal operation, BG TCV-129 controls letdown flow to the demineralizers. BG TCV-129 will bypass letdown flow to the demineralizers if the letdown heat exchanger discharge flow temperature is approximately 140 degrees Fahrenheit. This is to protect demineralizer resin from thermal damage. Application of the temporary instrument air defeats the automatic features of BG TCV-129. This temporary modification is valid only with Reactor Coolant System (RCS) temperature less than 130 degrees Fahrenheit and is a Mode 4 restraint. The Control Room is monitoring Chemical and Volume Control System letdown heat exchanger outlet flow temperature. The Control Room will divert flow to the VCT, bypassing the demineralizers, if RCS temperature reaches 130 degrees Fahrenheit or greater.

Safety Evaluation: BG TCV-129 is a safety related valve which serves to control reactor coolant water quality. It has no active or passive safety features. The cation and mixed bed demineralizers do not perform a safety related function and are special scope components which contain radioactive materials. The ability of the demineralizers to contain the radioactive materials (and ionic impurities) is maintained. The resin in the demineralizers is protected from overheating while BG TCV-129 is in manual by Control Room monitoring of RCS temperature.

SAFETY EVALUATION 88-SE-096 Revision: 0

Title: Use Of Spare Battery Charger To Support Maintenance Activities

Description: This temporary modification provides the means to maintain power on NK02 and NK04, 125V DC busses, during their normal charger's (NK22 and NK24, respectively) maintenance by utilizing the spare charger and temporary jumper cables. Currently, the plant is in Mode 6 entering Refuel III Outage. Maintenance on the chargers NK22 and NK24 is desired. During plant Modes 5 and 6, one offsite to onsite Class 1E system, one Emergency Diesel Generator and one division of electrical busses are required to be operable in the manner specified by technical specification. Currently B train electrical system (NB02) and its associated 125V DC systems are operable and A train is out of service. The modification is to support the maintenance work on the chargers associated with the operable busses and to maintain these busses operable.

Safety Evaluation: The spare charger has a rated output of 300 amps and is identical to the normal battery chargers, all of which are Class 1E. Each battery charger has sufficient capacity to restore the battery from the design minimum charge (one duty cycle) to its fully charged state while supplying the largest combined demand of the steady-state loads. The full load rating of the battery chargers is 59 amps. USAR Section 8.3.2 states that the spare battery charger is centrally located and is used in the event of failure of a charger or inverter by connecting it to the affected system. Therefore, by design, the spare charger is suited for this use; however, the electrical connections between spare charger and affected DC system are not provided by design. The above cables, spare charger to bus, Control Room metering cable, and NB02 power feed to the spare charger, are identical to the cables used by design for the respective connections. Therefore, since cables identical to design are used, the possibility or probability of equipment malfunction or creation of a different type of malfunction from previous analysis has not been increased. The modification actions do not degrade the operable Class 1E electrical power sources or distribution and provide the means to utilize the spare charger for its design use. The power feed cable from NB02 to the spare charger runs from the inoperable Engineered Safety Features (ESF) Switchgear Room Number 1 (NB01) to the operable ESF Switchgear Room Number 2 (NB02) on the 2000' elevation of the Control Building. The independence of redundant Class 1E power systems shall be maintained by removing this power feed cable prior to Mode 4 entry, when two independent Class 1E systems are required to be operable. This temporary modification invokes a Mode 4 plant restraint.

SAFETY EVALUATION 88-SE-097 Revision: 0

Title: Providing Instrument Air To Programmable Automated Remote Device

Description: This temporary modification allows the Programmable Automated Remote (PAR) device to receive its operating air from the plant Instrument Air System. The PAR is used to support the examination of the reactor vessel nozzles when the refueling pool is flooded for the Inservice Inspection (ISI) program.

Safety Evaluation: Providing instrument air to operate the PAR device does not degrade the quality of the instrument air system and the connection point is not seismically qualified. Upon a loss of this air, safety related pneumatically operated valves fail to their safe position.

SAFETY EVALUATION 88-SE-098 Revision: 0

Title: Procedure Change To Maintain Power During Supply Bus Outage

Description: This procedure change provides the means to maintain power on the 125VDC System during its supply bus outage through the use of temporary cables placing both DC busses in parallel and using the in-service charger.

Safety Evaluation: Steps in the procedure verify that systems are deenergized and isolated, polarity is checked, and fire impairments issued on all fire doors breached by the cable runs. These procedure requirements and controls maintain the probability, possibility and consequences of previously evaluated accidents within previous analyses. The procedure controls prevent an increase in the possibility or probability of a short circuit, ground fault, electrical fire, over-current degradation, or backfeed to the outage bus from occurring. Group separation is lost by this change, but is not required since the plant is in Mode 5 or 6 before the procedure can be used.

SAFETY EVALUATION 88-SE-099 Revision: 0

Title: Temporary Cables To Provide Temporary Power To Cubicles

Description: This temporary modification allows the use of temporary cables to energize portions of a deenergized electrical bus during maintenance. Currently the plant is in Mode 6 during Refuel III Outage. Outage maintenance work requires deenergizing Class 1E NB01 bus AC electrical power. In order to keep selected cubicles of the out-of-service train energized, temporary power via temporary cables is being provided to these cubicles. Three temporary cables are requested by the subject temporary modification. The first cable runs from an operable Class 1E spare 50 amp circuit breaker 52NG02ACF4 to the out-of-service Class 1E distribution transformer XNG01A (37 amps full load) through 50 amp circuit breaker NG01ABF5. The second cable runs from an operable Class 1E spare 50 amp circuit breaker 52NG04CMF4 to the out-of-service Class 1E distribution transformer XNG03C (37 amps full load) through 50 amp circuit breaker 52NG03CAF4. The third cable runs from a welding receptacle non-Class 1E PG20GDF6 100 amp fuse to the out-of-service Class 1E distribution transformer XNG01B (37 amps full load) through 50 amp circuit breaker 52NG01BAF4.

Safety Evaluation: The operability of the Class 1E power system is protected from fault or system overload by the circuit breakers. The loads added to the operable Class 1E Motor Control Centers (MCCs) via their spare breakers are small and do not affect their operability. The full load current of the MCCs is 600 amps and the maximum current they can draw off their bus is 800 amps. AC load group Class 1E separation is lost by these cable connections and a Mode 4 restraint is placed on the subject temporary modification to remove these cables prior to establishing Class 1E integrity as required during Modes 1 through 4. A reduction in the margin of safety as defined by Technical Specification 3.8.3.2 is not introduced by these cable runs since the operability of the specified AC power sources and associated distribution systems remains. The first cable run breaches fire door 33011 and affects the associated rooms' Halon protection, the second run breaches fire doors 14031 and 14101 and also affects the associated rooms' Halon systems, and the third run breaches fire door 15012. The appropriate fire impairment measures are to be implemented in accordance with the Wolf Creek Generating Station Fire Protection Manual when these fire protection features have been breached. The operability of the Control Building Heating, Ventilation, and Air Conditioning systems has not been affected by breaching the subject fire doors since these doors do not provide a pressure boundary. The expected duration of the modification is 3 days.

SAFETY EVALUATION 88-SE-100 Revision: 0

Title: Placement Of Day Tank And Storage Tank In Recirculation For Visual Examination

Description: This procedure is applicable to the Emergency Diesel Engine "A" Fuel Oil Storage and Transfer System. It places the day tank and storage tank in the recirculation mode by defeating the high level stop of the transfer pump to allow visual examination of the system to satisfy Inservice Inspection (ISI) program requirements.

Safety Evaluation: The procedure provides proper controls for electrical jumper installation and removal and subsequent Technical Specification Limiting Condition for Operation entry.

SAFETY EVALUATION 88-SE-101 Revision: 0

Title: Placement Of Day Tank And Storage Tank In Recirculation For Visual Examination

Description: This procedure is applicable to the Emergency Diesel Engine "B" Fuel Oil Storage and Transfer System. It places the day tank and storage tank in the recirculation mode by defeating the high level stop of the transfer pump to allow visual examination of the system to satisfy Inservice Inspection (ISI) program requirements.

Safety Evaluation: The procedure provides proper controls for electrical jumper installation and removal and subsequent Technical Specification Limiting Condition for Operation entry.

SAFETY EVALUATION 88-SE-102 Revision: 0

Title: Temporary Procedure To Tap Cooling Water From Essential Service Water Feed To Air Compressor

Description: This temporary procedure provides a means to temporarily tap cooling water from the Essential Service Water feed to Air Compressor A or B by way of a hose to Air Compressor C when Service Water is not available. This allows Compressor C to remain usable when its Service Water is secured during the outage.

Safety Evaluation: Automatic isolation of the safety related portion of the Essential Service Water supply line remains intact if the temporary line breaks. The hose will be of sufficient rating to retain pressure integrity. System cleanliness is the same for both systems.

SAFETY EVALUATION 88-SE-103 Revision: 0

Title: Defeat Of The Pressure Control Valve Interlock On The Volume Control Tank

Description: This temporary modification defeats the pressure control valve (BG PCV115) interlock on the Volume Control Tank (VCT) so the tank can be drained and vented to perform maintenance on the VCT relief valve. The VCT does have it's own vent, but to avoid the possibility of venting any potentially remaining fission gases from the tank to the room, it is desired to vent the tank via the gaseous vent line. This path to the gaseous radwaste processing system does not introduce a different pathway for gaseous waste processing or effluent release since it is the gaseous waste processing line used during normal operations. After the maintenance work is completed, the nitrogen cover gas on the VCT will be controlled manually slightly above atmospheric pressure until the VCT is restored for power operations.

Safety Evaluation: Only the 15 psig VCT isolation function of BG PCV115 is being defeated. The isolation function that this valve also provides on low waste gas compressor suction header pressure and on high hydrogen concentration in the Hydrogen Recombiner have not been affected. A check valve in the gaseous vent line prevents an unlikely backflow. The VCT is only open to the Auxiliary Building atmosphere during the maintenance work. At all other times during this modification the system will remain closed in its normal configuration. The check valve prevents backflow and the bypass line around the check valve is normally isolated. This bypass line is to be isolated when the modification is in effect. The potential to backflow out of the gaseous vent line has not been created by the actions of the modification. A lower VCT pressure does not increase the probability of Reactor Coolant Pump seal failure due to a backwash event since seal leakoff is isolated at Reactor Coolant System pressure less than 100 psig.

SAFETY EVALUATION 88-SE-104 Revision: 0

Title: Use Of Spare Battery Charger To Support Maintenance Activities

Description: This temporary modification provides the means to maintain DC power on the out-of-service NK02 125 volts DC (Class 1E) bus during maintenance of its normal charger (NK22) and battery bank (NK12) by utilizing the spare charger and temporary jumper cables.

Safety Evaluation: By design, the spare charger is suited for this use. To maintain Class 1E electrical integrity, the spare charger will be disconnected from NK02 prior to declaring it operable. The jumper cables will be of sufficient size and fire impairments in place to maintain fire protection features within allowable limits.

SAFETY EVALUATION 88-SE-105 Revision: 0

Title: Installation Of Test Gauges On Essential Service Water System

Description: This temporary modification, in support of Essential Service Water (ESW) system testing, has two test gauges installed on the vent valves of the ESW main supply line and a single test gauge on the cross connect line between trains.

Safety Evaluation: Gauges are connected via a metal flex hose, hence seismicity of ESW system is not altered. Failure of the gauge or connection would result in an insignificant flow loss.

SAFETY EVALUATION 88-SE-106 Revision: 0

Title: Application Of Freeze Seal To Discharge Line Of Relief Valve

Description: This temporary modification allows a freeze seal to be applied to the discharge line of a relief valve on the Fuel Pool Cooling Heat Exchanger "A". The discharge line of the relief valve connects to an operable portion of the Component Cooling Water (CCW) system, hence a boundary isolation is needed to allow maintenance activity on the relief valve.

Safety Evaluation: This is to be performed by freeze seal procedure. In that the event freeze seal fails, the loss of CCW water from the 1 inch line would not degrade the operability of the CCW system to an unacceptable level.

SAFETY EVALUATION 88-SE-107 Revision: 0

Title: Incorporation Of Recommendations For Half-Loop Operation Into Procedure

Description: The procedure GEN 00-007 provides instructions for maintaining the plant in Cold Shutdown and draining the Reactor Coolant System (RCS) to any desired level greater than or equal to half-loop level. The changes made to the procedure incorporate the recommendations for half-loop operation, with or without nozzle dams/open primary manways from the Westinghouse Owners Group study in response to regulatory concerns. Additionally, the changes made to this revision of the procedure include the method to displace the water from the Steam Generator U-tubes by introducing Nitrogen into the tubes during RCS drain down. RCS drain down and Residual Heat Removal (RHR) operations are described in USAR Sections 5.4.7 and 9.1.4.2.3.1.

Safety Evaluation: Using the Nitrogen to force out the water provides a smoother drain down with conservative level indications during final approach to half-loop or thereafter, and provides no interference with RHR operation. The malfunction probability or possibility of the RHR equipment has not been increased from previous analysis since pump suction and net positive suction head are not affected by the presence of the Nitrogen. All changes made to the subject procedure do not interfere with RHR operation, system level indication or system drain down. The possibility, probability or consequences of equipment important to safety malfunction occurrence have not been increased by the changes made to the procedure from previous analysis.

SAFETY EVALUATION 88-SE-108 Revision: 0

Title: Procedure Change To The Fire Protection 18 Month Operational Test

Description: Procedure STN FP-206 is a Fire Protection System functional test performed at least once per 18 months which provides verification that the automatic valves in the plant's sprinkler system's flow paths actuate to their correct positions on a simulated fire test signal. Fire Protection sprinkler systems inside Containment are designed with a fixed, manually charged, closed head pre-action sprinkler system over the cable trays in Zones RB-3 and 4. To protect the chloride sensitive piping and equipment from Fire Protection System leakage, the standpipes inside the Containment Building are normally dry. Control Room operator action is required to charge these standpipes. Zones RB-3 and 4 sprinkler systems are tested during a plant scheduled outage when the standpipes are flooded for testing. This surveillance test is a part of the Wolf Creek Generating Station Fire Protection Manual and is not a Technical Specification surveillance test.

The change to this surveillance procedure deletes the testing requirements of the sprinkler system in Zones RB-3 and 4 since there are no automatic valves in their systems. The deluge valves (two total) for Zones RB-3 and 4 are placed in the tripped condition of the Fire Signal Mode by operator action, i.e., no automatic function. USAR Table 9.5.1-2 lists the type of system for Zones RB-3 and 4 as a manual pre-action sprinkler system, and in USAR Section 9.5.2.1 the description of this type of system states that there are no automatic release devices. The purpose of the surveillance test is to verify automatic actuation, therefore, the test of the sprinkler systems in Zones RB-3 and 4 serves no purpose.

Safety Evaluation: The Containment isolation valve of the Fire Protection header has no automatic function on a fire signal. The Containment fire detectors and their alarms are not affected by the deletions made to the procedure. These fire protection features are tested by different test procedures. The capability to fight a Containment fire remains readily available and no decrease in fire protection capability has resulted. The change does not affect the operability of the Containment's fire detection and alarm system, hose stations, portable fire extinguishers, or the oil collection system for the Reactor Coolant Pumps.

SAFETY EVALUATION 88-SE-109 Revision: 0

Title: Install Temporary Blind Flange To Perform Weld Repair

Description: This Temporary Modification Order (TMO) is to facilitate weld repair of the 30" Essential Service Water (ESW) A train warming line downstream of EF V264 (ESW traveling water screen 1A warm water header downstream isolation). This valve and 6' of connected upstream piping between EF V264 and EF V262 (ESW traveling water screen 1A warm water header upstream isolation) are being removed. Valve EF V262 has been closed and tagged "do not operate" to isolate the warming line supply. To provide additional personnel safety a temporary blind is being bolted onto the mating flange connection of EF V262 via the subject TMO. The 30" warming line provides freeze protection of A train ESW during its operation, Reference USAR 9.2.1.2.2.3. Two trains of ESW are required to be operable in Modes 1 through 4 per Technical Specification 3.7.4.

Safety Evaluation: It has been determined that the warming line need not be operable when the cooling lake temperature is above 36^{O}F since the ice hazard is not present. Therefore, the isolation of the warming line above 36^{O}F does not render A train ESW inoperable or require entry into the Limiting Condition Operation (LCO). No reduction in the margin of safety defined in the bases of Technical Specification 3.7.4 will result and compliance with the LCO shall be adhered to should lake temperature fall to 36^{O}F. With only one train of ESW operable, the Technical Specification LCO allows normal plant operations for up to 72 hours.

SAFETY EVALUATION 88-SE-110 Revision: 0

Title: Installation Of Temporary Fuel Oil Supply Line To Auxiliary Boiler

Description: This temporary modification is on the fuel oil supply line to the Auxiliary Boiler System. This system is not essential to plant safety. The fuel oil supply line to the suction side of the booster pumps of the boiler is underground, located in the exterior yard area bounded by the Demineralized Water Storage Tank, the Containment/Auxiliary Building, and the Auxiliary Boiler Building. This line has developed a leak and the modification will bypass the portion of the line which is leaking by using hard pipe and a bypass hose. The bypass hose and associated piping shall be rated at or above 75 psig and 100^{o}F which will maintain or exceed system design. The bypass hose shall remain uninsulated for leak detection purposes. Should the external temperature fall below the fuel oil freeze/gel state, the worst case scenario would be shutdown of the Auxiliary Boiler.

Safety Evaluation: This system is not essential to plant safety and it does not impact any safety related systems. This modification does not degrade or alter fire barriers, zones or supression systems. Removal of the roof drain line to facilitate the excavation work has no significant flood consequences.

