WOLF CREEK
NUCLEAR OPERATING CORPORATION

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

Gentlemen:

This letter transmits the Annual Safety Evaluation Report for Wolf Creek Generating Station (WCGS), which is being submitted pursuant to 10 CFR 50.59(b)(2). Attachment I provides a summary of the evaluation results. Attachment II provides the WCGS Annual Safety Evaluation Report Attachment III identifies commitments made in this report.

This report covers the period from January 1, 1999, to December 31, 1999, and contains a summary of approved safety evaluations performed during this period. If you have any questions concerning this report please contact me at (316) 364-4034, or Mr. Michael J. Angus at (316) 364-4077.

Very truly yours,

Richard A. Muench

RAM/rlr

Attachments

cc: J. N. Donohew (NRC), w/a
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WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

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

ANNUAL SAFETY EVALUATION REPORT

Report No.: 15

Reporting Period: January 1, 1999 through December 31, 1999

SUMMARY

This report provides a brief description of changes, tests, and experiments performed at Wolf Creek Generating Station pursuant to 10 CFR 50.59(a)(1). This report includes summaries of the associated safety evaluations that were reviewed and found to be acceptable by the Plant Safety Review Committee (PSRC) for the period beginning January 1, 1999 and ending December 31, 1999. In addition, one safety evaluation, 59 1999-0133, Revision 1, is included that was approved by the PSRC on January 26, 2000. This report is submitted in accordance with the requirements of 10 CFR 50.59(b)(2).

A significant number of the safety evaluations summarized in this report are a result of the Updated Safety Analysis Report Fidelity Review described in letter WM 97-0009, dated February 9, 1997, from O. L. Maynard, WCNOC, to USNRC.

On the basis of these evaluations of changes:

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

Therefore, all items contained within this report have been determined not to involve an unreviewed safety question.

Safety Evaluation: 59 1995-0180 **Revision:** 1

Domestic Water System Cross Tie to Demineralized Water System

Description:

The portion of the domestic water (KD) system inside the radiological controlled area (RCA), is connected to the demineralized water system (AN) by Revision 0 of this design change. The KD system is used for wash down inside the RCA. The KD water has a high concentration of anions which deplete the resin beds of the Radwaste system and therefore increases the quantity of Radwaste to be shipped and buried. The KD system also provided water to emergency eyewash and showers. The eyewash and shower stations will be removed and portable eyewash stations with body spray will be placed at the needed locations.

Revision 1 of this USQD adds isolation valve ANV1001 in the AN-KD cross tie line and drain valve KDV0568 in the KD line of the Turbine Building. The isolation valve allows the Demineralized Water Makeup and Storage (AN) system to be out of service for a shorter time period during the cross tie connection. The drain valve will allow the KD line (in the Turbine Building which uses Domestic water) to be drained and flushed. The two systems will be connected with a spool piece.

Safety Summary:

The following Condition II (Faults of Moderate Frequency) fault was investigated to confirm that the event does not propagate to a more serious condition or event:

Neither the KD system or the AN systems can have any potential impact on any of the accidents or Conditions discussed in Chapter 2, 3, 6, or 15 of the USAR.

The change in flow rate in the AN system (non safety related system) can not affect any system, structure, or component (SSC) important to safety. The administrative controls and the instrumentation and equipment used to control the boration function and their failure modes will not change. The ability of the auxiliary feed water system to perform its safety related function if main feed water is lost will not change. The maximum quantity of water discharged from the system as wash down will not increase and create a flooding hazard because of this change since the supply pressure remains the same. The routing inside the RCA is the same and the pipe design requirements have not changed. Therefore, there can not be any affect to equipment important to safety. The portable eyewash stations will be located away from safety related equipment and the tanks do not create a missile hazard as identified in USAR Section 3.5.1.1.2.

No acceptance limits contained in the bases of technical specifications are identified that could be affected by this design change. The reactivity control system, condensate storage tank and the auxiliary feedwater systems requirements were reviewed to confirm that none of these systems or components were affected by this design change.

Safety Evaluation: 59 1996-0106 **Revision:** 0

Permanent Auxiliary Boiler Recirculation Line

Description:

Design Change Package (DCP) 06231 makes the existing auxiliary boiler Chemical Recirculation line part of the permanent design for the Wolf Creek Generating Station. The line was installed under an operating procedure to maintain the chemistry in the Auxiliary boiler to prevent corrosion and to extend the life of the boiler.

The auxiliary boiler is part of a non-safety related system providing steam for plant heating, radwaste processing, condenser sparging, and etc. The auxiliary boiler is normally only used during refueling outages, when heat is not available from the Main Steam System. Therefore, the boiler requires long-term wet lay-up during normal plant operation and appropriate chemistry control to prevent corrosion and to maximize the commercial operating life of the unit. Because of its location, the chemical recirculation line cannot affect the auxiliary feedwater pump turbine exhaust line that passes through the same area.

Safety Summary:

There are no design basis accident in the Updated Safety Analysis Report (USAR) Chapters 2, 3, 5, 6, 9, or 15 that will be affected by the addition of this permanent chemical recirculation line. This addition of a low flow 1/2 horse power pump is either separated by structures and distance or energy levels are too low to have any affect on any safety related or special scope systems, structures or components (SSCs).

Based on the physical barriers and distance between the components, there are no credible accidents scenarios affected by these material changes and additions.

There are no credible malfunctions of equipment affected by these material changes and additions.

There are no acceptance limits contained in the bases for the Technical Specifications affected by these material changes and additions.

Safety Evaluation: 59 1997-0008 **Revision:** 1

Clarification of Freeze Protection for Plant Tanks

Description:

Steam heating coils are provided by the design for the Condensate Storage Tank (CST), the Refueling Water Storage Tank (RWST), and the Reactor Makeup Water Storage Tank (RMWST) to prevent freezing during cold weather. The steam flow to each tank was designed to be automatically controlled by control valves. The valves would modulate to maintain a nominal minimum tank temperature of 50°F. During periods of cold weather, the heating coil condensate return lines have frozen preventing proper operation of the heating system. To prevent this, the control valve bypass lines are opened during cold weather to provide a continuous steam flow. Also, during 1990, the Safety injection pump return line became blocked with ice due to inadequate freeze protection. As a result, Wolf Creek committed to having provisions for placing the RWST in a continuous recirculation mode during cold weather to prevent the Safety Injection Pump return line from freezing. The changes described in Updated Safety Analysis Report (USAR) Change Request 97-044 are provided to clarify how the affected systems are designed and operated. Specifically, the clarification deals with the minimum temperatures for the tanks and how they are maintained.

Safety Summary:

The CST provides a non-safety grade water supply to the Auxiliary Feedwater (AFW) pumps. The minimum assumed design temperature for the CST is 50°F. The temperature control valve set point is a nominal 50°F. There is no Technical Specification temperature limit for the CST. The only limit is keeping the tank above freezing; however, the design is based on a 50°F minimum. The assumed maximum operating temperature for the CST is 95°F. The safety analysis calculations have conservatively used a maximum design temperature of 120°F. The consequences of having the CST temperature slightly below 50°F is not significant. The change in mass flow will be extremely small. The cooler temperature will improve the heat removal capability of the AFW system.

The RWST provides a safety grade supply of borated water to the Emergency Core Cooling System (ECCS) pumps. The minimum temperature limit for the RWST is 37°F per Technical Specification 3/4.5.5. The temperature control valve set point is a nominal 50°F. The tank is typically operated well above this value. The maximum temperature limit per the Technical Specifications for the RWST is 100°F.

The Reactor Makeup Water Storage Tank (RMWST) supplies water to the reactor water makeup system. The RMWST is non-safety related and has no temperature limit other than the need to prevent freezing. The temperature control valve set point is also a nominal 50°F. The tank is typically operated well above this value. The maximum design temperature is 120°F and the maximum normal operating temperature is 100°F.

The 50°F temperature setting will be described in the Updated Safety Analysis Report (USAR) as a nominal design value. The control valves will be described as having their associated bypass lines open during winter operation to prevent the condensate return lines from freezing. The RWST will be described as being placed in a continuous recirculation mode through its return lines during cold weather to ensure that no freezing occurs. Although the bypass lines are open, the control valves are still enabled and can still function if temperatures are cold enough to call for additional steam flow.

All of the design basis accidents were considered. The only accidents which were determined to have the remote possibility of being affected were the inadvertent operation of the ECCS or containment spray, loss of coolant accident (LOCA), Main Steam Line Break (MSLB), and Steam Generator Tube Rupture (SGTR). These accidents were examined in detail for possible affects that would result because of this change. This change has no impact on accident previously evaluated in the USAR and creates no new accident scenarios. The temperature limits for the tanks are not being changed. The function of the control valves is a non-safety function.

The operational impact of the change will be to maintain the equipment in the desired range of temperature control. Therefore, there can be no credible malfunctions of equipment affected by the change.

The Technical Specification 3/4.5.5 requires the RWST to be maintained between 37°F and 100°F. This basis is not affected by this change because the operational impact is to maintain the tank temperature in the same required range. There are no temperature limits in the Technical Specifications for the CST or the RMWST.

Safety Evaluation: 59 1997-0027 **Revision:** 0

Correction to the One Line Diagram

Description:

Performance Improvement Request (PIR) 97-4163 was written to identify discrepancies on the electrical one line diagram for the Wolf Creek Generating Station, with respect to the correct labeling of a normally used breaker. This Updated Safety Analysis Report (USAR)change corrects this inaccuracy in USAR Figure 8.2-4-01, "WCGS Electrical One-Line Diagram." During the review of the above discrepancy a minor component number change was also identified and corrected.

A summary of the corrections to the USAR involve only corrections to USAR Figure 8.2-4-01.

1. Remove the normally open designation to the west construction power loop breaker (13-4).
2. Remove seven of the future 345 kV circuit breakers not installed in the switchyard.
3. Correct the asset (component) number for the Guard House transformer.

This document change corrects USAR figure 8.2-4 to the as-built condition of the Wolf Creek substation, showing only eight 345 kV breakers. The additional seven breakers, that were inadvertently added, are intended for five future transmission lines and Unit 2, of which none are installed. A review of the AC system analysis for Wolf Creek (Calculation XX-E-006) reveals that only the original eight 345 kV circuit breakers are modeled in the analysis.

Safety Summary:

The correction to remove a designation on a switchyard breaker does not affect any previous analysis as these loads are not modeled. The remaining typo to correct the component number does not impact any previous analysis.

This document change corrects USAR Figure 8.2-4, back to the original analysis used for the FSAR. That is, only three 345 kV transmission lines and the Wolf Creek generator come into Wolf Creek switchyard via eight switchyard breakers.

The probability of occurrence of a malfunction of equipment important to safety is unchanged. This change returns the USAR Figure 8.2-4 to the original configuration for which the plant is analyzed (eight breakers). The closed breaker change for 13-4 does not affect the load analysis model for Wolf Creek. These are part of the 69 kV system.

The radiological consequences of a malfunction of equipment is unchanged. This change will return the figure to the original configuration licensed and analyzed. Only eight 345 kV breaker exist in the Wolf Creek switchyard.

The possibility of an accident of a different type than previously evaluated does not exist. This change will return the USAR Figure to the original configuration as previously licensed and analyzed.

The possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR does not exist with this change to the USAR Figure 8.2-4. This change returns the figure to the same configuration as was previously analyzed. This change also agrees with the 345 kV lines, into the Wolf Creek switchyard, as described in USAR Section 8.2.1.1 transmission network.

The margin of safety is not reduced in the basis for any technical specifications.

Safety Evaluation: 59 1997-0085 **Revision:** 2

Procedure GEN 00-006 Revision 42

Description:

This revision of Unreviewed Safety Question Determination (USQD) 59 97-0085 is written to evaluate normal reactor cooldown using the steam generators and the main steam dump valves (MSDV) or turbine bypass system to a point below 350° F and 360 psig before initiating residual heat removal (RHR) cooling. Revision 42 of procedure GEN 00-006, "Hot Standby to Cold Shutdown," allows this method of cooldown. The simultaneous use of the RHR system and the MSDV for Reactor Coolant System (RCS) cooldown after the RCS reaches 350° F and 360 psig was evaluated by Revision 1 of this USQD and is allowed by the USAR. Initiation of RHR cooling is prohibited when the RCS is above 350° F or 360 psig. Alignment of RHR cooling to provide cold overpressure protection when the RCS is cooled below 350° F and 360 psig may be required. There are no requirements to initiate RHR cooling when these values are reached. This change will remove the implied requirement to initiate RHR cooling when these values are reached during cooldown. This change will allow refueling and maintenance conditions to be established sooner after shutdown without exceeding allowable cooldown rates. By waiting to initiate RHR cooling, the adverse effects of thermal transients on the RHR system and components will be reduced.

Revision 11 to the USAR describes two phases for a normal reactor cooldown. The MSDV are used for the first phase. The second phase begins when RCS temperature and pressure are approximately 350° F and 360 psig and RHR is placed in service. Continued use of the MSDV after RHR is placed in service is allowed.

Revision 1 of this USQD was issued to evaluate USAR Change Request 97-135 which was incorporated into the USAR by Revision 11. The revision allowed both methods of cooldown to be used simultaneously, but did not provide the flexibility to cool down below 350° F and 360 psig using the MSDV without initiating RHR cooling. The flexibility currently provided in Revision 42 of GEN 00-006 allows cooling of the RCS to below 350° F and 360 psig using MSDV without initiation of RHR cooling which would make the USAR incorrect. The revision to GEN 00-006 does not provide for tests or experiments not described in the USAR, therefore adequacy of systems, structures, or components (SSCs) to prevent accidents or mitigate the consequences of an accident are not adversely affected.

Safety Summary:

Technical Specification 3.4.12 provides the option of using the RHR relief valves or the PORVs for LTOP. The PORVs are used until the RHR system is aligned. This USAR change allows the operators to operate the systems as designed. The change covers plant operation in Mode 4, RCS temperature below 350° F. A review of accidents in USAR Chapters 2, 3, 6, 9 and 15 which may be applicable at these Mode 4 conditions include feedwater malfunctions that increase feedwater flow (S/G overfill), faulted steam generator, main steam line break, uncontrolled rod withdrawal, Boron dilution, and rod ejection. The proposed changes have no potential impact on these accidents.

This change allows operation within normal design limits. No new or credible accidents are created by the subject change because it does not affect or challenge the plant's design in an adverse or unique manner to cause any accident.

This change does not alter the controls or conditions under which equipment is designed to operate. The level of qualification is not changed and there is no effect on any SSC. The change provides the option to choose the point at which RHR cooling is initiated within its normal operating range. No credible malfunctions to any equipment have been created by this change.

Allowable RCS cooldown rate and pressure-temperature limitations are not changed, just clarifying the combined or individual use of the RHR system and turbine bypass system during cooldown. The acceptance limits contained in Technical Specification 3.4.9.1 or 3.4.9.3 have not been affected by the proposed activity.

Since no acceptance limits have been affected, the margin of safety is not affected by this change.

Safety Evaluation: 59 1997-0085 **Revision:** 3

Procedure GEN 00-006 Revision 43

Description:

Revision 3 of Unreviewed Safety Question Determination (USQD) 97-0085 evaluates the use of the Condensate Pumps to feed the steam generators during normal reactor cooldown using the steam generators and the main steam dump valves (MSDV) or turbine bypass system. The flexibility to use the condensate pumps to feed the steam generators during normal reactor cooldown is not explicitly described in the Updated Safety Analysis Report (USAR), specifically Section 10.4.7.2.3.

Revision 2 of USQD 59 97-0085 evaluates normal reactor cooldown using the steam generators and the MSDV or turbine bypass system to a point below 350° F and 360 psig before initiating Residual Heat Removal (RHR) cooling. Revision 42 of procedure GEN 00-006, "Hot Standby to Cold Shutdown," allows this to occur. The simultaneous use of RHR system and the MSDV for RCS cooldown after the RCS reaches 350° F and 360 psig was evaluated by Revision 1 of this USQD and is allowed by the USAR. Initiation of RHR cooling is prohibited when the RCS is above 350° F or 360 psig. Alignment of RHR cooling to provide Cold Overpressure Protection when the RCS is cooled below 350° F and 360 psig may be required. There are no requirements to initiate RHR cooling when these values are reached.

Revision 2 of USQD 97-0085 evaluated the removal of the implied USAR requirement to initiate RHR cooling when these values are reached during cooldown. Revision 2 allowed refueling and maintenance conditions to be established sooner after shutdown without exceeding allowable cooldown rates. By waiting to initiate RHR cooling, the adverse effects of thermal transients on the RHR system and components will be reduced.

Revision 11 to the USAR describes two phases for a normal reactor cooldown. The MSDV are used for the first phase. The second phase begins when RCS temperature and pressure are approximately 350° F and 360 psig and RHR is placed in service. Continued use of the MSDV after RHR is placed in service is allowed.

Safety Summary:

Revision 1 of this USQD evaluated USAR Change Request 97-135, which was incorporated into the USAR by Revision 11. The revision allowed both methods of cooldown to be used simultaneously, but did not provide the flexibility to cool down below 350° F and 360 psig using the MSDV without initiating RHR cooling. Revision 2 of this USQD evaluated the flexibility provided in Revision 42 of GEN 00-006 which allows cooling of the RCS to below 350° F and 360 psig using MSDV without initiation of RHR cooling.

This revision of the USQD evaluates the flexibility provided in GEN 00-006 to use the Condensate Pumps to feed the steam generators during cooldown. This flexibility is not explicitly described in the USAR, specifically Section 10.4.7.2.3. The revision to GEN 00-006 does not provide for tests or experiments not described in the USAR, therefore adequacy of SSCs to prevent accidents or mitigate the consequences of an accident are not adversely affected.

Technical Specification 3.4.12 provides the option of using the RHR relief valves or the PORVs for LTOP. The PORVs are used until the RHR system is aligned. This USAR change allows the operators to operate the systems as designed. The change covers plant operation in Mode 4, RCS temperature below 350° F. A review of accidents in USAR Chapters 2, 3, 6, 9 and 15 which may be applicable at these Mode 4 conditions include feedwater malfunctions that increase feedwater flow (steam generator overfill), faulted steam generator, main steam line break, uncontrolled rod withdrawal, Boron dilution, and rod ejection. The proposed changes affect the Feedwater Isolation Signal (FWIS) for lo-lo Tavg (P-4) only. All ESF protective functions for feedwater isolation remain enabled. These changes have no potential impact on these accidents.

This change allows operation within normal design limits. No new or credible accidents are created by the subject change because it does not affect or challenge the plant's design in an adverse or unique manner to cause any accident.

The subject change does not alter the normal controls or conditions under which equipment is designed to operate. The level of qualification is not changed and there is no effect on any system, structure, or component (SSC). The change provides the option to choose the point at which RHR cooling is initiated within its normal operating range. It also provides the option of feeding the steam generators using condensate pumps. It provides the option to use main steam, auxiliary steam or extraction steam for preheating the feedwater during plant heatup. No credible malfunctions to any equipment have been created by the subject change.

Allowable RCS cooldown rate and pressure-temperature limitations are not changed, just clarifying the combined or individual use of the RHR system and turbine bypass system and method for feeding steam generators during cooldown. The acceptance limits contained in Technical Specification 3.4.9.1 or 3.4.9.3 have not been affected by the proposed activity.

Since no acceptance limits have been affected, the margin of safety is not affected by this change.

Safety Evaluation: 59 1997-0086 **Revision:** 4

Fiber Optic Containment Penetration

Description:

Revision 9 of Design Change Package (DCP) 07065 expands the scope of this modification to include communications cabling and equipment for remote radiation monitoring/dosimetry and for non-outage video monitoring. This expanded scope requires the routing of fiber optic cables to locations not previously identified in DCP 07065. This revision adds video monitoring equipment in the Health Physics (HP) Access Control office, Room 3222, and the installation of remote receiving and monitoring equipment in the 1974 and 2000 elevation of the Auxiliary Building and the 2047 elevation of the Fuel Building. The cables added to the auxiliary building and fuel building will be free air routed. Appendix R Safe shutdown cable separation and Regulatory Guide 1.75 cable separation requirements are maintained. The fire loading identified in Updated Safety Analysis Report (USAR) Section 9.5B will be affected because there will be an increase in the combustibles due to new cables. Different fire zones will be affected due to the video monitors, cable, receivers and transmitters.

Safety Summary:

The electrical separation criteria for permanent plant raceway at Wolf Creek was developed from IEEE 384-1974 and NRC Regulatory Guide 1.75, Revision 1. The criteria is applied to Wolf Creek Generating Station (WCGS) in documents E-0, E-11013 and E-1R8900. The separation criteria established in these documents is for maintaining the redundancy of Class 1E power systems and the redundancy of safety related protection systems. The regulatory commitments are identified in USAR Section 8.3.1.4.1. These requirements are being met by the cable installed under this revision All cabling identified under this modification is non-safety related and is not connected to any plant process systems. Cable jacketing is IEEE-383 qualified for flame retardance as required by electrical design criteria. The additional combustible load is very low and does not affect inputs, assumptions, or results of any previously performed fire hazards analysis.

All non-safety related cable and equipment added or affected by Revision 9 of DCP # 07065 is installed to meet fire safe shut down requirements and Regulatory Guide 1.75 separation criteria and seismic II/I requirements. None of the design basis accidents identified in the subject chapters involve non-safety related cable faults in these raceways. Therefore, the additions have no impact on accident discussions in the USAR.

Due to the restrictions placed on their installation, the non-safety related cables can not cause any credible accidents. The only credible failure mode is the failure to perform their own non-safety functions.

These functions are typically communications for non-process computers, outage video communications, outage microcell telephone system interconnection, remote radiation monitoring and dosimetry and communications for other outage activities such as eddy current testing and sludge lancing. Cable failures in these applications do not affect nuclear safety related systems. No new fire scenarios are created.

The cable and equipment additions identified on Revision 9 of DCP 07065 will adhere to design criteria specified in applicable design documents. Based on this, credible malfunctions of equipment important to safety are not affected.

Design margins for cable separation will not be exceeded since the applicable design requirements are being followed for this installation. The combustible loading additions have been evaluated, found to be acceptable and the information concerning fire loading in the USAR is being revised accordingly. There are no acceptance limits in the Technical Specifications or licensing basis documents affected by this change.

Safety Evaluation: 59 1997-0110 **Revision:** 0

Emergency Diesel Generator Starting Air System Valves KJ V715A and B

Description:

This Updated Safety Analysis Report (USAR) change revises the description of the valve in the interconnecting pipe between the two starting air tanks for the Emergency Diesel Generators (EDG) in USAR Section 9.5.6.2.3 from normally open to normally closed. In addition, this change adds a parenthetical "administratively controlled" to describe how the system functions as further described in the USAR. These valves are KJ-V715A and KJ-V715B for EDGs A and B respectively.

For each EDG, the system design has two compressors, two dryers (one dedicated to each compressor) and two starting air tanks. Each compressor/dryer can be aligned (via the above mentioned valves) to change one or both of the starting air tanks. During normal operation of the plant, the cross connect valve in the interconnecting pipe upstream of the starting air tank is preferred to be closed. This ensures that one compressor is aligned to maintain the required pressure in one starting air tank and not both tanks, i.e., with KJ-V715A closed, Starting Air Compressor CKJ01A will only charge Starting Air Tank TKJ02A; with KJ-V715A open CKJ01A will charge both TKJ02A and TKJ02B. The controls for the air compressor are capable of being aligned to the pressure switch associated with either or both of the starting air tanks. To maintain independence, and redundancy between the starting air trains, these controls are normally aligned to only the tank associated with that compressor.

Safety Summary:

The safety function of the starting air system is to provide air to roll the diesel at sufficient speed for it to start when required for accident mitigation. This functions is provided by the starting air tanks alone, either one of which meets the requirement for starting air by itself and without dependence on the compressors. The starting air compressors are non-safety related components which are provided to recharge the starting air tanks after use and to make up for the minor leakage in the starting air system.

Valves KJ-V715A and KJV715B are upstream of the starting air tanks, before the check valves which separate the safety related and non-safety related portions of the system. Therefore, these valves are non-safety related and are not required for the starting Air Tanks to perform their safety function. Changing the USAR to reflect the normal position of these valves from normally open to normally closed does not affect any system, structure, or component (SSC) required for accident prevention or mitigation and does not negatively affect the performance of activities that are important to the safe and reliable operation of the plant.

The EDGs are designed to provide power to the plant in the event of a loss of off-site power coincident with any of the accidents discussed in the USAR. However, this change does not affect the ability of the EDGs to start and operate on demand. Therefore no design basis accidents are impacted because the proposed change does not affect any SSC utilized to prevent or mitigate the consequences of an accident nor does the change affect any of the event or conditions associated with initiation of design basis accidents.

This change does not affect any safety related design function of the EDGs to start or operate when demanded. The EDG system is an accident mitigation system. No credible accidents will be created by changing the normal position of the valves described above.

The ability of the diesel generators to start or operate is unaffected by this change. Therefore, this change does not affect any SSC utilized to prevent or mitigate the consequences of an accident. This change is associated with non-safety related valves. No new equipment or components are being added. This change will not alter or create any new failure modes which directly or indirectly affect equipment important to safety.

Since this changed does not affect any SSC utilized to prevent or mitigate accidents and the change will not alter or create new failure modes for the identified valves, no acceptance limits are affected.

Safety Evaluation: 59 1997-0169 **Revision:** 1

Revision to Radiological Posting Practices

Description:

Revision 1 of this Unreviewed Safety Question Determination (USQD) corrects typographical errors found in USQD 97-0169, Revision 0. Revision 1 of this USQD corrects the abbreviation of Supplier Material Quality from "MSQ" to "SMQ". Revision 0 of this USQD which was reported to the NRC on March 11, 1998.

The corrected text of Revision 1 of USQD 97-0169 is provided below.

Revision 3 of AP 25A-001, "Radiation Protection Manual", will allow personnel outside of the Health Physics (HP) Department to establish and remove radiological postings. Specifically, Revision 3 allows Supplier Materials Quality and Quality Control radiography personnel who are appropriately trained, to make radiological postings, which are required by I-SMQ-008, "Material conformation (X-Met)" or AP 25B-200, "Radiography Guidelines". These procedures control activities performed with the alloy analyzer and other radiological equipment.

The effect of the procedure change to allow properly trained individuals to make radiological posting in accordance with approved procedures is to increase the efficiency of Health Physics (HP)`, Supplier Materials Quality (SMQ) and Quality Control (QC). Work by SMQ or QC will not be dependent on the availability of a HP technician to make the postings and the HP technician will not be pulled from other plant duties.

This requirement of USAR limits radiological posting responsibilities to only the HP staff.

Safety Summary:

The proposed procedure change does not have an impact on design basis accidents.

Allowing properly trained SMQ or QC personnel to make radiological postings for procedurally controlled activities does not create a new credible accident.

The posting of radiological signs by SMQ or QC rather than by HP does not effect the credible malfunctions of equipment important to safety.

The probability of a design basis accident (DBA) is not effected by the proposed procedural change. The proposed procedural change does not add or modify plant equipment, setpoints, performance requirements, or qualification requirements. It does not alter the conditions or assumptions used for evaluating design bases accidents. Allowing SMQ and QC to make radiological postings for their work activities will not effect the DBA analysis. The posting will still be done by trained, qualified individuals, in accordance with approved procedures.

The proposed procedural change does not modify plant equipment, setpoints, performance or qualification requirements. It does not alter the conditions or assumptions used for evaluating the radiological consequences of design bases accidents. The radiological consequences are not changed.

The proposed procedural change does not add or modify plant equipment, setpoints, performance or qualification requirements. The probability of a malfunction of safety related equipment is not changed by allowing SMQ or QC to make radiological postings for their work activities. It does not alter the conditions or assumptions used for evaluating the radiological consequences of design bases accidents. The radiological consequences of an equipment malfunction is not changed.

This change does not alter the conditions or assumptions used for evaluating the radiological consequences of design bases accidents. The posting of radiological signs by SMQ or QC rather than by HP does not create a different type of accident from those evaluated in USAR.

This change does not alter the conditions or assumptions used for evaluating the radiological consequences of design bases accidents. The posting of radiological signs by SMQ or QC rather than by HP does not effect the credible malfunctions of equipment important to safety. No acceptance limits are effected by the proposed procedural change.

Safety Evaluation: 59 1998-0020 **Revision:** 1

Evaluation of Thermal Relief Valve Piping Removal

Description:

Design Change Package (DCP) 07693, Revision 0, which was evaluated by Unreviewed Safety Question Determination (USQD) 59 98-0020, Revision 0, approved the deletion of thermal relief valve (BGV0020), which is scheduled to be removed in Refuel (RF)10. To maintain the structural integrity of the piping system and for minimum field changes, the relief valve was to be replaced with an elbow spool piece and capped with blind flanges.

An interim repair was performed due to the leakage on the Component Cooling Water (CCW) piping downstream of thermal relief valve BGV0020 where the one inch pipe intersects with the six inch pipe. The repair was performed per Configuration Change Package (CCP) 08019, Revision 0, by providing encapsulation on the joint connection. The change package also required the removal of the encapsulation during Refuel 10 along with the 1 inch cracked socket welded fitting.

After deleting the thermal relief valve, which was approved during Revision 0 of this change package, there is no design function of 1" and ¾" piping. Maintenance requested Engineering to evaluate the deletion of 1" and ¾" piping along with the thermal relief valve. Engineering found that it is acceptable to delete the thermal relief valve along with its associated piping. Updated Safety Analysis Report (USAR) Figures 9.3-8 -02, 9.2-15 -03 and Table 3.11(B)-3 are to be revised to incorporate the above changes. The thermal relief valve (BGV0020) and the associated piping will be deleted from the system.

The function of the thermal relief valve is to release the excess pressure of the system due to expansion of the fluid in an isolated component (i.e. the component is out of service, inlet & outlet isolated) and that the system is designed for the highest expected pressure it can be subjected to during operable modes.

The chemical and volume control letdown heat exchanger, EBG01 is designed to the requirements of ASME Section III. The ASME Section III Code (Summer 1975 Addenda) requires the designer to make consideration for over pressure protection. ND-7110 states "vessels, tanks, piping, pumps and valves shall be protected while in service from the consequences arising from the application of steady state or transient conditions of pressure and coincident temperature that are in excess of the design conditions specified for the system. Individual components which are isolable from normal system over pressure protection shall be reviewed to determine whether additional individual over pressure protection is necessary (ND-7155)".

The ASME Section III Code further states in ND-7110 that "overpressure protection is required at pump discharges except for centrifugal pumps which are designed and applied so that a pressure in excess of the maximum allowable working pressure cannot be developed." "Exhaust and pump suction lines shall have relief valves of suitable size unless the lines are designed for the maximum pressure to which they may be subjected."

The following question was asked to ASME Section III committee, "A portion of a piping system consisting of piping and isolation valves is out of service and performs no required function. Does Section III require the isolated portion of the system to be protected from overpressure during these conditions". Interpretation No. III-1-95-18 of ASME Section III Interpretation Volume 38, replied "No" to the above question.

It has been concluded that the relief valve in question only provide a relief function when the component is out of service (i.e. inlet & outlet isolated) and that the system is designed for the highest expected pressure it can be subjected to during operable modes. This conclusion means that the relief valve is not required by the ASME Section III Code nor required by the design of the system for normal operation.

Safety Summary:

There are no procedures, activities, administrative controls, or sequences of plant operations; or any plant structures, systems, components or equipment; or any requirements outlined, summarized or described in the USAR, which, if the proposed activity were implemented, would make information in the USAR no longer true or accurate or would violate a requirement stated in the USAR. There are no tests or experiments not described in the USAR which may adversely affect the adequacy of SSCs to prevent accident or mitigate the consequences of an accident.

The thermal relief valve and associated piping are no required for over pressure protection during normal operation. This function is over pressure protection when the heat exchanger is isolated. When shutdown for maintenance the heat exchanger is isolated. The thermal relief valve and its associated piping perform no safety function during operable modes and only provided pressure boundary integrity of the component cooling water. The new configuration will provide the same function and the same level of design margin as the relief valve and its associated piping.

Anytime the heat exchanger is isolated it is required by procedure, AP-21-001, "Clearance Orders", that the heat exchanger shall be either drained or vented to atmosphere to prevent the possibility of over pressure from a thermal transient event. The proposed activity can not create potential impact to design bases accidents as identified, discussed or referenced in USAR chapter 2, 3, 6, 9 or 15.

Because the heat exchanger is drained or vented to atmosphere to prevent the possibility of over pressure from a thermal transient event no new credible accident could be created by this proposed activity. Neither could any new credible malfunctions of equipment important to safety could be affected.

Deletion of the thermal relief valve and its associated piping can not create any type of credible accidents because the existing design basis safety function of the system, without the relief valve and its associated piping, is not affected during operable modes.

In addition, this change could not directly or indirectly affect any type of credible malfunctions of equipment important to safety because the existing design basis safety function of the system, without the relief valve and its associated piping, is not affected during operable modes.

This USAR change will not affect the acceptable limits of the bases of any technical specifications.

Safety Evaluation: 59 1998-0043 **Revision:** 1

Chemecial and Volume Control System Insulation

Description:

This modification is a result of Wolf Creek Nuclear Operating Corporation's response to NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity." The purpose of this modification is to replace portions of the insulation on the Chemical and Volume Control System (CVCS) normal letdown piping inside the containment with FOAMGLAS® Super K™ insulation. This will prevent thermally induced overpressurization in the pipe during a loss-of-coolant accident (LOCA) or high energy line break (HELB) condition. Thermally induced overpressurization of isolated water-filled piping sections could potentially cause ASME code components to experience stresses beyond the code allowable stresses.

Revision 0 of this Unreviewed Safety Question Determination (USQD) evaluated the safety aspect of installing safety related insulation on certain sections of the Chemical and Volume Control System. It also evaluated the safety aspect of using a computer code which is not listed in the Updated Safety Analysis Report (USAR) for performing the design evaluation of the insulation.

Revision 1 of this USQD removes the evaluation for the computer code. The evaluation performed in Revision 0 for the safety related insulation remains valid.

Revision 0 of this USQD indicated that a computer code named "PIPEPRESS" would be used to calculate the thermal and pressure response of isolated water solid piping sections to a postulated design basis accident event inside the containment. Therefore, "PIPEPRESS" was evaluated and a USAR change was prepared to include "PIPEPRESS" in the USAR as an acceptable computer code. During the analysis portion of the design, it was determined that the computer code named "ANSYS" would be a better code to use for this application due to a potentially non-conservative result using "PIPEPRESS". Therefore, the USAR change is being revised to remove "PIPEPRESS" as an acceptable code. "ANSYS" is already in the USAR as an acceptable code, therefore, there is no effect on the USAR.

Safety Summary:

Since "ANSYS" is already included in the USAR and it is an acceptable computer code to use at Wolf Creek, there are no design basis accidents or malfunctions of equipment which need to be reviewed as a result of this change to the USAR. There are no credible accidents which the use of ANSYS can create. There are no procedures, activities or acceptance criteria which are affected by the use of ANSYS for this evaluation.

Because no design basis accidents are identified, the probability of occurrence of an accident is not affected. Because no design basis accidents are identified, the consequences of accidents are not affected. Because no malfunctions are identified the probability of occurrence of a malfunction is not affected. Because no malfunctions are identified, the consequences of a malfunction are not affected. Because no credible accidents that could be created are identified no accidents of a different type can be created. Because no malfunctions are identified no malfunctions of a different type can be created. Because no acceptance limits are identified that could be affected, the margin of safety is not affected.

Safety Evaluation: 59 1998-0050 **Revision:** 0

Changes to Clarify and Update Information in the USAR

Description:

The proposed changes to the Updated Safety Analysis Report (USAR) Sections 3.11(B), 9.4 and USAR Table 3.11(B)-1 will clarify information, correct typographical errors, and coordinate temperature and humidity values given referenced in these sections. In some instances the USAR has referenced a specific temperature value instead temperature range. Office spaces, work spaces, laboratories and etc. are intended to be operated within a temperature and/or humidity range and are typically provided with a thermostat for temperature regulation.

In USAR Section 3.11(B).2.3.1, the Control Room (CR) temperature and humidity are referenced as a single digit. The single values will be changed to reference a range similar to the values given in Table 3.11(B)-1. The sentence will be changed to read "Normally, the temperature and humidity in the control room are maintained between 62 degrees to 84 degrees F and 20 percent to 70 percent respectively."

In USAR Section 3.11(B).2.4 the discussion for Essential Service Water (ESW) pump house temperature reads "It is normally heated to maintain 50 degrees F". This will be changed to agree with the temperature range given in Table 3.11(B)-1. It will be changed to read "It is normally heated to maintain a temperature equal to or greater than 50 degrees F."

The following revisions will be made to USAR Section 3.11(B).4:

1. Paragraph 1, sentence 1, will be grammatically changed to provide clarification.

2. In paragraph 2, sentence 1, the plant computer is referenced as the balance of plant (BOP) computer. The plant computer is no longer named BOP. The name BOP will be changed to simply state "plant computer".

3. In paragraph 2 sentence 2, will be revised to recognize that room temperatures are monitored either by a plant computer or a plant operator/building watch.

4. In paragraph 3, sentence 1, the Control Room (CR) temperature range is given as 78 degrees F +/- 6 degrees F. This range will be change to agree with the value given in Table 3-11(B)-1.

The following changes will be made to Table 3.11(B)-1:

1. On sheet 4, the temperature for room 3222 will be changed from 78/60 degrees F to 85/60 degrees F. The 85 degrees F temperature will be in

agreement with the rooms surrounding room 3222. This area is served by a central air unit and is regulated by a single thermostat. Since these rooms are open and communicate with each other it would not be practical to cool a specific room at a temperature different from the general area.

2. On sheet 4, the control building cable chase is identified as room 3220. This is a typo the room should be identified as 3230.

3. On sheet 5, the minimum temperatures for Control Room area rooms 3601, 3605 and 3609 is given as 72 degrees F. This temperature will be changed to 62 degrees F. It is desirable to maintain the Control Room at a temperature lower than 72 degrees F in order to enhance equipment, instruments, and control systems reliability. Engineering disposition to EER 88-GK-14 approved a Control Room and surrounding area temperature of 60 degrees F.

4. On sheet 5, Room 3613 is identified as a computer room. This space no longer houses a computer it has been converted into office space. The new values will be 84/62 degrees F and 70/10 percent for the temperature and humidity respectively.

USAR Page 9.4-9, Last Paragraph:

In the last sentence of this paragraph, a temperature range "temperatures between 60 and 70 degrees F" is stated for the access control area. This will be revised to comply with Table 3.11(B)-1. The revised version will read "temperatures above 60 degrees F".

USAR Page 9.4-11:

1. In the 2nd paragraph, the reference to the access control temperature value will be deleted since this value is already listed in Table 3.11(B)-1.

2. In paragraphs 5, 6, and 7 the temperature values of 78, 72 and 75 degrees F, respectively will be changed to 84, 62 and 84 degrees F in order to agree with the values given in Table 3.11(B)-1.

USAR Page 9.4-12:

1. The last sentence in paragraph 3 will be deleted since the temperature range is already listed in Table 3.11(B)-1.

2. Paragraph 4 is being revised to remove the specific range for relative humidity. The specific range is already listed in Table 3.11(B)-1.

There are no expected negative affects for the changes described above. The changes are being made to correct a typographical error, coordinate the information given in Table 3.11(B)-1 with the other sections of the USAR, and to revise temperature and humidity values to agree with how the plant operates.

Safety Summary:

Lowering the temperature range a few degrees in the Control Room; revising the temperature values in the access control and counting room areas to allow the open areas have the same temperature ranges; updating the old computer room data; and correcting the Essential Service Water (ESW) reference temperature will have no affect on procedures, activities, administrative controls, or sequences of plant operations. Additionally these changes will have no affect on any plant structure, systems, components, or equipment; or any requirement outlined, summarized or described in the USAR. There are no tests or experiments associated with the proposed changes.

There are no design bases accidents discussed or referenced in the USAR Chapters 2, 3, 6, 9, or 15 that are potentially impacted by the proposed changes. Lowering the temperature range in the Control Room has been previously evaluated and would actually enhance performance of safety related electronic equipment and/or instrumentation

No credible accidents will be created by the proposed activity. Expanding the temperature ranges a few degrees for the Control Room, access control and computer rooms will not change how personnel perform their duties in these areas. Personnel can still change the thermostat setting to a desired temperature. The temperature ranges are being expanded to account for temperatures that may normally be seen in these areas.

There are no credible malfunctions of equipment important to safety, which may be directly or indirectly affected by the proposed changes. The proposed changes are to clarify, coordinate, correct a typo, and correct information in the USAR. Operating at a slightly lower temperature range in the Control Room and Central Alarm Station (CAS) areas enhances the functionality of the equipment.

There are no acceptance limits contained in the basis for the technical specifications for licensing basis documents that could be affected by the proposed activities.

The changes will have a minimal effect on the Control Room operator comfort. The high temperature range in the Control Room and CAS areas have not been changed. The thermostat can still be adjusted to meet Control Room demands.

The proposed change will not increase the probability of occurrence of an accident previously evaluated in the USAR. This issue is dealing with adjusting the temperature ranges in the Control Room and access control areas. The proposed changes will not create a condition that would enhance the circumstances that would create an accident.

The proposed changes will not increase the radiological consequences of an accident previously evaluated in the USAR. These changes will not alter any radiation boundaries, monitoring or detection equipment, or change/alter a monitored release path.

The only area of change that contains safety related equipment or control of safety related equipment important to safety is the Control Room. Allowing the Control Room temperature to operate a few degrees lower will enhance equipment and instrumentation performance and increase life expectancy. The proposed changes will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

If a piece of equipment malfunctioned lowering the temperature range in the Control Room a few degrees would not further degrade the material or components in that equipment. Lowering the temperature a few degrees would actually enhance equipment functionality. Thus there would be no means of increasing the radiological consequences of a malfunction of equipment important to safety.

The proposed change will not create the possibility of an accident of a different type than any previously evaluated in the USAR.

The proposed change will not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

Safety Evaluation: 59 1998-0076 **Revision:** 1

Updated Safety Analysis Report Description Correction

Description:

Revision 0 of this Unreviewed Safety Question Determination (USQD) was previously reported on March 11, 1999. This change was later rejected because the Revision 0 USQD evaluation characterized the change as a "simple text change." Revision 1 of this USQD re-evaluates this change as a design change. Changing from packless to a packed valve should be treated as a design change. This revision incorporates that method of evaluation.

Performance Improvement Request (PIR) 98-1714 identified a discrepancy in Updated Safety Analysis Report (USAR) paragraph 12.1.2.5.d.1 on page 12.1-7, Revision 8. The USAR text states that to alleviate airborne radioactivity in containment, WCGS design provisions includes "A packless, low-leakage, ball-type pressurizer spray valve". The valves under discussion utilize packing, per approved plant design, to prevent leakage around the valve stem. The design is consistent with USAR Section 5.4.12.2, which states that throttling type control valves in the Reactor Coolant System (RCS), which includes the pressurizer spray valves, are provided with graphite packing. To help assure low leakage, the packing design was previously enhanced by installing live-loading packing under Plant Modification Request (PMR) 02232.

Safety Summary:

Therefore, the text is being changed to read; "A low-leakage, ball-type pressurizer spray valve." The USAR change is made to provide a correct description of the pressurizer spray valves. The proposed change does not affect existing plant design or other information provided in the USAR.

The change from a packless valve to a packed valve has no effect on any procedures, activities, administrative controls, or sequences of plant operations described in the USAR. The USAR does not address the details of valve packing other than material, and does not address procedures dealing with packing maintenance. The change is consistent with the USAR requirement in Section 5.4.12.2 which requires graphite packing in valves 3" or larger. Therefore, no requirement of the USAR is violated or made untrue.

Design Basis Analysis (DBAs) that involve the spray valves are found in Chapter 15 as follows:

15.2.2, Loss of External Electrical Load
15.2.3, Turbine Trip
15.3.3, RCP Shaft Seizure

15.5.2, CVCS Malfunction that Increases Reactor Coolant Inventory.

Each of the above DBAs include scenarios where the pressurizer spray valves (PSVs) are both operable and inoperable. The DBAs listed above are mitigated by the Power Operated Relief Valves (PORVs) and the pressurizer safety valves, and no credit is taken for the PSV for overpressure protection. The safety functions of the PORVs and the pressurizer safety valves are unaffected by the change. The change does not alter the way in which the valve operates. Therefore there is no impact on any DBA, and no DBAs are identified for which the pressurizer spray valves perform an accident mitigation function.

The change represents a minor change in the valves' configuration, but does not affect the way in which the valve operates. The pressure boundary of the valves is unaffected. The change brings the plant into its analyzed condition assumed in the Chapter 15 accident analysis. Therefore, no new accident could be created.

Packed valves are susceptible to minor leakage due to deterioration of the packing. These valves were modified several years ago and are now equipped with live loaded packing which minimizes the possibility of a packing leak. The valves are designed to operate with packing installed. Furthermore, a packless valve is also subject to leakage of a similar nature due to rupture or cracking of the diaphragm. A packing leak would not affect operability of the PSVs or any other safety-related equipment. Additionally, leakage of the spray valves is assumed in the USAR Section 12.1.2.5 "Examples of Radiation Protection Design Reviews."

The pressurizer spray valves are not used to protect the plant from an overpressure event. This change does not affect the operation of the valves, or any setpoint or parameter involving overpressure protection. Therefore no acceptance limit is affected.

Safety Evaluation: 59 1998-0080 **Revision:** 1

Updated Safety Analysis Report Description Changes

Description:

This is a revision to the Unreviewed Safety Question Determination (USQD) for Updated Safety Analysis Report Change Request (USARCR) 98-140. This supplemental revision is to delete Item 17 from USARCR 98-140. The original USQD (with Item 17 now shown as deleted) was as follows: This USARCR is corrective action for Performance Improvement Request (PIR) 97-0580 which addresses discrepancies associated with Updated Safety Analysis Report (USAR) Section 9.5 found during Self Assessment (SEL) 97-001, "Fire Protection Program USAR Compliance." This USARCR also serves as corrective action for Performance Improvement Request (PIR) 98-2334 which also deals with USAR discrepancies associated with Fire Protection. The USAR will be changed as follows:

1) Page 9.5-31: in the first paragraph under Section 9.5.1.7, change references to the Fire Protection Manual (FPM) to Fire Protection Administrative Procedures.
2) Page 9.5-33: change the title of Section 9.5.1.7.4 from Fire Protection Manual to Fire Protection Plan. In addition, change the first paragraph in Section 9.5.1.7.4 to read "AP 10-100, Fire Protection,..." as opposed to "The WCGS Fire Protection Manual..."
3) Page 9.5-34: in the first paragraph in Section 9.5.1.7.5.1.2.1 change "... the Fire Brigade Leader proceeds to the vicinity of the fire..." to "...the Fire Brigade Leader selects a command location...." In addition, change "as directed by the Shift Supervisor." to "as he deems appropriate." The fire brigade leader directs the off site fire department.
4) Page 9.5-35: change the first paragraph in Section 9.5.1.7.5.2.1.2 as follows:
"Before being assigned to the Fire Brigade, a person must successfully complete a formal training program established by the Supervisor Fire Protection. This program may consist of off-site training conducted by outside organizations, on-site training conducted by WCGS personnel, or a combination of both...."
5) Page 9.5-36: in Section 9.5.1.7.5.2.1.4 clarify the words "Section 2.1.2" by changing them to read "Section 9.5.1.7.5.2.1.2." Likewise, change "2.1.3" to read "9.5.1.7.5.2.1.3."
6) Page 9.5-36: in Section 9.5.1.7.5.2.1.5 clarify the words "Section 2.1.2" by changing them to read "Section 9.5.1.7.5.2.1.2." Likewise, change "Section 2.6" to read "Section 9.5.1.7.5.2.6."
7) Page 9.5-38: in the first paragraph in Section 9.5.1.7.5.2.6.2 change "Drill monitoring is performed by station management and training personnel supervised by the WCGS Fire Protection Coordinator" to "Drill monitoring is performed by the Fire Protection Staff with the assistance of other plant personnel as designated by the Supervisor Fire Protection." The FP Supervisor is responsible for all training and drills

8) Page 9.5-38: in the last paragraph on the page change Fire Protection Specialist to Supervisor Fire Protection.
9) Page 9.5-39: in Section 9.5.1.7.5.3.2 change "A permit system controls the storage and handling of combustible materials including welding and cutting and acetylene-oxygen gas systems inside or adjacent to safety-related areas of WCGS" to "A permit system controls the storage and handling of combustible materials. Welding, cutting and acetylene-oxygen gas systems, inside or adjacent to safety-related areas of WCGS are controlled by the Transient Ignition Source Program."
10) Page 9.5-39: change Section 9.5.1.7.5.3.6 as follows: "Smoking is prohibited in the Protected Area Boundary except where specifically designated."
11) Sheet 1 of Table 9.5A-1: in the Wolf Creek Generating Station (WCGS) column, change all references from "Fire Protection Supervisor" to "Supervisor Fire Protection." With the aforementioned changes incorporated change the second paragraph in the WCGS column as follows: "The President and Chief Executive Officer, Manager Resource Protection and Supervisor Fire Protection are supported by a Fire Protection Engineer who reports to the Vice President Engineering. This Fire Protection Engineer shall be a member..." An SFPE member grade or equal is available in Engineering.
12) Sheet 2 of Table 9.5A-1: in the WCGS column, change "Fire Protection Supervisor" to "Supervisor Fire Protection."
13)Sheet 9 of Table 9.5A-1: in the WCGS column, change "The Burlington Rural Fire Department" to "Coffey County Fire District #1."
14) Sheet 11 of Table 9.5A-1: in the WCGS column, delete the words "Section 2.0 of" from the following paragraph: Details concerning Fire Brigade Training and Drills can be found in Section 2.0 of the WCGS Fire Protection Program.
15) Sheet 13 of Table 9.5A-1: in the WCGS column, delete all references to the words "Section 2.0 of."
16) Sheet 49 of Table 9.5A-1: in the WCGS column, change "Burlington" to "Coffey County" and add "(Lyon County)" after Emporia.
17) Sheet 64 of Table 9.5A-1: in the WCGS column, change the following: The WCGS computer is not safety-related. The computer room is protected by detectors. Manual hose stations and portable extinguishers are provided.
18) Sheet 74 of Table 9.5A-1: in the WCGS column, delete the following: "(See section 3.2 of the WCGS Fire Protection Program.)"
19) USAR Section 13.1.2.5, delete the second paragraph in it's entirety. The Section 13.1.2.5 is entitled "Manager Training" and the second paragraph states:

"Through his staff, he assists the Fire Protection Specialist in developing and administering the training portions of the Fire Protection Program. He is responsible for documentation of all Fire Protection Program training."

The necessity for this supplement stems from a USARCR Review of the original USARCR 98-140. This review revealed that the paragraph referred to in Item 17 above should have been changed to "Not Applicable."

Safety Summary:

The above USAR changes clarify the general description, system operation, component description and administrative statements to reflect the way the fire protection system and program is operated. The changes do not affect the design basis function of the system. Therefore, there are no design basis accidents impacted by this change.

Since there are no physical changes and the design basis function of the system is not affected by this change, no new types of accidents not previously analyzed could be created.

Since the proposed change would not affect the system's failure mode, the systems design function, the level of qualification, or equipment important to safety, no credible malfunctions of equipment important to safety are identified.

There are no acceptance limits associated with these editorial changes, therefore, no acceptance limits could be affected.

Safety Evaluation: 59 1998-0081 **Revision:** 1

Changes to Reflect As-built Conditions in the USAR

Description:

The proposed activity involves removal of covers over the cask loading pit shown in Updated Safety Analysis Report (USAR) Figure 1.2-20, "Equipment Location Fuel Building Plan Elevation 2000'-0", 2026'-0" and 2047'-6", Figure 1.2-22, "Equipment Location Fuel Building Sections D, E, & F," Figure 3.8-96, "Fuel Building Plan - Elev. 2047'-6"," and Figure 3.8-98, "Fuel Building - East-West Cross Section." These covers were removed prior to initial start-up and efforts to locate documentation of the final disposition of these covers were unsuccessful. Based on an engineering evaluation, the function of the covers was to prevent objects (equipment/personnel) from accessing the cask loading pit, protect handling tools which are hung in the cask loading pit, and allow for additional space in the exclusion area. There is no design requirement for covers over the cask loading pit since the analysis of heavy load drop does not consider the pit covers. Planning and performing heavy lifts and transferring heavy loads within the power block including cask loading pit is governed under the Administrative Controls Program AP 14-001, "Control of Heavy Loads." Other Administrative Controls require workers to wear flotation devices or safety harnesses when working within six feet of the cask loading pit.

Safety Summary:

The fuel handling accident analysis in the fuel building is performed in USAR Section 9.1.4.3 and WCNOC-4, "Report on Control of Heavy Loads." The analysis considers a fuel cask drop from a vertical or a tipped position into the pit and does not take credit for the pit covers. Workers are required to wear flotation devices or safety harnesses when working within six feet of the cask loading pit. Therefore, the proposed activity has no effect on any procedure, activity, administrative control, or sequence of plant operations and does not violate any requirement stated in the USAR. Neither the proposed activity nor the design change are associated with any test or experiment.

The fuel handling accident analysis in the fuel building is performed in USAR Section 9.1.4.3 and WCNOC-4. The analysis considers a fuel cask drop from a vertical or a tipped position into the pit and does not take credit for the pit covers. Therefore, the proposed activity does not have any influence on equipment or parameters associated with any design basis accident discussed or referenced in the USAR.

The fuel handling accident analysis in the fuel building is performed in USAR Section 9.1.4.3 and WCNOC-4. The analysis considers a fuel cask drop from a vertical or a tipped position into the pit and does not take credit for the pit covers. Workers are required to wear flotation devices or safety harnesses when working within six feet of the cask loading pit. Therefore, the proposed activity will not create a potential for, or have influence on, equipment or parameters that may cause any new credible accidents.

There are no malfunctions of equipment important to safety which may be directly or indirectly associated with the proposed activity or the design change. As stated in 1.a above, there is no design requirement for the covers and the removal of the covers does not invalidate the conclusion regarding the analysis presented in USAR Section 9.1.3.4 or WCNOC-4.

The proposed activity does not affect acceptance limits as defined in the Technical Specifications or other licensing basis documents since the cask loading pit covers are not considered a part of the bases for acceptance limits.

Safety Evaluation: 59 1998-0084 **Revision:** 1

Changes to the Demineralizer Water Storage and Transfer System Description

Description:

Configuration Change Package (CCP) 07859 changes the Demineralizer Water Storage and Transfer (AN) System Description, drawing M-10AN, "Demineralizer Water Storage/Transfer System," and the Updated Safety Analysis Report (USAR) to address discrepancies and clarify the operation of the system. These are document changes only. These changes to the AN System Description are minor changes that clarifies the general description, system operation and component description section to reflect the way the system is operated to supply demineralized water to plant components and systems.

The following USAR sections are being changed:

Section 9.2.3.2.1, paragraph 1, change the third sentence to read: Check valves are provided to preclude backflow from the demineralized water transfer system to the demineralized water storage tank, assuring that contamination of the source is precluded.

Section 9.2.3.2.3, paragraph 5, change the first sentence to read: The supply of demineralized water to the demineralized water storage tank can be initiated manually or automatically controlled by the actuation of tank level switches which cycle pumps in the demineralized water makeup system.

The description changes do not affect the design bases function of the Demineralized Water Storage and Transfer System, nor the performance activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

Section 9.2.3.2.1, paragraph 1 last sentence states, "Redundant check valves are provided to preclude backflow from the DWSTS to the DWMS, assuring that contamination of the source is precluded."

The word "redundant" is being removed to accurately describe the plant's as-built configuration. Plant configuration has no check valves located in the supply piping between the demineralized water storage, transfer system (DWSTS) and the demineralized makeup water system (DWMS). The existing configuration has the check valves located down stream of the demineralized water transfer pumps which precludes any contamination from back flowing into the demineralized water storage tank (DWST).

The outlet isolation valves for the makeup water demineralizer trains precludes the back flow of contamination from DWST to the DWMS. These spring closed valves isolate the makeup to the DWST until a train has been rinsed to the required water quality specified by Chemistry. If the DWST became contaminated and back flowed into the DWMS, the contamination would be rinsed out prior to the train being placed in operation. Therefore, the importance of preventing contamination of the water source should be focused on the DWST not the DWMS.

Redundant check valves are installed in the majority of the piping branches downstream of the demineralized water transfer pump's check valves. In the branch lines without a redundant check valve, back flow is precluded by manual valves, solenoid valves, or control valves that isolate the branch line.

Changing the descriptions for the check valves will not affect the design function of the system.

USAR Section 9.2.3.2.3, paragraph 5 first sentence states, "The supply of demineralized water to the demineralized water storage tank is automatically controlled by the actuation of tank level switches which cycle pumps in the demineralized water makeup system (Section 9.2.3).

Safety Summary:

The control room monitors the water level in the Demineralized Water Storage Tank and initiates makeup to the tank at their discretion. The control room prefers to manually control the water level in the tank to ensure that the demineralized water chemistry is maintained. All system protective alarms and switches remain active for the protection of the system's equipment. This change clarifies the preferred method of controlling the water level in the demineralized water storage tank.

The manual Demineralized Water (DW) make up to the Demineralized Water Storage Tank has always been a part of the design. The automatic DW make up design aspect was provided as a convenience to maintain the level in Demineralized Water Storage Tank.

The automatic or manual DW make up does not affect any equipment important to safety or design basis accident.

The USAR was updated to encompass the full design capabilities of the Demineralizer Water Storage and Transfer System. No other USAR descriptions or conclusions would change due to this change.

Since there are no physical changes and the design basis function is not changed, no credible accidents that could be created are identified.

Since the proposed changes would not affect the system's failure modes, controls on activity performance, the level of qualification, or the effects on equipment important to safety, no credible malfunctions of equipment important to safety are identified.

Since the proposed changes would not affect the system's failure modes, controls on activity performance, the level of qualification, or the effects on equipment important to safety, no credible malfunctions of equipment important to safety are identified.

Safety Evaluation: 59 1998-0085 **Revision:** 0

USAR Revision to Update Terms and Correct Typographical Errors

Description:

This change revises portions of Section 2.3.3 of the Updated Safety Analysis Report (USAR) per Performance Improvement Request (PIR) 97-1936 as follows:

1) Revise typographical error on page 2.3-40 (date of fuel load)
2) Revise Table Number on page 2.3-42, first paragraph (Table 2.3-49)
3) Name the shed that houses the fiber optic equipment as the "Communications Shed" on page 2.3-42
4) Revise the distance and direction of the rain gage from the meteorological shed on page 2.3-42
5) Number the first paragraph under Section 2.3.3.4 as 2.3.3.4.1 "Phase 1 Instruments"
6) Revise Table 2.3-48 for the type of temperature sensor (RTD) for the accuracy of the RTD (0.3°C) and for the accuracy of the Reference Temperature Transmitters (0.1°C)
7) Revise Table 2.3-49 to revise the distance and direction of the meteorological shed from the tower, to revise the dimension of the shed, to revise the distance and direction of the rain gage from the shed, and to remove references to the mechanical weather station as it is no longer in use.

Safety Summary:

There are no expected effects of these USAR changes on any component, system or piece of plant equipment.

There are no procedures, activities or any other actions (e.g. administrative controls, system operation) which will be impacted by these USAR Changes.
These USAR changes only deal with typographical errors, correct distance and direction statements on existing equipment, to revise a table to accurately reflect the equipment that we already have and to remove equipment that we do not have. There are no field changes as a result of any of the USAR changes.

There are no accident scenarios within the USAR which need to be reviewed for impact by these USAR changes. The rationale behind this is based on the fact that these changes only deal with typographical errors, correct distance and direction statements on existing equipment, to revise two tables to accurately reflect the equipment that we already have. Again there are no field changes as a result of any of the USAR changes.

There are no credible accident scenarios that these USAR changes could create. The rationale behind this is based on the fact that these changes only deal with typographical errors, correct distance and direction statements on existing equipment, and revise two tables to accurately reflect the equipment that that is in place. There are no field changes as a result of any of the USAR changes.

These USAR changes will not affect in any manner any failure mechanism or mode for any safety related components of any plant system. These are document changes only. No field work will be done.

There is no impact on any acceptance limit for the technical specification bases. These are document changes only in the USAR, and no field work is involved. These changes only deal with typographical errors, correct distance and direction statements on existing equipment and to revise two tables to accurately reflect the equipment that we already in use.

Safety Evaluation: 59 1998-0089 **Revision:** 0

Pre-Fabricated Building Added to the New Radwaste Storage Building Slab

Description:

A prefabricated building has been added on top of the new Radwaste Storage Slab to protect the items stored on the slab from normal weather conditions, such as rain, wind, and snow. It is not designed to protect the items on the slab from extreme weather conditions, such as significant earthquakes and tornadoes. Also added with the building is a bridge crane for moving items around the slab. In addition, the following Updated Safety Analysis Report (USAR) figures are being revised to show the addition of the new Radwaste Storage Building:

This Unreviewed Safety Question Determination (USQD) is for the addition of the physical building only. Separate USQDs (59 92-0072, 59 92-0096, 59 93-0209, 59 94-0048, and 59 94-0082) have already been approved for the slab, electrical and mechanical interfaces, and the storage of Radwaste in the area. The only effect that the addition of the Radwaste Building has on the previous USQDs is to provide greater protection against the natural elements (wind, rain, etc.) for all items on or directly above the slab. Therefore, the USQDs listed above are all still valid. This USQD was written in response to Performance Improvement Request (PIR) 98-0624, which was written during Self Assessment (SEL) 97-044.

Safety Summary:

The addition of the new Radwaste Storage Building requires revision to the various USAR figures to show the building on site plans. Other changes to the USAR regarding the slab, electrical and mechanical interfaces, and the storage of Radwaste in the area have already been made under previous revisions to the USAR. There are no other items from this change which will make information in the USAR no longer true or accurate.

The area where the building is located is separated from all safety related structures by the original Radwaste Storage Building and the Waste Bale Drumming Area. The building contains no safety related equipment. Therefore, none of the design basis accidents in USAR Chapters 2, 3, 6, 9 or 15 would be affected by this change to the USAR.

Because the area where the building is located, it is separated from all safety related structures and the building contains no safety related equipment, there are no types of credible accidents that could be created as a result of this change.

Because of the area where the building is located is separated from all safety related structures and the building contains no safety related equipment, it is impossible for the addition of the building to affect equipment important to safety. Therefore, there are no credible malfunctions of equipment important to safety that could be created as a result of this change.

Because the building itself is non-safety related and is not related to any plant systems contained in the technical specifications, there are no acceptance limits which could be affected.