SAFETY EVALUATION 88-SE-111 Revision: 0

Title: Temporary Procedure To Allow Recirculation Of The Diesel Generator Storage Tanks

Description: This temporary procedure places the day tank and storage tank for the Emergency Diesel Engine Fuel Oil Storage and Transfer System in a recirculation mode by defeating the high level stop of the transfer pump. This mode of operation is desired to allow the vendor to verify flow measurement.

Safety Evaluation: The procedure provides proper controls for electrical jumper installation and removal and subsequent Technical Specification Limited Condition Of Operation entry. These actions maintain the plant within the allowable limits of design. Fuel oil spill has not been created by the procedure actions since by design the overflow/recirculation line is designed to provide this function. The procedure does not increase the malfunction probability of the diesel engine since the diesel's operability is retained with the day and storage tanks in a recirculation mode. The diesel's engine driven fuel oil pump provides fuel from the day tank to the fuel injectors and this system function is not affected by the procedure actions. Should the diesel generator start during the recirculation mode, the leakage from the injection nozzles is still capable of draining to the storage tank since the force of gravity and piping system pressure gradients have not been altered. The fire detection signal from the diesel building, which stops the transfer pump if it is running and the engine is not, remains unaffected by the electrical jumper.

SAFETY EVALUATION 88-SE-112 Revision: 0

Title: Change Of Testing Method For Non-Safety Filtration And Adsorption Units

Description: The subject USAR change supersedes USAR Change Request 87-064, Revision 0 and deals with the laboratory testing criteria for activated carbon used in the plants normal ventilation exhaust system air filtration and adsorption units. Specifically, the units are the Condenser Air Removal Filtration Unit FGE01, Radwaste Building Exhaust Filtration Unit FGH01, Access Control Exhaust Filtration Unit FGK03, Containment Atmospheric Control Filtration Units FGR01A and B, Containment Purge Filtration Unit FGF01, and the Auxiliary/Fuel Building Normal Exhaust Filtration Unit FGL02. These units are not seismically qualified and are not powered from Class 1E electrical sources.

The testing criteria for these units is contained in Regulatory Guide 1.140. This guide presents methods acceptable to the NRC staff for implementing the commission's regulations in 10 CFR Part 50 and in Appendices A and I of 10 CFR Part 50 with regard to the design, testing, and maintenance criteria for air filtration and adsorption units installed in the normal ventilation exhaust systems of light-water-cooled nuclear power plants. This guide applies only to atmosphere cleanup systems designed to collect airborne radioactive materials during normal plant operation, including anticipated operational occurrences, and addresses the atmosphere cleanup systems, including the various components and ductwork in the normal operating environment. This guide does not apply to post-accident Engineered Safety Feature (ESF) atmosphere cleanup systems that are designed to mitigate the consequences of postulated accidents. Regulatory Guide 1.52 provides the guidance for these systems. These systems are seismically qualified and powered from Class 1E sources.

USAR Table 9.4-3 (sheet 12) provides a design comparison of the Wolf Creek Generating Station (WCGS) systems to Regulatory Guide 1.140 and currently references a carbon laboratory test method that has been proven obsolete for the activated carbon used in these units. A more strict and reliable method has been developed by the Idaho National Engineering Laboratory (INEL). The subject USAR Change Request proposes to revise USAR Table 9.4-3 to allow representative samples to be tested in accordance with the recommendations of USNRC Information Notice 87-32, Deficiencies in the Testing of Nuclear Grade Activated Charcoal, and the associated Final Technical Evaluation Report of the NRC/INEL Activated Charcoal Testing Program, Report Number Eggshell. The new proposed test method places more control and specified conditions on the laboratory testing of the carbon samples.

Safety Evaluation: When using the new proposed test method, a 20% penetration limit on used carbon samples would be equivalent to approximately a 5% penetration limit when using the present test method. Compliance with Regulatory Guide 1.140 acceptance criteria of less than 10% penetration on newly installed or used carbon is maintained by the USAR change. WCGS is taking exception to only the test method recommended in the guide by the subject USAR change. The acceptance criteria of the guide is still adhered to by the submitted change. The new test method is therefore more conservation than the old method and at the same time provides for a more consistent and representative laboratory test analysis. This type of consistency will support the efforts to continue to improve the carbon filter test and replacement program. The new test method is more conservative and consistent than the test guidance given in Regulatory Guide 1.140. The new test method remains within the acceptable limits of the issued regulatory guidance for this subject area and does not result in an increase of any previously evaluated USAR accident consequences.

SAFETY EVALUATION 88-SE-113 Revision: 0

Title: Wording Change In USAR To Clarify Fulfillment Of Shift Technical Advisor Position

Description: This USAR change request has been submitted to replace the word "personnel" with the wording "at least one (1) person" to clarify that the Shift Technical Advisor (STA) position need only be fulfilled by one qualified Senior Reactor Operator (SRO) when the combined SRO/STA position option is exercised. The current wording, using the word personnel, is misleading since it implies that more than one person is required to fulfill the STA position.

Safety Evaluation: The subject change or text clarification is in agreement with the NUREG requirements and the STA program established and described in the USAR for Wolf Creek Generating Station. This change does not involve an unreviewed safety question. Shift crew composition requirements of Technical Specification Table 6.2-1 are maintained and not affected by the subject USAR change request. No margin of safety reduction has been created.

SAFETY EVALUATION 88-SE-114 Revision: 0

Title: Installation Of Equipment To Monitor Thermal Stratification In The Pressurizer Surge Line

Description: This temporary modification installs equipment to allow monitoring of the Pressurizer Surge Line temperature and movement of the Surge Line for determining the potential presence and extent of thermal stratification. This data will be multiplexed by a data logger and sent to a computer via spare cables through a Containment Penetration. Additional system process protection (safety related) and control (non-safety related) racks will be multiplexed in a similar fashion.

Safety Evaluation: No containment penetration conductor overcurrent protection is necessary due to the low energy levels of the signals passing through the cables. The signals that actually leave the Nuclear Steam Supply System (NSSS) process protection racks will be electronically isolated with existing rack isolators so as not to impact the signals' safety related function. The installation of additional equipment will add negligible amounts of aluminum and zinc and will not affect the hydrogen generation analysis. The weight of the additional equipment being installed shall have no affect on surge line stresses or the structural steel utilized. They will not add to the piping stresses as analyzed.

SAFETY EVALUATION 88-SE-115 Revision: 0

Title: Modification Of A Test Boundary On The Emergency Core Cooling System

Description: The procedure for performing a visual examination of the piping between the discharge of the low head Emergency Core Cooling System (ECCS) Pumps and the suction of the high head ECCS pumps has been changed to gag the existing relief valves in the suction piping being examined. Overpressure protection will be provided by temporarily installed relief valves using a flexible connection. The changes prevent lifting of the existing relief valves, with attendant possible failure to reseat. Failure of the relief valve to reseat would require undesirable system maintenance for restoration.

Safety Evaluation: The changes are not made to support the visual examination or introduce a system operational configuration which is abnormal. The temporary alternate relief valves installed on the suction piping have equal or greater relieving capacity because the discharge is directed to an Auxiliary Building drain. The potential to spray or flood equipment important to safety has not been introduced should the relief valves lift. They will not degrade or adversely affect the seismicity or stress analysis of the suction piping since the mass of the temporary relief valves are not borne by the piping and are not rigidly coupled to it due to the flexible connection used between the piping system and valves. The temporary relief valves will be adequately secured to prevent movement from a resultant discharge force, which will be of the same magnitude as the normal discharge force. The temporary flexible connection maintains pressure integrity and is not more restrictive than the valve's orifice.

SAFETY EVALUATION 88-SE-116 Revision: 0

Title: Adjustment Of Limit Switch On A Fuel Building Emergency Exhaust System

Description: This temporary modification adjusts the close limit switch on a Fuel Building Emergency Exhaust System (EES) intake damper allowing the intake damper to remain partially open during Safety Injection System (SIS) lineup. With this modification, the EES takes suction from both the Auxiliary Building and Fuel Building in this lineup. This temporary modification allows supplemental flow to the EES in SIS lineup to prevent the EES from pulling such a high negative pressure on the Auxiliary Building. This allows the Control Room Emergency Ventilation System Filtration System to maintain enough flow to meet the Control Room positive pressure requirements.

Safety Evaluation: The EES intake damper serves no other safety related purpose other than to ensure that following the SIS, the Auxiliary Building's negative pressure is greater than 0.25 inches of water gauge. The damper adjustment does not affect the EES following a Fuel Building Ventilation Isolation Signal because the damper will still full open, allowing a negative pressure of at least 0.25 inches of water gauge in the Fuel Building.

SAFETY EVALUATION 88-SE-117 Revision: 0

Title: Adjustment Of Limit Switch On A Fuel Building Emergency Exhaust System

Description: This temporary modification adjusts the close limit switch on a Fuel Building Emergency Exhaust System (EES) intake damper allowing the intake damper to remain partially open during Safety Injection System (SIS) lineup. With this modification, the EES takes suction from both the Auxiliary Building and Fuel Building in this lineup. This temporary modification allows supplemental flow to the EES in SIS lineup to prevent the EES from pulling such a high negative pressure on the Auxiliary Building. This allows the Control Room Emergency Ventilation System to maintain enough flow to meet the Control Room positive pressure requirements.

Safety Evaluation: The EES intake damper serves no other safety related purpose other than to ensure that following the SIS, the Auxiliary Building's negative pressure is greater than 0.25 inches of water gauge. The damper adjustment does not affect the EES following a Fuel Building Ventilation Isolation Signal because the damper will still full open, allowing a negative pressure of at least 0.25 inches of water gauge in the Fuel Building.

SAFETY EVALUATION 88-SE-118 Revision: 0

Title: Installation Of Temporary Ecolochem Portable Water Treatment Equipment

Description: This temporary modification affects the Demineralized Water Storage and Transfer System (DWSTS) which stores water for use upon demand for makeup within the plant. The DWSTS receives filtered and demineralized water from the Demineralized Water Makeup System (DWMS). The DWMS is being taken out of service. While the DWMS is out of service, temporary Ecolochem portable water treatment equipment will be utilized to provide demineralized water. Ecolochem equipment will be tied into the DWMS in the Shop Building east of the Turbine Building. The temporary Ecolochem water treatment equipment has the capability to adequately treat the raw water of the DWMS in meeting chemical specifications for demineralized water as given in USAR Table 9.2-16.

Safety Evaluation: The DWMS does not interfere with any equipment important to safety and this system is not needed to shut down the reactor. The DWMS is not 1E qualified or powered, serves no safety function and contains no equipment important to safety. All previously evaluated accidents evaluated in USAR Section 15.0 do not take any credit for operability of the DWMS. The DWSTS is located in the Shop Building outside of the power block. The quality of demineralized water has not been affected by the subject change and the cleanliness requirements of plant systems which use demineralized water will be retained. The probability and possibility of previously evaluated accidents and malfunctions of equipment important to safety has not been increased by this temporary change made to the water treatment system.

SECTION II

PLANT MODIFICATION REQUEST 00286 Revision: 0

Title: Extension To Main Feedwater Pump Suction Valve Work Platform

Description: The handwheels on main feedwater pump suction valves AE-V006 and AE-V009, located in Area 4 of the Turbine Building at Elevation 2015'-4", are beyond a safe, reachable distance for the valve operation from the existing platform. To access and operate these valves safely, an extension to the existing platform is to be provided.

Safety Evaluation: The proposed extension included in this Plant Modification Package will provide easier access to the valve operators for valves AE-V006 and AE-V009. It is located in a noncategory I building; consequently, there are no II/I concerns. The only proposed USAR change is to include the platform extension on the equipment location drawing for this area.

PLANT MODIFICATION REQUEST 00569 Revision: 0

Title: Addition Of Required Jumpers To General Atomics Radiation Monitor Preamplifier Boards

Description: The Wolf Creek application of General Atomics Particulate - Iodine - Gas (PIG) and Particulate - Iodine (PI) radiation monitors require that the preamplifier boards have a jumper on the circuit board to allow it to operate in the gross detection mode. These required jumpers are being added to the design documents/drawings.

Safety Evaluation: The "as built" jumper connections are in accordance with the instruction manual's recommendation for allowing the radiation monitors to operate in the gross detection mode rather than the window detection mode. Because these changes are in accordance with the manual, the probability of occurrence and consequences of an accident previously evaluated in the USAR, probability of occurrence of malfunctions of equipment important to safety and the consequences of such malfunctions will not be increased and a new malfunction is not created. The design change will not reduce the margin of safety defined for any Technical Specification.

PLANT MODIFICATION REQUEST 00831 Revision: 0

Title: Stop Turbine Loading Interlock Bypass Switch Added To The Main Control Board

Description: This modification adds a back-lit bypass switch to the main control board for the stop turbine loading interlock (C-16). The bypass will reset when Low
T-Error/Low Tave bistables have returned to normal.

Safety Evaluation: The subject modification is implemented to comply with USAR Figure 7.2-1, Sheet 16. In addition, an indication light is being installed to indicate that the manual test bypass for the C-16, Low Tave signal is being utilized. Upon review of this modification, it is concluded that there will not be any adverse effect on the safety of the plant operation at Wolf Creek Generating Station (WCGS). The probability or consequence of a previously evaluated accident is not increased, the possibility of a new or different type of accident is not created and the probability of malfunctions of any equipment, either safety or non-safety related will not result due to the subject modifications. The subject modification will not affect the bases of any WCGS Technical Specification.

PLANT MODIFICATION REQUEST 00831 Revision: 1

Title: Stop Turbine Loading Interlock Bypass Switch Added To The Main Control Board

Description: This revision modifies the wiring for the main annunciator board indication to retain indication of Low T-Error/Low Tave with C-16 bypassed. The C-16 bypass from the main control board was provided in Revision 0.

Safety Evaluation: The portion of the reactor instrumentation involving the C-16 stop turbine loading signal and associated bypass switch serves no safety function and has no safety design basis. The modification has no effect on the function and operation of the reactor instrument system as described in the USAR.

PLANT MODIFICATION REQUEST 01027 Revision: 1

Title: Centrifugal Charging Pump Measurements Out Of Tolerance

Description: The measurements taken to verify the squareness between the shaft and stuffing box of the Centrifugal Charging Pump (CCP) were found to be out of tolerance. The purpose of this revision is to address a Nuclear Safety Review Committee (NSRC) concern.

Safety Evaluation: USAR Section 6.3.2.5 discusses the Emergency Core Cooling System (ECCS) reliability. Under this section the USAR addresses leakage from ECCS components. For mechanical equipment leakage (i.e. pumps, valves, and flanged joints), it was concluded that the largest sudden leak potential is from the sudden failure of a pump shaft seal. It was determined that the resulting leakage would be less than 7.5 gpm. Engineering evaluated the slight out-of-roundness condition for the potential of exceeding the USAR analysis. Engineering has determined that the USAR analysis still represents the bounding case and, therefore, is still valid. If a failure should occur because of the out of tolerance between the shaft and the stuffing box on CCP "A", the redundant train "B" is available for core cooling in accordance with USAR Section 6.3.2.5. Therefore, the probability and/or consequences of an accident are not increased.

PLANT MODIFICATION REQUEST 01328 Revision: 0

Title: Local Indication Of Total Steam Generator Blowdown Flow Near Throttle Valves

Description: The Steam Generator Blowdown System is designed to control the secondary side water chemistry in conjunction with the condensate demineralizer system to meet the plant's chemistry specifications. The blowdown flow rate is controlled manually using throttling valves just upstream of the blowdown flash tank. The subject modification will provide local indication of total steam generator blowdown flow near the blowdown throttle valves to facilitate adjustment of blowdown flow rate by the plant operators.

Safety Evaluation: The subject modification is accomplished through the use of existing non-safety related process instrumentation, with the addition of other non-safety related components. The resulting configuration has no impact on the Technical Specifications or on the Safety Design Bases for the Steam Generator Blowdown System. The only effect is to USAR Figure 10.4-8, Sheet 1, which will be revised to reflect the additional local indicator.

PLANT MODIFICATION REQUEST 01350 Revision: 0 and 2

Title: Improvements In Safety Conditions, Drainage And Flow Measurements Of Wastewater Treatment

Description: This modification adds a weir and stairway from the top of the dike to the weir at the lime sludge pond, a platform at the circulating water discharge structure, a walkway at the weir structure at the oily waste separator, a storm sewer line to drain the area around the sewage plant, and raises the manhole near the sewage plant. This modification improves the safety conditions for the personnel who collect the samples needed to comply with discharge permit requirements, alleviates the drainage problem near the sewage plant, and provides flow measurements for lime sludge discharge.

Safety Evaluation: The only change made to the USAR is the addition of a manhole and drains to the storm sewer system at the sewage treatment plant. The drainage system is assumed to be inoperative during flooding conditions. Also, the sewage treatment plant is remotely situated with respect to the power block and the changes made to the drainage system are external to the safety related and special scope systems. Thus, the malfunction of equipment important to safety previously evaluated in the USAR will not be increased. In addition, the changes made to the drainage system are passive changes and only serve to improve the existing local drainage outside the north and west sides of the sewage treatment plant. Thus, the changes do not create the possibility of an accident or a malfunction of equipment important to safety previously evaluated in the USAR. Other changes, such as the weir, stairways, etc., do not involve changes to the facility as described in the USAR and will not affect the consequences of the accident, increase the probability of occurrence of an accident or malfunction of equipment, or create any new accident or new malfunction not evaluated. Revision 2 clarified the responses to three questions on the 10 CFR 50.59 safety evaluation forms, to indicate the scope of USAR and Technical Specification review, and indicate no unreviewed safety question existed.

PLANT MODIFICATION REQUEST 02489 Revision: 0

Title: Emergency Shower And Eyewash Station Addition
Description: This modification adds emergency shower and eyewash stations at several locations throughout the plant.