Safety Evaluation: 59 1998-0090 **Revision:** 0

Change Relating to Emergency Diesel Starting

Description:

The proposed activity involves a text change to the fourth paragraph on Updated Safety Analysis Report (USAR) to clarify a statement concerning periodic diesel generator tests. The USAR text currently reads; "During periodic diesel generator tests, subsequent to diesel start and synchronization to the preferred system, a switch in the control room allows parallel operation with the preferred system". Clarification is needed, as identified in Performance Improvement Request (PIR) 98-0810, because the "switch" under discussion is the Unit-Parallel Switch which must be operated prior to (rather than subsequent to) synchronization in preparation for synchronizing the diesel generator to another system. Diesel generator control logic is designed such that speed control to enable establishing synchronization is not available until after this switch has been operated to the 'parallel' position. Therefore, the text is being changed to read; "During periodic diesel generator tests, subsequent to diesel start and prior to synchronization to the preferred system, a switch in the control room allows parallel operation with the preferred system". The USAR change is made to provide clear and concise statements with respect to periodic testing of the diesel generators, consistent with approved plant design.

Safety Summary:

The clarification of USAR text has no effect on any procedures, activities, administrative controls, or sequences of plant operations because the clarification is consistent with existing test procedures and current plant design. The correction of USAR text does not constitute a test or experiment.

The proposed change is textual only, does not involve any hardware changes, nor impact the manner in which the equipment is designed, operated, or maintained. Therefore, this change does not create any new credible accidents.

The change of USAR text to correct a misleading statement does not affect, directly or indirectly, any malfunctions of equipment important to safety. The switch under discussion is operated only during testing. The emergency mode operation of the diesel generators is not affected by the change.

There are no acceptance limits associated with clarification of the USAR text. Operation of the Unit-Parallel Switch, as discussed in the revised text, allows testing of the diesel generators in accordance with Technical Specification Surveillance Requirement 4.8.1.1.2.a.5. This surveillance requirement cannot be performed unless the switch is operated as described in the text change.

Safety Evaluation: 59 1998-0093 **Revision:** 0

Correction of Radiation Zone for Radiation Monitor

Description:

Updated Safety Analysis Report (USAR) Table 12.3-2 and USAR Figure 12.3-2 Sheet-2, "Radiation Zones for Normal Operation," do not agree in regards to the radiation zone designation and room location for Area Radiation Monitor (ARM) 0-SD-RE-29.

USAR Table 12.3-2 incorrectly identifies ARM 0-SD-RE-29 as being located in the hot machine shop (room 1332) and lists the designated radiation zone as 'B'. The actual location of this ARM is in room 1333, which is the decontamination room and designated a zone 'C' radiation zone per USAR Figure 12.3-2 sheet-2 (see partial sketch below). Also this ARM location is not presently shown on this USAR figure.

The criteria for radiation zones is listed on USAR Figure 12.3-2 sheet-1. The Area Radiation Monitoring system has no safety related functions. It serves to warn personnel and plant operators of increasing or abnormal radiation levels in the plant (Reference USAR Section 12.3.4.1).

The decontamination equipment located in room 1333 has been abandoned in place since Refuel 1. Field walk down with a Health Physics supervisor, confirmed that Room 1333 can be designated as a radiation zone 'B'. The changes made to USAR Table 12.3-2 and USAR Figure 12.3-2 sheet-1 will not increase the risk to personnel exposure or over exposure.

Rooms 1332, 1333 and 1334 are all used in the same manner during plant work activities. Rooms 1332 and 1334 are designated radiation zone 'B' criteria with ARM 0-SD-RE-31 being located in room 1334, hot instrument shop.

The resolution to this conflicting USAR information is the following:

1) Change USAR Figure 12.3-2 sheet-2 to show room 1333 as a radiation zone 'B'. This is consistent with today's plant usage.

2) Add the location of 0-SD-RE-29 to the southwest corner of room 1333 for this USAR figure.

3) Correct USAR Table 12.3-2 to show ARM 0-SD-RE-29 being located in the "hot machine shop / decontamination room". The set points in this table will remain as-is for this ARM since they are presently applicable to a zone 'B' criteria. Also no change is require to the plant set point document which has the criteria for a radiation zone 'B' listed.

The expected results are both USAR documents, Table 12.3-2 and Figure 12.3-2 Sheet-2 will agree in regards to the radiation zone designation and room designation for ARM 0-SD-RE-29. USAR Figure 12.3-2 Sheet-2 will require radiation zone designation correction for room 1333 from 'C' to 'B' and show the location for ARM 0-SD-RE-29. USAR Table 12.3-2 will require the location to be corrected for ARM 0-SD-RE-29. The correct location is the Hot Machine Shop/Decontamination room.

Safety Summary:

There are no design basis accidents discussed or referenced in the USAR Chapters 2, 3, 6, 9, or 15 impacted by the proposed activity.
There are no credible accidents that the proposed activity could create.

There are no credible malfunctions of equipment important to safety which may be directly or indirectly affected by the proposed activity.

There are no acceptance limits which are contained in the basis for the technical that could be affected by the proposed activity.

Safety Evaluation: 59 1998-0117 **Revision:** 0

Hydrostatic Testing of Fire Hose

Description:

This Updated Safety Analysis Report (USAR) change is corrective action for Performance Improvement Request (PIR) 97-2281 which addresses discrepancies between Wolf Creek's policy for hydrostatically testing fire hose and the requirements in Appendix 9.5E of the USAR. The USAR currently states: "Fire hose shall be hydrostatically tested at a pressure of 150 psi or 50 psi above maximum fire main operating pressure, whichever is greater. Hose stored in outside hose houses shall be tested annually. Interior standpipe hose shall be tested every three years." Wolf Creek is in the process of implementing a program where the hose rack hoses are replaced every 5 years as allotted by NFPA 14-1976 in lieu of hydro testing. This practice eliminates the need to perform manpower intensive service tests. Therefore, the USAR will be changed as follows:

1. Table 9.5E-1, Sheet 4; item III. E. of 10CFR50 Appendix R reads as follows:

III. E. Hydrostatic Hose Tests

Fire hose shall be hydrostatically tested at a pressure of 150 psi or 50 psi above maximum fire main operating pressure, whichever is greater. Hose stored in outside hose houses shall be tested annually. Interior standpipe hose shall be tested every three years. The Wolf Creek response to this item will be changed from "Complies" to "Complies except interior hose is tested per NFPA 14-1976 for hose frequency."

2. Appendix 9.5A (APCSB 9.5-1 Appendix A) Sheet 52; makes an editorial change in the second paragraph of column two by changing NFPA 14-1974 to NFPA 14-1976.

Safety Summary:

The above USAR changes are minor changes that clarify the general description, system operation, component description and administrative statements to reflect the way the fire protection system and program is operated. The changes do not affect the design basis function of the system. Replacement of hose racks on a 5 year frequency in lieu of hydrostatically testing the hose racks will not effect Wolf Creek's ability to maintain fire safe shutdown. Therefore, there are no design basis accidents impacted by this change.

Since there are no physical changes and the design basis function of the system is not affected by this change, no new types of accidents not previously analyzed could be created.

Since the proposed change would not affect the system's failure mode, the systems design function, the level of qualification, or equipment important to safety, no credible malfunctions of equipment important to safety are identified.

There are no acceptance limits associated with these editorial/administrative changes, therefore, no acceptance limits could be affected.

Safety Evaluation: 59 1998-0131 **Revision:** 0

Changes to Testing Descriptions Resulting From the USAR Fidelity Review

Description:

USAR Section 6.2.6.1.1 states that the steam generators are considered an extension to the containment boundary and the secondary side of the steam generators is not vented to containment atmosphere during the Integrated Leak Rate Testing (ILRT). The USAR describes that the secondary side is vented outside containment during the stabilization period. However, the actual venting does not occur at this time. Therefore, the USAR is being revised to reflect that the venting does occur during the Type A (ILRT) test. This change has no effect on other information in the USAR.

USAR Section 6.2.6, "Containment Leakage Testing," states that during Type B, Local Leak Rate Testing (LLRT) clamps are installed on the inside closure device on the equipment and personnel hatches to ensure seating during testing. The USAR is being revised to reflect that these test clamps are installed on the inside closure device to support the closure device during the testing and that clamps are not installed on the equipment hatch. This change only clarifies the information in the USAR on the use of the test clamps.

Safety Summary:

The changes to the USAR do not affect any of the design basis accidents discussed or referenced in the USAR. No plant structures, systems, components or equipment are affected by this change. Now new accidents are created since the steam generators are vented.

Since the changes do not affect the operation or function of any systems, structure, or component (SSC) as described in the USAR, no credible malfunction of equipment important to safety are identified.

These changes to the USAR do not change the acceptance limits that are contained in the bases for the technical specification. The changes only clarify the use of clamps during the Local Leak Rate Testing of the personnel hatches for support and the states the secondary side is vented during the ILRT. Therefore, no acceptance limits are affected by this change.

Safety Evaluation: 59 1998-0132 **Revision:** 0

Changes to Government Organization Titles and Equipment Information Identified During the USAR Fidelity Review

Description:

This change, 1) corrects the Updated Safety Analysis Report (USAR) to reflect the current name of the governmental organization that controls primary calibration standards, 2) clarifies that some portable instrumentation is stored in the radiologically controlled area (RCA), and 3) corrects the list of equipment shared by Health Physics and Radiochemistry to reflect current usage.

These changes to USAR Chapter 11, "Radioactive Waste Management," and USAR Chapter 12, "Radiation Protection," replaces "National Bureau of Standards" with "National Institute of Standards and Technology." In Section 12.5.2.1 the storage location of portable instrumentation in the RCA is clarified and the description of equipment and instrumentation shared jointly by Health Physics and Radiochemistry is revised to reflect the equipment currently shared by the two groups. These changes are based on comments documented in Performance Improvement Request (PIR) 98-1713 which was initiated during the USAR Fidelity Review.

Safety Summary:

The USAR changes do not affect any system, structure, or component (SSC) nor do they change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

These USAR changes do not impact any procedures, activities, administrative controls, or sequence of plant operations nor are any SSCs impacted. No requirements outlined in the USAR are revised by these changes. No other USAR descriptions or conclusions will changed or be made untrue as a result of these changes. No test or experiments are involved with these changes. There are no design basis accidents impacted by this change.

These USAR changes provide the correct name of the governmental organization that controls primary calibration standards, clarifies the location of portable instrumentation used by Health Physics and corrects the list of equipment shared by Health physics and Radiochemistry. These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no credible Accidents that could be created and no acceptance limits are identified that could be affected by these changes.

Safety Evaluation: 59 1998-0133 **Revision:** 0

Administrative Corrections to Health Physics Information Resulting From the USAR Fidelity Review

Description:

This change to the Updated Safety Analysis Report (USAR) revises USAR Chapter 12, "Radiation Protection," USAR Table 3.11(B)-1, "Plant Environmental Normal Conditions," and USAR Table 3.11(B)-2, "Environmental Qualification Parameters for SNUPPS NUREG-0588 Review (LOCA, MSLB and HELB)," to reflect current organizational titles; changes to the designation of site buildings; changes in use and nomenclature of certain rooms used by Health Physics; and clarification of the use and description of Health Physics facilities at Wolf Creek. . These changes are based on comments documented in Performance Improvement Request (PIR) 98-1713 which was initiated during the USAR Fidelity Review. The specific changes are:

Changes reflecting that the individual in charge of the Radiation Protection program at Wolf Creek is the Manger Chemistry/Radiation Protection. The positions of Superintendent Radiation Protection and Superintendent Chemistry were combined into the new position of Manager Chemistry/Radiation Protection. This change was part of Amendment 115 to the Wolf Creek Technical Specifications.

Changes that clarify the description of Health Physics facilities. The specific changes are that the sinks in the toilet area are not stainless steel and the female clean showers are located in the respective toilet area.

Changes that clarify how modesty garments are now used at Wolf Creek.

Changes that reflect the current use of rooms utilized by Health Physics in the Control Building.

Changes to building designations.

Safety Summary:

These USAR changes provide a consistent and clear description of the manner in which the Health Physics facilities at Wolf Creek Generating Station (WCGS) are used. These changes also reflect previously approved changes in organizational structure. The USAR changes do not affect any system, structure or component (SSC), nor do they change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

These USAR changes do not impact any procedures, activities, administrative controls, or sequence of plant operations nor are any plant structures, systems, components or equipment impacted. No requirements outlined elsewhere in the USAR are revised by these changes. No other USAR descriptions or conclusions will change or be made untrue as a result of these changes. No test or experiments are involved with these changes. No acceptance limits are identified that could be affected.

Safety Evaluation: 59 1998-0134 **Revision:** 0

Reduction of Snubber Population

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the USAR sections which are affected by Design Change Package (DCP) 04697. The primary objective of this modification is to reduce the snubber population on piping systems inside Containment. The reduction of the snubber population will reduce the risk of an extended outage duration. In addition, a reduction in cost and personnel exposure will be realized due to the decrease in the required snubber test population.

The following specific systems have been re-analyzed:
· Accumulator Line, Loop 1 (Stress problem P-234)
· Accumulator Line, Loop 3 (Stress problem P-236)
· Accumulator Line, Loop 4 (Stress problem P-235)
· Seal Water Injection Line, Loop 1 (Stress problem P-250)
· Seal Water Injection Line, Loop 3 (Stress problem P-251)
· Seal Water Injection Line, Loop 4 (Stress problem P-249)
· Pressurizer Surge Line, Loop 4 (Stress problem P-257)

Safety Summary:

The reanalysis was performed using ASME Code Case N-411. Wolf Creek Generating Station (WCGS) committed three site enveloped response spectra, which are anchored at 0.20G for a safe shutdown earthquake (SSE) and 0.12G for operationing basis earthquake (OBE) ground motions, and the guidance provided in Generic Letter 87-11 (GL 87-11), "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements." Code Case N-411 allows the use of higher damping values at lower system frequencies.

The primary stress evaluation for the design, normal, upset, and faulted conditions shows the stresses are below the allowable limits established in the ASME Code, subsection NB. The Class 1 auxiliary piping stresses and fatigue usage factors are in conformance with the requirements of the Code for the fatigue damage evaluation performed under all normal, upset, and test conditions. The auxiliary piping evaluated for the faulted condition shows that the stresses for the Class 1 auxiliary lines are within the allowable faulted limits of the Code for the postulated auxiliary line branch nozzle breaks and main steam/main feedwater line breaks, and for the applicable seismic loads.

The pipe supports on the Class 1 auxiliary lines have been evaluated under the appropriate loading conditions and found to be acceptable in accordance with the criteria specified in the ASME Code, subsection NF.

The reactor coolant loop branch nozzles have been evaluated for all appropriate loading conditions. The results of the evaluations show that the nozzles remain within the appropriate allowable stress criteria of the ASME Code for the prescribed loading conditions.

Accidents identified in Chapters 2, 3, 6, 9 or 15 of the USAR are not impacted by this change.

New analysis has been performed for the Class I piping systems that meet the requirements of ASME Code section III, subsection NB and NF. The re-analysis also meets all requirements of the USAR and other design basis documents. Therefore, the proposed design changes will not create any credible accidents.

The re-analysis of Class I piping systems comply with industry criteria, Branch Technical position MEB3-1 Revision 2, and GL 87-11. This provides a basis for safety requirements of components designed in accordance with the requirements of Section III, subsections NB and NF of the ASME code. Therefore, the proposed change will not create any credible malfunction to equipment associated with the systems discussed above.

Acceptance limits as identified in the USAR or other documents are not affected by this change.

Safety Evaluation: 59 1998-0135 **Revision:** 0

Control Building HVAC Description Clarification

Description:

This Updated Safety Analysis Report (USAR) change revises USAR Section 9.4.1.2.2, "System Description," to clarify equipment nomenclature and make the USAR consistent with Piping & Instrument Diagrams (P&ID's), configuration control data, and specifications. These changes are based on comments documented in Performance Improvement Request (PIR) 98-2542 which was initiated during the USAR Fidelity Review.

(i) The Component Description for Control Room and Class IE electrical equipment air-conditioning units as shown on page 9.4-5 refers to "high efficiency pre-filters". This statement is to be revised to read "85% efficiency filters" to be consistent with drawing M-12GK01, Specification M-622.1, and Vendor Instruction Manual M-622.1-00061.

(ii) The Component Description for Nonessential Air Handling Units on page 9.4-5 refers to the "access control air-conditioning unit" in two places. These statements are to be revised to read "access control fan coil unit" to be consistent with drawing M-12GK03, Specification M-611 and Vendor Instruction Manual M-611-00137.

(iii) The Component Description for Nonessential Air Handling Units on page 9.4-5 refers to the "access control filtration unit". This statement is to be revised to read "access control exhaust filter adsorber unit" to be consistent with drawing M-12GK02, Specification M-621 and Vendor Instruction Manual M-621-00036.

(iv) The Component Description for Safety-Related Fans on page 9.4-6 refers to "the control room filtration system fans, and the control room pressurization system fans". The word "system" is to be deleted from both places in this statement to be consistent with drawings M-12GK01 & M-12GK03 and the configuration control description.

(v) The Component Description for Flow Control Dampers on page 9.4-7 is to be revised to include "single-blade-type" dampers to be consistent with Specification M-627A.

Safety Summary:

These USAR changes do not affect any system, structure, or component (SSC), nor do they change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

The activity described above involves changes that reflect the present plant equipment configuration and nomenclature. The changes ensure accuracy and consistency, do not affect other information in the USAR, and do not violate any requirements in the USAR. No tests or experiments are affected or introduced.

The proposed activity is a clarification of the text in the USAR and will not have any affect on equipment or parameters associated with any design basis accident that has been previously described in the USAR.

This activity makes no additional changes to the plant and does not affect any SSC. Therefore, no credible accident that could be created are identified.

This activity does not directly or indirectly affect the function of any equipment important to safety. Therefore, no credible malfunctions of equipment important to safety are identified.

Safety Evaluation: 59 1998-0136 **Revision:** 0

Revision 3 to Procedure AP 21-005

Description:

Revision 3 to procedure AP 21-005, "Control of Pipe Caps and Vent/Drain Flanges," allows for the removal of cap and blind flanges at vent, drain, test point and pressure point connections while any plant system is still in service. This revision also allows for the temporary installation of hardware at these connections to facilitate draining the system. This change provides the means to control these connection activities before the system is removed from service and does not manipulate the connection valve(s). This change applies to those connections that have at least one normally closed connection valve and does not apply to connections that do not contain valves for isolation.

The expected results of the procedure change are that it will shorten system outage time by allowing the setup portion of system drainage activities to occur at the subject connections before removing the system from service. Currently these setup drainage activities are begun after the system has been tagged out of service. The procedure change will allow removal of the caps and blind flanges and installation of the temporary drainage hardware before the system is removed from service. This change will have no adverse affect on system operation and will expedite the outage schedule.

The function of the caps and flanges at these connections is to support housekeeping by containing the system fluid in the event that the connection valve(s) leak by. Removal of the cap and flange will not be performed by the procedure change if the connection valve(s) is not providing isolation. The connection valve(s) support system pressure boundary and define the seismic boundary as shown on the design drawings. Removal of the cap and flange at the desired locations will not compromise the systems pressure boundary or impair the seismic capability of the connection. The temporary hardware installed at the connection, typically a coupling instead of a cap or an orifice flange instead of a blind flange, to facilitate draining or venting the system will be of the same relative mass as that removed thus no significant change in mass to the connection will be introduced. No rigid temporary hard piping from the connection to facilitate draining or venting the system is allowed by the procedure change that could compromise the seismic response of the connection. The use of flexible hoses are used at the connection and these hoses will not alter the seismic response of the connection.

This procedure change affects the Figure drawings in the Updated Safety Analysis Report (USAR) for each of the systems because these drawings reflect that the subject connections are capped or blind flanged. The change will thus make the drawing information in the USAR incorrect when the caps and flanges are removed. These USAR Figures will not be changed because the affects of the change being evaluated will be in effect only for a short time duration. These changes will occur prior to removing the system from service before other plant procedures take control. Additionally, because the USAR Figure drawings typically reflect normal plant system configuration when the plant is at power, no change to these drawings is desired to reflect the temporary configuration of the subject connections. The procedure already contains measures to ensure the permanent plant cap and flanges are restored when the draining and venting activities are completed, thus these control measures are only mentioned here for completeness.

Safety Summary:

A review of the accidents in USAR Chapters 2, 3, 6, 9, and 15 concluded that no accidents were impacted by the subject change. The assumptions and conditions assumed prior to, during, and after, these accidents are not changed by the proposed activity.

No reduction in double barrier protection has been allowed by this change. The ASME code pressure boundary and seismic boundary only goes out to the first valve on a drain or vent connection regards of the configuration downstream of this boundary valve (another valve, cap or flange). No ASME code double isolation valves were involved with this change. This change does not impact the quality or structural integrity of the safety related portion of the system. Removal of the non safety related cap or flange, installed for housekeeping concerns, (evaluated for II/I concerns only) by definition and as the evaluation concludes has no impact on the safety related barrier performance of the first valve. No credit is taken for the second valve, cap or flange for barrier protection. The ASME envelope defines the barrier protection limit and the likelihood of leakage through this envelope boundary whether ASME pipe or ASME valve is assigned the same probability. Thus the question becomes considering leakage through ASME pressure boundary that is controlled by the Technical Specification. This change does not change or adversely affect this boundary.

All leakage from radioactive or potentially radioactive systems, drain into a radwaste drainage collection system. Therefore, it is concluded that no new or credible accidents are created by the subject change. It lacks the potential to effect or challenge the plants design in an adverse or unique manner nor will it cause a new or different type of accident.

This change does not alter the pressure boundary or seismic integrity of the plants piping. The seismic integrity of the connection piping and supports will remain within the acceptable limits of the design. They will not be introduced to a condition outside of their design capability, that could cause a malfunction of the piping system considered to be, in a broad sense, equipment important to safety. No credible malfunctions to any equipment has been created by the subject change.

The acceptance limits contained in the licensing basis documents have not been changed by the proposed activity. Therefore, the margin of safety is not impacted by this change.

Safety Evaluation: 59 1998-0138 **Revision:** 0

Updated Safety Analysis Report Changes

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the Updated Safety Analysis Report (USAR) that are changes made to achieve consistency and enhance clarification compared with other USAR Sections. This change updates the USAR by providing information not previously explicitly provided, such as the maximum calculated Reactor Coolant System (RCS) pressure compared to the design value, and corrects RCS pressure response information, currently based on the RETRAN Statistical Core Design model instead of the Pressure Evaluation Model. In addition, this change corrects references to figures currently in the USAR.

Safety Summary:

The proposed activity does not change any administrative control which would reduce the effectiveness of existing programs, reduce the qualification of WCNOC personnel, nor does it affect any systems, structures, and components. The proposed activity does not change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS). No credible malfunctions of equipment important to safety are identified.

The USAR changes provide a consistent presentation, within the major subsections, of the transient RCS pressure response information generated using the RETRAN Pressure Evaluation model for accident analyses where the RETRAN Statistical Core Design model or the Revised Thermal Design Procedure (RTDP) methodology was used as a statistical approach to DNB, in addition to changes made to enhance clarification or correct references. No other procedures, descriptions, or conclusions would change or be untrue due to this change.

Since the proposed changes to the USAR merely reflect the presentation of the transient RCS pressure responses generated using the RETRAN Pressure Evaluation model that are compared to the design basis pressure value acceptance limit. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1998-0139 **Revision:** 0

Clarification of the Health Physics ALARA Group Function Identified During the USAR Fidelity Review

Description:

This change to the Updated Safety Analysis Report (USAR) Chapter 12, "Radiation Protection," clarifies the function of the Health Physics ALARA Group. This change is based on comments documented in Performance Improvement Request (PIR) 98-1713 which was initiated during the USAR Fidelity Review.

USAR Section 12.1.1.2, "Health Physics," second paragraph, first sentence, is changed to read "The Health Physics ALARA group is responsible for incorporating applicable regulatory criteria into the station ALARA program and providing general Health Physics support."

Safety Summary:

This USAR change clarifies that the Health Physics ALARA Group is responsible for incorporating applicable regulatory criteria into the station ALARA Program. The Licensing Group and the Nuclear Safety Engineering Group analyze changes in regulatory criteria and provide that information to the Manager Chemistry/Radiation Protection if it is deemed to have potential impact on the Health Physics program.

This USAR change does not affect any system, structure, or component, nor does it change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

This USAR change does not impact any procedures, activities, administrative controls, or sequence of plant operations nor are any plant structures, systems, components or equipment impacted. No requirements outlined in the USAR are revised by this change. No other USAR descriptions or conclusions will change or be made untrue as a result of this change.

No test or experiments are involved with this change.

No design basis accident is identified for review.

No credible accidents that could be created are identified.

No credible malfunctions of equipment important to safety are identified.

No acceptance limits are identified that could be affected.

Safety Evaluation: 59 1998-0145 **Revision:** 1

USAR Change to Reflect As-built Configuration of the Waste Water Treatment Sump Pump and Control Switch

Description:

Revision 1 to Unreviewed Safety Question Determination (USQD) 98-0145 provides clarification for the conclusions in this USQD. The control switch for the Waste Water Treatment System Sump Pump (PWT01) is currently shown as three components (by switch functions (WT-LSH-0001, WT-LSL-0001 AND WT-LSHH-0005) on Updated Safety Analysis Report (USAR) Figure 9.2-25-01, "Waste Water Treatment Facility." The switch serves three functions; however, there is only one physical component. USAR Figure 9.2-25-01 is being revised to show one component with a unique asset number (WTLS-15) as detailed in Drawing E-13WT10 (Waste Water Treatment Facility Sump Pump PWT01). This change will reflect the current design. The sump/sump pump symbol as shown on USAR Fig. 9.2-25-01 is also being changed to indicate the current configuration of the sump and sump pump motor. The sump is open and the motor is located outside the sump while the drawing shows the sump as covered and the motor contained within the sump. These are configuration changes only and do not affect the function or settings of the component.

Safety Summary:

The proposed USAR changes are based on comments contained in PIR 98-1681. The USAR changes do not affect any SSC nor do they change the performance of activities that are important to the safe and reliable operation of WCGS.

The proposed activity involves a configuration change only that reflects the present plant design and configuration. The change ensures accuracy and consistency, does not make any information in the USAR no longer true or accurate and does not violate any requirements in the USAR. No tests or experiments are affected or introduced. There is no additional impact on the performance of plant activities nor does the activity affect any SSC. Therefore, no design basis accident is identified for review.

No credible accidents that could be created are identified because the Waste Water Treatment Facility is not involved in any accidents.

No credible malfunctions of equipment important to safety are identified because the Waste Water Treatment Facility is non-safety related.

There are no acceptance limits identified that could be affected. Therefore, the margin of safety is not affected by this change.

Safety Evaluation: 59 1998-0149 **Revision:** 0

Change in Setpoint for Liquid Radwaste Monitor

Description:

This revision to procedure AP 07B-003, "Offsite Dose Calculation Manual," changes the Alert Alarm setpoint, for release points that have dilution, from one order of magnitude below the High Alarm/Trip setpoint to 80% of the High Alarm/Trip setpoint.

Step 2.4.4 is revised to state, "The Alarm/Trip Setpoints for the Liquid Effluent Radiation Monitors are based on instantaneous concentration limits of 10 CFR 20, Appendix B, Table II, Column 2 applied at the boundary of the restricted area. Specifically, the High Alarm Setpoint will correspond to the 10 CFR Part 20 limits at the Boundary of the restricted area; the Alert Alarm Setpoint is set to 80% of the High Alarm/Trip Setpoint." The background activity of some of the radiation monitors are too high to allow for the maximum discharge flowrate when the Alert Alarm setpoint is set to one order of magnitude below the High Alarm/Trip setpoint. Before the release is initiated the Alert Alarm annunciates in the Control Room, which is not consistent with the philosophy of sequential alarming of a radiation monitor. To release water from a radiation monitor with a high background activity requires a reduction in the release flowrate, which is less efficient. By setting the Alert Alarm to 80% of the High Alarm/Trip setpoint the maximum discharge flowrate may be accomplished without annunciator alarms in the Control Room.

By setting the Alert Alarm to 80% of the High Alarm/Trip setpoint there is less time for the High Alarm/Trip setpoint to annunciate once the Alert Alarm is reached. Updated Safety Analysis Report (USAR) Section 11.5.2.1.8 states, "the Alert Alarm is administratively established at a point sufficiently below the High alarm so as to provide additional assurance that Technical Specification limits are not exceeded." The High Alarm/Trip setpoint is set to conform to the regulatory requirements of 10 CFR 20 and the release is isolated automatically if the setpoint is exceeded.

Safety Summary:

USAR Table 11.5-2," Liquid Effluent Radioactivity Monitors," footnote (3), states "The alert alarm is set to one order of magnitude below the High Alarm/Trip Setpoint for release points that have dilution and up to the High Alarm value for those without dilution." The footnote in the Table should provide direction to set the alert alarm to 80% of the High Alarm/Trip setpoint. There is no other information in the USAR that is affected by this change.

There are no design basis accidents identified in USAR Chapters 2,3,6,9, or 15 affected by the proposed change. Increasing the Alert Alarm to 80% of the High Alarm/Trip setpoint does not impact the isolation function of the release, which protects the health and safety of the public by conforming to 10 CFR 20 limits. The dose calculations for postulated accidents are not affected by the proposed change.

No credible accidents could be created by changing the Alert Alarm setpoint. The operation, testing, design function and maintenance of the radiation monitors remains the same and the 10 CFR 20 limits for radiological releases of liquids are still applicable.

Changing the Alert Alarm setpoint does not impact the isolation function of the release, therefore no credible malfunctions of equipment may be induced.

Since 10 CFR 20 release limits for liquids are protected by the isolation function, the acceptance limits are not affected.

Since no acceptance limits were identified that could be affected, the margin of safety is not affected by this change.

Safety Evaluation: 59 1998-0150 **Revision:** 0

Containment Spray pH

Description:

Performance Improvement Request (PIR) 98-2804 describes Updated Safety Analysis Report (USAR) inaccuracies in defining the pH of the Containment Spray system during the injection phase. A review of the USAR and supporting calculation GS-M-004, "Hydrogen Generation Analysis," will result in the following USAR changes:

In USAR Section 6.2.5, "Combustible Gas Control in Containment," Step 6.2.5.2.3c, second paragraph, the second line is changed to read, "The containment during the injection phase, is injected with a borate solution adjusted to a pH between 9.0 and 11.0 with sodium hydroxide addition in operation, while a minimum pH of 4.0 could be experienced in one of the spray trains in the event of a single failure in the spray additive subsystem."

In USAR Table 3.11(B)-5, "Containment Spray Requirements," the value for Sprayed Fluid, Injection Phase, Aqueous Solution, pH (max.) is changed from "10.5" to "11.0" and Final Sump Fluid, Aqueous Solution, is changed from" 8.0/9.0" to "(min) 8.5".

Safety Summary:

The changes to Section 6.2.5.2.3c and Table 3.11(B)-5 are editorial in nature and will bring these USAR sections into agreement with other sections and supporting calculations. The changes do not affect the operation or function of any SSC. These USAR changes do not affect the performance of activities necessary for the safe and reliable operation of Wolf Creek Generating Station.(WCGS).

USAR Section 3.11(B)1.2.2, under Containment Spray paragraph states "The NRC Standard Review Plan indicates that single failures should be evaluated to determine the worst case chemical concentrations. The worst case concentrations, resulting from a single failure, are pH = 4.0 and pH = 11.0 ...". Calculation GS-M-004, "Hydrogen Generation Analysis," also supports these values and provides conclusions that support the licensing basis provided in the USAR.

USAR Section 6.1.1.2.1 "Control of pH During a Loss-of-Coolant Accident" states "The resultant basic pH range of 8.5-9.0 ..." This statement is based on containment reaching a chemical equilibrium after chemical addition has ceased. The Wolf Creek Technical Specifications also provide limits for the recirculation phase during chemical addition and post chemical addition. The Technical Specifications (Bases for Sections 3.0 and 4.0 Limiting Conditions for Operation and Surveillance Requirements) state:

Technical Specification Bases 3/4.5.5 "The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA."
Bases 3/4.6.2.2 "The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA."

In addition Calculation EN-03-W, "18 Month Fuel Cycle Four Boron pH," Revision 2, uses a minimum bounding limit of 8.5 pH for all recirculation conclusions and is the design basis limit for the containment sump in equilibrium during Main Steam Line Break and Loss-Of-Coolant Accidents.

Changing Section 6.2.5.2.3c and Table 3.11(B)-5 will correct USAR inaccuracies thus providing a consistent understanding of plant design between USAR sections. No other USAR descriptions or conclusions are affected by this change.

The changes being reviewed are resultants of Main Steam Line Break (MSLB) or Loss Of Coolant Accident (LOCA), therefore these accidents were reviewed for this change.

The proposed changes address a change in pH values of the containment environment as a result of a MSLB or LOCA. The changes are editorial and are being used to make USAR sections consistent. Therefore the changes being made will not create an additional credible accident.

The changes being proposed define the chemical parameters for the containment environment during a design basis accident and a possible malfunction of the chemical addition of one Containment Spray train. The changes will not increase the likelihood of the malfunction nor will they change its results. No other malfunction can be associated with this change. Therefore the change will not directly or indirectly cause a credible malfunction of equipment important to safety.

The proposed changes are being used to clarify design basis limits in USAR Section 6.2.5.2.3c and USAR Table 3.11(B)-5. These design basis limits are identified in other sections of the USAR and have not been changed. The changes being made will not change the acceptance limits contained in the bases for the technical specifications or the licensing basis documents.

Safety Evaluation: 59 1998-0151 **Revision:** 0

Corrections to Seal Water Flow and Volume Control Tank Pressure Resulting From the USAR Fidelity Review

Description:

This Unreviewed Safety Question Determination evaluates Revision 28 to procedure SYS BB-201, "Reactor Coolant Pump Operation," and the associated changes to the Updated Safety Analysis Report (USAR). Changes are identified below.

Change A: The Seal Water Injection flow specified in SYS BB-201 step 6.1.1.3, is changed from "6 to 13 gpm" to "8 to 13 gpm" to be consistent with operating procedures GEN 00-001, "Mode 5 - Fill and Vent of the RCS," SYS BB-114, "Venting the RCS," SYS BG-120, "Chemical and Volume Control System Startup," and ALR 00-041A, "Seal INJ to RCP Flow Low," all of which require a seal water flow of 8-13 gpm.

USAR Section 9.3.4.2.3.1, last paragraph on Page 9.3-56 is changed to read, "The manual throttle valves in each of the Reactor Coolant Pump (RCP) seal water supply lines are set to provide a flow of 8 to 13 gpm per RCP." This will provide consistency with the operating procedures.

Change B: The pressure range of the Volume Control Tank (VCT) as specified in SYS BB-201, step 5.4 is changed from "between 15 psig and 65 psig" to "between 15 psig and 50 psig".

USAR Section 9.3.4.2.3.1, fifth paragraph on Page 9.3-57, is changed to read, "During this operation, the VCT pressure is maintained at 15 to 50 psig by the pressure control valve in the gaseous vent line."

Change A is consistent with operating procedures GEN 00-001, SYS BB-114, SYS BG-120 and ALR 00-041A all of which require a seal water flow of 8-13 gpm. RCP Vendor Manual M712-00068 states that Normal Operating flow is 8-13 gpm with minimum and maximum flows of 6 and 20 gpm respectively. Westinghouse Project Information Package, Operating Instruction S-2 states "The seal injection flow to the pump (RCP) must be maintained greater than 6 gpm to avoid overheating of the water reaching the #1 seal. The seal injection flow should be maintained below about 13 gpm." The change to "8 to 13 gpm" is still within the range specified by the vendor and provides a greater operational margin to maintain the integrity of the Reactor Coolant Pumps' #1 seals.

Change B is based on an Engineering recommendation to change the upper gas pressure limit for the VCT to 50 psig to prevent gas entrapment. Design Basis M-10BG specifies a VCT maximum design pressure of 75 psig. The effect of the proposed change is to conservatively limit the maximum operating pressure and to increase the operational margin of the VCT while maintaining the operation of the tank fully within the operating range specified by vendor and design documents.

Safety Summary:

The proposed changes will correct the inconsistencies between SYS BB-201 and other plant operating procedures. No requirements outlined in the USAR are revised by these changes. No other USAR descriptions or conclusions will change or be made untrue as a result of these changes. No tests or experiments are involved with these changes.

The proposed changes will correct the inconsistencies between SYS BB-201 and other plant operating procedures. There is no additional impact on the performance of plant activities nor affect on any SSC. Therefore, no design basis accident is identified for review.

The proposed changes will correct the inconsistencies between SYS BB-201 and other plant operating procedures. These changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no credible accidents that could be created are identified.

The proposed changes will correct the inconsistencies between SYS BB-201 and other plant operating procedures. These changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no credible malfunctions of equipment Important to safety are identified.

The proposed changes will correct the inconsistencies between SYS BB-201 and other plant operating procedures. These changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no acceptance limits are identified that could be affected

Safety Evaluation: 59 1998-0152 **Revision:** 0

Updated Safety Analysis Report Clarification to Control Bank Operation

Description:

Currently, the sixth sentence of the first paragraph in the Updated Safety Analysis Report (USAR) Section, "Control Rod Patterns and Reactivity Worth" 4.3.2.5, implies that all the Control Rods (both the Control Banks and the Shutdown Banks) may be controlled automatically or manually. The first sentence in this paragraph states, "The rod cluster control assemblies are designated by function as the control groups and the shutdown groups." The sixth sentence states, "The axial position of the rod cluster control assemblies may be controlled manually or automatically". This USAR change expands this sixth sentence to state that "The Control Banks can be controlled manually or automatically, while the Shutdown Banks are only controlled manually".

The fourth paragraph in USAR Section 7.7.1.2.1, "Rod Control System," states, "The control banks are the only rods that can be manipulated under automatic control". This statement substantiates this USAR change.

Safety Summary:

This USAR change only discusses auto and manual rod control. The next sentence in Section 4.3.2.5 states, "The rod cluster control assemblies are all dropped into the core following actuation of reactor trip signals". The actual reactor trip function is not affected by this change.

This USAR change cannot create any credible accidents. This change only clarifies the description in the USAR, on how the Shutdown Rods are operated.

No equipment is directly or indirectly affected by this USAR change. This change only clarifies the description in the USAR.

This USAR change only discusses the control function of the Shutdown Banks. The tripping function, which is part of Technical Specifications, is not part of this change.

This USAR change only clarifies a description in the USAR. No physical or administrative changes are occurring. This change just clarifies that the Shutdown Rods can only be operated manually. As USAR Section 4.3.2.5 is currently written, it could be interpreted that they can also be automatically controlled. Therefore, the probability of occurrence of an accident previously evaluated in the USAR, is not increased. The radiological consequences of an accident previously evaluated in the USAR, is not increased.

This change only discusses how the Shutdown rods are controlled. This change does not affect the Engineered Safety Feature (ESF) reactor trip function of the Shutdown Rods. Therefore, this change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. The radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR is not increased.

This change is only making a wording change to the USAR. No physical or procedural change is occurring. Therefore, no accident of a different type than any previously evaluated in the USAR, is created. A different type of malfunction of equipment important to safety than previously evaluated in the USAR is not being created.

The accident analysis already assumes the operation as being corrected and no acceptance limits are being affected by this change. Therefore the margin of safety as defined in the basis for any technical specifications is not reduced.

Safety Evaluation: 59 1998-0153 **Revision:** 0

Corrections to the USAR Resulting From the Fidelity Review

Description:

The following changes to the Wolf Creek Updated Safety Analysis Report (USAR) are proposed in response to Performance Improvement Requests initiated by the USAR Fidelity Review team.

1) Revise Section 15.0.3.2, "Plant Characteristics and Initial Conditions Assumed in the Accident Analyses, Initial Conditions," to indicate a pressure uncertainty of ±30 psi and a temperature uncertainty ±4.85°F. The uncertainties are currently indicated to be in the positive direction only (i.e., +30 psi, +4.85°F). The pressure and temperature uncertainties are also applicable in the negative direction and are modeled as such in some analyses (e.g., Calculations AN-96-013, "Cycle 9 Turbine Trip With Reduced TFD," and AN-96-015 "Cycle 9 Loss of Normal Feedwater").

2) Revise USAR Section 15.2.2.1 "Loss of External Electrical Load, Identification of Causes and Accident," to correctly identify the instrument power supply voltage as 120 volts. The USAR currently indicates the voltage supply as 118 volts. No physical change to the power supply is proposed.

3) Revise the relief capacity description in Section 15.2.2.1 to correctly reflect the design basis of the Main Steam Safety Valves (MSSV). It appears that this text was not updated with implementation of the plant rerate, and hence the reference to the MSSV capacity with respect to rated thermal power is misleading. The revised description will reference USAR Section 5.2.2.1, which discusses the basis for the MSSV total relief capacity and is consistent with the discussion in the Technical Specification Bases Section B 3.7.1. No change to the MSSV relief capacity is proposed.

4) Revise USAR Sections 15.2.2.1 and 15.2.3.1, "Turbine Trip, Identification of Causes and Accident," to clarify the basis for why the turbine trip analysis presented in Section 15.2.3 bounds the loss of an external electrical load accident. The analysis of the turbine trip event (Section 15.2.3) clearly bounds the loss of external electrical load event (Section 15.2.2). The basis for this conclusion is clarified in the revised text.

5) Revise USAR text in seven locations to indicate "RETRAN-02" code only and remove specific reference to "Mod 5." The affected pages include 15.2-6, 15.2-12, 15.2-18, 15.2-22, 15.3-2, 15.4-8, and 15.6-2. The Mode 5 designation has not been consistently used in Chapter 15. Deleting the designation is intended to restore consistency to the USAR. This does not represent a change in the version or use of the RETRAN code.

6) Revise Item b in Section 15.2.6.1" Loss of Nonemergency AC Power to Station Auxiliaries, Identification of Causes and Accident Description," to remove reference to the power operated relief valves. The description is being revised to be consistent with the description in Section 15.2.7.1. This does not represent a change to the analysis model.

7) Revise Section 15.2.6.2, "Analysis of Effects and Consequences," Item d to clarify the components of the delay before AFW delivery and to indicate the assumed AFW system single failure assumptions to clarify the existing analysis assumptions. Table 15.2-1, "Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal by the Secondary System," Sheet 3, was also revised to be consistent with the stated single failure assumptions.

8) Add a description of the MSSV modeling assumptions for the loss of AC power analysis as a new item "g" in Section 15.2.6.2. Adding this item to the analysis assumptions listed clarifies the existing modeling assumptions in the loss of non-emergency AC analysis. This does not represent a change to the analysis assumptions.

9) Revise USAR Section 15.2.7, "Loss of Normal Feedwater Flow," and USAR Section 15.2.8, "Feedwater System Pipe Break," to reflect use of the DNB analysis code (VIPRE) and to state the DNB analysis results (i.e., DNBR design limit met). The loss of normal feedwater and feedwater system pipe break analyses each include a DNBR calculation which was not noted in the USAR. The loss of non-emergency AC analysis includes a similar calculation and briefly discusses this in the USAR description. The loss of normal feedwater and feedwater pipe rupture descriptions have been revised to be consistent with loss of AC description. This does not represent a change to the analysis method, models, or assumptions.

10) Revise Section 15.2.7.2, "Analysis of Effects and Consequences," item e to clarify the components of the delay before AFW delivery and to indicate the assumed AFW system single failure assumptions in order to clarify the existing analysis assumptions. Table 15.2-1 Sheet 3 was also revised to be consistent with the stated single failure assumptions.

11) Revise Section 15.2.7.2 item e to correctly reflect the assumed AFW total delay (433 seconds) as included in the accident analysis.

12) Revise assumption e in Section 15.2.8.2, , "Analysis of Effects and Consequences," to correctly reflect the time and conditions for the loss of feedwater. The USAR currently indicates 438 seconds versus 433 seconds. This discrepancy is likely the result of a typographical error.

13) Revise Sheet 4 of USAR Table 15.2-1, "Time Sequence of Events for Incidents Which Result in a Decrease in Heat Removal by the Secondary System," to reflect the time that the main steam safety valves (MSSVs) first open as 48.25 seconds. The analysis predicts initial (brief) opening

of the MSSVs at ~48 seconds. The sequence of events in Table 15.2-1 currently indicates MSSVs actuation at ~220 seconds, which is the time MSSVs actuate for the second time. This does not represent a change in the analysis.

Safety Summary:

Each of these proposed changes will correct existing discrepancies in the USAR by revising the analysis or plant description to be consistent with the existing safety analyses and plant configuration. No change is proposed to the existing licensing basis analyses or to any plant structure, system or component, or plant operating procedures.

There are no plant procedures, activities, administrative controls, or sequences of plant operations, that are impacted by the proposed changes. As stated above, the proposed changes will correct existing discrepancies in the USAR which were identified by the USAR Fidelity Review Team.

No credible accidents are created by the proposed changes. The proposed USAR changes are intended to remove discrepancies within the USAR and between the USAR and the current licensing basis accident analyses. No changes to the plant or to the accident analyses are proposed.

No malfunction of equipment important to safety is impacted by the proposed USAR changes. The proposed USAR changes are intended to remove discrepancies within the USAR and between the USAR and the current licensing basis accident analyses. No changes to the plant or to the accident analyses are proposed.

No acceptance limits are impacted by the proposed changes. Since the accident analyses are not being revised, there is no impact on the analysis results, acceptance limits, or margin to the existing acceptance limits.

Safety Evaluation: 59 1998-0154 **Revision:** 0

Established a Radiological Controlled Area for Material Storage

Description:

This Unreviewed Safety Question Determination (USQD) evaluates establishing a satellite Radiological Control Area (RCA) for radioactive material storage in the North Laydown Yard and revises Procedure AP 25A-001 "Radiation Protection Manual" accordingly. This involves storage of radioactive materials outside the current Restricted Area, defined as the security fence around the protected area. Procedure AP 25A-001, "Radiation Protection Manual," currently allows RCAs to be established by the Manager Chemistry/Radiation Protection.

Radioactive materials considered for initial storage include scaffolding, protective clothing (PCs), and communication equipment with fixed contamination enclosed inside containers (Sea-land containers w/7 at 1100 cubic feet and 1 at 2200 cubic feet capacity).

Historically, radioactive material, housed in semi-truck trailers, was stored in the North Laydown Yard until November 1994. In 1994 the definition of the Restricted Area was modified from an area out to 1200 meters to the present definition.

Safety Summary:

The Updated Safety Analysis Report (USAR) does not describe storage of Radioactive Material in the North Laydown Yard and will require revision. The proposed activity does not change any administrative control which would reduce the effectiveness of existing programs including the radiological control program, reduce the level of qualification of WCNOC personnel, nor does it affect any systems, structures, and components. The proposed activity does not change the performance of activities that are important to the safe and reliable operation of WCGS.

The USAR does not describe storage of Radioactive Material in the North Laydown Yard. This proposed activity would require modification of USAR Section 11.4 to describe storage of solid radwaste in containers located at the North Laydown Yard. No other USAR descriptions or conclusions would change or be untrue due to this change.

There are no design basis accidents identified as the proposed activity does not affect any systems, structures, or components, does not affect performance of activities, and does not change administrative controls that would adversely affect existing programs including the radiological control program. However, because of phenomenological similarities, the accidents described in USAR 15.7 were examined to ascertain whether they remain bounding.

A natural disaster/fire may cause the dispersal of radioactive material to the ground surface or the atmosphere, as the proposed satellite RCA in the North Laydown Yard doesn't meet the criteria of the main power block structures. Based on the results Calculation AN 98-035, the radiological consequences of a worst case incident would not exceed a small fraction of the 10 CFR 100 guidelines and would be bounded by the design basis accident described in the USAR. Therefore, this change does not create the possibility of an accident of a different type than previously evaluated in the USAR.

Since the proposed changes do not affect controls on activity performance, the level of qualification is not changed and there is no affect on any system, structures, or components, no credible malfunctions of equipment important to safety are identified.

The proposed activity places radioactive material in a location not explicitly designed and constructed to meet the criteria specified in various regulations for radioactive waste management systems, structures, and components. Acceptance limits potentially affected by the proposed activity of establishing a satellite RCA include those specified in the 10 CFR 20 and 10 CFR 100 guidelines, and those related administrative limits specified as part of the Wolf Creek radiological program.

Safety Evaluation: 59 1998-0155 **Revision:** 1

Corrections to Training Information Resulting From the Fidelity Review

Description:

This Unreviewed Safety Question Determination (USQD) evaluates the following changes to the Updated Safety Analysis Report (USAR) Sections 13.1, "Organizational Structure of the Operating Agent," and 13.2, "Training." Changes are as follows:

1. Add a reference to ANSI/ANS 3.1-1981 for Operators in Table 13.1-1 (Notes 1 & 15); also delete Note 2 for Shift Supervisor and Supervising Reactor Operator, and correct misspelling of license in Note 13.
2. Add ANSI/ANS-3.1-1978 as reference.
3. Due to the recommendation in Regulatory Guide 1.70, add words to the effect of "The duration will be administratively controlled.".
4. Add "Plan" following "Emergency."
5. Change "oral" to "walkthough.".
6. The words "Full compliance in the area of training" are added to USAR 13.2.3, Item 9 as required by Regulatory Guide 1.70
7. Reword USAR 13.2.2.8 to better define the subjects of the Engineering Support training and specifically address Supervisory Training.
8. USAR 18.1.5.2, 2nd paragraph is reworded to discuss selected requalification program topics and reference to simulator training is deleted. Program topics includes simulator.
9. Correct misspellings.
10. Change "Corporate Development" to "Corporate Services" to match the current organization title.
11. Delete redundant wording in USAR 13.2.2.6 and 13.2.2.7.
12. Reword 13.2.2.9 as necessary to reflect the current INPO accredited program.
13. Add words to 13.2.1.1.1.6 and 13.2.1.1.1.8 stating that topics needed for these programs may be included in other training programs.
14. Add "Loss of all feedwater (normal and emergency)" in 13.2.1.1.3.2 to be consistent with the list in 13.2.2.12.7.

The above listed changes are administrative in nature and are in agreement with existing plant commitments, procedures, Technical Specifications and other regulatory requirements.

Safety Summary:

The proposed changes to personnel qualification and training requirements as described in the USAR are to indicate compliance with existing commitments. The changes resolve discrepancies identified by PIR 98-2112. The changes do not establish new requirements in the areas of training and qualification, and are in agreement with the existing training program/procedures and regulatory requirements. The proposed changes have no affect on any plant structures, systems, components or equipment, and do not affect the sequence of any plant operations. The proposed USAR changes do not constitute a test or experiment.

The proposed changes do not affect any accident analyses in USAR Chapters 2, 3, 6, 9 or 15. The proposed changes do not affect systems/equipment design, maintenance, operation or testing. The proposed changes are made to indicate compliance with existing commitments and regulatory requirements with regard to personnel qualifications and testing. Therefore, the changes cannot create any credible accidents.

The proposed changes have no direct affect on equipment important to safety. An indirect effect could result from operation or maintenance of equipment by improperly qualified or improperly trained personnel. The changes proposed will revise personnel training as described in the USAR to indicate compliance with existing commitments and regulatory requirements. The revised descriptions of personnel training requirements do not require any corresponding changes to the existing training program or associated procedures. Therefore, the changes reflect a training/qualification program which ensures that the Wolf Creek staff consists of properly trained and properly qualified personnel. Therefore, the changes do not result in any credible malfunctions of equipment important to safety.

The proposed USAR changes do not affect any acceptance limits in the technical specifications or any other licensing basis document. The changes bring personnel training and qualifications as described in the USAR into compliance with existing commitments and regulatory requirements. The changes are consistent with the requirements of Technical Specification 6.3 which address Unit Staff Qualifications.

Safety Evaluation: 59 1998-0156 **Revision:** 0

Change to Auxiliary Feedwater Section of the Updated Safety Analysis Report

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a proposed change to the Updated Safety Analysis Report (USAR) Auxiliary Feedwater System section. The last paragraph in USAR Section 10.4.9.2.3 states that during a secondary side break, the operator utilizes Auxiliary Feedwater flow indication, in the Control Room, to determine which loop is broken. Emergency Operating Procedures (EMGs) procedures and the applicable Alarm (ALR) procedure, only use steam generator pressure indication, to determine which steam generator is faulted. Therefore, this reference to the use of the Control Room Auxiliary Feedwater flow indication is being deleted. The wording has been revised to state that the Operator determines which steam generator is faulted. This new wording is consistent with USAR Section 15.0.13, "Operator Actions."

The EMG procedures strictly use steam generator pressures to determine which steam generator is faulted. Procedure ALR 00-128A. "AFP DISCH PRESS LO," first has the Operator determine which pump has the low pressure, then check to see if their flow control valves are in the correct position and then check the steam generator pressures to determine if any steam generator is faulted. All of these procedures only use steam generator pressure to determine the location of a break.

Safety Summary:

The steam line break and feedwater line break accidents were reviewed with the Design Basis Accident (DBA) steam line break being the bounding accident. Continuing to add feedwater to a faulted steam generator longer than expected, or continuing to pump feedwater out of a feedwater break are concerns, associated to this proposed change.

The credible malfunction of equipment important to safety pertaining to this proposed USAR change, are the failure of Auxiliary Feedwater flow instrumentation and steam generator pressure instrumentation. There are no acceptance limits involved with this proposed USAR change. Per Technical Specification 3.3.3.6 and Table 3.3-10, two steam pressure instruments per steam generator are expected to be available, while only one Auxiliary Feedwater flow instrument per steam generator is available.

This proposed USAR change pertains to identifying which steam generator is faulted, after an accident has already occurred. Therefore, this proposed change could not increase the probability of occurrence of an accident previously evaluated in the USAR.

This proposed USAR change only involves a Main Steam or Main Feedwater break and the identification of which steam generator is involved with that break. Once the faulted steam generator is identified, Auxiliary Feedwater is isolated to that steam generator. Since this change only involves a secondary break and not a steam generator tube rupture, this proposed change does not increase the radiological consequences of an accident previously evaluated in the USAR.

This proposed USAR change involves determining which steam generator is faulted, during a secondary break accident. This change is just changing the wording in USAR Section 10.4.9.2.3 to be consistent with Chapter 15 of the USAR and eliminates the conflict with the Operations' Emergency Procedures. There are no physical or procedural changes being made, only a wording change to the USAR, to enhance consistency. Therefore the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR is not increased.

This change only involves a wording change to USAR Section 10.4.9.2.3, to make it consistent with USAR Chapter 15 and eliminate the conflict with the Operations' Emergency Procedures. No physical or procedural changes are being made. Therefore, the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR is not increased; the possibility of an accident of a different type than any previously evaluated in the USAR is not created, and the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR is not created.

As USAR Section 10.4.9.2.3 is currently written, it indicates the Operator uses Auxiliary Feedwater flow indication to identify a faulted steam generator. However, operations Emergency Procedures utilize steam generator pressure indication to identify a faulted steam generator. Steam generator pressure is a more direct indication that makes identifying a faulted steam generator more efficient than using auxiliary feedwater flow, since many factors can affect auxiliary feedwater flow. Steam generator pressure is a direct and positive indication. There are diverse pressure instruments for each steam generator available for this determination. The method used by the Operator to determine where the secondary break is located, does not reduce the margin of safety as defined in the basis for any technical specification.

Safety Evaluation: 59 1998-0157 **Revision:** 0

Emergency Diesel Fuel Oil Safety Classification

Description:

This Unreviewed Safety Question Determination (USQD) evaluates an Updated Safety Analysis Report (USAR) text and figure change. This change is made to appropriately reflect the safety classification boundaries for the cross-connection piping between the two fuel oil storage tanks for Trains "A" and "B" of Emergency Diesel Generators (EDG). The pipes in question are 029-HBD-2² (cross-connection pipe) and 059-HBD-1² (the drain pipe) shown on USAR Figure 9.5.4-1, "Emergency Fuel Oil System,". This USAR figure shows the correct break at valves V007 and V008 which classifies these lines as non-safety related, consistent with "HBD" designation of the pipes. Contrary to this, the class break shown at the drain valve V082 renders this classification to be incorrect. This change package deletes the class break shown at drain valve V082 eliminating the contradiction. The piping is supported seismically per the Hanger Location Isometric M-19JE02. Note No. 6 is being added to the subject USAR figure to indicate that these pipes are supported seismically. In USAR Section 9.5.4.2.1., 'Emergency Diesel Engine Fuel Oil Storage and Transfer System, System Description", 2nd sentence in paragraph 3, where it refers to this piping as "Seismic Category I", is being changed to "Seismically Supported". This will make the description in USAR Section 9.5.4.2.1 consistent with the USAR Figure 9.5.4-1. This sentence will read as " Figure 9.5.4-1 indicates that the cross-connection piping between the two fuel oil tanks is Seismically Supported".

A typographical error is being corrected in third sentence of the first paragraph in USAR Section 9.5.4.2.1, where the word "values" should read as "valves". The sentence will now read as "The EDEFSTS for each diesel engine has an underground storage tank with a transfer pump, day tank, strainers and filters, piping, valves, instruments, and controls."

This change affects USAR Figure 9.5.4-1 and USAR Section 9.5.4.2.1. No other USAR descriptions were found that are affected.

Safety Summary:

The proposed activity has no effect on any procedure, activity, administrative control, or sequence of plant operations and does not violate any requirement stated in the USAR. The proposed activity is not associated with any test or experiment.

This change appropriately reflects the safety classification of the cross-connection piping between the two fuel oil storage tanks for Trains "A" and "B" of Emergency Diesel Generators. The proposed activity has no affect on equipment or parameters associated with any design basis accident discussed or referenced in the USAR.

The Piping is seismically designed and supported and due to nature of the change the proposed activity does not create a potential for, or has influence on, equipment or parameters that may cause any new credible accidents.

There are no malfunctions of equipment important to safety which may be directly or indirectly associated with the proposed activity.

The proposed activity does not affect acceptance limits as defined in the Technical Specifications or other licensing basis documents.

Safety Evaluation: 59 1998-0159 **Revision:** 0

Fire Protection Drawing Changes Resulting From the Updated Safety Analysis Report Fidelity Review

Description:

Configuration Change Package (CCP) 07933 revises Fire Delineation drawings 10466-A-1801 through -1804 to show the Auxiliary Building stairways in Fire Areas A-5 & A-6 as diagonally crosshatched to indicate a three hour fire resistive rating. The current revisions of these drawings show these stairways with vertical crosshatching to indicate a two hour fire resistive rating. Drawings 10466-A-1801 through 10466-A-1804 are USAR Figures 9.5.1-2 Sheet 1 through 4, respectively. These changes are based on comments documented in Performance Improvement Request (PIR) 98-1858 initiated during the USAR Fidelity Review.

The Auxiliary Building stairways are designated as rooms 1119 and 1127 on drawings 10466-A-1801 through -1804. As shown on drawings 10466-A-1801 through -1804, all walls separating the stairwell rooms 1119 and 1127 in Fire Areas A-5 and A-6 are three hour fire resistive rating. The following architectural design drawings also show the fireproofing of both Auxiliary Building stairways to be a three hour fire resistive rating:
a.) 10466-A-0310, Revision 5, Auxiliary Building, Stairs A-1, A-1A - Plans & Sections
b.) 10466-A-0311, Revision 4, Auxiliary Building, Stair No. A-2 - Plans & Sections

Safety Summary:

The Fire Hazards Analysis (FHA), USAR Section 9.5B, "Fire Hazards Analysis," for Fire Area A-5 states: "The stairwell is enclosed by 3-hour-rated fire barriers (walls and floors). ... All of the stairwell doors are also fire rated for 3 hours." The Fire Hazards Analysis (FHA), USAR Section 9.5B, for Fire Area A-6 states: "The stairwell is separated from the rest of the plant by 3-hour rated fire barriers (walls and floors)."

The proposed changes were evaluated against the Fire Protection Program for potential impact. The approved evaluation determined that the change will not effect the ability to achieve and maintain fire safe shut down. This change will not result in a significant reduction in fire protection.

There is no field work associated with CCP 07933. The drawing changes made by CCP 07906 and the associated USAR changes are documentation changes that do not affect any SSC nor do they change the performance of activities that are important to the safe and reliable operation of WCGS.

The drawing changes made by CCP 07933 do not impact any procedures, activities, administrative controls, or sequence of plant operations nor are any plant structures, systems, components or equipment impacted. No other USAR descriptions or conclusions will change or be made untrue as a result of these changes. No test or experiments are involved with these changes. Therefore, no design basis accident is identified for review.

CCP 07933 makes no additional changes to the plant, does not affect the performance of plant activities and does not affect any SSC. Therefore, no credible accidents that could be created are identified and no credible malfunctions of equipment important to safety are identified. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1998-0160 **Revision:** 0

Water Treatment Changes

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the Updated Safety Analysis Report (USAR) Section 13, "Training," and in procedures associated with qualifications/responsibilities of non-licensed Radwaste Operators (RWOs) and Water Treatment Operators (WTOs). These changes are described below.

1.) In Section 13.1.2.2.1 the following paragraph is added after the sixth paragraph in the section "Radwaste Operators (RWOs) and Water Treatment Operators (WTOs) work under the supervision of the Shift Supervisor. The RWOs and WTOs responsibilities include the operating of equipment associated with their respective watch stations. This is done under the supervision and direction of the Main Control Room operating personnel, with input from the Treatment Systems Supervisor."

2.) Section 13.2.2 is revised to specifically recognize the Radwaste Operator and Water Treatment Operator positions, and discuss training/requalification training requirements for these positions. These changes consist of the following:

The first change adds a paragraph to USAR section 13.1.2.2.1 to describe the responsibilities of the Radwaste Operators and the Water Treatment Operators as well as describing the requirements for supervising these non-licensed operators.

The changes included in change 2 above adds the Radwaste Operator and the Water Treatment Operator to USAR Section 13.2.2, Non-Licensed Plant Staff Training, which describes the initial training program as the requalification training program for these non-licensed operators. Previously this USAR section addressed primarily the Nuclear Station Operator initial and requalification training programs.

In addition to the USAR changes described above, a new administrative procedure, AP 17C-030, "Radwaste Operator/Water Treatment Operator Qualifications and Responsibilities," is being issued to provide additional procedural controls of qualification and responsibilities for the Radwaste Operator and Water Treatment Operator positions. Additional procedures that are already in effect that relate to this issue are: AP 30B-002, "Nuclear Station Operator (NSO) Requalification Training" and AP 17C-015, "Operations Watchstation Qualifications."

Safety Summary:

These USAR changes are administrative in nature and are in agreement with existing plant commitments, Technical Specifications and other regulatory requirements. No other USAR sections are impacted by these changes nor is any plant equipment affected by these changes.