Safety Evaluation: These stations are piped to the potable water system. They are located so as to not impact any seismic design. They will not affect the operation of any safety related equipment.

PLANT MODIFICATION REQUEST 02505 Revision: 0

Title: Watertight Door Removal From Security System
Description: This modification removes twelve watertight doors from the Security System due to frequent replacement of the electric deadbolt which is necessary to keep watertight doors on security system. This is a high maintenance item and these are non-vital doors, not required to be on-line per USAR commitment. All the security hardware will be left in place. This change will modify the subject doors by removing electric deadbolt and permanently engaging drive hub and actuator hub assembly with bolts.

Safety Evaluation: The doors are not required for security purposes. The modification does not affect their fire protection or watertight function.

PLANT MODIFICATION REQUEST 02512 Revision: 0

Title: Nuclear Instrument System Parts Replacement

Description: Westinghouse has supplied replacement parts for the Nuclear Instrument System (NIS) which have part numbers different from those used in the original equipment. Implementation of this modification will allow the use of alternate replacement parts used for maintaining the NIS. A Model KHP-17D11 relay is used in the NIS flux deviation and miscellaneous control and indicator drawer. The replacement has been changed to a model KHU-17D11, which has a polycarbonate case instead of nylon. The relay provides signals when flux deviations occur in the reactor core when reactor power is above 50%. A Model CD1147 diode is used in the NIS source range start-up rate circuitry and in the rate calibration module assembly.

Safety Evaluation: The replacement relay is equivalent in form, fit and function and will not change the operating characteristics of the NIS in any manner. The replacement diode will perform exactly the same function as the original and has been certified by Westinghouse to be equivalent in form, fit and function. Westinghouse has reviewed the known changes in characteristics of form, fit, function and materials of the components described above and has certified that interchangeability has been maintained, therefore, the new components may be used for replacement as-is.

PLANT MODIFICATION REQUEST 02529 Revision: 0

Title: Control Room Window Modification And Fire Barrier Corrections

Description: This modification removes the middle panel of the window between the Shift Supervisor office and the foyer for easy handling of correspondence. It also removes an apparent discrepancy in the USAR which states that "peripheral rooms in the Control Room have automatic smoke detection and are separated from the Control Room by a 1-hour barrier".

Safety Evaluation: USAR Figure 9.5.1-2 Sheet 4 does not identify any 1-hour barriers for the Control Room. None are installed by design. The Fire Hazards Analysis takes no credit for any 1-hour barriers. The modification to the window does not affect the function, operation, structural integrity or reliability of the Control Room or any other room.

PLANT MODIFICATION REQUEST 02531 Revision: 0

Title: Documentation Update For Reactor Cavity Cooling Fans CGN02A/B

Description: Initially, the cavity cooling fans were Buffalo Forge fans with 25 hp. Westinghouse motors. They were changed to Joy fans with 50 hp. Reliance motors via Interim Design Change Package (IDCP) M-110-W. Vendor documents, as well as a USAR Table, were not revised at that time to incorporate the replacements. Therefore, via the disposition to Engineering Evaluation Request (EER) 87-GN-01, the appropriate documents are being revised to indicate the current configuration of the plant. As for the USAR revision, Table 9.4-12 (Sheet 4) is being revised using information from USAR Figure 9.4-6, as well as from the fan data sheet.

Safety Evaluation: This change updates documentation to reflect the current configuration of the plant. This does not impact safety analysis.

PLANT MODIFICATION REQUEST 02535 Revision: 0

Title: Pressurizer Spray Valve BB PCV-455B Packing Box Assembly Material

Description: Valve BB PCV-455B has a Packing Box Assembly installed without the appropriate material certifications or correct material. Continued operation of the plant is justified based on the information provided in Engineering Disposition to WR 01285-88 and the NRC approval for operation. However, the Packing Box Assembly shall be replaced at the next outage of sufficient duration but no later than Refuel III Outage.

Safety Evaluation: The valve packing box has been shown to be stable under assumed severely flawed conditions. Significant margins against conservative criteria have been demonstrated. The packing box should satisfy its service requirements without incident.

PLANT MODIFICATION REQUEST 02538 Revision: 0

Title: Wide Range Reactor Coolant System Cold Calibration Level Indication Addition

Description: In preparation for refueling and/or steam generator (SG) and reactor coolant pump (RCP) maintenance, Reactor Coolant System (RCS) level must be decreased below reactor vessel head and/or below the inside top of the hot leg, also known as mid-loop level. The present wide range pressurizer level indication will be replaced by a "Wide Range RCS Cold Calibration" level indication that will cover the blind spot between the hot leg and lower tap of the present level indicating system. The hot leg level is presently monitored by a separate level indicating system and alarms are provided to alert the operator of high coolant level in the hot leg during SG/RCP maintenance. A low alarm is provided for operator action to preclude uncovering the Residual Heat Removal (RHR) suction. The wide range cold calibration level system will be extended to the bottom of the hot leg to provide overlapping indication in this region. Local indication will be duplicated in the main control room for the above described wide range RCS cold calibrated level monitoring. Diversification will be achieved by using a transmitter for control room indication and mechanical local indicator. All of the above level indicators will be "pressure compensated" to eliminate pressure/vacuum influence on level measurement due to the nitrogen blanket pressure (or vacuum during rapid drain down) in the pressurizer steam space.

Safety Evaluation: The instrument tubing used for pressure compensation will use the existing pressurizer upper nozzle for the safety related transmitters BB LT-461, PT-457 and PT-458. A portion of the tubing added by this modification shares the existing seismic category 1 tubing supports provided for the above safety related transmitters. These supports are designed to carry seismic loading for two tubing runs, therefore, this additional tubing will not affect the existing safety related functions of pressurizer level or pressure. The remainder of the tubing supports are either designed to meet the II/I criteria or determined to have no impact on the safety related items in their vicinity. During plant operational modes 1 through 4, the above instruments will be isolated with normally closed manual isolation valve(s). These valve(s) will only be open after RCS depressurization, prior to drain down or refill, in preparation for, or after, refueling and/or maintenance operations. None of this instrumentation is required to perform a safety or post accident monitoring function. It is not required for hot or cold safe shutdown, either from the control room or outside the control room. The present design "Pressurizer Wide Range" indication does not perform any safety function nor is it a post accident or safe shutdown level indicator. The resulting configuration has no impact on Technical Specifications or on the safety design bases for the RCS. Table 3.11(B)-3 Page 25 in the USAR must be updated to delete the presently installed pressurizer wide range level transmitter, BB LT-462. The RCS cold calibrated level transmitter is not required for hot/cold shutdown, nor for Loss Of Coolant Accident/Main Steam Line Break/High Energy Line Break monitoring/mitigation. It is not required for the reactor coolant pressure boundary and therefore should not be included in this table. Figure 5.1.1, Sheet 1 and 2, in the USAR, will require update to show the new RCS Cold Calibrated Level indication.

PLANT MODIFICATION REQUEST 02543 Revision: 0

Title: Residual Heat Removal Sample Lines Non-Conformance

Description: After removal of the freeze plugs on lines EJ-034-ECB-3/4" and EJ-056-ECB-3/4", a slight bulge was noted on each line. The subject lines are routed from their respective Residual Heat Removal (RHR) pump minimum flow recirculation lines to the Nuclear Sampling System. The design function of these lines is to provide a pressure boundary for the RHR System which contains reactor coolant. The reported non-conformance has potentially created a weakened area in the piping. The concern is that this weakened area could develop into a leak during either an overpressurization of the line or a Safe Shutdown Earthquake event.

Safety Evaluation: Engineering has verified that the piping can withstand the maximum expected pressure (i.e., shut off pressure of the RHR Pump) and the design normal, upset, and seismic induced stresses in the piping system are low compared to the design allowable stresses. Although this bulge was discovered after the removal of a freeze plug, it would appear yielding had occurred. Engineering has determined that the piping was not over-stressed based upon an acceptable Penetrant Test (PT) of the entire freeze plug area after removal of the seal (i.e. no cracks were found). If the piping had been taken to its yield point, cracks in the area of the bulge would have been found. The difference in the outside diameter dimensions taken before and after freeze sealing was small.

PLANT MODIFICATION REQUEST 02561 Revision: 0

Title: Removal Of Valves From Steam Generator Feed Pump Turbine Exhaust Casing

Description: Valves FC V020, V021, V040 and V041 are auxiliary test connection isolation valves attached to the Steam Generator Feed Pump turbines exhaust casings. These valves are capped and were provided to facilitate installation of test equipment during turbine performance testing. The subject modification deletes valves FC V020 and V021 from Steam Generator Feed Pump "A", and valves FC V040 and V041 from Steam Generator Feed Pump "B" and associated piping. It also provides seal welded pipe plugs at their connection points on the turbine casings. Plant performance Testing Valves FC V020, 021, 040, and 041 were removed to eliminate possible condenser air inleakage and to eliminate the tripping hazard.

Safety Evaluation: The modification will have no impact on the operability of the Steam Generator Feed Pumps and will also reduce the possibility of air inleakage into the condenser and reduce obstructions during turbine maintenance. The Steam Generator Feed Pump Turbines are not safety related and serve no safety design function as described in USAR Section 10.4. The subject modification has no impact on any WCGS Technical Specification or associated bases. The only proposed change to the USAR involves revising USAR Figure 10.4-6 sheets 7 and 8 to reflect the deletion of the above valves.

PLANT MODIFICATION REQUEST 01422 Revision: 0

Title: Removal Of Automatic Opening Of Main Steam Drain Trap Bypass Valves And Main Turbine Stop Valve On Turbine Trip

Description: This modification involves the removal of the automatic opening of the main steam drain trap bypass valves and the main turbine stop valve above seat drain on turbine trip. The purpose of these valves is to drain water from the steam lines following a turbine trip. Once open, these valves currently cannot be reclosed until the turbine trip feature is reset at the Electro-Hydraulic Control (EHC) panel. This feature was originally provided to ensure these valves are open prior to plant restart, thereby precluding moisture carry over into the turbine. However, opening these valves provides additional heat removal capabilities to the extent that the plant approaches Technical Specification limits for cooldown. To prevent this from occurring, the automatic opening of these valves on turbine trip is deleted. To ensure these valves are open during plant startup, Operations Procedures SYS AC-120 and GEN 00-003 specifically require these valves to be open. These valves can be operated from the Control Room.

Safety Evaluation: The Main Steam and Main Turbine Systems are non-safety related. The subject drain valves support operation of these two systems and are also non-safety related. The circuits being modified are non-Class 1E. There are no modifications to structures, systems, or components required for plant safety. The automatic opening of the subject drain valves is causing excessive Reactor Coolant System cooldown following a turbine trip. Deletion of this automatic function removes the potential for these valves to cause the Reactor Coolant System cooldown rates specified in Technical Specification 3/4.4.9 to be exceeded. No other Technical Specifications or their bases are affected.

PLANT MODIFICATION REQUEST 01513 Revision: 0

Title: Modification Of Three Missile Doors To Emergency Exits

Description: This modification serves to modify doors 52011, 52031 (diesel generator room exterior missile doors) and 14032 (rod control cabinet room exterior missile door) to emergency exits from their present lockable state. This modification will allow emergency egress in case of a fire or accident from both "ends" of rooms 5201, 5203 and 1403.

Safety Evaluation: The three subject doors are Q-listed (safety related) because they have been assigned a seismic category I design basis (refer to USAR Table 3.2.1 Sheet 21, Section 8.2). As a result of this, these doors are required to be maintained in a safety related state while the plant is operational. Assurance must be provided that this modification will not invalidate the safety related criteria maintained on the subject doors at the present time. The following modifications provide that assurance:

1) Doors 52011 and 52031 are changed from type B doors to type G doors (changes the method the computer monitors the doors to provide additional security). Door 14032 is already a type G door.

2) On all three doors, the locking assemblies will be disabled and a "dummy" lock cylinder will be installed in place of the present locking cylinders.

3) A lighted emergency exit sign will be installed over door 14032. The other two doors already have lighted exit signs over them.

As scoped above, this project will not impact the missile resisting capabilities of each of the doors.

PLANT MODIFICATION REQUEST 01541 Revision: 0

Title: Replacement Of Level Switches, Solenoid Valves And Relays For The Emergency Diesel Generators

Description: This modification replaces the existing level switches, solenoid valves and relays for the Emergency Diesel Generators with vendor supplied environmentally qualified components. The Emergency Diesel Generators were procured to a specification which required the electrical components to be seismically and environmentally qualified. At the time of equipment delivery, some of the components had not yet received environmental qualification. Subsequently, Colt Industries has shipped qualified components to Wolf Creek Generating Station. It should be noted that the change out of these components is an enhancement and not a regulatory requirement as the components are located in a mild environment.

Safety Evaluation: Change out of the air start solenoid valves, level switches, and the static exciter voltage relays for the Emergency Diesel Generators with components which meet environmental qualification will not increase the probability of an accident which has been analyzed in the USAR. These components are not involved in any accident scenario as an initiator or component whose failure would lead to an accident initiation.

PLANT MODIFICATION REQUEST 01544 Revision: 0

Title: Installation Of Gates To Control Access To Areas Greater Than One Rem Per Hour

Description: Four locked wire mesh carbon steel gates are to be installed in the Containment Building controlling access at each of the four entry points from floor elevation 2000'-0" to floor Area 2201. These gates are being added to provide positive access control to an area having a potential radiation level greater than 1 Rem per hour in accordance with Technical Specification Section 6.12.2 and USAR Section 12.3.1.2. The modification has no impact on any Wolf Creek Generating Station Technical Specification Limiting Condition for Operation or associated bases. It also enables Health Physics personnel to more effectively control personnel access in Containment General Area 2201, as discussed in the Administration controls Section 6.12.2 of the WCGS Technical Specifications.

Safety Evaluation: The added gates have been seismically designed for both the open and closed gate position, thereby assuring the integrity of safety related equipment in the vicinity of the gate. Flooding has been evaluated due to the addition of the wire mesh gates. The gates will be painted in accordance with Specification A-125(Q) meeting the criteria of ANSI N101.2 assuring coating integrity following a Design Basis Accident. As a result, no clogging of the containment sump screens (Reference USAR Page 6.1-4) will occur. The gates will not impact any previous flooding hazard analysis as the gates do not contribute to any significant volume reduction, thereby having no impact on the flood level in the containment. The addition of the subject gates will not impact the fire hose stations and portable fire extinguishers located outside the secondary shield wall. Although the gates will be locked from the outside, the fire brigade will access the gates to elevation 2001'-4" by the use of keys or wire cutters. Therefore, the subject gates will not impact the fire protection features described in USAR Section 9.5B, Section RB.4. Although the gates are locked from the outside, they are designed such that they can be opened from the inside, thereby assuring unrestricted emergency egress from inside the secondary shield wall. Administrative control of keys for the subject gate shall be maintained by the Shift Supervisor and/or Health Physics in accordance with Technical Specification 6.12.2.

PLANT MODIFICATION REQUEST 01573 Revision: 3

Title: Replacement of 1 1/2 Inch Steam Generator Blowdown Piping

Description: The lines to Steam Generator Blowdown Flash Tank, TBM03 are subject to erosion because of high velocity, two phase flow. Replacement of the 1 1/2 inch steam generator blowdown piping with 4 inch piping will reduce the velocity in the lines and the possibility of failure because of erosion. Valve BM V-003 (Steam Generator "A" blowdown flow throttling valve) is difficult to access by some operators. Engineering has been requested to improve the accessibility by providing either a platform or relocating (within original installation tolerances) piping, pipe supports and conduit obstructions.

Safety Evaluation: Piping is being replaced with 4 inch schedule 160 (EBD) piping to reduce the velocities and excessive erosion in the Steam Generator blowdown piping. The piping is located in the non-safety related portion of the Steam Generator blowdown system between manual throttle valves and the Steam Generator Blowdown Flash Tank. The above piping change results in a design improvement to the non-safety related portion of the Steam Generator Blowdown Piping System.

PLANT MODIFICATION REQUEST 01609 Revision: 3

Title: Increase In Listed Design Pressure Of Central Chiller Condenser And Evaporator, And Chilled Water Expansion Tank

Description: This revision increases the design pressure listed in USAR Table 9.4-19 for the Central Chiller condenser and evaporator, and for the Chilled Water Expansion Tank. These corrections were noted while preparing the USAR Changes for the modification which provides removable sections on the shell of the Central Chiller condenser.

Safety Evaluation: The Central Chiller and the Chilled Water Expansion Tank are not required for safe shutdown of the plant or for mitigating an accident. Their operation does not impact any safety related equipment.

PLANT MODIFICATION REQUEST 01614 Revision: 1

Title: Instrument Isolation Valve Substitution On Moisture Separator Reheater Drain Tanks

Description: The isolation valves for the level controllers and the level switches on the Mositure Separator Reheater (MSR) Drain Tanks A and B, the first Stage Reheater Drain Tanks A and B, and the second Stage Reheater Drain Tanks A and B are leaking excessively around the bonnet gasket and packing area. The valves require a long lead time if they are purchased through the original 10466-M-003 (General Electric) specification. Specification 10466-M-234 (Bechtel) was reviewed and found to be acceptable for use in replacement of the subject valves. The impact of the new valve on the tanks, level switches and controllers has been coordinated with General Electric. General Electric response was that no impact to the tanks, level switches, or level controller would result.

Safety Evaluation: These valves are not safety related. They have no affect on safety related equipment. This modification provides Engineering approval to substitute one set of purchase requirements for non-safety related valves (General Electric) with another set of requirements (Bechtel) for these valves.

PLANT MODIFICATION REQUEST 01672 Revision: 0

Title: Addition Of Steam Dump To Auxiliary Boiler

Description: This modification provides a means of dumping steam to ensure that minimum load requirements are met during periods when the plant is shutdown and the Auxiliary Boiler is needed but the load on the Boiler is not enough to keep the Boiler running properly. The steam dump is required for low load conditions to ensure that the boiler is above its 8000 pounds mass per hour minimum load rating. Automatic control of the steam dump valves is to be provided to allow for steam dump to atmosphere when steam demand is less than 9000 pound mass per hour. New and existing flow transmitters should be used to sense steam demand for control of the steam dump valve. Provisions shall also be made for the installation of a silencer on the vent to atmosphere to mitigate noise problems when steam is being relieved to atmosphere.

Safety Evaluation: The Auxiliary Steam and Auxiliary Steam Generation Systems serve no safety function. The Auxiliary Boiler is used during plant shutdown to supply plant steam requirements. The steam dump will be used during periods of low load on the boiler to establish an artificial load to prevent the boiler from going below its minimum design load. The addition of the silencer on the auxiliary steam vent line does not change the function of the system. The silencer will reduce noise level while dumping auxiliary steam to the atmosphere. These changes will not affect any other system. A reduction in the margin of safety as defined in the bases for any Technical Specification does not result since the subject modification affects a non-safety related system and has no impact on any Technical Specification.

PLANT MODIFICATION REQUEST 01679 Revision: 0

Title: Installation Of Three Inch Flush Connection In The Floor And Equipment Drains System

Description: The subject modification provides for the installation of a three inch flush connection in the ANSI B31.1, II/I supported portion of the Floor and Equipment Drains (LF) System to allow the localized flushing of two valves (LF HV-105 and LF HV-106). The flush connection will allow for the decontamination of these valves and reduce the dose received by personnel walking by or working in the area of the valves.

Safety Evaluation: This modification has no impact on the structural integrity, function, and/or operability of the LF System or any other system and does not affect the ability to fulfill any safety related function as described in the safety design basis in USAR Section 9.3.3. The use of the above described flush connection has no impact on any Wolf Creek Generation Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 01697 Revision: 0

Title: Update Of Condensate And Condensate Demineralizer Piping Drawings

Description: The subject modification involves updating condensate and condensate demineralizer piping drawings to incorporate as-built conditions to reflect the existing physical arrangement of a pipe and instrument tubing.

Safety Evaluation: No operational or functional changes result. The subject modification does not involve any physical changes to piping or equipment. As indicated in USAR Sections 10.4.6.1.1 and 10.4.7, the condensate demineralizer and the condensate systems serve no safety function and have no safety design basis. The only impact to the USAR is the proposed revision to Figure 10.4-5, Sheet 2.

PLANT MODIFICATION REQUEST 01704 Revision: 0

Title: Addition Of Audible Alarm For Auxiliary Feedwater Pump Turbine Mechanical Overspeed Trip

Description: This change is provided for system operability enhancement by providing an annunciator window in addition to the existing indicating light for the Auxiliary Feedwater Pump Turbine Mechanical Overspeed Trip. The annunciator will clear only when the trip has been reset and the operator resets the annunciator.

Safety Evaluation: The existing indicating light is Class 1E and no credit is taken for the non-Class 1E annunciator. The non-Class 1E annunciator is electrically isolated from the Class 1E indication, therefore it will have no adverse effect on the existing Class 1E light indication.

PLANT MODIFICATION REQUEST 01735 Revision: 1

Title: Addition Of Chemical Addition Pot And Piping To Plant Heating System

Description: The subject modification adds a chemical addition pot and associated piping to the startup test flanges of the plant heating system. The chemical pot will allow for easier addition of corrosion inhibitor to the system. The system is a closed system which transports heated water to space heaters throughout the plant. This modification revision provides for a chemical addition pot design for the plant heating system and supersedes the eductor design of the previous revision. The present chemical addition pot has unsatisfactory materials and welds and is replaced along with the required minor piping reconfiguration. A vent valve is added to eliminate excessive air introduction into the system. The existing funnel is designated in the new design. A field fabricated drip pan should be installed under the chemical addition pot, vent, drain and funnel area to contain spills which occur and prevent spillage through the 2033'-0" level Turbine Building grating. The drip pan should be removable for disposing of spilled liquids, or the means should be available for pumping the drip tray through small tubing to a container.

Safety Evaluation: The plant heating system does not perform any safety related function and has no safety design basis.

PLANT MODIFICATION REQUEST 01760 Revision: 0

Title: Addition Of Ultrasonic Flowmeters To The Boric Acid Transfer Pumps And Emergency Fuel Oil Transfer Pumps

Description: This modification adds ultrasonic flowmeters for ASME Section XI Inservice Inspection Testing of the Boric Acid Transfer (BAT) Pumps PBG02A/B and Emergency Fuel Oil (EFO) Transfer Pumps PJE01A/B. Design features of this system are:

1) Light weight flow transducers are clamped onto the pipe. They are not inserted into the ASME Section III pipe which forms the pressure boundary.

2) Flow indicators for BAT pumps are located outside the shield walls that enclose the pumps/tanks.

3) Conduits, raceways and indicators are installed II/I using seismically designed supports.

These features minimize the time required to conduct/evaluate performance tests of the subject pumps, reduce radiation exposure and therefore are consistent with ALARA considerations.

Safety Evaluation: The addition of the ultrasonic flow transducers has been evaluated and determined to have no impact on the structural integrity of the associated piping. The subject flow transducers have been evaluated for II/I considerations and determined to pose no hazards. The flow transducers are clamped on to the diesel fuel oil piping. This eliminates fire hazards. The subject modifications will not affect the ability of the Chemical and Volume Control System and the Emergency Fuel Oil System to fulfill their design bases as described in USAR Sections 9.3.4 and 9.5.4, respectively. The resulting configuration has no impact on the Wolf Creek Generating Station Technical Specifications or any associated bases. USAR Section 9.5.4.4, Page 9.4-48; Fig. 9.3-8, Sheet 5, and Figure 9.5.4-1 will require update to reflect the addition of the subject flowmeters.

PLANT MODIFICATION REQUEST 01843 Revision: 1

Title: Removal Of Reactor Vessel Head With No Water In The Refueling Pool

Description: This revision to the subject modification allows the removal of the Reactor Vessel Head with no water in the refueling pool for the Reactor Vessel Head O-Ring Outage only.

Safety Evaluation: This disposition limits the maximum elevation the Reactor Vessel Head can be lifted during Reactor head disassembly and reassembly. The imposed maximum drop of the head in free air of 18' - 8" above the Vessel flange ensures that if a drop were to occur there will be no consequential damage to the structural integrity of the Reactor Vessel and core cooling capability and the integrity of the fuel cladding will be maintained. This is based on a resultant total vertical deformation of the Reactor Vessel and supports of 0.72 inches being within prescribed limits. In addition, any postulated drop of the head onto the refueling floor during travel to the Head storage stand will not affect the capability of core cooling due to physical separation of both Engineered Safety Features (ESF) trains.

PLANT MODIFICATION REQUEST 01843 Revision: 2

Title: Removal Of Reactor Vessel Head With No Water In The Refueling Pool

Description: This revision to the subject modification allows the removal of the Reactor Vessel Head with no water in the refueling pool for the Refueling Outage.

Safety Evaluation: This disposition limits the maximum elevation the Reactor Vessel Head can be lifted during Reactor head disassembly and reassembly. The imposed maximum drop of the head in free air of 18' - 8" above the Vessel flange ensures that if a drop were to occur there will be no consequential damage to the structural integrity of the Reactor Vessel and core cooling capability and the integrity of the fuel cladding will be maintained. This is based on a resultant total vertical deformation of the Reactor Vessel and supports of 0.72 inches being within prescribed limits. In addition, any postulated drop of the head onto the refueling floor during travel to the Head storage stand, will not affect the capability of core cooling due to physical separation of both Engineered Safety Features (ESF) trains.

PLANT MODIFICATION REQUEST 01858 Revision: 0

Title: Addition Of AMSAC

Description: This modification provides a redundant system, Anticipated Transient Without Scram (ATWS) Mitigating System Actuation Circuitry (AMSAC), to sense the need for Auxiliary Feedwater actuation. It will cause a Turbine trip and actuate Auxiliary Feedwater. Current design provides for a Reactor Trip upon a Turbine Trip Signal.

Safety Evaluation: This system is isolated from Class 1E equipment and is built with reliability to ensure actuation occurs when needed without compromising systems important to safety. The addition of ASMAC is a requirement of 10 CFR 50.62 and does not increase or decrease the margin of safety as defined in the bases for any Technical Specification. AMSAC is a non-safety, redundant, diverse backup to the plant Reactor Protection System.

PLANT MODIFICATION REQUEST 01974 Revision: 0

Title: Establishment Of Inspection Criteria For Safety Related Water Structures

Description: This modification adds a specification into the system to establish criteria for inspection of safety related water structures.

Safety Evaluation: This specification is a compilation of the commitments and current practices/procedures that are used to inspect the safety related water structures as set forth in Dames and Moore, "Procedures for Periodic Inspection of the Main Dam and Reservoir, and the Ultimate Heat Sink", the USAR, and regulatory correspondence. The only actual methodological change is the height measurement of the Ultimate Heat Sink (UHS). The companion markers are replaced by setting a level rod on the submerged benchmark by a diver. This change shall not detract from the capabilities of measuring the height of the UHS dam.

PLANT MODIFICATION REQUEST 02018 Revision: 3

Title: Replacement Of ASCO Solenoid Valves

Description: This plant modification involves the replacement of the originally supplied ASCO solenoid valves with ASCO solenoid valves that have internal terminal blocks in lieu of pigtails for various air-operated valve applications. All work is complete. This revision was issued to resolve Nuclear Plant Engineering Licensing Supplement concerns identified during the recent Safety Systems Outage Modification Inspection (SSOMI).

Safety Evaluation: This revision updated the safety evaluation for this modification. The replacement solenoid valves were purchased in accordance with requirements of the purchase order that was used to procure the original solenoid valves. The Material Received Report for the replacement parts includes a Certificate of Conformance documenting their compliance with the requirements of the original purchase order, including seismic and environmental qualification. Based on this discussion, the replacement solenoid valves are demonstrated to be equivalent or superior in form, fit, function, material and qualification to the original solenoid valves.

PLANT MODIFICATION REQUEST 02040 Revision: 0

Title: Addition Of Check Valve To Prevent Backpressure On Radiation Monitor

Description: Valve FB V218 is being added to the discharge of radiation monitor FB RE-50, to prevent back pressure caused by steam flashing in the Auxiliary Steam Condensate Recovery Tank (TFB03) from being transmitted to the flow meter of FB RE-50. This modification, located in the Radwaste Building, does not alter the function of the radiation monitor and does not affect system performance.

Safety Evaluation: The above described modification will not affect the operability or function of the Auxiliary Steam System as described in USAR Section 9.5.9. The Auxiliary Steam System serves no safety function and has no safety design bases. The only proposed change to the USAR involves revising USAR Figure 9.5.9-1 Sheet 3 to reflect the associated changes.

PLANT MODIFICATION REQUEST 02045 Revision: 0

Title: Addition Of Two Ionization Smoke Detectors

Description: Technical Specification Table 3.3-11 (Fire Detection Instruments) is being updated to reflect the addition of two new ionization smoke detectors. The two detectors are being added in the new Instrument And Control Hot Shop, Room 1414, in the existing fire detection zone 118. The addition of the detectors is required to provide adequate detection commensurate with 10 CFR 50 Appendix R requirements and previous commitments in the USAR.

Technical Specification Table 3.8-1 (Containment Penetration Conductor Overcurrent Protective Devices) is also being updated to reflect the deletion of the power feeds for the Steam Generator Instrument Test System (SGITS) which are no longer required.

Safety Evaluation: The addition of the above described detection instrumentation assures adequate warning capability is available for new Room 1414 to enable prompt detection of fires that could occur. The subject modification improves the effectiveness of the fire detection system for the associated area as described in the revised Fire Hazards Analyses.

The SGITS cables have been determinated from their associated Containment Electrical Penetration Assembly (EPA). Therefore, since the subject components are no longer connected to an EPA, Regulatory Guide 1.63 is no longer applicable for these components and they are deleted from Table 3.8-1.

PLANT MODIFICATION REQUEST 02057 Revision: 0

Title: Addition Of Condenser Air Inleakage Measurement System

Description: The subject modification provides a permanent condenser air inleakage system to get a more accurate airflow measurement at the condenser vacuum pump discharge than provided by existing instruments.

Safety Evaluation: The system is a non-safety related system. As advised by the condenser pump manufacturer, the system is designed such that the pump will not have a back pressure of greater than 1 psig. Under this condition the system will be capable of measuring a maximum total flow of 35 scfm. If the subject system is required to be used to measure a flow greater than 35 scfm, the time period shall be limited to less than 15 minutes. The subject system modification shall never be used during establishment of condenser vacuum. The subject modification provided in this package will not adversely affect the operation of the condenser vacuum pumps or the safety of plant operation.

PLANT MODIFICATION REQUEST 02068 Revision: 0

Title: Replacement Of Safety Related MDA Scientific Chlorine Monitors

Description: In order to improve performance and reduce malfunction of the existing safety related MDA Scientific Inc. chlorine monitors, it is proposed to replace them with Class 1E commercial grade (dedicated) Xertex Delta chlorine sensor/transmitter. This design change package is subjected to Xertex Delta chlorine sensor/transmitter passing the Class 1E qualification test to be performed by Wyle Laboratory.

Safety Evaluation: Following qualification of Xertex Delta chlorine monitors as Class 1E instruments, the quality of the monitor will be equal to or better than the existing MDA chlorine monitors. In addition, their reliability/maintainability is expected to be superior. Since Xertex Delta chlorine monitors do not have any moving parts and have faster response times, compared to the existing MDA chlorine monitors, it is expected that the installation of Xertex Delta chlorine monitors shall enhance the performance of the control room emergency isolation system. This will reduce the unwanted challenges to the Engineered Safety Features Actuation System and isolation dampers and shall not degrade any safety related or special scope systems.

PLANT MODIFICATION REQUEST 02094 Revision: 0

Title: Revision To The Method Of Initiation Of The Foam Spray System At The Fuel Oil Storage Tank

Description: This modification revised the method in which the foam spray system at the Fuel Oil Storage Tank was initiated once a fire was detected. The original design had an automatic initiation of the foam spray system upon detection of a fire. This modification revised the design to provide a Control Room alarm upon detection and a Control Room suppression alarm upon manual initiation of the foam spray system.

Safety Evaluation: The location of this non-safety related site structure is such that a fire involving the Fuel Oil Storage Tank will not affect any safety related or special scope system or structure because of the remoteness of the tank, the design provisions for a full capacity dike, and the foam suppression system. Therefore, a fire at this structure will not impact site fire hazards analysis as required by Appendix R of 10 CFR 50.

PLANT MODIFICATION REQUEST 02131 Revision: 1

Title: Repair Procedure For Motor Leads For The Hydrogen Mixing Fans

Description: This modification inserts a repair procedure for motor leads in the instruction manual for the hydrogen mixing fans. The preferred method for repairing damaged motor leads is to cut off the damaged portion and reterminate the lead in accordance with the manufacturers recommended practices or site accepted practices for termination. In this case, it is recommended that the leads be replaced. If replacement of the leads is unacceptable and if the damage to the cable is only sleeving or insulation damage (not to the conductor), then and only then, will this repair procedure be used.

Safety Evaluation: The repair procedure very nearly returns the cable to its original state. The electrical function of the subject repair does not affect the ability of the repaired cable to perform its required function, under all conditions. The ability of the hydrogen mixing fans to perform their required function, as discussed in USAR Section 6.2.2.2, is not affected by the implementation of the repair procedure. The subject repair does not affect the margin of safety of any Wolf Creek Generating Station Technical Specification. The materials used are environmentally qualified in the Motor Qualification Report and are recommended by the manufacturer. Because the repaired cable is functionally the same as the original cable and because the procedure is only to be used as specified above, there is no major reduction in the degree of protection provided to the public health and safety. Therefore, no unreviewed safety question exists.

PLANT MODIFICATION REQUEST 02167 Revision: 7

Title: Reinstatement Of Design For Three Way Temperature Control Valve

Description: This modification revision to the electrical equipment room fan coil unit (Auxiliary Building HVAC) reinstates the design for the three way temperature control valve. This valve had to be replaced with a globe valve for manually throttling the supply air registers and the screen on the suction side of the fan coil unit. Also, this modification provides the temperature settings of the temperature switches associated with the electrical equipment room fan coil units.

Safety Evaluation: The electrical equipment room fan coil unit and associated controls are non-safety related and there are no safety design bases associated with the unit as defined in USAR Section 9.4.3.1.1. The modification has no impact on the safety of the plant as the Electrical Equipment Room Cooling System and Chilled Water System serve no safety function, and its failure does not affect safe shutdown of the plant. The replacement three way control valve, GL TV-185, and supply air registers and screens do not affect the II/I analysis. The revised setpoint for the temperature switch ensures parallel operation of electrical equipment room fan coil units to maintain the Motor Generator (MG) Set Room temperature within the design limits. The components being installed are not safety related and do not interface with any other safety related system. Operation of the associated fan coil unit is not required to support the operation of any equipment important to safety. There will be no hazard created by this modification. Failure of the modified components and associated fan/coil unit does not impact any safety related component, ystem, or structure and does not affect the safe shutdown of the plant. The modification has no impact on any Wolf Creek Generating Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 02185 Revision: 0

Title: Flush Connections And Double Isolation Valves For Steam Generator Blowdown System

Description: This modification facilitates cleaning of Process Sampling System (RM) sample lines from Steam Generator Blowdown System by providing flush connections and double isolation valves.

Safety Evaluation: The Process Sample System serves no safety function. Also, the portion of the Steam Generator Blowdown piping affected by this design change has no safety function. This modification does not change the safety related function of the Steam Generator Blowdown System and does not affect any interconnected system.