The proposed USAR changes are made to more specifically address the Radwaste Operator and Water Treatment Operator staff positions. The changes resolve discrepancies identified by PIRs 98-1381 and 98-1384. The changes provide information in the areas of training and qualification of specific non-licensed operator positions, and are in agreement with existing commitments and regulatory requirements. The proposed changes have no affect on any plant structures, systems, components or equipment, and do not affect the sequence of any plant operations. Procedures associated with the proposed text changes are:

AP 30B-002, Nuclear Station Operator Requalification Training
AP 17C-015, Operations Watchstation Qualifications

The proposed changes do not affect any accident analyses in USAR Chapters 2, 3, 6, 9 or 15. The USAR text changes are made to more specifically address personnel qualifications and training of certain non-licensed operator positions. This USARCR does not change the performance of activities that are important to the safe and reliable operation of WCGS. Therefore, no design basis accidents identified.
The proposed changes do not affect systems/equipment design, maintenance, operation or testing. The USAR text changes are made to more specifically address personnel qualifications and training of certain non-licensed operator positions. This CCP does not change the performance of activities that are important to the safe and reliable operation of WCGS. Therefore, no credible accidents that could be created are identified.

The proposed changes have no direct affect on equipment important to safety. An indirect effect could result from operation of equipment by improperly qualified or improperly trained personnel. The USAR changes proposed will revise the USAR text to more specifically address personnel qualifications and training of certain non-licensed operator positions. The revised text does not require any corresponding changes to the existing training program. A new procedure (AP 17C-030) is being issued to administratively address qualifications and responsibilities of the Radwaste Operator and Water Treatment Operator positions. Therefore, the changes reflect a training/qualification program which insures that the Wolf Creek operations staff consists of properly trained and properly qualified personnel. Therefore, the changes do not result in any credible malfunctions of equipment important to safety.

The proposed changes do not affect any acceptance limits in the technical specifications or any other licensing basis document. The changes are administrative in nature and are made to correct discrepancies in USAR text with regard to personnel training and qualifications of specific non-licensed operator positions. The changes are consistent with the requirements of Technical Specification 6.3 which address Unit Staff Qualifications. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1999-0001 **Revision:** 0

Automatic Signal to Motor Operated Valves

Description:

This modification provides an automatic signal to motor operated valves EGHV0101 (Component Cooling Water to Residual Heat Exchanger "A" Isolation Valve), EGHV0102 (Component Cooling Water to Residual Heat Exchanger "B" Isolation Valve), ECHV0011 (Fuel Pool Heat Exchanger "A" Component Cooling Water Discharge Isolation Valve), and ECHV0012 (Fuel Pool Heat Exchanger ""A Component Cooling Water Discharge Isolation Valve) to go to their respective safety related positions following a safety injection signal simultaneous with a RWST LO-LO-1 level signal. Valves EGHV0101 and EGHV0102 will be caused to open. Valves ECHV0011 and ECHV0012 will be caused to close.

Safety Summary:

Performance Improvement Request (PIR) 98-1008 identified a discrepancy between the Updated Safety Analysis Report (USAR) and the procedures used by operations for post-LOCA actions. USAR Tables 6.3-11, "RWST Outflow (Large Break) - No Failures," and 6.3-12, "RWST Outflow (Large Break) - Worst Single Failure (9)," list six manual steps which the operators need to complete to place Emergency Core Cooling System (ECCS) in the recirculation phase. These steps are given times in the tables which show that the manual actions can be accomplished before the Refueling Water Storage Tank (RWST) is drained. In review of procedure EMG-ES-12, "Transfer to Cold Leg Recirculation," eleven steps are indicated. Several of the steps not listed in the USAR are related to the manual realignment of the CCW system. It was shown that, in following the procedure on the simulator, operations crews could not meet the times indicated in the two USAR tables. With the times not being met, a safety concern was raised that the ECCS pumps and the containment spray pumps could be damaged by taking suction from an empty RWST. The safety concern was addressed in an operability evaluation sent to the Shift Supervisor on April 10, 1998.

Performance Improvement Request (PIR) 97-3483 had earlier identified a related discrepancy between the USAR and the procedures relating to USAR Table 6.3-8 "Sequence of Changeover Operation from Injection to Recirculation." USAR Table 6.3-8 states that Component Cooling Water (CCW) flow to the Residual Heat Removal (RHR) heat exchangers is established, and flow to the Fuel Pool Cooling heat exchangers is terminated as the level in the RWST "nears" the LO-LO-1 setpoint. However, the operator is not directed to perform these actions, via procedure EMG ES-12, until the LO-LO-1 setpoint is reached. Evaluation of PIR 97-3483 determined that damage could occur to components in the RHR and CCW systems if the CCW realignment is not performed in a timely manner. Operations revised the procedure to immediately begin the CCW realignment as soon as the RWST LO-LO-1 setpoint alarm is received. With this revision to the procedure, it was assured that the CCW system would be aligned before any damage occurred to any safety-related component.

By automating the alignment of the CCW valves, the above two concerns are alleviated. The operators will have fewer manual steps to perform, making it easier to accomplish the tasks required for the alignment of the ECCS pumps to the containment sump for recirculation mode. The operators will not have to concern themselves with potential damage to safety related equipment while trying to manage an accident scenario.

Several sections on the USAR discuss the "manual" operation of the CCW valves during the realignment for the recirculation phase. By making these valves automatic, USAR Sections 6.3.2.2, "Equipment and Component Descriptions," 6.5.2, "Containment Spray System," 9.1.3, "Fuel Pool Cooling and Cleanup System," and 9.2.2, "Cooling System for Reactor Auxiliaries," will no longer be true. USAR Tables 6.3-8, 7.1-2, "Identification of Safety Criteria," 9.1-6, "Spent Fuel Pool Cooling and Cleanup System Single Active Failure," 9.2-13, "Component Cooling Water System Single Active Failure Analysis," and USAR Figures 9.1-3-01, "Fuel Pool Cooling and Cleanup System," and 9.2-15-02, "Component Cooling Water System," have information which also will no longer be true. All of these sections, tables and figures will be revised to show the new configuration. The only tests which may be affected are STS-IC-740A, "RHR Switchover to Recirc Sump Test - Train A," and STS-IC-740B, "RHR Switchover to Recirc Sump Test - Train B." However, these tests are performed to verify the functionality of the logic circuitry, and will not adversely affect the adequacy of any system, structure, or component (SSC) to prevent accidents or mitigate the consequences of an accident.

The design basis accidents that will be reviewed for potential impact are those which require ECCS pumps or containment spray pumps to be switched over from injection phase to recirculation phase. Those accidents are the large and small break LOCAs, the main steam line break, and the Rod Cluster Control Assembly (RCCA) ejection accident. The RCCA ejection accident results in a small break LOCA. If the large and small break LOCAs are reviewed, the RCCA ejection accident is bounded. These are all discussed in USAR chapter 15.

If a spurious actuation of the valve results in isolating CCW supply to Spent Fuel Pool Cooling Water HX (SFPCWHX) with a full core offload, approximately 3.6 hours are available for operator action to restore cooling to the spent fuel pool before pool temperature will reach design limit temperature of 200 F. The effect of this failure mode is no different than tripping of a Spent Fuel Pool Cooling Water (SFPCW) pump. Control Room operators are alerted of high temperature in the SFP at 130° F and respond in accordance with applicable alarm procedures. When the fuel is in the reactor, RHR is used for cooling.

The initiation of switch over from injection to recirculation mode is started at RWST level Lo-Lo-1 as at that RWST level, sufficient water has been transferred to the containment recirculation sumps for providing adequate net positive suction head (NPSH) to the RHR pumps. The switch over of the containment spray pump is initiated at RWST Lo-Lo-2,at that level sufficient water has been transferred to the containment recirculation sumps for providing adequate NPSH to the containment spray pumps.

No identified credible accidents exist that the automation of these valves can create.

The automation of the CCW valves can result in a failure of the logic circuitry to realign any of the valves. This failure is no different than the possible failure of these valves when operated by the control room hand switches. The failure of the valves to realign is analyzed in USAR Tables 9.1-6, and 9.2-13. The effect of the logic circuitry to fail is no different than the failure of the valve to realign.

In accordance with Calculation EJ-M-032, "Evaluate Time to Boil CCW in the RRH HX, " the CCW system must be aligned to the recirculation alignment within 90 seconds of receipt of the RWST LO-LO-1 signal. This is difficult to accomplish with manual action. By automating the valves upon receipt of the LO-LO-1 signal, the 90 second requirement is easily met. The maximum acceptable Inservice Inspection (ISI) Program valve stroke times are 60 seconds for all four affected valves. Data from the ISI Program testing shows that the valves actual stroke times are closer to 50 seconds.

Changes to failure modes and effects for valves ECHV0011, ECHV0012, EGHV0101, and EGHV0102 are limited to utilization of the K740 and K741 relays to automate valve actuation. Failures which could occur during normal plant operation and design basis accident conditions include failure of the relay contacts to open and spurious unintended actuation of a relay.

If the relay contacts fail to close, the affected valves will not change state automatically upon SI and RWST Lo-Lo-1 signals. However, since the relay contacts will be connected in parallel with the hand switch contacts, the Control Room operator can still manually move the valve to the required state the same as the existing design.

In the new design, one contact on the K740 relay is connected in series with one contact on the K741 relay for each valve. Under normal plant conditions in the absence of SI and RWST Lo-Lo-1 signals, the spurious unintended actuation of one relay would have no affect since the contacts on both relays must be closed in order to actuate the valve.

If the spurious unintended actuation of one relay were to occur after the other relay has been actuated by an invalid signal, then four valves on that train would move to their automated positions. An the A train, valves ECHV0011 and BNHV8812A would close and valves EGHV0101 and EFHV8811A would open. (Valves BNHV8812A and B are actuated via limit switch contacts on valve EJHV8811A and B). On the B train, valves ECHV0012 and BNHV8812B would close and EGHV0102 and EJHV 8811B would open.

The plant could be affected in two ways by the unintended valve actuations. If the valid signal were a safety injection (SI) signal, and the unintended actuation were on the RWST Lo-Lo-1 relay, the RHR pump on that train would be automatically lined up to take suction from the containment sump when there could be insufficient water available in the sump to provide the proper NPSH. The RHR pump could be damaged by not having sufficient suction head. The effect of this failure is bounded by the failure of the RHR pump to start, previously analyzed in the USAR Table 15.0.7.

At the same time, CCW water flow would be stopped to the spent fuel pool heat exchanger and directed to the RHR heat exchanger. This would leave only one spent fuel pool heat exchanger to remove heat from the spent fuel pool. However, the design of the spent fuel pool cooling system is such that one operating heat exchanger is sufficient to remove the maximum heat load.

If the valid signal were a RWST Lo-Lo-1 signal, and the unintended actuation were on the safety injection relay, the RHR pump on that train would be automatically lined up to take suction from the containment sump. However, the RHR pump would not be running, and there would be no effect on the system. If the RHR were already running for some other reason, or if the unintended actuation of the safety injection relay were to occur at a point in the logic system such that it caused the ECCS pumps to start, then the effect would be similar to the scenario previously described when insufficient suction head is available. The effect on the CCW system would be identical to that previously discussed for insufficient RHR suction head, i.e., CCW water flow would be stopped to one spent fuel pool heat exchanger and directed to the RHR heat exchanger.

When the K740 and K741 relays close and the valves move to their automated positions, at least one of the signals (SI or RWST Lo-Lo-1) must be removed before the valves can be manually returned to the opposite position. If a valve is manually returned to the opposite position prior to removal of at least one of the signals, the valve will immediately return to the automated position.

The only credible malfunction of equipment important to safety is a failure of the valves to realign (due to any cause). This failure has been previously analyzed and is documented in the USAR.

This modification increases the probability that the valves will be realigned within the amount of time required to prevent thermally induced damage to the RHR heat exchanger. The modification increases the probability that the ECCS pump swapover evolution will be complete within the time required to prevent damage to any ECCS pump. There are no reduction to any margins of safety as a result of this modification.

Safety Evaluation: 59 1999-0002 **Revision:** 0

USAR Revision to Correct Inconsistency

Description

This Updated Safety Analysis Report (USAR) change revises USAR Section 15.4.9, "References," Reference 15 to read "TR-95-0001 METCOM Manual, Westinghouse Electric Corporation."

This proposed change removes the METCOM manual revision specification to allow presentation of the reference in a manner consistent with existing references. No change is proposed to the existing licensing basis analyses or to any plant structure, system or component, or plant operating procedures. Note: The Section 15.4.9 METCOM manual reference is cited in Section 15.4.3, ",Rod Cluster Control Assembly Misoperation," in regard to the Dropped Rod Cluster Control Assembly (RCCA) analysis.

Safety Summary:

There are no plant procedures, activities, administrative controls, or sequences of plant operations, etc., that are impacted by the proposed change. The proposed change will correct an existing inconsistency in the USAR presentation identified by the Fidelity Review Team.

The USAR Chapter 15 sections being revised and affected include the following:
Section 15.4.9. References Section 15.4.3

Based on the review, no potential impact from the proposed activity was determined to exist.

No credible accident is created by the proposed change. The proposed USAR change is intended to remove an inconsistency in the presentation of a reference in the USAR. No change to the plant or to the accident analyses is proposed.

No malfunction of equipment important to safety is impacted by the proposed USAR change. The proposed USAR change is intended to remove an inconsistency in the presentation of a reference in the USAR. No change to the plant or to the accident analyses is proposed.

No acceptance limits are impacted by the proposed change. Since the accident analyses are not being revised, there is no impact on the analysis results, acceptance limits, or margin to the existing acceptance limits.

Safety Evaluation: 59 1999-0003 **Revision:** 0

Technical Requirements Manual Revision

Description:

This Unreviewed Safety Question Determination (USQD) evaluates administrative changes to Wolf Creek Technical Requirements Manual (TRM) requirements that are being reformatted and revised during the conversion of Wolf Creek Technical Specifications to NUREG-1431 (Standard Technical Specifications for Westinghouse Plants), Revision 1. The proposed changes involve reformatting, renumbering, and rewording of the TRM with no change in intent. License Amendment No. 89, dated October 2, 1995, relocated specific Technical Specification requirements to the USAR and subsequently to the TRM. This USQD also provides justification of administrative revisions to Technical Specification requirements that are being relocated to the TRM as part of License Amendment No. 123. However, additional changes are being made to those items that have been relocated.

This USQD is required because the TRM is incorporated into the USAR by reference. The TRM contains a wide variety of information on, and requirements for, various systems and processes, most of which existed in the Technical Specifications previously. The TRM is used primarily by Operations to provide operating guidance for various plant equipment (similar to the Technical Specifications).

The scope of this USQD is to justify administrative revisions to the CTRM requirements during the conversion to the Improved Technical Requirements Manual (ITRM) and to justify the administrative changes made to current Technical Specifications requirements in the process of relocating Technical Specifications to the TRM.

Discussion:

Technical Specifications relocated to the USAR and subsequently to the TRM are those items in the current Wolf Creek Technical Specifications that did not meet at least one of the four criteria in 10 CFR 50.36(c)(2)(ii). In accordance with the NRC Final Policy Statement on Technical Specification Improvements, items in the current Technical Specifications that fail to meet the criteria are to be relocated from Technical Specifications to a licensee-controlled document (e.g., the USAR, the QA Plan, or ITS Bases). Wolf Creek has relocated the majority of the items not retained in Technical Specifications to the TRM, consistent with Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications." The format of the current TRM (CTRM) is also being revised to be consistent with the Improved Technical Specification (ITS) format and is in accordance with NUMARC 93-03, "Writers Guide for Restructured Standard Technical Specifications." In the process of revising the CTRM, changes are proposed that involve reformatting, renumbering, and rewording with no change in intent.

These changes, since they do not involve technical changes to the TRM, are administrative. These changes are connected with the movement of requirements within the current requirements, or with the modification of wording that does not effect the technical content of the current TRM. These changes will include non-technical modifications of requirements to conform to the Writers Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1431. Administrative changes are not intended to add, delete, or relocate any technical requirements or the current TRM.

Technical and administrative control of the revisions of CTRM requirements during the conversion to the ITRM was maintained by marking CTRM and relocated Technical Specification pages to show: where in the ITRM the existing requirement will be located; how it will be numbered; and changes to the requirements. Each change is marked by a numbered designator which relates back to a discussion of the change and the justification for that change. To facilitate evaluation, each marked change was categorized as follows: Administrative (A), More Restrictive (M), and Less Restrictive - General (LG), and Less Restrictive - Specific (LS). Attachments 1 through 16 (listed below) contain the markups of the CTRM requirements and the current Technical Specifications for each of the Technical Specifications relocated to the TRM during the ITS conversion and Amendment No. 89 and a discussion and justification for the changes. This USQD only evaluates the administrative changes to the TRM.

Safety Summary:

Each of the changes covered by this USQD is categorized as an administrative (A) change. Each of the A changes was reviewed and the justification for each of the A changes is provided in its associated discussion of changes.

The changes described above are changes made to the Technical Requirements Manual which is incorporated into the USAR by reference. As such, the proposed changes themselves result in a change to the USAR and the specific change justifications are provided in the associated discussion of changes in each TRM section.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. These changes do not have a detrimental impact on the integrity of any plant structure, system or component. The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The Technical Requirements Manual requirements that govern OPERABILITY, or Technical Surveillance Requirement testing and verification of plant components and variables are not assumed to be initiators of any analyzed event.

The TRM contains requirements relocated from the current Technical

Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirements would not impact the initial conditions of a design basis accident (DBA) or transient analysis that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier; is not part of the primary success path, and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the requirements in the TRM would not affect a DBA.

The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirement would not impact the initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analysis and licensing basis. As such, the requirements in the TRM would not affect a DBA or create a credible accident.

The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSC). The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The reformatting, renumbering, and rewording process involves no technical changes to the current TRM. There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes.

Safety Evaluation: 59 1999-0004 **Revision:** 0

Technical Requirements Manual Revision

Description:

This Unreviewed Safety Question Determination (USQD) evaluates more restrictive changes to Wolf Creek Technical Requirements Manual (TRM) requirements that are being reformatted and revised during the conversion of Wolf Creek Technical Specifications to NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1. The proposed changes involve adding more restrictive requirements to the TRM by either making current requirements more stringent or by adding new requirements that currently do not exist. License Amendment No. 89, dated October 2, 1995, relocated specific Technical Specification requirements to the USAR and subsequently to the TRM. This USQD also provides justification of more restrictive revisions to Technical Specification requirements that are being relocated to the TRM as part of License Amendment No. 123. However, additional changes are being made to those items that have been relocated.

This safety evaluation is required because the TRM is incorporated into the USAR by reference. The TRM contains a wide variety of information on, and requirements for, various systems and processes, most of which existed in the Technical Specifications previously. The TRM is used primarily by Operations to provide operating guidance for various plant equipment.

The scope of this USQD is to justify more restrictive revisions to the CTRM requirements during the conversion to the Improved Technical Requirements Manual (ITRM) and to justify the more restrictive changes made to current Technical Specifications requirements in the process of relocating Technical Specifications to the TRM. The proposed changes involve adding more restrictive requirements to the TRM by either making current requirements more stringent or by adding new requirements that currently do not exist.

DISCUSSION:

Technical Specifications relocated to the USAR and subsequently the TRM are those items in the current Wolf Creek Technical Specifications that did not meet at least one of the four criteria in 10 CFR 50.36(c)(2)(ii). In accordance with the NRC Final Policy Statement on Technical Specification Improvements, items in the current Technical Specifications that fail to meet the criteria are to be relocated from Technical Specifications to a licensee-controlled document (e.g., the USAR, the QA Plan, or ITS Bases). Wolf Creek has relocated the majority of the items not retained in Technical Specifications to the TRM, consistent with Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications." The format of the current TRM (CTRM) is also being revised to be consistent with the Improved Technical Specification

(ITS) format and is in accordance with NUMARC 93-03, "Writers Guide for Restructured Standard Technical Specifications." In the process of revising the CTRM, changes are proposed that involve adding more restrictive requirements. These changes include additional commitments that decrease allowed outage times, increase the frequency of technical surveillance requirements, impose additional technical surveillance requirements, increase the scope of technical requirements to include additional plant equipment, increase the applicability of technical requirements, or provide additional actions.

Technical and administrative control of the revisions of CTRM requirements during the conversion to the ITRM was maintained by marking CTRM and relocated Technical Specification pages to show: where in the ITRM the existing requirement will be located; how it will be numbered; and changes to the requirements. Each change is marked by a numbered designator which relates back to a discussion of the change and the justification for that change. To facilitate evaluation, each marked change was categorized as follows: Administrative (A), More Restrictive (M), and Less Restrictive - General (LG), and Less Restrictive - Specific (LS). This USQD only evaluates the more restrictive changes to the TRM.

Safety Summary:

Each of the changes covered by this USQD is categorized as more restrictive (M). Each of the M changes was reviewed and the justification for each of the M changes is provided in its associated discussion of changes.

The changes described above are changes made to the Technical Requirements Manual which is incorporated into the USAR by reference. As such, the proposed changes themselves result in a change to the USAR and the specific change justifications are provided in the associated discussion of changes in each TRM section.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The changes do not have a detrimental impact on the integrity of any plant structure, system or component. These changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The Technical Requirements Manual requirements that govern OPERABILITY, or Technical Surveillance Requirement testing and verification of plant components and variables are not assumed to be initiators of any analyzed event. The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirements would not impact the initial conditions of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the requirements in the TRM would not affect a DBA.

The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirement would not impact the initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analysis and licensing basis. As such, the requirements in the TRM would not affect a DBA or create a credible accident.

The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSC). The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. As provided in the discussion of changes, each change in this category is, by definition, providing additional restrictions to enhance plant safety and operations. There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes.

Safety Evaluation: 59 1999-0005 **Revision:** 0

Technical Requirements Manual Revision

Description:

This Unreviewed Safety Question Determination (USQD) evaluates generic, less restrictive changes to Wolf Creek Technical Requirement Manual (TRM) requirements that are being reformatted and revised during the conversion of Wolf Creek Technical Specifications to NUREG-1431, "Standard Technical Specifications for Westinghouse Plants," Revision 1. The proposed changes involve moving details out of the TRM or Technical Specifications and into the TRM Bases, the USAR, or other documents under regulatory control. License Amendment No. 89, dated October 2, 1995, relocated specific Technical Specification requirements to the USAR and subsequently to the TRM. This USQD also provides justification of generic, less restrictive revisions to Technical Specification requirements that are being relocated to the TRM as part of License Amendment No. 123. However, additional changes are being made to those items that have been relocated.

This safety evaluation is required because the TRM is incorporated into the USAR by reference. The TRM contains a wide variety of information on, and requirements for, various systems and processes, most of which previously existed in the Technical Specifications. The TRM is used primarily by Operations to provide operating guidance for various plant equipment (similar to the Technical Specifications).

The scope of this USQD is to justify generic, less restrictive revisions to the CTRM requirements during the conversion to the Improved Technical Requirements Manual (ITRM) and to justify the generic, less restrictive changes made to current Technical Specifications requirements in the process of relocating Technical Specifications to the TRM. The proposed changes involve moving details out of the TRM or Technical Specifications and into the TRM Bases, the USAR, or other documents under regulatory control.

Discussion:

Technical Specifications relocated to the Updated Safety Analysis Report (USAR), and subsequently to the TRM are those items in the current Wolf Creek Technical Specifications that did not meet at least one of the four criteria in 10 CFR 50.36(c)(2)(ii). In accordance with the NRC Final Policy Statement on Technical Specification Improvements, items in the current Technical Specifications that fail to meet the criteria are to be relocated from Technical Specifications to a licensee-controlled document (e.g., the USAR, the QA Plan, or ITS Bases). Wolf Creek has relocated the majority of the items not retained in Technical Specifications to the TRM, consistent with Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications." The format of the current

TRM (CTRM) is also being revised to be consistent with the Improved Technical Specification (ITS) format and is in accordance with NUMARC 93-03, "Writers Guide for Restructured Standard Technical Specifications." In the process of revising the CTRM, the removal of detail to the TRM Bases, USAR or other licensee controlled documents is proposed. The removal of information from the Technical Specifications is considered less restrictive because it is no longer controlled by the Technical Specification change process. Removal of information from the TRM Technical Requirement to a licensee controlled document is also considered as a less restrictive change, even though the TRM Technical Requirement and the licensee controlled document are subject to the same change evaluation provisions. Changes to the TRM, TRM Bases, USAR or other licensee controlled document is evaluated in accordance with 10 CFR 50.59. The USAR is subject to the change control provisions of 10 CFR 50.71(e). Other licensee controlled documents are subject to controls imposed by Technical Specifications or regulations.

Safety Summary:

Each of the changes covered by this USQD is categorized as less restrictive - Generic (LG). Each of the LG changes was reviewed and the justification for each of the LG changes is provided in its associated discussion of changes.

The changes described above are changes made to the Technical Requirements Manual which is incorporated into the USAR by reference. As such, the proposed changes themselves result in a change to the USAR and the specific change justifications are provided in the associated discussion of changes in each TRM section.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The changes do not have a detrimental impact on the integrity of any plant structure, system or component. The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The Technical Requirements Manual requirements that govern OPERABILITY or Technical Surveillance Requirement testing and verification of plant components and variables are not assumed to be initiators of any analyzed event. The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (C)(2)(ii). These criteria indicate that the relocated requirement would not impact the initial conditions of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary.

The TRM Bases and USAR contain details moved out of the TRM Technical Requirement and Technical Specifications that is not necessary to ensure proper application or the Technical Requirement or Technical Specification. As such, the requirements in the TRM would not affect a DBA or create a credible accident.

The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSC). The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. As such, no new failure modes are being introduced.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The changes have no effect on any safety analysis assumptions since the changes are only moving information. There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes.

Safety Evaluation: 59 1999-0006 **Revision:** 0

Technical Requirements Manual Revision

Description:

This Unreviewed Safety Question Determination (USQD) evaluates specific, less restrictive changes to Wolf Creek Technical Requirement Manual (TRM) requirements that are being reformatted and revised during the conversion of Wolf Creek Technical Specifications to NUREG-1431 (Standard Technical Specifications for Westinghouse Plants), Revision 1. License Amendment No. 89, dated October 2, 1995, relocated specific Technical Specification requirements to the USAR and subsequently to the TRM. This USQD also provides justification of specific less restrictive revisions to Technical Specification requirements that are being relocated to the TRM as part of License Amendment No. 123. However, additional changes are being made to those items that have been relocated.

This safety evaluation is required because the TRM is incorporated into the USAR by reference. The TRM contains a wide variety of information on, and requirements for, various systems and processes, most of which existed in the Technical Specifications previously. The TRM is used primarily by Operations to provide operating guidance for various plant equipment (similar to the Technical Specifications). The format of the current TRM (CTRM) is also being revised to be consistent with the Improved Technical Specification (ITS) format and is in accordance with NUMARC 93-03, "Writers Guide for Restructured Standard Technical Specifications."

The scope of this USQD is to justify specific Less Restrictive revisions to the CTRM requirements during the conversion to the Improved Technical Requirements Manual (ITRM) and to justify the specific Less Restrictive changes made to current Technical Specifications requirements in the process of relocating Technical Specifications to the TRM. This USQD is not intended to justify that the items being relocated to the TRM do not meet the criteria of 10 CFR 50.36 (c)(2)(ii). The determination that each of the requirements in the TRM do not meet the criteria 10 CFR 50.36 (c)(2)(ii) or are adequately addressed by the ITS and, therefore, can be relocated out of Technical Specifications is provided in NRC Safety Evaluations associated with License Amendment 89, dated October 2, 1995, and Amendment 123, dated March 31, 1999. The determination that each of the TRM requirements could be relocated to another licensee-controlled document was approved by the NRC per these License Amendments.

Discussion:

Technical Specifications relocated to the USAR and subsequently the TRM are those items in the current Wolf Creek Technical Specifications that did not meet at least one of the four criteria in 10 CFR 50.36(c)(2)(ii). In accordance with the NRC Final Policy Statement on Technical Specification Improvements, items in the current Technical Specifications that fail to meet the criteria are to be relocated from Technical Specifications to a licensee-controlled document (e.g., the USAR, the QA Plan, or ITS Bases). Wolf Creek has relocated the majority of the items not retained in Technical Specifications to the TRM, consistent with Administrative Letter 96-04, "Efficient Adoption of Improved Standard Technical Specifications."

Technical and administrative control of the revisions of CTRM requirements during the conversion to the ITRM was maintained by marking CTRM and relocated Technical Specification pages to show: where in the ITRM the existing requirement will be located; how it will be numbered; and changes to the requirements. Each change is marked by a numbered designator which relates back to a discussion of the change and the justification for that change. To facilitate evaluation, each marked change was categorized as follows: Administrative (A), More Restrictive (M), and Less Restrictive - General (LG), and Less Restrictive - Specific (LS). Attachments 1 through 16 (listed below) contain the markups of the CTRM requirements and the current Technical Specifications for each of the Technical Specifications relocated to the TRM during the ITS conversion and Amendment No. 89 and a discussion and justification for the changes. This USQD only evaluates the Less Restrictive - Specific changes to the TRM.

Safety Summary:

Each of the changes covered by this USQD is categorized as Less Restrictive - Specific (LS). Each of the LS changes were reviewed and the justification for each of the LS changes is provided in its associated discussion of changes.

The changes described above are changes made to the Technical Requirements Manual which is incorporated into the USAR by reference. As such, the proposed changes themselves result in a change to the USAR and the specific change justifications are provided in the associated discussion of changes in each TRM section.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The changes do not have a detrimental impact on the integrity of any plant structure, system or component. The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The Technical Requirements Manual requirements that govern OPERABILITY or Technical Surveillance Requirement testing and verification of plant components and variables are not assumed to be initiators of any analyzed event. The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirement would not impact the initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the requirements in the TRM would not affect a DBA.

The TRM contains requirements relocated from the current Technical Specifications that did not meet the criteria of 10 CFR 50.36 (c)(2)(ii). These criteria indicate that the relocated requirement would not impact the initial conditions of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path and which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the requirements in the TRM would not affect a DBA or create a credible accident.

The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSC). Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced.

The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. The changes do not adversely impact these factors. There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes.

Safety Evaluation: 59 1999-0007 **Revision:** 0

Technical Specification Bases Revision 1

Description:

License Amendment No. 123, dated March 31, 1999, converts the current Technical Specifications (CTS) for Wolf Creek Generating Station (WCGS) to the improved Technical Specifications (ITS). The overall objective of the amendment was to rewrite, reformat, and streamline the CTS to improve safety and the understanding of the Bases underlying the Technical Specifications. With the issuance of the ITS, the Technical Specification Bases were also rewritten and issued as Revision 0. Technical Specification 5.5.14, Technical Specifications Bases Control Program, allows changes to the Bases without prior NRC approval provided the changes do not involve either of the following:

1. A change in the Technical Specification incorporated in the license; or

2. A change in the Updated Safety Analysis Report (USAR) or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

Summary of Changes:

1. Specific typographical, editorial, and formatting changes that have been identified since the issuance of Amendment No. 123 on March 31, 1999, were corrected.

2. During the NRC review of the ITS and Bases, SR 3.3.2.14 was created from SR 3.3.2.6 in response to NRC RAI Q 3.3.19. In developing the Bases for SR 3.3.2.14 a sentence that was previously in SR 3.3.2.6 was inadvertently omitted from SR 3.3.2.14. The sentence being restored states: "The SLAVE RELAY TEST of relay K620 does not include the circuitry associated with the main feedwater pump trip solenoids since that circuitry serves no required safety function."

3. During the development of the ITS, traveler TSTF-285 revised the Note to the Technical Specifications, but did not specifically revise the Technical Specification Bases Section 3.4.12. The Technical Specification Bases were modified during the development, but did not correctly reflect the wording of the Note in the Technical Specifications. Revision 1 correctly reflects that the accumulator discharge isolation valve surveillance requirement (SR 3.4.12.3) is not required when the accumulator is unisolated (accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing cold leg temperature as allowed by the P/T curves).