PLANT MODIFICATION REQUEST 02211 Revision: 0

Title: Safety Classification Of Main Turbine Stop Valve Position Indication Switches And Hydraulic Control Oil Pressure Transmitters

Description: This modification changes the safety classification of the Main Turbine Stop Valve position indication switches and the hydraulic control oil pressure transmitters to safety related.

Safety Evaluation: Changing the component classification of the subject components from special scope to safety related will help to ensure that the commitments for safety related equipment located in a mild environment are met. Meeting these commitments will ensure a high degree of system reliability. Therefore, this change will ensure a high degree of system reliability.

PLANT MODIFICATION REQUEST 02238 Revision: 0

Title: Drawing Revision To Reflect Enhanced Reactor Coolant Pump #1 Seal Design

Description: Westinghouse modified the Reactor Coolant Pump (RCP) #1 seal to improve seal performance during pump startup. This modification consisted of incorporating new drawings and revising existing drawings in the Wolf Creek Generating Station Document Control System as applicable per Westinghouse letter SAP-87-535. RCP #1 Seal Faceplate material has been revised to Silicon Nitride.

Safety Evaluation: The subject change entails revision/incorporation of the Reactor Coolant Pump #1 Seal drawings. The seal material change is an enhancement for better reliability with no change in design function or leak-off flows. No safety feature of the plant is impacted.

PLANT MODIFICATION REQUEST 02252 Revision: 0

Title: Emergency Diesel Generator Modification

Description: The emergency diesel engine lubrication system provides essential lubrication and cooling for the components of the Emergency Diesel Generators. The subject modification exchanges the inlet and outlet on filters FKJ10A and FKJ10B. This exchange provides proper flow direction through the filter housings to allow strainers with a higher micron rating to replace the existing filter elements. The Rocker Arm Lube Oil filters FKJ10A and FKJ10B each contain two 5 micron filter elements. The filter high differential pressure alarm sounds if the diesel or the motor driven rocker arm lube oil pump is running. This is due to the cellulose filter elements absorbing condensate from the lubricating oil and increasing the differential pressure across the filter enough to activate the alarm. After 4-6 hours of running, the lubricating oil becomes warm enough to eliminate the condensate and the alarm clears. The Rocker Arm Lube Oil is a separate system which lubricates the rocker arm assemblies of the diesel. After consulting with Colt Industries (manufacturer), the 5 micron filter element will be replaced with a 30-40 micron strainer to lower the differential pressure. The filter and strainer housings are physically identical and the change can be accomplished by exchanging the inlet and outlet to obtain the proper flow through the strainer. Colt Industries has reviewed these changes and verified that the change does not affect the existing seismic or environmental qualifications for the emergency diesel generators and does not require any support/hanger changes.

Safety Evaluation: This change does not affect the ability of the Rocker Arm Lube Oil System to fulfill the Safety Design Bases as described in USAR Section 9.5.7.1 or the seismic and environmental qualifications of the diesel engines.

PLANT MODIFICATION REQUEST 02254 Revision: 3

Title: Control Building Staircase High Security Locking System

Description: This modification installs high security locking systems on hollow metal doors in the Control Building Staircase, Number C-1. This change permits quick secondary access to the vital equipment areas in the event of plant emergencies and thus greatly enhances Operations' ability to deal with plant emergencies and accident mitigation.

Safety Evaluation: This design change will not result in a reduction in the margin of safety as defined in any Technical Specification.

PLANT MODIFICATION REQUEST 02255 Revision: 0

Title: Main Steam Isolation Valve and Feedwater Isolation Valve Low Air Pressure Supply Alarm

Description: The Main Steam Isolation Valves (MSIVs) and Main Feedwater Isolation Valves (FWIVs) low air pressure supply alarm plant modification package is being issued. This package will install a pressure switch between the air pressure regulators (A and B) on each of the MSIVs and FWIVs and provide a single Control Room annunciator window for "MSIV/FWIV supply air pressure low" using the pressure switch contacts to alarm. The Panalarm logic card to be installed in Panel RK045 is a Panalarm Standard Sequential Visual Logic Card Model 70-A10. Initial considerations indicate no spurious alarm signals will be generated at the pressure switches due to various mode testing of the MSIVs and FWIVs. In the event these spurious alarm signals are generated, Panalarm recommends a Panalarm variable time delay logic card, modified to add a dropping resistor to the card, be installed to replace the Standard Sequential Visual Logic Card in Panel RK045. The variable timer shall be set at the minimum delay required to prevent the spurious alarms. In such case the Plant Modification Request will be revised accordingly.

Safety Evaluation: The pressure switches being added for the MSIV and FWIV air supply headers are for annunciation only and serve no safety function. The portion of the valves to which the subject pressure switches are being added is non-safety related and non-Class 1E.

PLANT MODIFICATION REQUEST 02261 Revision: 0

Title: Main Steam Isolation Valve Pressure Switch Isolation From DC Power Supply

Description: A review of the environmental qualification for the pressure switches on the nitrogen accumulators for the Main Steam Isolation Valves (MSIVs) has identified a potential failure mechanism. Therefore, the pressure switches will be isolated from their safety related DC power supply. The isolation of the subject pressure switches will be accomplished by the use of Class IE auxiliary relays. It should be noted that the Feedwater Isolation Valves (FWIVs) are not affected in a similar manner due to the fact that they fail closed, whereas the MSIVs fail as-is.

Safety Evaluation: The subject modification provides electrical isolation between the DC power supplies for the Main Steam Isolation Valves and their associated accumulator pressure switches. The present configuration provides a series input to the Engineered Safety Features (ESF) status panel utilizing the subject pressure switches in direct connection with the control circuit fuses for the MSIVs. The new configuration provides Class 1E interposing relays (the added relays meet the requirements of IEEE 384 and Reg. Guide 1.75) to provide isolation between the subject switches and fuses such that any postulated failure of the switches will not impact the fuses and associated control circuits. This modification is considered an upgrade since there has been no pressure switch failure mode identified that could degrade (i.e. blow) the subject control circuit fuses. Additionally the subject switches are used solely for Engineered Safety Features status panel input and perform no safety related control function (used for indication only), although they are qualified to maintain the pressure boundary. This modification has been designed to be fail-safe. Loss of power to the subject isolation relays will provide the associated status panel actuation for added assurance.

PLANT MODIFICATION REQUEST 02268 Revision: 0

Title: Containment Cooling Fan Vibration Switch Reset
Description: Containment Cooling Fan vibration switches occasionally actuate due to momentary high vibration levels (setpoint is 0.1g above normal). Since the switches can only be reset locally, a continuous alarm remains until the next time containment can be entered. This modification will allow vibration switch setpoints to be set up to 0.2g above normal and provide the capability to reset the switches outside containment.

Safety Evaluation: This modification provides remote reset capability and specifies the setpoint for the vibration monitors located on the Containment Coolers 1SGN01A, B, C and D. Remote reset will allow the computer point alarm to be reset without containment entry and enhances the operator's ability to distinguish nuisance alarms from alarms that would remain following reset. The setpoint is within the manufacturer's recommended allowable vibration levels and will minimize nuisance alarms from unrelated sources. The vibration switch monitors vibration level at the Containment Cooler Fan housing. The vibration switch contact status is monitored by the plant computer and a combined computer alarm is generated in the Control Room. The vibration monitoring circuits are non-safety related, provide computer alarm only and are not associated with the containment cooling fans control circuits. The subject modification enhances the present design by providing remote reset instead of recurring containment entry to locally reset the vibration monitor on spurious actuation. This will reduce radiation exposure to personnel. This modification has no impact on the ability of the containment cooling system to fulfill its safety design bases as discussed in USAR Section 6.2.2.2.

PLANT MODIFICATION REQUEST 02284 Revision: 0

Title: Electro-Hydraulic Control Cabinet Air Conditioner Replacement

Description: The existing Electro-Hydraulic Control (EHC) Cabinet Room Air Conditioner SGE13 (24,000 BTU/hr) in the Turbine Building can no longer be properly maintained to provide the proper cooling effect. Parts for the existing York unit are no longer manufactured and the existing York unit is required to be replaced with a new unit. A Plant Modification Package has been issued to install a new Carrier Room Air Conditioner for the existing unit SGE13 which is a York Model. The York unit is a 24,000 BTU/hr (nominally 2-ton) unit, whereas the Carrier unit is 32,500 BTU/hr (nominally 1 1/2-ton).

Safety Evaluation: The Engineering disposition provides for the addition of a non-safety related component in a non-safety related area, therefore no impact to any safety aspects of the plant are involved. The subject modification replaces existing York unit, SGE13, with a Carrier unit for the EHC Cabinet Room.

PLANT MODIFICATION REQUEST 02287 Revision: 0

Title: PORV Block Valve Logic Modification
Description: This modification revises the pressurizer Power Operated Relief Valve (PORV) Block Valve logic to delete the automatic opening feature of the block valves, provide seal-in circuitry to preclude mid-travel reversal of the valves, and provide a two second time delay upon completion of the valves opening stroke before the valve can be driven in the close direction. This will allow the motor to come to rest before it receives a reverse direction signal.

Safety Evaluation: This modification will reduce the possibility of the block valve actuator motor circuit breakers opening on overcurrent due to valve cycling and/or mid-travel reversal of the valve. This will help ensure that the block valves are available to isolate a stuck-open PORV.

PLANT MODIFICATION REQUEST 02289 Revision: 0

Title: Diesel Fire Pump Coolant Pressure Gauge Replacement

Description: This modification involves the replacement of 0-300 psig Diesel Fire Pump coolant pressure gauge with a 0-60 psig pressure gauge so that an accurate reading of the coolant pressure, usually around 8-10 psig, can be obtained.

Safety Evaluation: The Diesel Fire Pump is part of the Fire Protection system, which is non-safety related. This modification does not affect the function or operating characteristics of the Diesel Fire Pump or Fire Protection System and does not affect any safety related system.

PLANT MODIFICATION REQUEST 02302 Revision: 0

Title: Auxiliary Steam Reboiler Sample System Modification

Description: Auxiliary Steam Reboiler pH Analyzer (AE-55) and grab sample point is 12 inches downstream of the Chemical Addition System tap. Mixing of ammonia and hydrazine is not complete at this point and as a result sample test results do not reflect the true water chemistry. Inaccurate sample data makes it difficult to keep the Auxiliary Steam Reboiler within water chemistry specifications. Another sample point is available at FB V020 (Auxiliary Steam Reboiler Sample point Isolation Valve). This modification reworks the sample system in the most economical manner such that a sample may be taken from FB V020.

Safety Evaluation: The Auxiliary Steam System is not safety related and does not perform any safety related function. Changing the sample point downstream of Level Control Valve Station for the Auxiliary Steam Reboiler will assure that mixing of chemicals added to the feedwater is complete prior to sampling for water chemistry requirements. The subject modification does not affect the operability of the Auxiliary Steam System or any interconnected system.

PLANT MODIFICATION REQUEST 02307 Revision: 0

Title: Updated Vendor Instruction Manual For Fisher Controls
Description: Fisher Controls has replaced its 4150/4160 series pressure controllers with the 4150K/4160K series and has changed several part numbers. This change updates vendor instruction manuals so that the Fisher Controls 4150K and 4160K series pressure controllers and spare parts may be used to replace or repair 4150 and 4160 series controllers.

Safety Evaluation: This change updates vendor instruction manuals so that the Fisher Controls 4150K and 4160K series pressure controllers and spare parts may be used to replace or repair 4150 and 4160 series controllers. All are used in non-quality applications including the controller used in the special scope fire protection system. The new 4150K and 4160K controllers perform the same function and are direct replacements for the 4150 and 4160 series. Spare parts for 4150 and 4160 series controllers are compatible with 4150K and 4160K series controllers. This change does not affect any safety related system and does not affect the function or operating characteristics of the fire protection system.

PLANT MODIFICATION REQUEST 02308 Revision: 0

Title: Miscellaneous Building and Auxiliary Building HVAC Systems Drawing Revisions

Description: The subject change revises Piping and Instrument Drawings for the Miscellaneous Building and Auxiliary Building HVAC Systems to reflect the actual physical locations of the subject dampers with respect to their associated rooms and to agree with existing HVAC location drawings. Drawings will be revised to note actual damper location with respect to their associated room for drawing M-12GF01 and M-02GL01. The modification also provides correct room numbers for drawing M-OH1111.

Safety Evaluation: The subject change revises Piping and Instrument Drawings for the Miscellaneous Building and Auxiliary Building HVAC Systems to reflect the actual physical locations of the subject dampers with respect to their associated rooms and to agree with existing HVAC location drawings. No physical modifications to the plant are necessary.

PLANT MODIFICATION REQUEST 02317 Revision: 0

Title: Filter Holder Drawings for Radiation Monitors

Description: This modification incorporates parts lists for filter holders into the technical manuals for Radiation Monitors containing particulate and/or iodine sensors. This change also incorporates particulate and/or iodine subassembly drawings and parts lists.

Safety Evaluation: This modification incorporates parts lists for filter holders into the technical manuals for Radiation Monitors containing particulate and/or iodine sensors. No change to the form, fit or function of these monitors is involved with this addition to the technical manuals.

PLANT MODIFICATION REQUEST 02334 Revision: 0

Title: Typographical Error In Westinghouse Drawing and Instruction Manual Correction
Description: This change is issued to correct a typographical error on Westinghouse Drawing 1226E61 (M-716-00134) and in Instruction Manual (I/M) M-716-00183. These documents pertain to the Refueling machine and indicate the Master Overload setpoint at 400 lb. over the overload. The correct setpoint is 150 lb. over the overload. A revision to USAR Section 9.1.4.2.2, Items 9 and 10, is proposed to accurately state the various load setpoints.

Safety Evaluation: At no time has the plant had an incorrect setpoint due to the error in the above drawing. A review of all revisions to Surveillance Test Procedure STS KE-001 has determined that the correct master overload setpoint has always been utilized. A revision to USAR Section 9.1.4.2.2, Items 9 and 10, is proposed to accurately state the various load setpoints.

PLANT MODIFICATION REQUEST 02337 Revision: 0

Title: Containment Cooler Repair
Description: A leak has been reported in one of the cooling coil units on Containment Cooler SNG01A. One of the following options will be used to correct the leak:

<u>Option 1</u> - Locate the coil circuit of the Cooling Coil Unit in which the leak is located. Remove this coil circuit from service by cutting both ends on the supply and return side of the coil circuit at the cooling coil unit manifolds. Plug the supply and return points on the cooling coil unit manifolds per the Section XI repair/replacement program;

<u>Option 2</u> - If the above repair can not be accomplished, remove the subject Cooling Coil Unit from the Containment Cooler for shipment to America Air Filter to repair/replace the damaged tube(s).

Safety Evaluation: Option 1 - The removal of one circuit from one cooling coil unit does not significantly reduce the heat removal capability of the Containment Cooler below the heat removal rate in Fig. 6.2.1-15 of the USAR. The Containment Coolers are just one of the means of removing heat from containment during post accident conditions. The overall post accident pressure temperature analyses considers two (2) Containment Coolers (worst case), containment sprays, passive heat sinks, etc. to generate peak temperatures, pressures and long-term cooldown. The reduction of heat removal capability created by removing one coil circuit on one coil unit in one containment cooler against the total available heat sinks will not significantly increase the peak containment pressure and temperatures, nor significantly change the long-term cooldown. Removing one coil circuit out of a total of 384 coil circuits per containment cooler would result in a proportionately small increase over the pressure and temperature of the original analysis for two (2) Containment Coolers. Therefore, the ability of the Containment Cooling System to remove the required heat from the containment atmosphere following a Loss of Coolant Accident or Main Steam Line Break as discussed in Safety Design Basis Seven of USAR Section 6.2.2.2 is unaffected by the above described repair.

Option 2 - This option involves a complete repair of the Cooling Coil Unit by the original manufacturer, American Air Filter. This is a standard repair which restores the entire cooling coil unit to its original configuration, therefore the containment cooling system is unaffected by this option. Neither repair method will affect the function or operability of the containment cooling system or any other system. The ability of the containment cooling system to fulfill its safety bases, as described in USAR Section 6.2.2.2, is not affected. These repairs have no impact on any Wolf Creek Generating Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 02363 Revision: 0

Title: Raychem Control Circuit Splices

Description: Raychem splices identified had seal length, usage range and bend radius which deviated from the installation criteria in the manufacturers instructions. Raychem splices are acceptable to "use-as-is".

Safety Evaluation: Raychem splices identified have been evaluated by site Nuclear Plant Engineering (NPE) and are considered qualified for their application in control circuits at Wolf Creek Generating Station.

PLANT MODIFICATION REQUEST 02372 Revision: 3

Title: Door Seal Improvement Performance

Description: This Modification permits substitution of an improved weather stripping material in place of the existing weather stripping on Control Room Boundary doors. The air leakage through the subject doors appears to be the result of wear and tear on existing weather strips and door hardware caused by frequent door usage. This modification also removes one card reader from service due to the recurring problem of excessive wear on the electric strike often making it very difficult to close the door tightly.

Safety Evaluation: These modifications will improve door seal performance and ultimately enhance airtightness through the control room boundary.

PLANT MODIFICATION REQUEST 02379 Revision: 1

Title: Reactor Vessel Stud And Flange Hole Thread Damage
Description: The modification scope has been enlarged to include evaluation of thread damage of studs and flange holes at Reactor Vessel Flange hole locations 21 and 48. The function of the Reactor Vessel Flange, Head, and Studs is to provide a pressure boundary for the Reactor Coolant System. Adequate torque and proper thread engagement are required of the stud and nut to ensure the pressure boundary remains intact. The reported nonconformance is potentially affecting the proper thread engagement between the stud and the flange hole. If too many threads at a particular location are damaged, the allowable shear stress of the stud/flange interface may be exceeded. This could lead to failure of the connection and, as a minimum, development of a leakage path into Containment. No pressure boundary leakage is allowed per Technical Specification 3/4.4.6.2.

Safety Evaluation: Engineering has reverified the shear stresses induced on both the flange hole and the stud at each reported location, the shear stress being the most limiting value in keeping the pressure boundary intact. For one complete thread assumed removed, the shear stresses were calculated and were found to increase slightly. However, both shear stresses are still well below their respective allowable shear stress.

PLANT MODIFICATION REQUEST 02384 Revision: 1

Title: High Pressure Turbine First Stage Drain Piping Replacement
Description: To reduce piping erosion and corrosion this modification replaces a short section of High Pressure Turbine first stage drain piping located downstream of flow restricting orifices FO-160 and FO-161. The new piping material will be low alloy
steel.

Safety Evaluation: The piping is non-safety related and is located in the Turbine Building. The normal operation or failure of this piping has no adverse effect on the function of any plant safety related system, structures or components. This modification has no impact on any Wolf Creek Generating Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 02402 Revision: 0

Title: Penetration Closure Drawing Changes

Description: This Modification provides clarifying "as-built" information for various penetrations. None of the affected penetrations are being modified in any way. In researching this modification, Engineering realized that additional clarification of the design intent of the Resin Loading chute is required in the USAR.

Safety Evaluation: The proposed USAR change is provided to give clarification of the design intent of the Resin Loading chute. The descriptions contained in the USAR Appendix 9.5B for Fire Areas A-8 and A-26 are being revised to address the Resin Loading Chute. Section 9.5.1.2.2.3 is also updated to add a paragraph that discusses the Resin Loading Chute. The Resin Loading Chute, which is located in the 2026' El. floor slab, between rooms 1405 and 1319, is currently mentioned in Section A.26.7 of Appendix 9.5B, Fire Hazards Analyses. However, the USAR currently does not contain a description of this particular penetration or chute, which clarifies the justification of why the penetration/chute is not fitted with a 3-hour rated fire seal. The crux of the justification is that the Resin Loading Chute is fitted with a heavy steel cover plate that is not susceptible to warping, thus the likelihood of propagation of fire/smoke is minimized and the amount of fixed combustibles in either of the rooms above or below is minimal.

PLANT MODIFICATION REQUEST 02413 Revision: 0

Title: Control Room Emergency Ventilation System Flow Rates

Description: Site Nuclear Plant Engineering will evaluate whether the flows measured below would cause the toxicity limits imposed by Reg. Guides 1.78 and 1.95 to be exceeded.

Maximum Flows: 1760 cfm outside air flow into the control building. 10800 cfm air flow into the Control Room through the pressurization system filter/adsorber units.

The maximum flow values includes a 10% tolerance for measurement uncertainty.

Safety Evaluation: An analysis of the worst case scenario for the new flow rates was made under Revision 1 to calculation CL FH-002. This calculation showed that the maximum chlorine concentration obtained in the Control Room is below the 15 ppm maximum given in Regulatory Guide 1.95.

PLANT MODIFICATION REQUEST 02427 Revision: 0

Title: Diesel Generator Breaker Control Circuits Breaker Reclosure

Description: The scope of this modification is to add an additional handswitch contact from NE HIS25 (NE HIS26) to the diesel breaker NB0111 (NB0211) control circuit in series with the existing permissives and interlocks for the "close" portion of the circuit. The purpose of the new handswitch contact is to allow deenergization of the breaker anti-pumping relay in order to enable the breaker to be reclosed from the Control Room after the breaker is manually tripped from the Control Room via NE HIS25 (NE HIS26) actuation. Prior to this modification, breaker reclosure could only be accomplished locally following a Control Room manual trip via NE HIS25 (NE HIS26). The circuit is not changed with respect to the normal (automatic) breaker closure function associated with a design basis event. Consistent with the original design, this modification does not allow breaker reclosure from the Control Room following a protective trip. All previous circuit permissives and interlocks to permit breaker closure are preserved and are unaffected by this modification, including the breaker "trip-free" feature.

Safety Evaluation: This modification provides Diesel Generator Breaker control circuit enhancements that will facilitate re:o.ery from misoperation of NE HIS25 (NE HIS26). The subject modification will not affect the ability of the Diesel Generators to fulfill their safety design basis or design criteria, including compliance with the applicable regulatory requirements as discussed in USAR Sections 8.1.4.2 and 8.1.4.3.

PLANT MODIFICATION REQUEST 02440 Revision: 0

Title: Structural Weld Inspection

Description: This modification allows the option of using NCIG-01 visual weld acceptance criteria for structural weld inspection.

Safety Evaluation: Criteria NCIG-01 addresses some of the difficulties associated with the inspection of structural welds and will help to minimize unnecessary repairs without lowering quality standards. The structural effect of each discontinuity for which acceptance criteria are provided has been subjected to critical engineering evaluation and has been reviewed by the NRC.

PLANT MODIFICATION REQUEST 02450 Revision: 0

Title: Residual Heat Removal Pump Motor Connection Box Cover Bolts

Description: Two of the four bolts supporting the Residual Heat Removal (RHR) Pump motor connection box cover were identified as being stripped. The stripped bolt holes were drilled and tapped to the next larger size. The safety related design function of the main conduit box on the RHR Pump Motors is to provide a location for the termination of the motor pigtails to the Class 1E Power Supply. The main conduit box must also be designed to prevent water intrusion into the RHR Pump Motor windings so that a high degree of system reliability is maintained.

Safety Evaluation: The structural integrity of the main conduit box is not abated by enlargement of the subject bolts and bolt holes. Therefore, the ability of the main conduit box to perform its design function is not degraded and the high degree of system reliability is maintained. Enlargement of the subject bolts and bolt holes provides a high degree of confidence that the RHR Pump motor connection box cover will remain securely fastened to the motor connection box. Engineering has reviewed the applicable documents and determined that the enlargement of the subject bolts and bolt holes will not prevent the pump motor or any other safety related equipment from performing its safety related design function.

PLANT MODIFICATION REQUEST 02453 Revision: 0

Title: Atmospheric Relief Valves Direct Mounted Pressure Gauges

Description: This modification allows the use of pressure gauges mounted directly on the Bailey positioners of atmospheric relief valves AB PV-1,2,3, and 4 in lieu of gauges tied into the pneumatic lines leading from the positioner. The gauges are installed on the positioners as an option from Bailey and perform the same function as the gauges usually tied into the pneumatic lines.

Safety Evaluation: The change has no significant effect on the total weight of the valve, does not alter the operating characteristics of the valve, and therefore has no effect on the existing qualification of the valve.

PLANT MODIFICATION REQUEST 02455 Revision: 0

Title: Floor And Equipment Drain System

Description: To eliminate a potential tripping hazard from stub-up-345 of equipment drain line LF-345-HCD-4" in Auxiliary Building aisle AK-2047, a modification is being made to cut off the stub-up flush with the floor and weld a plate flush inside the pipe.

Safety Evaluation: The subject design modification does not impact any hazards analysis (i.e., flooding, II/I, fire hazards, etc.). The subject equipment drain is not required for any system function, and therefore it is acceptable to plug.

PLANT MODIFICATION REQUEST 02459 Revision: 0

Title: Engineer i Safety Features Actuation System (ESFAS) Logic Circuit Board Assembly Drawings

Description: BETA Products, Inc. has changed its part numbering system and several replacement part numbers for the Dual Category "B" Logic Card (drawing 308300). BETA has also recommended that a diode be removed from the power input circuit of all Category "A" and "B" logic cards (drawings 308299 and 308300, respectively) and all System Level cards (drawing 308301). This modification revises BETA drawing 308300 to show the new part numbers and removes the diode from all existing 308299, 308300 and 308301 circuit boards.

Safety Evaluation: All parts ordered using the new part numbers will be identical to the old parts (except for the number) with the exception of two parts on the 308300 circuit board. Two replacement parts on the 308300 circuit board, 2N2222 transistors and 1N4007 diodes, are now being supplied as replacements for 2N5449 transistors and 1N4003 diodes. Utilization of these replacement parts does not adversely affect the ability of the 308300 circuit board to perform its safety related function. A zener diode was installed on 308299, 208300, and 308301 circuit boards to protect components used on earlier model boards from voltage spikes on the DC supply. The components used on the present boards are not as susceptible to voltage spikes and the failure mode of these diodes was to short, thereby affecting the power supply to all other boards using the same power supply. These diodes have not been removed. This increases the reliability of the circuit boards and, therefore, does not adversely affect the ability of the circuit boards or ESFAS system to perform their safety related function.

PLANT MODIFICATION REQUEST 02462 Revision: 0

Title: Valcor Solenoid Valve Cover Change

Description: This modification changes the standard nickel plated carbon steel cover with a non-magnetic stainless steel solenoid cover for the Valcor solenoid valves SJHV003 (Reactor Coolant System Loop 1 Hot Leg Sample Solenoid Isolation) and SJHV020 (Pressurizer Liquid Space Sample Solenoid Isolation). The operability of the valve is unchanged, however, the reliability of the valves indication will be improved.

Safety Evaluation: This modification changes the solenoid covers to allow for a more reliable setting on the magnetically actuated position indicator switches on both valves, which were located within three inches of a concrete wall and steel imbeds. The valve indication reliability is improved. Thus, the probability of equipment malfunction is actually decreased. Changing the valve solenoid covers will have no effect on the consequence of malfunctions of equipment.

PLANT MODIFICATION REQUEST 02467 Revision: 0

Title: Component Cooling Water Return Header Water Hammer Inspection

Description: The Component Cooling Water (CCW) return header to CCW Pumps was examined for any after-effects of a reported water hammer. The elbow examined is in line EG-178-HBC-20" which is a Class 3 piping system. A linear surface defect was found during the requested MT examination. By virtue of this type of exam, the depth of the defect cannot be determined. Therefore, a ASME Section XI repair shall be performed.

Safety Evaluation: The elbow is designed to have the same nominal wall thickness as the piping. The nominal wall thickness (including manufacturing tolerances) provides the system's pressure boundary. As long as this pressure boundary is not violated, the system can perform its safety design bases outlined in USAR Section 9.2.2. The Section XI repair, when completed, shall ensure that the Component Cooling Water System can perform its Safety Design Bases.

PLANT MODIFICATION REQUEST 02562 Revision: 0

Title: Elimination Of Loop Seals In Condensate Demineralizer System
Description: The modification deals with piping in the Condensate Demineralizer System (CDS). Specifically, the modification reroutes the demineralizers vent/drain headers to eliminate existing loop seals. The loop seals serve no function but impede the venting/draining process. The piping modification is designed per ANSI B31.1.

Safety Evaluation: The CDS is designed to maintain the required purity of feedwater for the steam generators by filtration to remove corrosion products and by ion exchange to remove condenser leakage impurities. The CDS components are located in the Turbine Building at elevation 2000'. The CDS serves no safety function and has no safety design bases. The subject modification will have no affect on the ability of the CDS to fulfill its design function as described in USAR Section 10.4.6.

PLANT MODIFICATION REQUEST 02568 Revision: 0

Title: Proposed Change To HVAC Technical Specifications

Description: Concurrent testing of the Emergency Exhaust System (GG) and the Control Room Pressurization System (GK) during Refuel II Outage demonstrated that the Emergency Exhaust Fans (CGG02A/B) were producing excessive negative pressure in the Auxiliary Building. This negative pressure adversely affected the Control Room Pressurization testing. Subsequent testing determined that an Emergency Exhaust flow of 6500 cfm ± 650 cfm was sufficient to produce the negative pressure in the Auxiliary Building necessary to contain any radiological releases during an accident scenario (e.g. Loss of Coolant Accident (LOCA)) and produce the negative pressure in the Fuel Building necessary to contain any radiological releases during an accident scenario (e.g. Fuel Handling Accident). A Technical Specification change request has been submitted to the Nuclear Regulatory Commission to revise sections 3/4.7.6, 3/4.7.7 and 3/4.9.13. The modifications are contingent upon approval of the requested changes to the Technical Specifications.

Safety Evaluation: The Emergency Exhaust System maintains a negative pressure of no less than 1/4 in. water gauge (w.g.) in the Fuel Building to prevent unprocessed exfiltration following a fuel handling accident which releases radioactivity both upstream and downstream of the filter adsorber unit prior to release to the site. The filter adsorber unit limits the radiological consequences of a fuel handling accident to less than 10 CFR 100 limits. The safety evaluation will still be valid after implementation of the modification. Testing during the Refuel II Outage demonstrated that a lower flow rate through the system with a Fuel Building Ventilation Isolation Signal (FBVIS) line-up produced a minimum 1/4 in. w.g. negative pressure in the Fuel Building. Therefore, the criteria for preventing unprocessed exfiltration is still met. Also, the reduced flow through the filter adsorber provides for longer residence time in the filter and thus improves removal efficiency.

The Emergency Exhaust System maintains a negative pressure in the Auxiliary Building of not less than 1/4 in. w.g., following a LOCA. The system collects and processes potential Emergency Core Cooling System (ECCS) leakage and the effluent purged from the Containment via the Hydrogen Purge System. The system is monitored for radioactivity upstream of the filter adsorber unit prior to release through the unit vent. This safety evaluation will still be valid after implementation of the modification. Testing during the Refuel II Outage demonstrated that a lower flow rate through the system with a Safety Injection Signal (SIS) line-up produced a minimum 1/4 in. w.g. negative pressure in the Auxiliary Building. Therefore, the criteria for preventing unprocessed infiltration is still met. Also, the reduced flow through the filter adsorber provides for longer residence time in the filter and thus for improved removal efficiency. The radiological consequences would, therefore, not be increased as a result of the modification.

PLANT MODIFICATION REQUEST 02589 Revision: 0

Title: Encapsulation Of Pressurizer Spray Valve

Description: This modification encapsulates the valve body to packing box flange on one pressurizer spray valve BB-PCV-455B. Valve BB-PCV-455B has a body to packing box leak. Attempts to stop the leak with Furmanite have failed. To control this leakage a structural attachment is being added to the valve.

Safety Evaluation: This structural attachment was designed to the requirements of the ASME B&PV Code Section III-Subsection NB (Class 1) and was analyzed for pressure to control and contain the leakage from the mechanical joint (i.e., this attachment was designed with capability of withstanding Reactor Coolant System design pressure). The original pressure boundary is still the ASME Code pressure boundary. This structural attachment will not hinder the operating characteristics of the valve nor impact the seismic qualification of the valve. The stresses under all conditions (including upset and faulted) of the piping and supports do not exceed the ASME B&PV Code allowables. An analysis was previously performed on the catastrophic failure of the pressurizer spray valve packing box per PMR 02535 and the results revealed that the core is not expected to be uncovered in the unlikely event that there is a catastrophic failure. Also, it was noted that this valve is not required to function in the ultimate mitigation of the consequences from accidents.

PLANT MODIFICATION REQUEST 02616 Revision: 0

Title: Replacement Of Drain Piping And Valve For Radwaste Building Supply Heating Coil

Description: This modification replaces the drain piping and valve for the Radwaste Building Supply Air Unit heating coil to accommodate a new heating coil supplied by the vendor. This modification replaces piping and a gate valve (GH-V003) for Air Unit SGH01 heating coil drain with 3/8" tubing and a ball valve due to relocation of connection per replacement of new coil section.

Safety Evaluation: This change has no impact on the operability and function of the air unit or the Radwaste Building HVAC System. The operation of this system is not required for the safe shutdown of the plant or for mitigating the consequences of a design basis accident.

PLANT MODIFICATION REQUEST 02620 Revision: 0

Title: Clarification Of Design Requirements Concerning Fire Barriers

Description: This modification clarifies the USAR design requirements to show that Missile Door 15041, external Auxiliary Building Door, is a fire barrier and to agree with the fire protection commitments of the Fire Hazards Analysis and the Fire Delineation Figures in USAR Section 9.5B. There are no physical modifications required. USAR Section 9.5.1.2.2.3e describes the design, construction, test method, and acceptance criteria for Missile Doors which are fire barriers. This section must be revised to include Missile Door 15041 as a fire barrier.

Safety Evaluation: This modification has no impact on the WCGS Technical Specification or on the Safety Design Bases for any system. This clarification satisfies the fire protection commitments of the Fire Hazards Analysis.

PLANT MODIFICATION REQUEST 02636 Revision: 0

Title: Replacement Of Control Rod Drive Mechanism Cooling Fan Unit With Equivalent Unit

Description: This modification replaced the Control Rod Drive Mechanisms (CRDM) cooling fan unit (CGN01B) with an essentially equivalent unit due to unavailability of original type. Following a trip of the existing CRDM Cooling Fan Unit, procurement of a replacement was initiated. A quote from the vendor indicated that they could not support rework/replacement of the unit during the upcoming Refueling III Outage. An alternate cooling unit was located in storage at another facility. The unit was retrieved from storage and modified by the vendor to Wolf Creek Generating Station requirements.

Safety Evaluation: The CRDM fan has no safety design basis and therefore, the only safety related impact of this modification would be from an adverse interaction or effect on safety related equipment inside the containment or in contact with the containment sump water during recirculation phase of a Loss of Coolant Accident or Main Steam Line Break. Potential interactions which were considered were hydrogen generation sources, seismic interaction and debris generation which might affect the containment sumps. Each of these issues was evaluated and determined not to constitute a significant safety concern.

PLANT MODIFICATION REQUEST 02642 Revision: 0

Title: Clarification Of Design Requirements Concerning Fire Barriers

Description: The purpose of this modification was to delineate Fire Barrier Requirements for the Control Building and Auxiliary Building roofs based on the Fire Hazards Analysis (Section 9.5B of the USAR).