4. The Bases of ITS SR 3.7.7.2 is modified to indicate that the automatic Component Cooling Water (CCW) valve testing applies to those CCW valves

that receive a Safety Injection signal, and the CCW valves associated with the Reactor Coolant Pump (RCP) thermal barrier cooling coils (valves BB HV-0013, -0014, -0015, -0016, EG HV-62) that receive a High CCW Flow signal. This change is made since the CCW System high flow and surge tank level instrumentation was relocated to the Technical Requirements Manual (TRM). Technical Surveillance Requirement (TSR) 3.7.7.2, which requires a CHANNEL CALIBRATION of CCW System surge tank level and flow instrumentation, requires testing of all required components in the high flow and surge tank level circuitry, including valve actuation, every 18 months consistent with the requirements stated in CTS 4.7.3.b.1. The requirements of Technical Requirement (TR) 3.7.7 and associated TSRs continue to assure that each automatic valve isolates the non-nuclear safety-related portions of the CCW System on a simulated high flow and surge tank level test signal.

CTS 4.7.3.b.1, in part, required verification that each automatic valve isolating non-nuclear safety-related portion of the CCW System be actuated to its correct position on a simulated high flow and surge tank level test signal. The details of requiring valve isolation to occur on a simulated high flow and surge tank level test signal were relocated to the ITS Bases (See ITS DOC 8-03-LG) per the ITS Amendment (License Amendment 123, dated March 31, 1999). However, the CCW System high flow and surge tank level instrumentation was relocated to a licensee controlled document (TRM) since the instrumentation is not assumed in any safety analyses (See ITS DOC 8-01-LG). During the ITS conversion, it was determined that the CCW System surge tank level and flow instrumentation requirements specified in CTS 3.7.3 were not required to ensure the CCW System could perform its intended support functions. Relocation of the CCW System surge tank level and flow instrumentation requirements specified in CTS 3.7.3 was reviewed and approved by the NRC in License Amendment 123 (See ITS DOC 8-01-LG). Since the CCW System high flow and surge tank level instrumentation is not assumed in any safety analysis and was approved by the NRC, the CCW System high flow and surge tank level instrumentation is not required to meet ITS LCO 3.7.7 (CTS 3.7.3).

The CCW System surge tank level and flow instrumentation only provides a backup to the Safety Injection and containment isolation of the non-essential CCW System loads and the CCW System surge tank level and flow instrumentation is not assumed in any previous analyzed accident or transient. This change does not impact the ability for the CCW System to perform its intended support functions. Technical Specifications 3.7.7, "Component Cooling Water (CCW) System," and 3.6.3, "Containment Isolation Valves," requirements continue to ensure the CCW System can perform its intended support functions under all postulated events. In addition, TSR 3.7.7.2, which requires a CHANNEL CALIBRATION of CCW System surge tank level and flow instrumentation, requires testing of all required components in the high flow and surge tank level circuitry including valve actuation every 18 months consistent with the requirements stated in CTS 4.7.3.b.1. This change continues to assure that each automatic valve isolates the non-nuclear safety-related portions of the CCW System on a simulated high flow and surge tank level test signal.

5. The LCO and SR 3.4.7.2 Bases are revised to delete the design information that any narrow range level indication above 6% will ensure the SG tubes are covered since Technical Specification 3.4.7 requires the use of wide range instrumentation. The statement, "Any narrow range level indication above 6% will ensure the SG tubes are covered," was added to the LCO and SR section of the Bases as part of the response to NRC RAI Q 3.4.5-2 (See letters ET 98-0078 and ET 98-0107). In the first response Wolf Creek had proposed to add the word "or equivalent" into the ITS LCO and SR to allow the use of wide range level instrumentation in MODES 3 and 4 or the use of narrow range level instrumentation in MODE 5 if the one wide range indicator were unavailable. However, in subsequent discussions with the NRC, the NRC would not accept the word "equivalent" in the Technical Specifications, and the Bases were revised to reflect that any narrow range level indication above 6% would ensure the top of the SG tubes are covered. This is a valid statement since MODE 5 temperature conditions will not induce a 100 inch level measurement error on the narrow range. Subsequently, the concern is that with this statement in the LCO Bases, it could be interpreted that the use of the narrow range level indication could be used to comply with the Technical Specification. Therefore, the statement is being deleted to prevent any confusion as to what is required to meet LCO 3.4.7.

6. The LCO 3.7.5 Bases is clarified to refer to the Turbine Driven Auxiliary Feedwater (TDAFW) train versus pump were appropriate to be consistent with Technical Specification LCO 3.7.5 that requires three Auxiliary Feedwater (AFW) trains OPERABLE. NRC RAI Q 3.7.5 (letter ET 98-0085) indicated that the Bases were being revised to provide more detailed guidance regarding system flow paths after the withdrawal of a license amendment request (ET 97-0075) that had proposed an action statement for an inoperable Essential Service Water (ESW) supply. In providing the additional clarification to the Bases, the Bases wording did not accurately reflect the terminology in the LCO that required trains to be OPERABLE with the Bases specifying what constitutes the train.

7. The Applicable Safety Analyses, LCO, and Applicability Bases in Section 3.3.1 for Function 19, Reactor Trip Breakers, is revised to clarify the OPERABILITY requirements of a trip breaker train. The wording in the Technical Specification Bases, Rev. 0, states: "A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed and capable of supplying power to the Rod Control System. Thus, the train may consist of the main breaker or bypass breaker USAR Section 7.2.1.1, "System Description," and Figure 7.1-1, "Protection System Block Diagram," indicate that a single Reactor Trip System logic train supplies the logic to the main trip breaker (52/RTA) and the opposite train bypass breaker (52/BYB). Therefore, with the main trip breaker closed and only the opposite train bypass breaker closed, only one Solid State Protection System (SSPS) logic train would be providing the trip logic to the trip breakers. The Technical

Specification Bases are revised to state: "Thus, the train may consist of the main breaker or main breaker and opposite train bypass breaker. If a bypass breaker is closed and supplying power to the Rod Control System, that specific trip breaker train is considered inoperable." The proposed revision clarifies the Bases consistent with the current design of the plant as described in USAR, and is consistent with the operational practices and requirements for operating the reactor trip breakers and bypass breakers.

8. Bases 3.4.11 LCO are revised to provide clarification regarding Power Operated Relief Valve (PORV) block valve OPERABILITY. Specifically, the Bases is revised to state: "An OPERABLE block valve may be either open and energized with the capability to be closed, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation." Additionally, Bases 3.4.11 Required Actions C.1 and C.2 are revised to clarify the actions when the block valve is restored to OPERABLE status. Specifically, the Bases are modified to indicate that with the restoration of the block valve to OPERABLE status within the specified Completion Time, the PORV may be restored to automatic operation. This change is consistent with Technical Specification Required Action C.1 that requires the PORV be placed in manual control with the block valve inoperable. The following sentence is deleted: "Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status." The sentence is deleted to prevent any confusion regarding the OPERABILITY of a PORV if the block valve is inoperable. The proposed changes more accurately reflect current plant philosophy and operation of the PORVs and are consistent with the design basis described in USAR Section 5.2.2, "Overpressure Protection."

9. Bases 3.8.3, Background, is revised to more accurately reflect the basis for the fuel oil capacity for each Diesel Generator (DG). The changes are made for consistency with USAR Sections 9.5.4.1, ",Design Bases," 9.5.4.2, System Description," and Calculation M-JE-321, "Emergency Diesel Storage Tank/Day Tank Volume and Level Set." The Bases is revised to state: "Each diesel generator (DG) is provided with a storage tank having a fuel oil capacity sufficient to operate that diesel for a period of seven days at rated continuous capacity). The post accident load demand is below the rated continuous capacity of the DG, which ensures the DG has sufficient fuel oil to perform its design function after an accident." Calculation M-JE-321 determined that 85,300 gallons of fuel is required for 7 days of continuous operation at full load (SR 3.8.3.1). This calculation assumes that the DG operates at rated full load (6201 kW) for seven days continuously. The calculation does not take into account post accident (LOCA or Loss of Offsite Power) loads since the continuous rated full load is greater than post accident loads. Drawing E-11005 specifies that the maximum loading in the event of a LOCA is 5258 kW with additional non-safety related loads of 132 kW for an approximate total loading of 5400 kW. For a station blackout event, the maximum loading is 4548 kW with additional non-safety related loads of 1275 kW for an

approximate total loading of 5823 kW. This is consistent with USAR Sections 9.5.4.1 and 9.5.4.2.2 which indicate that the capacity of each tank is based on the fuel consumption by one diesel engine for operation at continuous rating for seven days. Additionally, the Bases statement that all outside tanks, pumps, and piping are located underground is revised to be consistent with USAR Sections 9.5.4.2.1 and 9.5.4.3 to indicate that the oil fill connections for the underground storage tanks are located above grade.

Safety Summary:

The proposed changes to the Technical Specification Bases provide for consistency between the USAR and the Bases. The proposed changes would not make information in the USAR no longer true or accurate. The proposed changes would not violate a statement in the USAR.

Analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The changes do not have a detrimental impact on the integrity of any plant structure, system or component. The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. Therefore, the proposed changes to the Bases do not affect any design basis accidents.

The proposed changes would not impact the initial conditions of a DBA or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; is not part of the primary success path which functions or actuates to mitigate a DBA or transient; or is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the proposed changes would not affect a DBA or create a credible accident.

The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSC). The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced.

The proposed changes to the Technical Specification Bases do not affect any acceptance limits contained in the bases of the Technical Specifications. The margin of safety is determined by the design and qualification of the plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated. There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes.

Safety Evaluation: 59 1999-0039 **Revision:** 0

Discussion to Include Instrument Lines in Containment Penetrations

Description:

This Updated Safety Analysis Report (USAR) change pertains to the discussion portion of Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," on page 3A-4, in Appendix 3A of the USAR. A discussion on the Wide Range Containment Pressure instrument lines and the Reactor Vessel Level (RVLIS) instrument lines is being added. Currently, the discussion for Regulatory Guide 1.11 states that the only instrument lines that penetrate Containment are the Containment pressure sensing lines, as described in USAR Section 7.3.8.1.1, "Description,". This USAR section only describes the normal Containment pressure instruments that are part of the protection system. Therefore, this USAR Change is being made to make the discussion for Regulatory Guide 1.11, in Appendix 3A, accurate and complete.

Safety Summary:

The instruments discussed above were not part of the original Wolf Creek Design. NUREG-0737 required the addition of these instruments. The addition of these instruments did not get incorporated into the discussion portion of Regulatory Guide 1.11, in Appendix 3A. USAR Figure 6.2.4-1, "Containment Penetrations," page 43a, clearly shows that the RVLIS instrument lines penetrate Containment. These lines are closed on the inside of Containment, by hydraulic isolators, and closed on the outside of Containment, by the instruments. Page 43a of this Figure, also states that this arrangement is similar to page 72 of this same Figure. Page 72 shows all of the Containment pressure instrument lines. A review of Surveillance Procedure STS PE-018, "Containment Integrated Leakrate Test," indicates that these containment penetrations are accounted for in the leakrate testing.

No Design Basis Accidents need to be reviewed. This USAR change is only adding wording to the USAR on how Wolf Creek complies to Regulatory Guide 1.11. The instrument lines being added to the discussion, are filled and sealed both inside and outside Containment.

There are no credible accidents that could be created by adding this wording to Appendix 3A of the USAR. There are no credible malfunctions of equipment Important to safety which may be directly or indirectly affected by this proposed USAR change. There are no acceptance limits that could be affected by this proposed USAR Change. These instruments are part of Technical Specification 3.3.3.6.

Safety Evaluation: 59 1999-0040 **Revision:** 0

Cycle 11 Reload Design

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the Updated Safety Analysis Report (USAR) associated with the Cycle 11 Reload Design as documented in Configuration Change Package (CCP) 07715.

The reload design was performed utilizing NRC approved methodologies as described in NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station," as well as the Technical Specification Amendment 92 which documents the use of Westinghouse design codes and analysis methodologies. The use of and documented adherence to approved codes, methodologies, and acceptance criteria precludes any new challenges to components and systems that could increase the probability of any previously evaluated malfunction of equipment important to safety. The methodologies employed in the Cycle 11 Reload Design have been reviewed and approved by the NRC.

The Wolf Creek Generating Station (WCGS) Cycle 11 reactor core is comprised of 193 fuel assemblies. The core inventory will consist of 1 Region 1 assembly, 20 Region 11 assemblies, 88 Region 12 assemblies, and 84 Region 13 assemblies. The single Region 1 assembly will be reused in Cycle 11 in the center core location. This assembly is the Westinghouse STANDARD fuel design, and was previously used in the Cycle 1 core. The 20 Region 11 assemblies are Vantage 5 Hybrid with Performance+ features. The Region 11 assemblies feature Improved Zircaloy-4 cladding, guide tubes, and grids. Eight assemblies in Region 12 are the Robust Fuel Assembly (RFA) design. The RFA design has been previously evaluated and found to be acceptable for use in reload core designs at WCGS. The remaining assemblies in Region 12 and 13 are the Westinghouse Performance+ V5H design with rotated mid-grids. The RFA and Performance+ V5H design are hydraulically identical. The Region 1 and other resident fuel is designed such that there is no difference in the pressure drop across the bottom nozzle, thus preventing any additional coolant crossflow when the Region 1 assembly is placed in an adjacent position to the Regions 11, 12 or 13 fuel. However, the Region 1 assembly does not have the intermediate flow mixing (IFM) grid and therefore the standard transition core Departure from Nucleate Boiling (DNB) penalties were applied in the reload analyses.

All the design features being used in the Cycle 11 Reload (i.e., Fully Enriched Annular Axial Blankets, ZIRLO IFM grids, Mid-span grids, Guide thimble, Fuel Clad, and Instrumentation Tubes) are justifiable under 10 CFR 50.59 and require no prior NRC approval or exemptions. Use of ZIRLO fuel cladding has been previously approved for use in the WCGS core by the NRC. A full safety assessment which supports the Cycle 11 Reload Design may be found in the Reload Safety Evaluation (RSE).

Safety Summary:

The evaluations of the proposed fuel modifications and analyses to support the Cycle 11 reload design confirm that they will not result in a potential unreviewed safety question, as defined in 10 CFR 50.59.

The Cycle 11 Reload Design affects and initiates appropriate changes to Chapters 3 and 4 of the USAR. These changes incorporate cycle specific values of design parameters which were determined using approved codes and methodologies. All cycle specific design values are bounded by licensing analyses reviewed and approved by the NRC.

Documentation for the Cycle 11 Reload Design includes analyses to insure that the interface criteria for the reload fuel continues to be met for all affected systems. Specifically, evaluations of interface criteria for the reload fuel and the reactor internals, fuel handling equipment, spent fuel storage, and Rod Cluster Control Assemblies (RCCAs) have been performed.

Impacts to Design Basis Accidents for all of the fuel types featured in the Cycle 11 Reload Design have been previously evaluated and found to be acceptable. There are no new fuel features incorporated as part of this design. Use of approved analysis methodologies and demonstrated equivalency of identified design parameters insures that the loading pattern developed for Cycle 11 has no impact to any Design Basis Accident. The Cycle 11 Reload Safety Evaluation documents the absence of any LOCA related issue associated with the design. Documented acceptability of all NON-LOCA safety analyses parameters demonstrates there is no impact to any NON-LOCA Design Basis Accident as documented in the USAR.

The safety evaluation presented in the Cycle 11 RSE and Cycle 11 Core Operating Limits Report (COLR) demonstrates that the Cycle 11 Reload Design does not create the possibility of malfunction of equipment important to safety other than any previously evaluated in the USAR. All design and performance criteria continue to be met and no new failure modes have been introduced for any system, component, or piece of equipment as a result of the design. The implementation of the Cycle 11 Reload Design will not impact either the normal plant operation or the response to accident conditions. Therefore, the Reload Design does not create the possibility of any new type of accident different from those already evaluated in the USAR.

The Cycle 11 Reload Design does not result in a different response of safety related systems and components to accident scenarios than those postulated in the USAR. No new equipment malfunctions have been introduced that will affect fission product barrier integrity. In addition, there is no affect on the mitigation of the radiological consequences of an accident described in the USAR . Therefore, the Cycle 11 Reload Design will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The Cycle 11 Reload Design does not create any failure mode that could adversely impact safety related equipment or cause the initiation of any accident. The proposed design does not result in any event previously deemed incredible being made credible. In addition, the safety function of safety related systems and components, which are related to accident mitigation, have not been altered. Therefore, the Cycle 11 Reload Design will not create the possibility of a malfunction of equipment important to safety different than those previously evaluated in the USAR.

The Cycle 11 Reload Design has no affect on the availability, operability, or performance of the safety-related systems and components and does not affect the plant Technical Specification requirements. The Cycle 11 Reload Design has no impact on inspections or surveillance required by the Technical Specifications.

The Cycle 11 Reload Design and COLR establish that all design and safety analysis limits continue to be met and that these limits are supported by the applicable Technical Specifications. Evaluation of the Cycle 11 Reload Design accounts for both normal operation and postulated accident conditions for the WCGS. The LOCA evaluation demonstrates that all 10CFR50.46 criteria are met. The non-LOCA safety analysis acceptance criteria remain unchanged and continue to be met. The core design parameters and assumptions incorporated in the safety analysis remain bounding. The margin of safety as defined in the BASES is not reduced for any USAR accident and thus, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0041 **Revision:** 0

Drawing Revision to Correct Component Designation for the Temperature Controls of Hot Water Heater

Description:

Essential Drawing M-12HB04, Revision 2, "P&ID Liquid Radwaste System," incorrectly identifies the temperature controls for the Washing Machine Hot Water Heater (THB-11) as TSHL-1091 and TE-1091. The as-built controls are TI-1091, TSH-1091, and TSHH-1091. The controls are used for indicating the hot water temperature in the heater and for high and high high temperature limit alarms.

The existing P&ID M-12HB04 (UASR Figure 11.2-1-04), Revision 02, identifies the temperature controls for the Washing Machine Hot Water Heater (THB-11) as TSHL-1091 and TE-1091. However, the as-built plant condition and the Total Plant Set Document (K05-004) show instrument TI-1091, TSH-1091, and TSHH-1091 for the water heater.

Safety Summary:

The subject heater is non-safety related. This change only corrects the drawing to show the as-built condition in the plant. No new instrument/component has been added. No existing instrument/component has been modified by this change.

The proposed activity is non-safety related and does not alter the physical condition of the plant. The original design basis will not be affected due to this drawing revision.

The proposed activity is non-safety related and no credible accidents are created. The existing physical condition of the subject water heater and the instruments are unchanged. No safety related equipment will be affected by the proposed change.

There are no acceptance limits in the technical specification nor in the licensing basis documents for Washing Machine Hot Water Heater (THB-11) that could be affected by the proposed activity.

Safety Evaluation: 59 1999-0042 **Revision:** 0

Removal of Auxiliary Steam System Feedwater Pump Gauges

Description:

The proposed change will remove Auxiliary Steam System Feedwater Pump suction pressure gauges (FBPI-0069, FBPI-0070) permanently from service. These gauges are throw-away type instruments and are located at the suction of the auxiliary steam feedwater pumps. These gauges are not required by operation or system engineering for the operation, or surveillance of the pumps. The gauges have to be maintained and calibrated like any other active equipment while serve no useful purpose for the plant operation or surveillance.

The proposed activities are as follows:

- Remove the pressure gauges (FBPI-0069, FBPI-0070).
- Close the Root Valves (V1001, and V1002).
- Plug or cap the instrument tubing with appropriate fittings.

The proposed change will result in the reduction of maintenance and calibration of the instruments that are not required.

Safety Summary:

The proposed change will affect Updated Safety Analysis Report (USAR) Figure 9.5.9-1, "Auxiliary Boiler System," and Figure 9.5.9-2, "Auxiliary Steam System".

The auxiliary steam feedwater pump is non-safety related. The removal of these gauges will not impact directly or indirectly the design basis accidents as discussed or referenced in the USAR Chapters 2, 3, 6, 9 ,or 15.

The removal of these gauges will not create any credible accidents. The operation of the non safety related auxiliary steam system will not be effected by the proposed activities.

The removal of these gauges will not affect directly or indirectly any safety related SSCs. Also, the operation of the auxiliary steam system will not be affected by the changes.

The operation of the non-safety related auxiliary steam system will not be affected by the proposed activities.

Safety Evaluation: 59 1999-0043 **Revision:** 0

Updated Safety Analysis Report Revision to Correct Typographical Errors

Description:

This revision to the Updated Safety Analysis Report (USAR) changes USAR Section 9.4.3.2.2, "Auxiliary Building HVAC, System Description," to correct typographical errors and make the text consistent with Piping and Instrument Diagrams (P&ID's), configuration control systems, and specifications. These changes are based on comments documented in Performance Improvement Request (PIR) 98-2608 which was initiated during the USAR Fidelity Review.
The fourth paragraph of USAR Section 9.4.3.2.2, Component Description, Page 9.4-36, the text, "the component cooling water pump room fan coil unit," is being changed to read "the component cooling water pump room fan coil units." This change corrects of a typographical error. There are two component cooling water pump rooms, each with a fan coil unit, and the word "unit" in the text of USAR Section 9.4.3.2.2 is being change to reflect a plural (i. e., "units").

In addition, this change provides consistency with Specification M-627A, "Dampers," which specifies the use of single-blade dampers for rectangular dampers less than 12-inchs in height. A review of P&IDs, Specification M-627A, and vendor drawings indicates that dampers were installed in accordance with the requirements of M-627A for flow control dampers. The Design Basis for the Auxiliary Building HVAC (GL) system does not address specific damper types. This change adds text to provides a more accurate description and does not change the plant nor the NRC's basis for approval of the plant license.

Safety Summary:

These USAR changes are for consistency and accuracy only. They do not affect any system, structure, or component (SSC), nor do they change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station.

No requirements outlined in the USAR are revised by these changes. No other USAR descriptions or conclusions will change or be made untrue as a result of these changes. No tests or experiments are involved with these changes.

These USAR changes correct typographical errors and maintain consistency between the USAR descriptions and specified configurations. Therefore, no design basis accident is affected and no new credible accidents could be created.

These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no credible Malfunctions of Equipment Important to Safety are identified.

These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no acceptance limits that could be affected are identified.

Safety Evaluation: 59 1999-0045 **Revision:** 0

USAR Figure Revision to Correctly Identify a Line Designation Number

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a change to Updated Safety Analysis Report (USAR) Figure 9.2-2-04, "Essential Service Water System." This figure is being revised to correctly identify a line designation number from "085-HBC-30" to "083-HBC-30".

Safety Summary:

The proposed activity described above is the correction of a typographical error and does not impact any procedures, activities, administrative controls or sequence of plant operations nor are any plant structures, systems, components or equipment affected. No requirements outlined in the USAR are revised by these changes.

No other USAR descriptions or conclusions are affected as a result of these changes. No tests or experiments are involved with these changes. No design basis accident is identified is affected by this change. This change has no impact on the performance of plant activities nor any affect on any system, structure or component. Therefore, no credible accidents that could be created are identified. This change makes no additional changes to the plant, does not affect the performance of plant activities and does not affect any SSC. Therefore, no credible malfunctions of equipment important to safety are affected.

There are no acceptance limits that could be affected by this change. Therefore, the margin of safety is not affected by this change.

Safety Evaluation: 59 1999-0046 **Revision:** 0

Technical Requirements Manual Revision 2

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Revision 2 to the Wolf Creek Technical Resource Manual (TRM).

Technical Requirement (TR) 16.1.2 will be changed to clearly require a flowpath from the Boric Acid Storage System and a flowpath from the Refueling Water Storage Tank (RWST).

TRM 16.1.1 Bases will be changed to incorporate information from the Westinghouse generated "Wolf Creek Boration Systems Design Report SAP-99-102, 2/1/99."

The Updated Safety Analysis Report (USAR) Fidelity Review identified that LCO TR 16.1.2, the operating condition boration flow path LCO, is not consistent with design and licensing basis documentation. The original Technical Specification wording was ambiguous, and was interpreted to allow the flowpath from the Boric Acid Storage System to be out of service with no ACTION statement applicable. After the Technical Specification became an Operational Requirement, it was changed by USAR Change Request 96-085 to remove the ambiguity, but reflected the erroneous interpretation. This change will provide wording that will clearly communicate the original intent of having an operable flowpath from each required Borated Water Source. The requirement for Borated Water Sources does not change.

Upon a Wolf Creek Nuclear Operating Corporation's (WCNOC) request, Westinghouse provided a collation of design basis information of the Chemical and Volume Control System (CVCS) Boration Systems in the Westinghouse generated report "Wolf Creek Boration Systems Design Report, SAP-99-102, 2/1/99." The TRM 16.1.1 Bases is revised to include information from this document, i.e., required boration systems safety functions, emergency and normal boration with flowpath requirements and solubility/precipitation issues, diversity and redundancy requirements, boron equivalency for Shutdown requirements, and the single failure capability and assumptions in the design of the boration systems.

The TRM sections being revised and affected include the following:
Section 16.1.2
Section 16.1.1 Bases

Safety Summary:

There are no plant procedures, activities, administrative controls, or sequences of plant operations that are impacted by the proposed changes.

As stated above, the proposed changes will correct an existing erroneous interpretation of the operating condition boration flow path LCO (TR 16.1.2) in the TRM presentation identified by the USAR Fidelity Review Team and expand the Boration Systems Bases description.

No credible accident is created by the proposed change. The proposed TRM changes removes an erroneous interpretation regarding the operating condition boration flow path LCO (TR 16.1.2) and expands the Boration Systems Bases description. No change to the plant or to the accident analyses is proposed.

No malfunction of equipment important to safety is impacted by the proposed TRM changes. No change to the plant or to the accident analyses is proposed. No acceptance limits are impacted by the proposed change. There is no impact on the analysis results, acceptance limits, or margin to the existing acceptance limits.

Safety Evaluation: 59 1999-0047 **Revision:** 0

Technical Requirements Manual Revision 2

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the Technical Requirements Manual (TRM) to remove information from the TRM Section 16.6.1.1.a that is duplicated in Wolf Creek Generating Station (WCGS) Technical Specifications.

WCGS Technical Specification 3.6.1.1 requires Containment integrity to be maintained in Modes 1, 2 , 3 and 4 or to restore Containment integrity within one hour. WCGS Technical Specification 1.7.e, requires the containment leakage rates, determined by Technical Specification 4.6.1.1.c are within defined limits to demonstrate Containment integrity. Technical Specification 4.6.1.1.c states that the Containment leakage rates shall be accordance with Technical Specification 6.8.4.i. This section requires that the leakage rates for Type B and C tests be less than 0.60 La and less that .75 La for Type A test prior to startup.

The Containment integrity information in the TRM is being removed to eliminate duplication of WCGS Technical Specifications. The TRM is revised to reference WCGS Technical Specifications for Containment integrity requirements.

Safety Summary:

The Wolf Creek Technical Specification imposes the same requirements as the TRM 16.6.1.1, and the TRM is being revised to reference WCGS Technical Specifications.

This change ensures that no other criteria can be implemented without violating WCGS Technical Specifications. Therefore, the accident analyses are supported by this change.

Since this change does not affect the operation or function of any systems, structures, or components, no credible malfunction of equipment important to safety is created.

This changes to the TRM do not change the acceptance limits that are contained in the bases for the technical specification. This change only makes reference to the WCGS Technical Specifications.

Safety Evaluation: 59 1999-0048 **Revision:** 0

Boron Recycle System Description Clarification

Description:

CCP 07991 revises System Description M-10HE, "System Description Boron Recycle System", to clarify when the Recycle Holdup Tanks should be vented. The specific changes to M-10HE are:

In M-10HE, Section 3.2.2., "Recycle Holdup Tank (RHT) Venting," Sheet 8, delete the last sentence in the second paragraph which states: "The recycle holdup tank should also be vented before and after a RCS loop drain or a drain from the Fuel Pool Cooling and Cleanup system (FPCCS) since these are not deaerated drains."

In M-10HE, Section 3.2.3.3, "Maintenance Drains," Sheet 10, delete the first paragraph which states: "When large amounts of water must be drained from the RCS or the spent fuel pool (or fuel transfer canal) to the BRS, the recycle holdup tank should be drained of water and vented to the gaseous waste processing system. The RHT's may then be used to store the drained water until maintenance is complete. After verifying that the chemistry is within specifications the water may be returned to the RCS or refueling canal, with no processing required. After returning the water, the gases collected under the tank diaphragm should again be vented."

Safety Summary:

The two recycle holdup tanks provide storage for radioactive fluid which is discharged from the Reactor Coolant System (RCS) during startup, shutdown, load changes, and boron dilution. Each tank has a diaphragm which prevents air from dissolving in the water and prevents hydrogen and fission gases in the water from mixing with the air. The volume in the tank above the diaphragm is continuously ventilated with building supply air, and any gas which accumulates below the diaphragm is intermittently vented to the gaseous waste processing system via the recycle holdup tank eductor.

Replacement diaphragms were installed in the recycle holdup tanks, THE02A and THE02B, by Plant Modification Request (PMR) 04714. Since the replacement of the diaphragms in the tanks, the operating procedures have evolved in order to ensure the maximum operating lifetime for the diaphragms in the recycle holdup tanks. These operating procedures are based on venting the recycle holdup tanks after a specified amount of water has been input to the tanks from sources of water that contain dissolved gases such as the CVCS and the RCS. In addition, these procedures specify that the level in the tank must be at the Recycle Holdup Tank Low Level Alarm setpoint prior to venting; specify a venting rate (1 SCFM) that is consistent with the system flow diagram M-11HE02;

and specify a minimum pressure of 0.5 in. H2O for tank and diaphragm protection.

Performance Improvement Request (PIR) 98-2405 was generated as part of the Updated Safety Analysis Report (USAR) Fidelity Review (SEL 97-044). This PIR noted that USAR Section 9.3.6.2.3, "Boron Recycle System, System Description," contained no procedural guidance to vent the Recycle Holdup Tanks (RHUTs) after draining them. A review of USAR Section 9.3.6.2.3, M-10HE and the operating procedures, determined that the USAR section and System Description were not consistent with the venting requirements detailed in the operating procedures. As detailed in the operating procedures, the Recycle Holdup Tanks are vented based on inputting a specified amount of water into the tanks from sources such as the Reactor Coolant System that contain dissolved hydrogen as well as fission gases. By venting the Recycle Holdup Tanks based on the amount of water input to the tanks instead of before and after certain evolutions such as RCS loop drains or draining from the spent fuel pool (or fuel transfer canal), the number of times the Recycle Holdup Tanks need to be vented should be reduced which should extend the operating lifetime of the diaphragms installed in the tanks.

A USAR change is required to make the same changes described above for System Description M-10HE to USAR Section 9.3.6.2.3, System Operation. There is no field work associated with CCP 07991. The changes to M-10HE and USAR Section 9.2.6.2.3 will make the description of recycle holdup tank venting consistent with that used in the plant operating procedures to ensure the maximum operating lifetime for the diaphragms in the tanks. These changes do not affect any system, structure, or component (SSC), nor do they change the performance of activities that are important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

The current descriptions of venting operations for the recycle holdup tanks contained in System Description M-10HE and USAR Section 9.2.6.2.3 indicates that the tanks are to be vented after a sufficient amount of water has been passed to the recycle holdup tanks to require venting of accumulated gases. These descriptions are consistent with the current plant operating procedures for venting the recycle holdup tanks. However, both M-10HE and USAR Section 9.2.6.2.3 state that the recycle holdup tanks should be vented before and after a RCS loop drain or a drain from the spent fuel storage area (or fuel transfer canal). The venting requirements for the recycle holdup tanks in the plant operating procedures are based strictly on the amount of water input to the tanks from sources that contain dissolved gases. These venting requirements are based on ensuring the maximum operating lifetime for the tank diaphragms. No tests or experiments not described in the USAR are involved with these changes. No other USAR descriptions will change or be made untrue by these changes.

The changes proposed by CCP 07991 will make the description of the venting requirements for the recycle holdup tanks consistent with that utilized in the plant operating procedures. As detailed in the operating procedures, the Recycle Holdup Tanks are vented based on inputting a specified amount of water into the tanks from sources such as the Reactor Coolant System that contain dissolved hydrogen as well as fission gases. By venting the Recycle Holdup Tanks based on the amount of water input to the tanks instead of before and after certain evolutions such as RCS loop drains or draining from the spent fuel pool (or fuel transfer canal), the number of times the Recycle Holdup Tanks need to be vented should be reduced which should extend the operating lifetime of the diaphragms installed in the tanks. A USAR change will also be required to make the description of the venting requirements for the recycle holdup tanks consistent with that used in the plant operating procedures. There is no additional impact on the performance of plant activities nor affect on any SSC.

Based on the above discussion, no design basis accident is affected by this change.

Based on the above discussion, no credible accidents could be created by this change.

Based on the above discussion, no credible malfunctions of equipment important to safety are created by this change.

Based on the above discussion, there is no impact on the performance of plant activities nor any affect on any SSC resulting from this change. Therefore, no acceptance limits are identified that could be affected and the margin of safety is not impacted by this change.

Safety Evaluation: 59 1999-0049 **Revision:** 0

Auxiliary Building Ventilation Changes Resulting From the USAR Fidelity Review

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a change to Updated Safety Analysis Report (USAR) Section 9.4.3.1.2, "Auxiliary Building, Design Bases." This change resulted from the USAR Fidelity Review.

The auxiliary building ventilation system maintains the auxiliary building sample room, the hot machine shop, and the hot instrument shop between 60° F and 85° F. This statement is to be revised to read: "The auxiliary building ventilation system maintains the auxiliary building sample room between 60° F and 104° F; the hot machine shop, and the hot instrument shop between 60° F and 85° F."

The hot machine shop sensing instrument, GL TS 0036B, and instrument hot shop sensing instrument, GL TS 0036A, have set points of 80° F, which are within the limits of 60° F and 85° F, stated in USAR Section 9.4.3.1.2. These settings do not represent a change to the operating temperature of the hot machine shop or the instrument hot shop sensing, as described in the USAR. No other sections of the USAR are impacted by this change.

The containment isolation valves are located outside the sample room. Therefore, a change to the sample room temperature does not impact a safety design basis.

The existing set point of the thermostat (GL TS 0024) for Ground Floor Fan Cooling Unit, SGL05, represents an increase in the maximum operating temperature of the auxiliary building sample room, as described in USAR Section 9.4.3.1.2. However, this setting is within the normal operating temperatures described in USAR Table 3.11 (B) -1. No other USAR descriptions or conclusions are affected as a result of proposed change. No tests or experiments are involved with the proposed change.

The auxiliary building sample room operating temperature does not affect any design basis accidents discussed or referenced in USAR Chapters 2,3,6,9 or 15. Therefore, no design basis accident is identified for review.

The auxiliary building sample room operating temperature does not affect the creation of any credible accidents. Therefore, no credible accidents that could be created are identified.

The auxiliary building sample room operating temperature does not support or impact equipment important to safety. Therefore, no credible

Malfunctions of Equipment Important to Safety are identified.

The auxiliary building sample room operating temperature is not related to any acceptance limits which are contained in the bases for the technical specifications or in licensing basis documents. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1999-0050 **Revision:** 0

Assessment of Combustible Loading for Plant Fire Areas

Description:

This change revises Updated Safety Analysis Report (USAR) Section 9.5B, "Fire Hazards Analysis," and Section 9.5.1 "Fire Protection System," to incorporate information provided by Calculation XX-X-004, "Combustible Loading for Each Room in the Various Fire Areas at WCNOC." Calculation XX-X-004 is a new calculation which will provide an accurate accounting of the combustible fire loads in the plant Fire Areas included in USAR section 9.5B. The calculation identifies the significant combustible materials on a room or area basis. The combustible loading provides a quantitative assessment of the fire severity in the fire area in the event of an exposure fire involving in-situ combustibles. The USAR description of the fire load severity is being revised so that it is defined in terms of a range of British thermal units/square feet (Btu/Sq. Ft.) in all USAR 9.5.1, "Fire Protection System," sections, tables, and appendices.

In conjunction with issuance of Calculation XX-X-004, a USAR change is being made to delete the Fire Hazards Analysis (FHA) reference to materials and specific Btu/Sq. Ft. values for all fire areas in USAR Section 9.5B.7. The fixed combustibles fire loading in each room will be changed to describe the fire load based on the three classifications discussed above. The classification level identified in the USAR will correspond with the combustible loading for that room or area as identified in Calculation XX-X-004. This change will alleviate the need to change the USAR each time minor amounts of combustible materials are added to or subtracted from a fire area.

Safety Summary:

Review of the USAR and performance of a plant walkdown verify that there are no areas that exceed the three-hour rating for fire barriers. Therefore, this change will not increase the probability of an accident. Neither will it increase the radiological consequences of an accident.

Review of the USAR and performance of a plant walkdown verify that the combustible material in the various rooms in the plant did not exceed the upper range for the areas. Therefore, the probability of occurrence of a malfunction of equipment important to safety is not increased. The existing analysis already assumes the loss of all equipment for any fire scenario.

There is no change in the in the probability of, or the effects of a malfunction of equipment important to safety as a result of this change. Therefore, there is no potential to increase the radiological consequences of an equipment malfunction.

A fire hazards analysis has been performed for each room and documented in the USAR. No material is introduced to the fire areas that has not been evaluated. Therefore, this change will not create the possibility of a different type of accident than any previously evaluated.

Based on the FHA, this change will not create a different type of malfunction of equipment important to safety tan any previously evaluated in the USAR.

The change in wording for the fire load in the various rooms to does not reduce the margin of safety as this wording is consistent with the values provided for fire loading in the National Fire Protection Associations Fire Protection Handbook, Sixteenth Edition.

Safety Evaluation: 59 1999-0051 **Revision:** 0

Evaluation of Electrical Penetration Cooler Out of Service

Description:

This Unreviewed Safety Question Determination (USQD) provides an evaluation of the non-conforming condition (past and future) of having an Electrical Penetration Room Cooler out of service. This evaluation is not a change to the design or licensing basis but rather an evaluation of a non-conforming condition. Configuration Change Package (CCP) 07986 provides a detailed description of the limiting conditions and evaluation.

The penetration room coolers (SGL015A and SGL015B) are located in Rooms 1409 and 1410 respectively on elevation 2026 of the Auxiliary Building. This CCP provides the bounding conditions for the electrical penetration room cooler operation which will insure that the specified safety function of the supported equipment will be unaffected if the penetration room coolers become non-functional during normal plant operation, for example during planned or emergent maintenance activity.

The Updated Safety Analysis Report (USAR) Section 9.4.3 "Auxiliary Building" suggests that each electrical penetration room is provided with a room cooler and it is a required component. For example, USAR page 9.4-42 states "Operation of the penetration room coolers is controlled by a handswitch or SIS." USAR section 9.4.3 states "Operation of the penetration room coolers is controlled manually or automatically by a safety injection signal (SIS)". This CCP would allow a penetration room cooler to not start upon an SIS if it were taken out of service under the limiting conditions of this CCP. USAR Table 9.5B-2, "Equipment Required for Shutdown Following a Fire" indicates that both SGL15A and SGL15B are required components. This CCP disposition would allow a penetration room cooler to not be available to start upon an SIS if it were taken out of service under the limiting conditions of this CCP. USAR section 9.4.3.3 states.." SAFETY EVALUATION TWO - The safety-related portions of the auxiliary building HVAC systems are designed to remain functional after a SSE". This CCP disposition would allow a penetration room cooler to not be functional following an SSE if it were taken out of service under the limiting conditions of this CCP.

Safety Summary:

This change is not a change to the design/licensing basis because the supported systems safety functions will be unaffected during normal operation, Safe Shutdown Earthquake (SSE) or Design Basis Accident (DBA). It is a temporary limited condition of operation allowed by the guidance provided in NRC Generic Letter 91-18 "Information to Licensees Regarding Two NRC Inspection Manual Sections On Resolution Of Degraded And Nonconforming Conditions And On Operability," Section 6.12 which provides guidance allowing licensees to make determinations as to which support equipment is required under various conditions.

The Loss of Coolant Accident (LOCA) was reviewed. The USAR section 9.4.3.2 discusses operators taking manual actions in the south penetration room following a LOCA. The ability to access the room by operators is unaffected by this CCP. Safe shutdown capability following a fire was also reviewed. The electrical cable separation design will be unaffected by this CCP. The fire protection function of the room coolers is the same safety function as the normal operation & DBA function and that is to maintain room ambient temperature. Given the limiting conditions of this disposition the room ambient temperature will be maintained and the supported systems unaffected.

Given the limiting conditions of this disposition the required cooling function of the room cooler is not required as support equipment and thus the safety related equipment located in the affected room will still perform its specified safety function. No new credible accidents are created by this operating condition.

A failure of a penetration room cooler has already been taken into consideration in the Auxiliary Building HELB analysis. No credit is taken in the analysis for the cooling function of the room cooler. Thus, no credible malfunction of equipment is affected.

There are no Technical Specifications that specifically address an out of service penetration room cooler. Technical Specification 3.4.3 addresses the inoperability of the pressurizer back-up heaters. The basis of this specification is unaffected by this disposition. The Technical Requirements Manual (TRM) section 16.7.4 addresses area temperature monitoring. This disposition imposes more restrictive area temperature requirements than what is specified in the TRM. The basis of the TRM are unaffected by this CCP.

The specified safety functions are unaffected by this CCP. Therefore, this CCP does not increase the probability of any accident, nor are the radiological consequences of any accident affected. Neither is the probability of occurrence of a malfunction increased, nor are the radiological consequences of a malfunction of equipment increased. This CCP does not create an accident of a different type and does not create a different malfunction of equipment. Because the specified safety functions are unaffected, this CCP has no impact on any margin of safety defined in the Technical Specification bases.

Safety Evaluation: 59 1999-0052 **Revision:** 0

Temporary Modification to Stator Cooling Instrumentation

Description:

This Temporary Modification will prevent a Turbine Trip from an erroneous Stator Cooling Flow/Current comparitor. The Stator Cooling system instrumentation is providing a false low stator inlet flow indication which actuates protective action signals to the turbine circuitry. Troubleshooting has identified that the stator cooling flow square root extractor (CEFY0016B) is indicating erroneously low. The flow transmitter that CEFY0016B feeds is associated with the Stator Cooling Current Flow Comparator-Turbine Runback (CEFSL0016), and feeds both flow alarms and turbine runback circuitry. A false low flow alarm signal is locked in from this component and intermittent turbine runback signals are being generated. The turbine has been shifted to standby control to prevent actual reduction of load during the runback signals. A field wire in AC-119 bay 4 at ITB4-J404 terminal 11 has been lifted to disable the loss of stator cooling turbine trip that will be generated if a stator cooling runback signal occurs and locks in. This temporary modification is to allow this signal to be defeated until troubleshooting and repairs are complete and confidence in the stator cooling system protective circuitry has been demonstrated.

Safety Summary:

Updated Safety Analysis Report (USAR) Section 10.2.2.3.4, "Turbine Disk Design," lists loss of stator coolant as one of many turbine trips. Lifting of this field wire will defeat the turbine trip for loss of stator coolant.

USAR Section 15.2, "Decrease in Heat Removal by the Secondary System," discusses, decrease in heat removal by the secondary system accidents. Turbine Trip is identified in Section 15.2.3, "Turbine Trip," as one of the initiating events to this type of accident. Turbine Trips are considered Condition II - Faults of Moderate Frequency. Defeating the turbine trip on loss of stator coolant will not change the frequency class of a turbine trip to a Condition I - Normal Operation and Operational Transients. Rather, defeating the turbine trip on loss of stator coolant decreases the probability of having a design basis accident associated with a turbine trip.

The non-safety related loss of stator water cooling trip was installed by the turbine manufacturer for commercial reasons. Therefore, this change is not related to any of the events or conditions associated with the initiation of credible accidents.

There is no equipment important to safety associated with the non-safety related stator cooling turbine trip circuitry. This circuitry was installed by the turbine manufacturer for commercial reasons. Therefore, this change does not pose any credible malfunctions to any equipment important to safety, either directly or indirectly.

There are no acceptance limits associated with stator cooling turbine trip circuitry in the technical specifications or licensing basis documents.

Safety Evaluation: 59 1999-0053 **Revision:** 0

Updated Safety Analysis Report Revision for an Organization Change

Description:

The change described in this Updated Safety Analysis Report (USAR) change reflects the designation of the Plant Manager as a member of the board of directors for Wolf Creek Operating Corporation and as an officer of the company by adding Vice President Plant Operations to the title Plant Manager. In addition, some places in the USAR where the position of Plant Manager have been designated as responsible will be changed to lower case to indicate the position rather that the title/person. These places are those that are a reflection of the responsibilities described in the Technical Specifications.

There are several places in the USAR where the Plant Manager is designated as responsible for certain activities. These references will be revised to indicate the new title or be revised to be "plant manager."

Safety Summary:

The responsibilities of the Plant Manager are delineated in USAR Chapters 12, 13, 17 and 18. This title change will not affect physical plant equipment nor pose a test or experiment.

Design bases accidents do not depend on the title of the person filling the role of Plant Manager. Since the Plant Manager's responsibilities have not changed, no accidents are affected.

Since no responsibilities are changing, there are no credible accidents created.

Equipment operation does not depend on the title of the person filling the role of Plant Manager. Since the Plant Manager's responsibilities have not changed, no equipment malfunctions are directly or indirectly affected by the title change.

There is no change in responsibilities of the plant manager and communication paths remain the same. Therefore, what has been accepted by the NRC is not affected.

Safety Evaluation: 59 1999-0055 **Revision:** 0

Updated Safety Analysis Report Change to the Hydrogen Purge System

Description:

This change to the Updated Safety Analysis Report (USAR) removes paragraph 6.2.5.2.2.4 from the USAR. Paragraph 6.2.5.2.2.4 states:

"The hydrogen purge subsystem is used on a limited basis to maintain containment atmospheric pressure within Technical Specification limits when the containment shutdown and/or mini-purge system are out of service. When this is done, administrative controls are placed on the containment isolation valves to ensure closure in the event of a Loss-of-Coolant Accident. The flow path is the same as that described for accident operation above."

The Performance Improvement Request (PIR) 98-3833 evaluated this condition and concluded that if the Hydrogen Purge System was used as a containment pressure reduction system, then it should be included under the Purge Valve Technical Specification 3.6.1.7. This was reported to the NRC under LER 86-073-00, on January 18, 1999. Part of the corrective action for LER 86-073-00 is to remove this paragraph from the USAR.

Safety Summary:

This change will bring the USAR into compliance with the operating license. All chapters were reviewed and no accidents were identified that could be potentially impacted by the proposed activity.

Removing the function from the USAR of using the Hydrogen Purge System as a Purge System can not be related to any credible accidents. No credible accidents are affected.

Removing the function from the USAR of using the Hydrogen Purge System as a Purge System can not be related to any credible malfunctions. No credible malfunctions are affected.

The paragraph identified above violates the bases for Technical Specification 3.6.1.7. Deleting the paragraph will bring the USAR in compliance with the Technical Specifications. Therefore, the margin of safety is not negatively impacted by this change.

Safety Evaluation: 59 1999-0058 **Revision:** 0

Technical Requirements Manual Revision

Description:

The Technical Requirements Manual (TRM) is being revised to delete the "reject" line from Figure 16.7-1 for the 37 snubber test plan. The reference to the "reject" line will also be eliminated from TRM Surveillance Requirement 16.7.2.1.1e.2 and Bases Section 16.7.2.1.2. The reference to the 55 snubber test plan will be deleted from TRM Surveillance Requirement 16.7.2.1.1e.3 and TRM Bases Section 16.7.2.1.2. The 10 percent snubber test plan contained in TRM Surveillance Requirement 16.7.2.1.1e.1 and Bases Section 16.7.2.1.2 will be revised to require subsequent sample sizes to be at least 5 percent of the total population of a snubber type for each snubber test failure instead of the current 10 percent requirement. Also, the concept of "Failure Mode Groups" (i.e., focusing on the specific failure mechanisms) will be incorporated in TRM Surveillance Requirement 16.7.2.1.1e. These changes are consistent with Subsection ISTD of ASME OM Code-1995 Edition with the 1996 Addenda.

Background:

The Snubber Program requirements and surveillances were moved from the Technical Specifications to Chapter 16 of the USAR per Operating License Amendment Number 89. This was accomplished to improve plant technical specifications as endorsed by the NRC in its Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, 58FR39132, July 22, 1993. Subsequently, the requirements and surveillances contained in Chapter 16 of the USAR were relocated to the Technical Requirements Manual. The information contained in the TRM is considered part of the USAR and is thereby controlled as such with changes controlled per Administrative Control Procedure AP 26A-002, "Implementation and Revision of the Technical Requirements Manual".

Relief Request I2R-15 for the Second Interval ISI Program Plan allowed Wolf Creek to implement the Snubber Inspection and testing requirements contained in the Technical Requirements Manual in lieu of the requirements of OMa-1988, Part 4. In response to NRC questions about the process for future changes to the Snubber Program, WCNOC submitted letter ET 95-0126, dated November 17, 1995. This letter stated, in part, that any changes made to the snubber inspection program would be made in accordance with the requirements of 10 CFR 50.59. The NRC subsequently approved Relief Request I2R-15 on October 24, 1997.

The 37 snubber test plan option described in TRM Surveillance Requirement 16.7.2.1.1e.2 requires that a representative sample of snubbers be tested periodically in accordance with Figure 16.7-1. Figure 16.7-1 provides the acceptance criteria for the functional test results and denotes a "reject" region and a "continue testing" region. If at any time the plotted functional test results fall within this "reject" region, then all snubbers are to be functionally tested. The 37 snubber test plan is the current plan utilized at Wolf Creek. The proposed change would eliminate the "reject" line from Figure 16.7-1 and its reference in TRM Surveillance Requirement 16.7.2.1.1e.2 and Bases Section 16.7.2.1.2. Figure 16.7-1 was developed using "Wald's Sequential Probability Ratio Plan", as described in "Quality Control and Industrial Statistics," by Acheson J. Duncan. As long as the "reject" line remains in the 37 snubber test plan there is the possibility of requiring an unnecessary 100 percent functional testing of snubbers. This change will alleviate this possibility and still ensure continued or additional testing if snubber quality or failed snubbers is equal to or greater than 5 percent. This change will also make the 37 snubber test plan consistent with the 10 percent snubber test plan since the "reject" line is not contained in the 10 percent snubber test plan. Also, the "reject" line is not contained in Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants," of American Society of Mechanical Engineers (ASME) OM Code-1995 Edition with the 1996 Addenda, since no substantial basis was identified for its existence. The NRC has endorsed Subsection ISTD of ASME OM Code-1995 Edition with the 1996 Addenda through a proposed change to 10 CFR 50.55a (62FR63892) which would revise the requirements for construction, inservice inspection, and inservice testing of components.

The 55 snubber test plan option described in TRM Surveillance Requirement 16.7.2.1.1e.3 and Bases Section 16.7.2.1.2 requires that an initial representative sample of 55 snubbers be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The 55 snubber test plan is being eliminated from TRM Surveillance Requirement 16.7.2.1.1e.3 and TRM Bases Section 16.7.2.1.2 since the Wolf Creek Snubber Program does not utilize the 55 snubber test plan. The plan requires additional unnecessary testing of snubbers which would result in additional outage times, maintenance costs, and unnecessary worker radiation exposure. Also, other acceptable plans will still remain for functional testing of snubbers (37 and 10 percent snubber test plans). The 55 snubber test plan is not a Wald sequential plan and is not contained in Subsection ISTD of ASME OM Code-1995 Edition with the 1996 Addenda.

The 10 percent snubber test plan option described in TRM Surveillance Requirement 16.7.2.1.1e.1 and TRM Bases Section 16.7.2.1.2 requires 10 percent of the snubbers to be tested periodically. It also requires testing of an additional 10 percent of the snubbers for each snubber not meeting the acceptance criteria of Specification 16.7.2.1.1f. Although this plan is not currently utilized at Wolf Creek, it could potentially be used in the future for Failure Mode Groups. The proposed change modifies this plan to require only an additional 5 percent sample group for each snubber that fails functional testing. This change would make the 10 percent snubber test plan and the 37 snubber test plan consistent concerning the requirements for acceptability of snubber performance (i.e., the slope of the acceptance line would be consistent). For example, with the identified change to the 10 percent snubber test plan, a group of 370 snubbers would require the same number of additional snubbers to be tested for each snubber not meeting the acceptance criteria as that of the 37 snubber test plan. The difference in the quantity of snubbers in subsequent sample groups between the 37 snubber test plan and the 10 percent snubber test plan becomes greater with smaller general sample populations. Reducing the percentage of snubbers to be tested does not undermine the effectiveness of this surveillance since the initial test sample remains the same and is sufficient to provide an adequate sampling of the snubbers. In the future, this change could reduce the amount of additional testing required and thus reduce man-rem exposure and safety concerns associated with unnecessary functional testing. The original form of the 10 percent plan was not based on statistics so adaptation of the revised form will allow this plan to be consistent with current industry practices. Also, this change is consistent with Subsection ISTD of ASME OM Code-1995 Edition with the 1996 Addenda which has been formally endorsed by the NRC.

TRM 16.7.2.1.1.e currently requires that if during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing. TRM Surveillance Requirement 16.7.2.1.1e would be revised to include the concept of "Failure Mode Groups". The change will allow the option of the categorization of unacceptable snubbers into "Failure Mode Groups". A test failure mode group would include all unacceptable snubbers that have a given failure mode and all other snubbers subject to the same failure mode. It allows independent testing of failure mode groups based on the number of unacceptable snubbers and requires one additional test sample group from the general population for each failure mode group to provide assurance that failure mode groups have been properly established. The concept of "Failure Mode Groups" permits not only the correction of deficiencies related to specific types, but also other deficiencies related to location, environment, service, etc., regardless of the type of snubber. This change is also consistent with Subsection ISTD of OM Code-1995 Edition with the 1996 Addenda which has been formally endorsed by the NRC.

Safety Summary:

As discussed above, the Technical Requirements Manual, which is considered part of the USAR, will require revision. The above changes will have no affect on any current procedures or any other information contained in the USAR.

The proposed changes to the Technical Requirements Manual do not affect any of the design basis accidents discussed or referenced in USAR Chapters 2, 3, 6, 9 or 15. No plant structures, systems, components or equipment are affected by this change.

The proposed changes to the Technical Requirements Manual do not affect the operation of any SSC as described in the USAR, and therefore will not create any type of credible accident.

Since the proposed changes do not affect the operation or function of any SSC as described in the USAR, no credible malfunction of equipment important to safety are identified.

This change to the Technical Requirements Manual does not change the acceptance limits which are contained in the bases for the technical specifications. This change will have no affect on the criteria that define the performance of the fission product barriers.

Safety Evaluation: 59 1999-0059 **Revision:** 0

Updated Safety Analysis Report Change to Correct Discrepancies

Description:

This change to the Updated Safety Analysis Report (USAR) revises USAR Section 10.4.3.2.2, "Turbine Gland Sealing System, System Description," to clarify the location of the steam packing exhauster discharge. The exact location of discharge is inaccurate. The USAR is revised to read: "The steam packing exhauster is maintained at a slight vacuum by a motor operated blower, which discharges to the atmosphere."

In addition, USAR Section 11.3.2.1, "System Descriptions, General Description," states that the "steam packing exhaust discharges into the turbine building." This is also incorrect and will be changed to "... discharges outside the turbine building." Figures 11.3-2 and 11.1A-3 that illustrate the potential gaseous release points in a block diagram format will be corrected accordingly.

Safety Summary:

Since the original assumption in USAR Section 11, "Radioactive Waste Management," correctly assumed the turbine building ventilation system is an open system (i.e., effluents are not filtered) the calculation results in this section are not impacted. No other related discrepancies were found in the USAR.

The loss of the function of Turbine Gland Seal System may cause a loss of condenser vacuum, which if severe enough could cause a turbine trip due to loss of condenser vacuum. This is a Condition II fault of moderate frequency described in USAR Section 15.2, "Decrease in Heat Removal by the Secondary System,". However, the discrepancies noted in the location or routing of the steam packing exhauster discharge does not affect design function of the system, failure modes, or reliability. Therefore this design basis accident is not impacted by this proposed change.

No other design basis accidents were identified that could potentially be impacted by the Turbine Gland Seal System.

Since the design basis function of the system is not affected by this change, no new types of accidents not previously analyzed could be created.

Since the proposed change would not affect the system's failure modes, the systems design function, the level of qualification, or equipment important to safety; no credible malfunctions of equipment important to safety are affected

Since no acceptance limits were identified that could be affected, this change does not create a reduction in the margin of safety.

Safety Evaluation: 59 1999-0060 **Revision:** 0

Updated Safety Analysis Report Change to Describe Location of ALARA Coordinator

Description:

This change to the Updated Safety Analysis Report (USAR) provides for a change to the physical location of the ALARA Coordinator, as described in USAR Section 12.5.2.2, "Health Physics Facilities," to the first floor of the Administration Building. USAR section 12.5.2.2, states "The ALARA Coordinator is located on the 1984' elevation of the Control Building at Access Control." The sentence in the USAR is changed to describe the location of the ALARA Coordinator as being on the first floor of the Administration Building.

Safety Summary:

Moving the location of the ALARA Coordinator allows for improved communication/interaction with other work groups for better planning of work activities. The change does not affect the responsibility of the individual. Therefore, there are no effects on plant processes, functions, or systems as a result of the proposed change. There is no other information in the USAR that conflicts with this change.

There are no design basis accidents identified in USAR Chapters 2, 3, 6, 9, or 15 affected by the proposed change. Moving the physical location of the ALARA Coordinator to the Administration Building does not impact the Radiation Protection Program, which ensures the radiation doses to personnel and to the public are maintained as low as reasonably achievable by conforming to 10 CFR 20 regulations. The radiological consequences for postulated accidents are not affected by the proposed change.

No credible accidents could be created by changing the location of the ALARA Coordinator. The responsibilities of the ALARA Coordinator remains the same and meets all applicable Technical Specification requirements, as outlined in the Radiation Protection Manual. Changing the location of the ALARA Coordinator will not directly or indirectly affect the ability of a system, component or structure important to safety to perform its design function or induce a credible malfunction of that equipment. The proposed change does not affect the exposure limits as set forth in 10 CFR 20. Therefore, acceptance limits are not affected.

Safety Evaluation: 59 1999-0061 **Revision:** 0

Installation of a 150 kW Standby Diesel Generator and Switchgear in the Wolf Creek Switchyard

Description:

Western Resources is installing a 150 kW emergency generator (D/G) and dual automatic throwover (ATO) switches in the Wolf Creek Switchyard. This D/G addition will enhance the Wolf Creek switchyard's ability to maintain 208/120 volt three-phase AC power to the 345 kV Control Building and associated Load Centers. The generator will remain inactive until a loss of AC power to the transformer occurs. The generator is designed to extend the ability of the switchyard to restore offsite power during an extended system blackout.

The dual automatic throwover (ATO) switch, also known as a three-source automatic transfer system, that was installed on DCP 07092 is an ASCO Catalog No. 530A65C. It contains two ASCO 940 automatic transfer switches. As in all ASCO transfer switches, each ATO is a true double-throw, inherently interlocked switch; that is, the switch contacts are always closed on one source or the other and never in-between. Therefore, according to its design, no off position is credible. Also, the ATO switch is designed with wide main contact air gaps between sources to assure total isolation during inductive load interruption to 600% of rating.

The existing ATO was designed to switch between two 208/120 VAC station power transformers, with loss of the preferred transformer initiating throwover of all 208/120 volt loads in the 345-kV Control Building and associated Load Centers to the standby transformer. Western Resources has had problems at other locations with a similar type of ATO switch, and for this reason has replaced it with the above described dual automatic throwover switch.

In the new design, the first automatic throwover switch is an ATO between the preferred and standby sources, and the second is an ATO between the standby and standby diesel generator sources. The preferred, or normal, transformer is fed from an offsite 13.8 kV bus, which is designated as "SWYD" in the simplified sketch below. The standby transformer is fed from an onsite 13.8 kV bus, and is designated as "PLANT".

Upon review of the Updated Safety Analysis Report (USAR), only one sequence of plant operations would be affected by implementation of this modification. The description of automatic throwover of all switchyard load in USAR Section 8.2.1.3, "Compliance with Design Criteria and Standards," would need to be expanded to include the second automatic throwover switch and the standby diesel generator.

USAR Figure 8.2-3, "Electrical One-Line Diagram of Wolf Creek 345 kV Switchyard and Adjacent Subs," must be updated to indicate the revised location of the Station Power Throwover. Finally, USAR Figure 8.2-4, "Ultimate One-Line Diagram," which indicates the arrangement of the existing automatic throwover switch, must be revised to show the second automatic throwover switch and the standby diesel generator.

Safety Summary:

The installation, operation, testing and maintenance of the diesel generator and dual automatic throwover switch will not require any tests or experiments that may adversely affect the adequacy of systems, structures, or components (SSCs) to prevent accidents or mitigate the consequences of an accident.

If the second ATO were to fail closed in its normal position, its effect would be the same as if the new diesel generator had not been installed. The standby power for the AC switchyard loads would be fed from the onsite 13.8 kV bus. If the second ATO were to fail closed in its emergency position, its effect would be to start the diesel generator and have it continuously provide a source of power. In either case, no new failure modes are introduced. The reliability of offsite power is not affected.

In conclusion, the new design removes the single problematic ATO switch and installs a standby (emergency) diesel generator with a three-source automatic transfer system, containing two ATO switches of proven reliability.

USAR Chapter 3, "Design of Structures, Components, Equipment, and Systems," defines the scope of SSCs that are required to have adequate missile protection. The location of the proposed activity is in the 345 kV switchyard, which is well outside the analyzed areas.

USAR Section 6.2, "Containment Systems," assumes various accident conditions for containment systems. The proposed activity is situated remote from any containment system. Any normal or abnormal operating, testing or maintenance activity involving the proposed change could not impact any containment system. There are no direct or semi-direct electrical, mechanical, physical, chemical, radiological, or thermal connections between the switchyard and areas of the plant closely monitored or analyzed for containment system accidents.

The same reasoning as used for USAR Section 6.2 above can also be applied to USAR Section 6.3 "Emergency Core Cooling System", USAR Section 6.4 "Habitability Systems", and USAR Section 6.5 "Fission Product Removal and Control Systems". None could be impacted by the proposed modification.

Accidents in USAR Chapter 9, "Auxiliary Systems," pertain to the plant auxiliary systems. There is no connection with the switchyard modification in these systems.

The accident analyses for Wolf Creek Generating Station (WCGS) contained within USAR Chapter 15 assumes a loss of offsite power coincident with certain postulated events, such as a Loss of Coolant Accident (LOCA), and with certain abnormal conditions, such as a turbine or reactor trip. A loss of offsite power is analyzed throughout the USAR. Although all references to loss of offsite power were reviewed, no situation was found where the meaning would be changed by the proposed activity.