Safety Evaluation: Based on the Fire Hazards Analysis and the roof construction materials these roofs are not required to be fire rated barriers.

PLANT MODIFICATION REQUEST 02647 Revision: 0

Title: Piping And Instrument Drawings Revised To Show Proper Interface

Description: This modification corrects the continuation zones shown on two Piping and Instrument Drawings joining the Instrument Air and Fire Protection Systems. Drawing M-02KA02(Q), Rev. 14 indicates as reference continuation zones downstream of valve KA-V450 and KA-V630, Drawing M-02KC05, F-4 and H-4, respectively. It was determined that the continuation references did not reflect the actual design and as-built condition. Changes were issued to correct the discrepancy and clarify the cross-references.

Safety Evaluation: These drawing revisions are required only to show the correct interface between the two drawings. They do not affect the function of any safety related system or equipment.

PLANT MODIFICATION REQUEST 02652 Revision: 0

Title: Installation Of Coupling In Chemical And Volume Control System

Description: This modification installs a pipe coupling in the Chemical and Volume Control System (CVCS) to facilitate rework of a 2 inch 6000 pound elbow socket weld. The socket weld on the elbow immediately downstream of BG-V002 has developed a leak.
Safety Evaluation: The installation of this pipe coupling will not reduce or hinder the CVCS in performing the required functions. In addition, adequate safety margins are available in the existing piping to accommodate the small weight increase (approximately 5 lbs.) due to the installation of the coupling. Specifically, the stresses will not exceed the ASME Code allowable stress values. There is no relocation of the high energy break points because of this addition of the pipe coupling. All of the adjacent pipe supports are capable of handling the additional weight.

PLANT MODIFICATION REQUEST 02662 Revision: 0

Title: Exemption Of Containment Atmospheric Control System From Leak Testing Requirements

Description: This modification exempts the Containment Atmospheric Control System (CACS) charcoal adsorbers from in-place leak testing. The non-safety related CACS filter units are 100% recirculating and an adsorber leak would only extend containment clean up time. The deletion of requirement for in-place leak testing will decrease potential outage time, worker radiation doses, radwaste, and money spent throughout the plant life.

Safety Evaluation: The deleted requirement to perform adsorber in-place leak testing could result in an increased containment cleanup time, but this is not a reduction in the margin of safety as defined in any Technical Specification. CACS adsorber leakage will not cause malfunctions of other safety equipment and therefore, there will be no consequences of a malfunction to increase.

PLANT MODIFICATION REQUEST 02664 Revision: 1

Title: Modification Of Polar Crane Snubber Assembly

Description: The newly ordered long pin (axle) for the wheel on the Polar Crane Snubber Assembly cannot be installed without removal of the snubber assembly due to insufficient room above the pin hole. The reactor containment polar crane is equipped with seismic restraints to limit bridge motion under earthquake loading conditions. A seismic restraint is installed at each of the four corners of the bridge. Each seismic restraint consists of two wheels which roll on the face of the bridge rail girder and a spring loaded hydraulic snubber which keeps the wheels in contact with the rail girder under normal operating conditions yet locks up to provide a rigid support under seismic loading. The four snubbers/restraints were brought into their design operating range by machining down the existing clevises and shock absorbers of each of the four snubbers.

Safety Evaluation: The above modification does not change the safety related function of the polar crane, and does not affect the function, operation, structural integrity or reliability of the polar crane or any other system. The design of the subject modification will have no affect on the ability of the polar crane to fulfill its associated safety design basis as described in the USAR.

PLANT MODIFICATION REQUEST 02677 Revision: 0

Title: Modification Of Gears For The Auxiliary Feedwater Pump Governor Hydraulics

Description: The subject change entails revising the mating driver and driven gears for the turbine (KJC02) governor on the Turbine Driven Auxiliary Feedwater Pump. These gears drive the hydraulics for the turbine governor. The gear drive was changed from a spiral to worm gear type by the Turbine manufacturer.

Safety Evaluation: Both the new and the old gear types are sized to provide sufficient hydraulic capacity for this governor. The speed sensing for the turbine is electromagnetic and is not affected by this change. As a result, there is no change to governor overall operation or turbine operation due to this change. There is no change in setpoint or calibration for KFC02. No safety feature of the plant is impacted.

PLANT MODIFICATION REQUEST 02679 Revision: 0

Title: Increase In Sodium Hydroxide Content Of Containment Spray

Description: The 18 month fuel cycle will require higher boron concentrations in the Refueling Water Storage Tank (RWST) and Safety Injection Accumulators. In order for the containment spray system to achieve the proper pH with the higher boron concentrations, this modification revises the setpoints for EN-LBL-0017 and EN-LSL-0019 to increase the amount of Sodium Hydroxide (NaOH) that would be added to the containment sump in the event of a Loss of Coolant Accident. To ensure that the required amount of NaOH solution is added to the sump, spray additive eductor isolation valves, EN HV-15 and EN HV-16, are provided with an interlock from the spray additive tank level transmitters to preclude their closure prior to the addition of the required amount of NaOH solution. The setpoint of the subject low level switches has been lowered to ensure that the close permissive for valves EN HV-15 and EN HV-16 is received only after approximately 2960 gallons of NaOH solution is added into the containment sump, post-Loss of Coolant Accident (LOCA).

Safety Evaluation: This setpoint change does not impact the automatic isolation provision of the containment spray additive subsystem upon a receipt of a low-low level signal from the spray additive tank level transmitters. In addition, the lower setpoint will not impact the operability of the spray additive eductor. Since a greater quantity of NaOH solution is needed to achieve the long-term post-LOCA containment sump pH of at least 8.5, it will take approximately 20 minutes longer to achieve this pH based on a worst-case single failure of one of the motor operated valves in the spray additive line. This additional time has been determined to be insignificant. The capability of the Containment Spray System to perform its safety related functions as described in USAR Sections 6.2.2.1 and 6.5.2 is unaffected.

PLANT MODIFICATION REQUEST 02688 Revision: 0

Title: Replacement Of Gasket With Equivalent Gasket In Residual Heat Removal System

Description: The present design requirement for the shell and bonnet gaskets on the Residual Heat Removal (RHR) heat exchangers are for spiral wound gaskets with asbestos filler material. Asbestos gaskets are prone to leak under normal thermal cycling. These gaskets will be changed to a graphite filler material in the required spiral wound shell and bonnet gaskets. These replacement gaskets are considered to be an equivalent gasket with no change in the required bolt torque value.

Safety Evaluation: The use of the stainless steel spiral wound gasket with the graphite filler material in lieu of the stainless steel spiral wound gasket with asbestos filler material will provide an equivalent or better seal of the mechanical joint. The material change out of the gaskets does not change the form, fit or function of the RHR heat exchangers.

PLANT MODIFICATION REQUEST 02689 Revision: 0

Title: Rerouting Of Demineralized Water Makeup System Piping

Description: This modification changes the design of the Demineralized Water System Makeup (WM). The modification removes the acid and caustic day tank transfer pumps, associated interconnecting piping and valves. It also provides for direct acid and caustic metering from the acid and caustic storage tanks to the acid and caustic regeneration pumps, caustic feedpump, pH adjustment pumps, and the acid supply pumps. Also included is the addition of a 1'-0" standpipe inside the acid storage tank. The standpipe is located at the tank's suction nozzle to prevent debris from entering the new acid suction piping.

Safety Evaluation: The WM System is non-safety related. The system serves no safety design bases. This modification does not impact any Wolf Creek Generating Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 02691 Revision: 0

Title: Increase In Breaker Size For Radiation Monitor Vacuum Pump Motor Feeders

Description: This modification allows for an increase in the breaker size from 15 amps to 20 amps for the process radiation monitor vacuum pump motor feeders to eliminate tripping during motor starting. The motors will retain their 15 amp slow-blow fuse supplied with the skid.

Safety Evaluation: The safety related design function of the 15 amp breakers, being replaced by 20 amp breakers, is to protect the safety related low voltage power cables used for radiation monitor vacuum pump motor service. The radiation monitor vacuum pump motors are protected by a 15 amp slow-blow fuse supplied with each radiation monitor skid. The worst case ampacity for the low voltage power cables providing power to the vacuum pump motors associated with this PMR is 55 amps. Therefore, the 20 amp replacement breaker will adequately protect the affected safety related cables.

PLANT MODIFICATION REQUEST 02700 Revision: 0

Title: Replacement Of Frame Breakers With Equivalent

Description: This modification allows use of type HE Frame Breakers as replacements for the type EF Frame Breakers due to discontinuance from manufacturer. If procured as Class 1E, Q components, the HE3 breakers are suitable replacements for the existing EF3 breakers. These breakers are safety related and are seismically and environmentally qualified. There will not be any effect on the design basis operation of the breakers since they have identical form, fit, and function. Since the HE3 breakers have a higher interrupting rating, the substitution of HE breakers for EF breakers is considered an upgrade.

Safety Evaluation: The use of HE frame breakers as replacements for the EF frame breakers has been evaluated and it has been determined that the HE frame breakers can be used as direct replacement breakers for the EF frame breakers. There is no change in the form, fit, or function except that HE3 breakers have a higher interrupting rating, which is acceptable. Environmental and seismic qualification is not adversely affected.

PLANT MODIFICATION REQUEST 02702 Revision: 0

Title: Erection Of Steel Hoist Beams Over Reactor Coolant Pumps

Description: This modification allows erection of additional steel hoist beams over the Reactor Coolant Pumps (RCP) to allow removing RCP covers without having to remove the Hydrogen Mixing Fans. Present arrangement requires removal of Hydrogen Mixing Fans in order to remove the RCP Covers. The RCP Covers need to be removed on a regular basis to support Inservice Inspection and routine maintenance.

Safety Evaluation: The subject beams have been designed to withstand seismic occurrences and have been checked against different failure modes. The stresses in the existing beams are within design allowable limits.

PLANT MODIFICATION REQUEST 02703 Revision: 0

Title: Plant Heating Expansion Tank Vent And Drain Valves

Description: This modification replaces a drain valve on the Plant Heating System expansion tank with two valves and a tee to allow for venting and draining.

Safety Evaluation: The reconfiguration and replacement of the drain valve will not affect the operability or function of the Plant Heating System as described in USAR Section 9.4.9. As described in USAR Section 9.4.9 the Plant Heating System serves no safety function, has no safety design bases and its failure does not compromise any safety related system or prevent safe shutdown. The subject modification has no impact on any Wolf Creek Generating Station Technical Specification or associated bases.

PLANT MODIFICATION REQUEST 02708 Revision: 0

Title: USAR Revision For Drawing Clarity

Description: This modification involves a USAR change to correct illegible numbers on Piping and Instrument Drawing M12EC01(Q).

Safety Evaluation: This does not impact safety analysis.

PLANT MODIFICATION REQUEST 02718 Revision: 0

Title: Modification Of Neutron Shield Water Can Hinge Pins

Description: This modification allows use of neutron shield water can grating hinge pins cut into short lengths versus one long pin to allow easier removal and reinstallation. Due to misalignment of the neutron shield water can grating, some hinge pins are extremely tight for removal and reinstallation increases the exposure doses for workmen. Replacing the long hinge pin with pins cut into short lengths or use of hex head bolts and nuts was approved as alternate configurations.

Safety Evaluation: The subject change involves replacing one long rod with seven short length bolts or rods of the same diameter and material for supporting the grating bars. The grating bars support the water cans used for neutron shielding. The subject change does not change the original design conditions. The stresses in the rod remain unaltered.

PLANT MODIFICATION REQUEST 02721 Revision: 0

Title: Correction Of Typographical Error In USAR Concerning Filter Efficiency

Description: This modification addresses the USAR, Section 9.3.1.2.3, change regarding the Compressed Air System after filters, FKA02A and FKA02B, particle retention size. The subject USAR change revises the present particle retention of 0.03 microns and larger to the minimum required 3.0 microns and larger. The present particle retention of 0.03 microns and larger, as discussed in the USAR, has been determined to be a typographical error and therefore considered an incorrect figure.

Safety Evaluation: This change to 3.0 microns and larger for acceptable particle retention will not affect the design bases. USAR descriptions are established from the applicable design specifications which are in accordance with industry codes and standards. The design bases are generated for these codes and standards. Therefore, this change will eliminate a typographical error and reflect the originally invoked standard for the design bases of the Compressed Air System.

PLANT MODIFICATION REQUEST 02723 Revision: 0

Title: Material Modification In Waste Gas Compressor

Description: This modification addresses increased corrosion resistance on Gaseous Radwaste System (HA) waste gas compressor internals by replacing existing iron parts with bronze. This modification changes the material for the wetted parts of the Waste Gas Compressors (specifically the lobe and cone) to bronze. The bronze rotor should experience better corrosion resistance with the process inlet stream than the ductile iron.

Safety Evaluation: This modification is only to change the material of the cone and lobe to one that has demonstrated better corrosion resistance than the originally supplied ductile iron. No other design parameters are being changed (i.e. individual part dimensions including tolerances, running clearances between the rotor, cone and lobe, etc.). The use of a corrosion resistant material should actually help the waste gas compressors meet their design rated flow and discharge pressure. Thus, the use of a different material will in no way affect postulated exposures to radiation resulting from waste gas system leakage as analyzed in the USAR (see Sections 11.1, 11.3 and Appendix 11.1A).

PLANT MODIFICATION REQUEST 02725 Revision: 0

Title: USAR Change To Reflect As-Built Conditions

Description: This change to the USAR is necessary because Drawings M-12FC02(Q) and M-13FC01(Q) conflict in the area of the drain line and clean out on Valve FCV-095 in the Auxiliary Turbines (FC) System. Drawing M-12FC02(Q) shows this line to be open, while Drawing M-13FC01(Q) shows a threaded cap. The as-built condition is that the line is capped.

Safety Evaluation: Engineering has determined that this condition is not detrimental to the system and is acceptable to use-as-is.

PLANT MODIFICATION REQUEST 02739 Revision: 0

Title: Installation Of Pipe Unions To Facilitate Removal Of Auxiliary Turbine Governor Valve

Description: This modification installs unions in the Auxiliary Feedpump Turbine speed governing valve FC FV-313 leak off line to provide convenience in the removal of the speed governing valve.

Safety Evaluation: The installation of the unions in this line does not exceed the code allowable stress values for the normal, upset, or faulted conditions, nor does it result in a reduction in the margin of safety as defined in the bases for any Technical Specification.

PLANT MODIFICATION REQUEST 02740 Revision: 0

Title: Modification Of Cabinets To Prevent Interaction During Seismic Event

Description: This modification provides for the addition of structural ties at the top of Class 1E electrical/instrument cabinets closely spaced with Class 1E or non-Class 1E cabinets to prevent interaction during a seismic event. The subject cabinets are located in the Main Control Room and the Rod Drive Motor Generator Set Room.

Safety Evaluation: The new configuration of the cabinets (structurally connected) will not invalidate the original equipment seismic qualification. There is sufficient margin of safety between the required floor response spectra and the qualifying test spectra to account for any new torsional modes or frequency shifts in overall modes to account for resonance with local panel modes. The addition of structural ties at the top of Class 1E Cabinets closely spaced with Class 1E or non-Class 1E Cabinets to make them act together during a seismic event will preclude any potential interaction, transferring of kinetic energy or resonance. This method is a reconciliation of the as-built situation and does not invalidate the original equipment test results.

PLANT MODIFICATION REQUEST 02763 Revision: 0

Title: Substitution Of Lead Wire For Various Valcor Valves

Description: This modification allows the use of a substitution lead wire when the original lead wire (#18, 150°C) is not available for various Valcor valves.

Safety Evaluation: The evaluation of this substitution determined that the lead wire has been certified per the required J-603A qualification report and specification. In addition, this substitution lead wire is a #14 gauge wire with a design temperature rating of 200°C, while the original lead wire is a #18 gauge wire rated for 150°C. Therefore, the design function of the lead wire shall not be jeopardized by the subject substitution.

PLANT MODIFICATION REQUEST 02775 Revision: 0

Title: Replacement Of Stem Nut In Auxiliary Feedwater System Valves With Equivalent Nut

Description: The motor operated Auxiliary Feedwater Regulating Valves were supplied with a stem nut originally made from nylon. The nylon stem nut has not been effective as desired and is no longer provided as a standard for modulating service. The nylon appears to deform under load and wear. A standard bronze stem nut is now being used as a replacement.

Safety Evaluation: The bronze stem nut is equal to or better than the nylon type, therefore the reliability of the Auxiliary Feedwater System is not degraded. The added weight will have no adverse impact on seismic considerations.

PLANT MODIFICATION REQUEST 02789 Revision: 0

Title: Modification Of Main Feedwater System Snubber Support Member

Description: This modification restores and enhances the structure of the existing damaged snubber support member. The subject support AE05-R003 is installed in the Class 2 portion of the Main Feedwater line (AE080-EBB-14") located within the Reactor Building and supplies feedwater to Steam Generator "D".

Safety Evaluation: The configuration and operating characteristics of the associated snubber will not be affected in any manner. This design change will not result in a reduction in the margin of safety as defined in basis for any Technical Specification.

Section III

TECHNICAL SPECIFICATION AMENDMENT REQUEST 6.5.1.2

Title: Addition Of Manager Nuclear Plant Engineering To Plant Safety Review Committee

Description: This amendment request revises Wolf Creek Generating Station (WCGS), Unit No. 1, Technical Specification Section 6.5.1.2, Plant Safety Review Committee Composition, to add the Manager Nuclear Plant Engineering Wolf Creek as a committee member. The proposed change provides additional design engineering expertise on the Plant Safety Review Committee (PSRC).

Safety Evaluation: The proposed revision to the WCGS Technical Specifications does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report; or create the possibility for an accident or malfunction of a different type than any previously evaluated in the Safety Analysis Report; or reduce the margin of safety as defined in the basis for any technical specification. Therefore, the proposed revision does not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

TECHNICAL SPECIFICATION AMENDMENT REQUEST 3.4.9.1

Title: Revision Of Heatup And Cooldown Pressure/Temperature Limits

Description: The heatup, cooldown and Cold Overpressure Mitigation System (COMS) Power-Operated Relief Valve (PORV) setpoint pressure/temperature limits are being revised as required by 10 CFR 50 Appendix H and Technical Specification 4.4.9.1.2. Capsule "U" of the Wolf Creek Generating Station (WCGS) reactor vessel radiation surveillance program was withdrawn from the reactor during the Refueling I Outage. The results of the analysis of Capsule "U" were submitted to the NRC on November 4,1987.

Safety Evaluation: Results of surveillance testing show that the measured shift in the 30 and 50 ft. lb. transition temperature for the Wolf Creek Nuclear Operating Corporation (WCNOC) pressure vessel radiation surveillance samples from the lower shell plate and weld material, after exposure to 3.39×10^{18} N/cm^2 at $550^{\circ}F$ (1.08 Effective Full Power Years) was less than or equal to that predicted on the basis of proposed Reg. Guide 1.99 Rev. 2. The results also indicated that the lower shell material with the highest unirradiated initial RT_{NDT} of $40^{\circ}F$ should be the new material property control basis for the revised heatup, cooldown and COMS PORV setpoint pressure limit curves.

This change incorporates the revised heatup (Figure 3.4-2) and cooldown (Figure 3.4-3) pressure/temperature limit curves and the revised COMS setpoint curve (Figure 3.4-4) effective up to 7 Effective Full Power Years (EFPY) for the WCGS reactor pressure vessel into WCGS Technical Specifications 3/4.4.9.1 and 3/4.4.9.3. A technical specification change limiting the RCS heatup rate to less than or equal to 60^{o}F/hr for indicated Reactor Cooling System (RCS) T_{avg} (average temperature) less than 200 ^{o}F is also included.

Incorporating the revised heatup (Figure 3.4-2) and cooldown (Figure 3.4-3) pressure/temperature limit curves into the WCGS Technical Specifications, along with a change limiting the RCS heatup rate to less than or equal to 60^{o}F/hr for indicated RCS T_{avg} less than 200^{o}F, maintains the margin of safety required for prevention of non-ductile failure of the WCGS reactor pressure vessel as evaluated for normal operations and transients specified in WCGS USAR Chapters 3.9(N), and accidents/transients analyzed in USAR Chapter 15.

The revised curves along with the 60^{o}F/hr heatup rate limit for indicated RCS T_{avg} less than 200^{o}F, do not impact the probability or the consequences of an accident or malfunction of equipment important to safety as analyzed in the WCGS USAR.

The revised heatup and cooldown limit curves ensure that the margin for protection against non-ductile failure is maintained. This is accomplished by reducing the maximum allowable RCS pressures for operations at low RCS temperatures to compensate for the reduced ductility of the pressure vessel. This reduction in maximum allowable pressure for the vessel for indicated RCS T_{avg} less than 350^{o}F reduces the probability or possibility that the composite minimum Appendix G limits for the reactor pressure vessel will be challenged.

There are no significant changes in plant operating procedures other than to administratively reduce the maximum allowable heatup rate to less than or equal to 60^{o}F/hr for indicated RCS T_{avg} less than 200^{o}F.

The revised heatup and cooldown pressure limit curves when used in conjunction with a heatup rate limit of 60^{o}F/hr for indicated RCS T_{avg} less than 200^{o}F, do not create the possibility of accidents not previously analyzed in the WCGS USAR.

The revised curves are based on a more limiting value for RT_{NDT} including the radiation induced shift. The revised limits are more restrictive in that they decrease the maximum allowed pressure at the same measured RCS temperature for any heatup or cooldown rate. This reduction of maximum allowable pressure also lowers the total thermal-mechanical stresses experienced by the reactor vessel during a heatup or cooldown transient. Incorporating the revised COMS PORV setpoint pressure/temperature limit (Figure 3.4-4) curve into the WCGS Technical Specifications along with the revised heatup and cooldown limit curves and a change limiting the RCS heatup rate to less than or equal to 60^{o}F/hr for indicated RCS T_{avg} less than 200^{o}F, maintains the margin of safety required for prevention of non-ductile failure of the WCGS reactor pressure vessel as evaluated for normal operations and transients specified in WCGS USAR Chapters 3.9(N), 5.2.2.10, and accidents and transients analyzed in USAR Chapter 15.

The proposed revision of the COMS PORV setpoint pressure limit curve along with the change to limit the heatup rate to less than or equal to 60^{o}F for indicated T_{avg} less than 200^{o}F, will not impact the probability or the consequences of an accident or malfunction of equipment important to safety as analyzed in the WCGS USAR. The combined changes do not alter any other transients or the availability of safety related equipment used to prevent or mitigate overpressure transients as described in WCGS USAR Chapter 5.2.2.10.

The revised COMS PORV setpoint limit curve ensures that the margin of protection against non-ductile failure is maintained per 10 CFR 50 Appendix G requirements. This is accomplished by reducing the maximum allowable RCS pressures for operations at low RCS temperatures to compensate for the reduced ductility of the pressure vessel. This reduction in maximum allowable pressure (leading to lower pressure stresses for the vessel) for indicated RCS T_{avg} less than 350^{o}F reduces the probability that the composite minimum Appendix G limits for the reactor pressure vessel will be challenged.

A review of the RT_{PTS} values for WCGS was performed. The review indicated that the current value of 140^{o}F remains bounding. It should be noted that the transition to low leakage loading patterns for WCGS will lead to added conservatism in the currently bounding RT_{PTS} value of 140^{o}F. Based on this, no change to RT_{PTS} is required.

Based on the above discussions, the proposed revisions to the WCGS Technical Specifications do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create a possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the bases for any technical specification. Therefore, the proposed revisions do not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

TECHNICAL SPECIFICATION AMENDMENT REQUEST 5.3.1

Title: Replacement Of Fuel Rods With Filler Rods

Description: This license amendment request proposes to revise Technical Specification 5.3.1, Fuel Assemblies, to allow the replacement of a limited number of fuel rods with filler rods or vacancies if such replacement is acceptable based on the results of a cycle-specific reload analysis.

Safety Evaluation: Technical Specification 5.3.1 currently states that each fuel assembly shall contain 264 fuel rods clad with Zircaloy-4. The proposed license amendment will allow for a reduction in the number of fuel rods per assembly and replacement of defective rods with filler rods consisting of either Zircaloy-4 or stainless steel, or with vacancies. The ability to replace defective rods with filler rods or vacancies will permit utilization of the energy in the remaining non-leaking rods of the effected fuel assemblies. In addition, the proposed amendment allows added flexibility to provide for improved fuel performance by permitting the timely removal of individual fuel rods which are found to be leaking during a refueling outage.

In general, substitution of a limited number of fuel rods with filler rods or vacancies has a negligible effect on core physics parameters and consequently on the safety analysis. A safety evaluation for the replacement of fuel rods will be made on a cycle-specific basis as part of the reload safety evaluation process. The core reload analysis is performed to ensure that the safety criteria and design limits, including peaking factors and core average linear heat rate effects, are not exceeded. An explicit model with each discrete rod identified is utilized to predict core performance based on actual core inventory. The core reload methodology does not change when filler rods or vacancies are used. The filler rods or vacancies in a fuel assembly that is used in a core design will be modeled as required for the specific replacement.

Based on the above discussions, the proposed revision to the WCGS Technical Specifications does not increase the probability of occurrence or the consequences of an accident or malfunction or equipment important to safety previously evaluated in the safety analysis report; or create a possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the bases for any technical specification. Therefore, the proposed revision does not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

TECHNICAL SPECIFICATION AMENDMENT REQUEST 5.3.1 AND 5.6.1

Title: Increase In The Weight Percent Of Uranium In Each Fuel Assembly

Description: Technical Specifications 5.3.1 and 5.6.1.1 are being changed to allow storage of fuel assemblies of up to 4.5 weight percent (w/o) U-235, which is an increase from the current limit of 3.5 w/o U-235. Technical Specification 5.6.1.1 is also being revised to reflect the actual spent fuel pool storage rack nominal cell pitch of 9.236 inches. In addition, the acceptable/unacceptable regions of Figure 5.6-1 and Figure 3.9-1 are being changed on the burnup versus enrichment graphs to reflect the higher possible enrichments. Wolf Creek Generating Station (WCGS) Cycle 4 will be an 18-month cycle which will require fuel of a higher U-235 enrichment than is currently allowed for storage in the Spent Fuel and New Fuel Storage Racks. The current maximum enrichment allowed by Technical Specifications is 3.5 w/o U-235.

Criticality analyses of the WCGS fuel storage racks were performed for Wolf Creek Nuclear Operating Corporation (WCNOC) by Pickard, Lowe and Garrick, Inc. (PL&G). The conclusion of these analyses is that Westinghouse standard fuel assemblies with U-235 enrichment as high as 4.5 w/o can be safely stored in the WCGS Spent Fuel and New Fuel Storage Racks.

The spent fuel pool storage rack nominal cell pitch is being revised to reflect actual dimensions as shown on design drawings. The original analysis performed by PL&G utilized a SNUPPS cell pitch value of 9.14 inches. This resulted in a conservative estimate of K infinity at a cell pitch of 9.236 inches, since the effective fuel density decreases as the pitch increases.

Safety Evaluation: Based on the criticality analyses, WCNOC has determined that this proposed amendment does not involve an unreviewed safety question, because:

1) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not increased. An increase to a maximum enrichment of 4.5 w/o does not involve a significant increase in the probability or consequence of an accident or other adverse conditions over previous evaluations. The small increase in fuel enrichment has only a very minimal effect on the fuel handling of the conservative techniques and assumptions used to evaluate the maximum possible neutron multiplication factor, there is more than reasonable assurance that no significant hazards based on criticality safety is involved in storing fuel assemblies of up to and including 4.5 w/o in the spent fuel storage racks under both normal and postulated accident conditions.

2) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created. An increase to a maximum enrichment level of 4.5 w/o does not create the possibility of a new or different kind of accident or condition from previous evaluations. An increase to the enrichment level of 4.5 w/o from 3.5 w/o involved extending the previous evaluations to cover more realistic situations. The same calculational techniques and computer codes were used.

3) The margin of safety as defined in the bases for any technical specification has not been reduced. An increase in the maximum enrichment to 4.5 w/o does not involve a significant reduction in a margin of safety. As discussed above, in all cases the multiplication factors for worst case approximations fall considerably below the regulatory limit and do not represent significant reductions in a margin of safety.

Based on the above discussions , the proposed revisions to the WCGS Technical Specifications do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create a possibility for an accident or malfunction of a different type than any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the bases for any technical specification. Therefore, the proposed revisions do not adversely affect or endanger the health or safety of the general public or involve a significant safety hazard.

TECHNICAL SPECIFICATION AMENDMENT REQUEST 6.2

Title: Removal Of Organizational Charts

Description: The proposed amendment would remove the operating corporation and onsite organization charts (Figure 6.2-1 and 6.2-2) from Section 6 of the WCGS Technical Specifications and would incorporate essential organization requirements, such as lines of authority, responsibility, and communication. The amendment would also make additional editorial changes to delete references to the removed organization charts. The proposed changes are in accordance with NRC Generic Letter 88-06, "Removal Of Organization charts From Technical Specification Administrative Control Requirements".

Safety Evaluation: The regulatory requirements for Technical Specifications in 10 CFR Part 50.36 do not require that organization charts be included in the administrative controls section of plant Technical Specifications. Part 50.36 requires that necessary organization and management be in place to assure safe operation of the plant. As long as the necessary administrative controls are in place, the presence or absence of organization charts does not affect safe plant operation. To assure that the necessary administrative controls are available, Generic Letter 88-06 identifies certain provisions that must remain in the Technical Specifications. The proposed changes to the Wolf Creek Generating Station (WCGS) Technical specifications reference Chapter 13 of the WCGS Updated Safety Analysis Report (USAR), which contains the organizational information described in the Generic Letter, the required aspects of administrative controls will remain in the WCGS Technical Specifications the removal of the organization charts represents no reduction in safety commitments nor an adverse impact on a safe operation of WCGS.

Based on the above discussion, the proposed amendment would not increase the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report; or create a possibility of an accident or malfunction of a different report; or create a possibility of an accident or malfunction of a different type than any previously evaluated in the safety analysis report; or reduce the margin of safety as defined in the bases for any Technical Specification. Therefore, the proposed changes do not adversely affect or endanger public health and safety nor do they involve an unreviewed safety question.

UNREVIEWED SAFETY QUESTION USQ 88-003

Title: Cycle 4 Reload Design Evaluation

Description: This report presents an evaluation of the Wolf Creek Generating Station Unit 1, Cycle 4 reload design, and demonstrates that the core reload will not adversely affect the safety of the plant.

Safety Evaluation: The effects of the reload on the design basis and postulated incidents analyzed in the USAR were examined. In all cases, it was found that the effects were accommodated within the conservatism of the assumptions used in the applicable safety analyses.

USAR CHANGE REQUEST 87-046

Title: Evaluation Of Cycle 3 Reload Design

Description: As a result of Cycle 2 nuclear fuel depletion, the core will be loaded with fresh fuel for Cycle 3 operation. This evaluation is being performed for the WCGS Cycle 3 reload design.

Safety Evaluation: The incidents analyzed in the Wolf Creek Generating Station (WCGS) USAR which could be affected by this fuel load have been reviewed. It was found that the effects are accommodated within the conservatism of the safety analysis assumptions. All safety related parameters for the reload are bounded by the previously evaluated accident analyses. The margin of safety defined in the Technical Specifications has not been reduced.

USAR CHANGE REQUEST 87-049

Title: Add Valve Terminal Boxes To USAR
Description: Valve terminal boxes for the Reactor Vessel Head Vent Valves were inadvertently not added to USAR. This change also reflects that either Barton or Rosemount Transmitters are qualified for the Reactor Coolant System wide range pressure application.

Safety Evaluation: The terminal boxes are qualified for 40 years without contingencies, so no maintenance is required and omission of this resulted in no consequence. Either transmitter is qualified for this application hence there is no effect on the accident analysis.

USAR CHANGE REQUEST 87-051

Title: Revision To Equipment Qualification Categories

Description: This change was originated to correct USAR Sec. 3.11(b).5 to be consistent with Equipment Qualification documentation.

Safety Evaluation: A change in the status of a component from "harsh" to "mild" environment equipment based on an analysis for low-level post-accident radiation effects does not result in a change to the component itself. The change is to the method of determining the post-accident environment status as it currently exists in the USAR. A change in the post-accident environmental status from "harsh" to "mild" will result in the exemption of the component from the qualification requirements of NUREG-0588. The safety-related function and classification of the component is not affected.

The probability of occurrence or consequences of an accident or possibility of creation of an accident of a different type from any previously evaluated in the USAR will not be affected by a change in the post-accident environmental designation. The consequences of a malfunction of the equipment previously evaluated in the USAR will not be affected.

USAR CHANGE REQUEST 88-007

Title: Revision Of USAR Section Concerning Dilution During Hot Shutdown

Description: The Wolf Creek Generating Station (WCGS) USAR Section concerning dilution during hot shutdown is being revised to remove a non-conservative assumption. The original analysis assumed perfect mixing in the reactor vessel's upper head volume. Removal of this non-conservatism results in the minimum active Reactor Coolant System (RCS) volume being reduced from 4,900 cubic feet to 3,935 cubic feet, and alters the expected time of flux doubling.

Safety Evaluation: Revision of the effective mixing volume and effective time of flux doubling does not impact or violate safety limits because the combination of RCS average temperature and pressure, reactor power, and containment pressure during a Boron Dilution Event (BDE) is well within the safety limits. The change does not impact performance of systems responding to mitigate a BDE. The margin of safety as assumed in the analysis is not reduced by this modification.

USAR CHANGE REQUEST 88-020

Title: Document Change Of Equipment Qualification Work Package

Description: This change documents a change of an Equipment Qualification Work Package (EQWP) to reflect recent qualification testing which extends the qualified life of existing splice designs.

Safety Evaluation: This change involves no design change or physical change to the splices. This has no reduction in the margin of safety as determined in the bases for Technical Specifications.

USAR CHANGE REQUEST 88-021

Title: Evaluation Of Chlorine Event

Description: This change provided an evaluation of chlorine contaminated in-leakage of air, Control Room Chlorine alarm response, and changes to the chlorine gas accident scenario required for air flow rates. This USAR change request supersedes USAR change request 88-008 and its associated safety evaluation.

Safety Evaluation: Control Room habitability has not been reduced by allowing infiltration into the Control Room, and increasing air flow rates during post-isolation mode of operation. Also, the revisions to high chlorine response procedures for donning Self Contained Breathing Apparatus will enhance Control Room habitability.

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Bart D. Withers
President and
Chief Executive Officer

March 29, 1989

WM 89-0077

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: 1988 Wolf Creek Generating Station Annual Safety Evaluation Report

Gentlemen:

Attached is the 1988 Annual Safety Evaluation Report for Wolf Creek Generating Station. This report is being submitted pursuant to 10 CFR 50.59 (b)(2). This report covers the period of January 1, 1988, to December 31, 1988.

Very truly yours,

BDWithers

Bart D. Withers
President and
Chief Executive Officer

BDW/llk

Attachment

cc: B. L. Bartlett (NRC), w/a
E. J. Holler (NRC), w/a
R. D. Martin (NRC), w/a
D. V. Pickett (NRC), w/a

IEA7
1/1

P.O. Box 411 / Burlington, KS 66839 / Phone: (316) 364-8831