No accident analysis is impacted by the propose activity. Therefore, the probability of occurrence of any previously evaluated accident is not increased by the addition of a small diesel generator in the Wolf Creek Switchyard. Neither has any accidents of a different type than those previously analyzed in the USAR been identified

There are no design basis accidents discussed or referenced in the USAR that were impacted by the proposed activity. Therefore, there are no radiological consequences associated to the addition of a small diesel generator in the Wolf Creek Switchyard.

No malfunctions of equipment important to safety were identified. Therefore, there will be no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. Neither is there any potential to create the possibility of a different type of malfunction of equipment important to safety than any previously analyzed in the USAR or increase the radiological consequences of a malfunction of equipment previously evaluated in the USAR.

There is no direct or indirect affects on any equipment important to safety. Therefore, the proposed activity will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The margin of safety, as defined by the bases for any technical specification, is not reduced by the addition of a small diesel generator in the Wolf Creek Switchyard.

Safety Evaluation: 59 1999-0062 **Revision:** 0

Evaluation of Procedure MGM GT-004

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Revision 2 of MGM GT-004, "Temporary Installation of Containment Purge Exhaust Filter," which will allow a larger filter unit to be installed on the shutdown purge exhaust system.

Procedure MGM GT-004 provides instructions for installation and removal of a temporary filter unit installed over the inlet of the containment purge exhaust system during plant outage activities. This filter prevents dust and dirt from outage work activities from entering containment purge exhaust system which reduces the amount of work required to clean the containment shutdown purge valve seats prior to entering Mode 4. The dust and dirt that enters the duct can affect the leak tightness of the purge valves which is required in Modes 1 through 4 by Technical Specifications. The installation of the filter unit is restricted to Modes 5, 6, and 6 Defueled.

Safety Summary:

This procedure allows two different filter housings to be installed on the exhaust path of the shutdown purge system. The heaviest of the two is under 200 pounds. The pipe that this filter housing is mounted to will not be affected since 200 pounds is less than the blind flange that is normally installed. The blind flange is removed prior to the filter unit being installed. Nothing is located below the installation point of the filter unit that could be affected if the filter unit should fall.

The performance of this procedure is being limited to Modes 5, 6, and 6 Defueled. Since this filter housing is being installed in the containment, only accidents that can occur in Modes 5 and 6 were reviewed. No equipment in containment is required in Mode 6 Defueled..

The safety function of the system is to isolate on a Containment Purge Isolation Signal (CPIS). The single failure analysis for the isolation valves is a purge valve fails to close. A second valve is provided in series to ensure containment is isolated. Installing a temporary filter unit will not affect the single failure analysis assumed. The proposed change will not increase the radiological consequences of an accident previously evaluated in the USAR since installing a filter unit will not change the radiological assumption used on a fuel handling accident of the isolation provisions of the system.

The containment purge exhaust inlet is equipped with a debris screen. This prevents items from entering the purge system that could prevent closure of the valves. This installation will install the filter unit upstream of the debris screen and any debris that could be made from the filter unit and get past the screen would be transported through the system as designed. The inlet screen will remain installed. Therefore, there is no increase in the probability of occurrence of a malfunction of equipment important to safety.

The proposed change will not increase the probability of malfunction of equipment since the weight of the normally installed flange is greater than the temporary filter unit.

The shutdown purge containment isolation valves are required to close for a fuel handling accident. This accident is discussed in USAR Section 15.7, "Radioactive Release From a Subsystem or Component." This change will not increase the radiological consequences of a malfunction of equipment important to safety since the purge isolation valve will be able to close assuming the other has failed.

This change will not create the possibility of an accident of a different type than previously evaluated since the floor will prevent the filter unit from damaging other plant equipment should it fall.

No acceptance limits were identified that could be affect by installing a temporary filter unit. Therefore, the margin of safety is not impacted by the installation of this temporary filter unit.

Safety Evaluation: 59 1999-0063 **Revision:** 0

Security Procedure Change

Description:

The Unreviewed Safety Question Determination (UQSQ) evaluates a change to Security Procedure SEC 50-130, "Compensatory Requirements." The change involves a change in compensatory measures for partial degradation of a camera as follows:

Partial degradation of a camera that does not affect the ability to accurately assess intrusion alarms will be compensated by a roving patrol rather than posting a member of the Security Force.

Safety Summary:

Changing the requirement to use roving patrols rather than posting a camera degradation, that does not affect the capability of Security to assess alarms, does not change the previously evaluated accidents. Since Security will still be able to identify any unauthorized entry into the Protected Area and will be able to assess any tampering within the isolation zone and fence line; there is no potential increase of an accident previously evaluated. Therefore the proposed change does not increase the probability of the occurrence of an accident previously evaluated in the USAR.

Changing the requirements to use roving patrols rather than posting a camera degradation, that does not affect the capability of Security to assess alarms, does not increase the radiological consequences of an accident previously evaluated. Since Security will still be able to identify any unauthorized entries and roving patrols will be able to identify any tampering within isolation zone and the fence line there is no potential increase in a radiological accident. Therefore the proposed change does not increase the consequences of a radiological accident previously evaluated in the USAR.

Changing the requirements to use roving patrols rather than posting a camera degradation, that does not affect the capability of Security to assess alarms, does not increase the probability of occurrence of a malfunction of equipment important to safety. Security will still be able to identify any unauthorized entries and any tampering with equipment located within in the isolation zone and the fence line. Therefore, the proposed change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

This change does not increase the radiological consequences of a malfunction of equipment important to safety. Since Security will still be able to identify any unauthorized entries, and identify any tampering within the isolation zone and the fence line, there is no potential increase in a radiological malfunction of equipment. Therefore, the proposed change does not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

Since this change does not lesson our effectiveness to identify unauthorized entries and tampering, no new type of accidents could be identified. Therefore, the proposed change does not create the possibility of an accident different from that which was previously evaluated in the USAR.

Since this change does not lesson our effectiveness to identify unauthorized entries and tampering, no different type of malfunction could be identified. Therefore the proposed change does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

Since this change does not lesson our effectiveness to identify unauthorized entries tampering, the proposed change would not reduce the margin of safety as defined in the basis for any technical specifications.

Safety Evaluation: 59 1999-0064 **Revision:** 0

Security Computer Replacement

Description:

The digital processing portion of the Security Computer System is being replaced with upgraded equipment. The scope of the replacement will include the main servers, field multiplexer internals, operator workstations, video switching equipment and interconnecting cabling. The modification will not include replacement of the field wiring between the multiplexers and the alarm detection equipment, the video cameras nor console cabinetry. Nine of the twelve multiplexer (mux) locations will have a new mux cabinet installed adjacent to the existing mux. The field wiring will be jumpered to the new cabinet for these nine locations. In the remaining three mux locations, the internals of the old mux will be replaced with the new equipment. The workstations in the Central Alarm Station (CAS), Secondary Alarm Station (SAS), Access Screening, Badging and at the Security Administrative Coordinator's location, will be replaced/added. The main servers will be installed in the power block and the Video Switching equipment in the Charles Evans Whittaker building will be replaced. All major components of the new system will communicate over a dedicated Local Area Network (LAN).

This modification does not involve a significant functional change in the basic type of equipment being used to provide security monitoring. The field devices will continue to monitor the same points and communicate with the multiplexer concerning the status of these points and ultimately provide alarm annunciation to the Secondary Alarm Station (SAS) operators and the Central Alarm Station (CAS) operators. No reduction in this functionality will be incurred with the upgrade to more modern computer equipment. However, some enhanced functionality will exist. The new "smart mux" capability will allow ingress/egress through doors when a mux to host communications failure occurs. This functionality does not currently exist with the old system. A second enhancement is the utilization of a "Star" wiring configuration for the mux-to-host communications, rather than the currently used loop configuration. This will reduce the adverse affects of a cable/communications failure, should one occur.

Numerous field devices that are currently off-line can placed back online by a software command. This will no longer be possible with the new computer system. This is acceptable since the devices are not required to be monitored. However, the physical description of these devices in the Security Plan will change.

Safety Summary:

The Security Computer System is non-safety related. This modification will not affect safety related equipment as no electrical or spatial interfaces exist between the Security Computer equipment and safety related equipment. Existing raceway and mounting requirements are being used on this installation in order to assure that the new equipment does not adversely affect safety related equipment. The new mux cabinets are being installed in locations directly adjacent to the existing mux cabinets and these locations have been evaluated as suitable for this non-safety related equipment. Based on these facts, no new considerations exist for accidents identified in USAR Chapters 2, 3, 6, 9 or 15.

No credible accident can be created by the non-safety related Security Computer System due to its lack of interface with safety related equipment. Separation requirements for non-safety related and safety related components and cabling, ensures that an electrical fault within the Security Computer System will not affect safety related equipment. Location and mounting requirements for the new mux cabinets ensures that they can not be a II/I concern.

Because the mux cabinets are being installed in accordance with the approved separation requirements, there is no affect on malfunctions of equipment important to safety, either direct or indirect.

The acceptance limits for functions associated with the Security Computer System are identified in USAR Section 13.6, "Industrial Security." The limits were identified in the procurement process and the new systems' ability to meet them are being verified during factory and site acceptance testing. Since no acceptance limits are impacted by this change, the margin of safety is not affected.

Safety Evaluation: 59 1999-0065 **Revision:** 0

Updated Safety Analysis Report Change

Description:

This change to the Updated Safety Analysis Report (USAR) Section 9.5B, "Fire Hazards Analysis," adds Room 1403 to Fire Area A-16, "Auxiliary Buiding, El. 2026, General Area, Rooms 1401, 1402, 1406, 1408. USAR Section 9.5.B, Fire Area A-16, designates rooms which surround Fire Area A-16 (Room 1408) and contain equipment required for safe shut down of the plant. A 3-hour fire rated barrier is required for rooms which contain such equipment. Room 1403 contains Reactor Trip Switch Train A and B (SB102A and SB102B) but does not appear in the above USAR Section 9.5.B, Fire Area A-16 as an adjoining room containing safe shut down equipment. USAR Section 9.5.B, Fire Area A-27, "Reactor Trip Switchgear Room, Room 1403," and USAR Figure 9.5.1-2, "Fire Protection System (site)," both indicate that room 1403 has a 3-hour fire barrier. Room 1403 will be added to the list of rooms separated from Fire Area A-16 and meet the requirement of a 3-hour fire barrier for rooms containing equipment required for safe shut down of the plant.

Safety Summary:

The proposed changes will correct the inconsistencies between sections of USAR Section 9.5.B Fire Area A-27, USAR Figure 9.5.1-2 and USAR Section 9.5.B, Fire Area A-16. The changes do not impact any procedures, activities, administrative controls, or sequences of plant operations nor are any plant structures, systems, components or equipment impacted. No requirements outlined in the USAR are revised by these changes. No other USAR descriptions or conclusions will change or be made untrue as a result of these changes. No tests or experiments are involved with these changes.

The proposed changes will correct the inconsistencies between sections of USAR Section 9.5B, Fire Area A-27, USAR Figure 9.5.1-2, and USAR Section 9.5.B, Fire Area A-16. There is no additional impact on the performance of plant activities nor affect on any SSC. Therefore, no design basis accident is identified for review.

These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any system, structure, or component (SSC). Therefore, no credible accidents that could be created are identified.

These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no credible malfunctions of equipment important to safety are identified.

These USAR changes make no additional changes to the plant, do not affect the performance of plant activities and do not affect any SSC. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1999-0068 **Revision:** 0

Updated Safety Analysis Report Change to Correct Table Inaccuracies

Description:

Performance Improvement Request (PIR) 98-2728 identifies Updated Safety Analysis Report (USAR) inaccuracies in various tables. These inaccuracies resulted in the following USAR changes:

Change 1: USAR Table 7A-3, "Data Sheets," Data Sheet 1.2, indicates that there is a recorder for control rod full-in or not full-in. A review of design documentation and drawings indicates that this recorder was never installed in the Wolf Creek Control Room. The proposed change will delete the Recorder Panel and the Class IE entry from Data Sheet 1.2. This change will not result in a change to existing plant equipment and does not change plant operation.

Change 2 This change to USAR Table 7A-3, Data Sheet 2.3, revises the range of flow transmitter FT-917A and 917B from 0 to 1,000 gpm to 0 to 570 gpm. Also in Section II, the ranges are changed from 0 to 280% to 0 to 160%. The expected maximum flow is 550 gpm. This change will bring the USAR into agreement with plant configuration.

The proposed change allows for a more conservative and accurate measurement of charging pump flow through the boron injection tank (BIT) path. Changing the instrument range will not change values being read, nor will it alter any action taken as a result of the readings.

Change 3 In USAR Table 7A-3, Data Sheet 4.2, Remarks 1, 3rd paragraph, 1140 psig to is changed 1125 psig, 114 percent to is changed 115 percent and 45 psi is changed to 60 psi. In USAR Table 10.3-2, under Atmospheric Relief Valves, Normal set pressure is changed from 1,140 psig to 1,125 psig.

This change was previously evaluated by USQD 59 97-0036 for USARCR 97-086. Additionally, when USAR Revision 11 was issued, the number "1,125" was inadvertently transcribed as the Main Steam Isolation Valves (MSIV) closing time on the same table, replacing the true value of "1.5 - 5" seconds. The 1140 psig setpoint was unchanged. The 1,125 second stoke time is clearly erroneous. This change corrects a transcription error that occurred when USAR Revision 11 was issued.

Safety Summary:

Change 1 Deletion of control rod full-in or not full-in recorder from USAR Table 7A-3, Sheet 1.2 brings the USAR into agreement with the physical plant. There are no changes to operational procedures, activities, or conditions. It is obvious that deleting a recorder does change the plant as described in the USAR, but it does not reduce the ability of physical plant equipment to supply information or data. In the case of this recorder, indicators, alarms and the plant computer still supply the necessary information needed for plant operation and they fully meet the requirements associated with obtaining this data. This change does not address or affect test or experiments.

The proposed change deletes a recorder from the USAR. This recorder was never installed in the plant and would have only supplied redundant information that is now being provided by control room indicators. The deletion of this recorder from the USAR will have no impact on plant performance or activities nor have an affect on any system, structure, or component, (SSC). Therefore, no design basis accident is identified for review.

The proposed activities described above are being submitted to update the USAR. These changes are minor corrections that are fully supported by plant design. There is no impact on plant activities nor affect on any SSC. Therefore, there are no credible accidents that could be created.

Change 2 This change deleted an unused or dead area of the instrument range for FT-917A and FT-917B described in the USAR. Changing instrument limits to actual plant conditions creates better accuracy and eliminates useless data. The information provided does not change and will not affect plant operation. The change does not affect instrument calibration methods. Therefore, there are no additions to test or experiments defined in the USAR.

The proposed change adjusts the range of flow transmitters 917A and 917B. The changing of this range does not limit the data being provided, nor will it prevent normal plant operations. Therefore, no Design Basis Accident is identified for review.

The proposed changes will clarify the USAR bringing it into agreement with the physical plant. No malfunctions are associated with these changes. Therefore, the changes will not directly or indirectly cause a credible malfunction of equipment important to safety.

Change 3 This change corrects a transcription error that occurred when USAR revision 11 was issued. Before USAR, Revision 11, was issued the closing time was correctly identified as 1.5 - 5 seconds. This change does not alter the ability of any SSC's from performing their required function. No test or experiments are added due to this change.

The proposed change corrects a transcription error, and does not impact any design basis accident

A search of technical specifications and licensing basis documents identified no acceptance limits that would be affected by these changes. The change in MSIV stroke time brings the USAR into agreement with the

Technical Specifications.

Safety Evaluation: 59 1999-0071 **Revision:** 0

Removal of Transient Combustible Tracking Permit From Procedure

Description:

Revision 3 of procedure AP 10-102 "Control Of Transient Combustible Materials" step 6.1.1 requires a Transient Combustible Permit if a transient combustible load for a room/fire area exceeds 100 pounds of Class A combustibles.

AP 10-102 Revision 4 is being issued to remove the transient combustible tracking permit requirement for Class A combustibles when in plant operational Modes 5, 6, and Defueled. This change includes all safety related areas except the Essential Service Water (ESW) screen house. To provide an equivalent level of protection, two actions are being added. For plant operational Modes 5, 6, and Defueled, procedure AP10-102 is being revised to require hourly roving fire watches for all safety related fire areas except ESW.

Additionally, AP 10-109, "Fire Protection Inspections," is being revised to require fire protection to conduct a daily plant tour of safety related areas, except ESW, for review of transient combustible hazards during Modes 5, 6, and Defueled. The existing permit system for Class A combustibles used at the ESW screen house will remain in effect for all plant modes.

Updated Safety Analysis Report (USAR) Section 9.5.1.7, "Other Auxiliary Systems, Equipment Operability," Step 5.3.2 states" "A permit system controls the storage and handling of combustible materials, including welding, and cutting, and acetylene-oxygen gas systems, inside or adjacent to safety related areas of the WCGS.". This USAR Section is being revised to identify that a permit system is in place with the exception of tracking Class A combustibles during Modes 5, 6, and Defueled.

The hazards from transient combustibles is heightened during Modes 5, 6, and Defueled which typically only occur during scheduled refueling outages due to increased work activities in the plant. However, that is offset by the fact that the plant is already in a shut down mode. In lieu of a permit process, the procedure change will implement a hourly roving fire watch in addition to a daily plant tour by the fire protection group.

Safety Summary:

Roving hourly fire watches and daily walkdowns provide and equivalent level of fire protection as a permit system for Class A combustibles when in plant Modes 5, 6, and Defueled.

Design basis accidents discussed or referenced in USAR Chapter 9, "Auxiliary Systems, and USAR Chapter 15, "Accident Analysis," were reviewed for potential impact by the proposed activity.

The hourly roving fire watches and daily plant tours by fire protection provides administrative controls to assure that combustible loads due to transient fire hazards are maintained at levels that will not challenge any fire barriers. The assumptions regarding the capability of fire barriers to contain a fire to a single fire area remains valid. No new accidents can be created, existing analysis in USAR Chapter 9 bounds the postulated effects of fire due to transient fire hazards.

The assumptions regarding the capability of fire barriers to contain a fire to a single fire area remains valid. All postulated equipment malfunctions that can be caused by fire in a single fire area have been previously evaluated by analysis in USAR Chapter 9. This analysis bounds the postulated effects of fire due to transient fire hazards.

Fire protection requirements have been removed from the Technical Specifications. This change has no impact on acceptance limits in the USAR or the Operating License regarding the control of or tracking transient combustible materials. The probability of a transient combustible fire is not changed. Administrative controls will continue to be in place which provide reasonable assurance that acceptable levels and hazards from a transient fire are maintained.

The existing analysis for a design basis fire bounds any fire initiated by transient combustibles. The administrative controls provide reasonable assurance that any fire will be contained to a single fire area and the assumptions of the design basis fire remain valid.

The probability of a transient combustible fire is not changed. Administrative controls will continue to be in place which provide reasonable assurance that acceptable levels and hazards from a transient fire are maintained. Therefore, the probability of occurrence of an accident previously evaluated is not increased. Neither will this change increase the probability of occurrence of a malfunction of equipment important to safety.

The existing analysis for a design basis fire bounds any fire initiated by transient combustibles. The administrative controls provides reasonable assurance that any fire will be contained to a single fire area and the assumptions of the design basis fire remain valid. Therefore, the radiological consequences of an accident previously evaluated are not increased. Neither will this change increase the radiological consequences of a malfunction of equipment important to safety.

The existing design basis analysis for a fire assumes all credible failures for all equipment and circuits in a fire area. That existing analysis bounds the proposed changes. Therefore, this change cannot create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated.

This procedure change impacts no acceptance limits in the bases for any technical specifications or licensing basis documents. Therefore, the margin of safety is not reduced.

Safety Evaluation: 59 1999-0072 **Revision:** 0

Reactor Coolant Pump Changeout

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Configuration Change Package (CCP) 08039. CCP 08039 provides a change to the Updated Safety Analysis Report (USAR) to reflect the change in the Best Estimate Flow (BEF) as a result of the Reactor Coolant Pump (RCP) change out. Changes to the USAR include the following.

- Revise Thermal Hydraulic Comparison Table (Table 4.4-1) to reflect new DP
- Revise System Performance Characteristics (Section 5.1.4) to reflect new differences between BEF and thermal design flow and mechanical design flow.
- Revise System Design and Operating Parameters (Table 5.1-1) to reflect new flows.
- Revise Reactor Coolant Pump Design Parameters (Table 5.4-1) to reflect new BEF.

Safety Summary:

This change will have not adversely affect the conclusions of the USAR. There are no procedures, structures, systems, components outlined, summarized or described in the USAR that are impacted by the RCP change out, including the change in the BEF and the associated change in the vessel/core pressure drop. All of the accidents presented in Chapter 15 have been reviewed for the RCP change out, including the change in BEF and the associated change in the vessel/core pressure drop, and there are no accidents that are adversely impacted.

The RCP change out, including the increase in the BEF and the associated change in the vessel/core pressure drop, will not create any new potential failure modes nor will it create any new credible malfunctions of equipment important to safety.

The proposed change will not impact any acceptance limits in the bases for the technical specifications or in any licensing basis documents. Therefore, there will be no reduction in the margin of safety.

Safety Evaluation: 59 1999-0073 **Revision:** 0

Fuel Building Hvac Updated Safety Analysis Report Corrections

Description:

The proposed activity consists of changing the description of the fuel building HVAC flow control dampers so that the Updated Safety Analysis Report (USAR) accurately describes the designed and installed equipment.

The affected equipment are GGD0069, 70 (non-safety related Fuel Building Supply Air flow control dampers), and GGD0071 and 72 (safety related Emergency Exhaust Fan flow control dampers). The USAR description of these dampers indicates that they may be motor operated, versus their actually being manually operated. Further, the applicable USAR Figure does not accurately reflect the fact that these dampers are of the opposed blade type.

Safety Summary:

The design basis accident reviewed for impact is the Fuel Handling Accident. Making this change does not affect any analyses or procedures. No type of credible accident has been identified as being created as a result of accurately reflecting the plant's designed and installed configuration in the USAR.

No credible malfunctions of equipment important to safety have been identified as potentially affected by accurately reflecting the plant's designed and installed configuration in the USAR nor is any new malfunction created.

No Technical Specification Bases (or licensing basis document) acceptance limits are affected by the proposed change to the USAR. Therefore, there is no reduction in the margin of safety associated with this change.

Safety Evaluation: 59 1999-0074 **Revision:** 0

Modification to Battery Loading Profiles

Description:

This proposed change modifies the battery loading profiles (Amperes required per time interval) for 125 VDC Class IE batteries NK11, NK12, NK13, and NK14. Battery loads and loading profiles are used to size the batteries and to perform required testing on the batteries. This change is being made because of inconsistencies discovered between the design drawing (NK System Description) and the design calculations supporting the values in the drawing, and because of errors discovered in the design calculations. The battery loading profiles being revised are documented in the NK System Description, and are also described in Updated Safety Analysis Report (USAR) Table 8.3-2 "125 V DC Class IE Battery Loading Cycle (Amperes Required per Time Interval per Battery After Loss of AC Power) Subsystems 1 and 4," and USAR Table 8.3-3 "125 V DC Class IE Battery Loading Cycle (Amperes Required per Time Interval per Battery After Loss of AC Power) Subsystems 2 and 3."

Safety Summary:

The calculations supporting the new loading profiles confirm that the batteries are (and always were) properly sized for the design loads and that the testing performed in the past adequately tested the batteries in accordance with the applicable testing requirements. The new loading profile calculations resulted in a lower total current demand; thus, past testing of the batteries was conservative.

The design calculations supporting this change address two events to ensure the worst case loading scenario for the Class IE batteries: (1) Loss of Offsite Power concurrent with Loss of Coolant Accident and, (2) Station Blackout. The results of the calculations demonstrate that the total battery current demand is lower than that currently described in the design drawings. Therefore, neither of these accidents (nor any of the bounded USAR accidents) is affected by this change.

Since the newly calculated total battery current demand is lower than that currently described in the design drawings, no new accidents are identified that this change could create.

This change modifies the loading profiles for the batteries, but makes no changes to any equipment. The loading profiles (old and new) continue to be well within the design capacity of the batteries; therefore, no equipment is identified that is directly or indirectly affected by this proposed activity.

This change does not increase the total battery loads nor does it increase the total load demand required to be verified during testing. No accidents are affected and no fission product barriers are affected by lowering calculated total battery current demand. Therefore no acceptance limits could be impacted by this change and there would be no reduction in the margin of safety.

Safety Evaluation: 59 1999-0075 **Revision:** 0

Borated Refueling Water System Description Change

Description:

Configuration Change Package (CCP) 08037 provides for changes to System Descriptions M-10EJ, "System Description Residual Heat Removal," M-10EN, "System Description Containment Spray System," and M-10BN, "Borated Refueling Water Storage System." The changes are the result of calculations performed to address Performance Improvement Requests (PIR) 98-1008, 97-4026, and 97-4018. PIR 98-1008 identifies discrepancies pertaining to the time required to switchover Emergency Core Cooling System (ECCS) and Containment Spray System (CSS) pumps from injection to recirculation mode as described in Updated Safety Analysis Report (USAR) Table 6.3-11 and USAR Table 6.3-12 verses actual times required by the operators. PIRs 97-4026, and 97-4018 identify discrepancies related to Refueling Water Storage Tank (RWST) volumes referenced in design calculations, the USAR, and M-10BN. The resolution to these issues resulted in the revision of the USAR. These USAR changes revise minimum and maximum RWST water volumes that will be transferred to the containment during injection, at ECCS pumps switchover, and at CSS pumps switchover, containment flood levels following a loss of coolant accident (LOCA) or main steam line break (MSLB), and net positive suction head (NPSH) available for ECCS and CSS pumps during recirculation modes.

The effects of these changes are:

The maximum containment flood level following a LOCA/MSLB does not increase. Thus, there is there is no impact on the environmental qualification of electrical equipment.

There is sufficient NPSH available for the Residual Heat Removal (RHR) and CSS pumps during recirculation. The NPSH available for CSS pumps has marginally decreased, but the NPSH margin is greater than 10 percent.

The operator times available for ECCS and CSS pumps switchover form injection mode to recirculation mode have increased in the conservative direction. Operators have more time for manual actions.

The minimum and maximum volumes that will be transferred to containment during a LOCA have changed marginally. These volumes have no adverse impact on the post-LOCA sub-criticality analysis or pH analysis.

Safety Summary:

There are no physical changes required to implement this change. The existing RWST level setpoints have not changed. The minimum water volume maintained in the RWST during normal plant operation has not changed. The post-LOCA/MSLB containment maximum flood levels remain within the previously specified values. The changes confirm adequate NPSH margin for the RHR and CSS pumps. Operations has verified that the switchover of the ECCS and CSS pumps from injection mode to recirculation mode can be accomplished within the revised available times being provided by this change. Therefore, it is concluded that the proposed changes will not create any new types of credible accidents.

USAR Section 6.2.1.5 discusses minimum containment pressure analysis. The minimum containment backpressure for the limiting case (double ended cold leg guillotine break) is provided in Figure 6.2.1-86. This figure shows that the containment pressure will be approximately 27 psia after 300 seconds following the accident. Though the analysis was done only for 300 seconds, extrapolating this graph for 800 seconds, one can expect the pressure to be less than the RWST head. The earliest time at which the recirculation mode will be initiated is 13 minutes assuming both ECCS trains in operation.

Hence this single failure assumed RWST water to flow into the containment via the open valve. This approach is consistent with the original USAR analysis and is reflected in the operating procedures.

There are no credible malfunctions of equipment important to safety affected by this change.

There are no acceptance limits related to the proposed change defined in the Technical Specifications. The Technical Specification LCO 3.5.5.a requires that a minimum volume of 394,000 gallons be maintained in the RWST during Modes 1 through 4. This LCO is not impacted by this change. USAR Section 6.2.2.1.3, "Safety Evaluation Eleven," states that the NPSH available for CSS pumps meet the requirements of Regulatory Guide 1.1, Revision 0. This requirement continues to be met.

Based on the above discussion, there is no increase in the probability of occurrence of an accident previously evaluated in the USAR. Since there is no impact on accidents previously evaluated, this change will not increase the radiological consequences of any accident.

Based on the above discussion, this change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. Therefore, this change will not increase the radiological consequences of a malfunction of equipment important to safety.

There are no physical changes to the plant as a result of this change. Therefore, this change will not create the possibility of any accident of a different type. Neither will this change create the possibility of a different type of malfunction of equipment important to safety that previously evaluated in the USAR.

Base on the above discussion, there is no impact to the margin of safety.

Safety Evaluation: 59 1999-0076 **Revision:** 0

Essential Service Water Flow Requirements

Description:

This Updated Safety Analysis Report (USAR) change makes changes to USAR Table 9.2-2, "Essential Service Water System (ESW) Flow Requirements Normal Power Operation," which are consistent with previously implemented Plant Modification Request (PMR) 02149, "Minimum Flow To Standby EA Components." The exit ESW temperature for the Component Cooling Water (CCW) Heat Exchangers is being changed to show 111 °F for Train A (in use) and 90 °F for Train B (ESW flow, no heat load). Prior to this change, this USAR Table showed an exit ESW temperature of 111 °F with no differentiation between Train A and Train B. This change is consistent with other components in this USAR Table with one train in service and ESW flow to the second train but with no heat load. For the Control Room A/C Unit Condenser, the Number/In Use column is being changed from "2/2" to "2/1". This change is consistent with the other information for this components which shows no heat load on Train B.

Safety Summary:

This USAR Table was originally modified by USAR Change Request (CR) 91-002 to show one Train of the Control Room A/C Unit Condenser in service as a result of PMR 02149. The other changes to Table 9.2-2 are editorial to make corrections to the summation of current values for flow and heat duties. This USAR CR is in partial response to Performance Improvement Request (PIR) 98-3787 which was generated during the USAR Fidelity Review.

This USAR CR makes USAR Table 9.2-2, Essential Service Water System Flow Requirements Normal Power Operation, consistent with the manner in which the Service Water System/Essential Service Water System has been operated since the implementation of PMR 02149. In addition, editorial corrections are to the summation of flow rates and heat duties in the USAR Table. No other descriptions of plant procedure, structure, systems, or components in the USAR are affected by this change. There are no tests or experiments involved with this USAR CR.

Since PMR 02149 was previously determined to establish an acceptable mode of operation for the Service Water/Essential Service Water System based on a maximum inlet temperature of 90 °F for Normal and Normal Shutdown operation, no design basis accident is affected. PMR 02139, Revision 03, was evaluated and determined to have provided an acceptable basis for this change.

Based on a maximum inlet temperature of 90 °F for Normal and Normal Shutdown operation, no new credible accidents are created. PMR 02139, Revision 03, was evaluated and determined to have provided an acceptable basis for this change.

Based on a maximum inlet temperature of 90 °F for Normal and Normal Shutdown operation, no credible malfunctions of equipment important to safety are created. No acceptance limits are identified that could be affected by this USAR change.

Safety Evaluation: 59 1999-0077 **Revision:** 0

Installation of 150 kW Emergency Diesel Generator in the Wolf Creek Switchyard

Description:

To facilitate installation of a permanent 150 kW emergency diesel generator in the Wolf Creek Switchyard under Design Change Package (DCP) 07092, a second diesel generator will be brought into the switchyard to provide temporary backup for power to the Control Building 120/208 volt load center. This generator will be staged for connection at the Control Building prior to performing trenching and conduit work in the vicinity of the preferred and standby station service power transformers. Should both sources of AC power be lost during installation activities, the generator will meet needed station service power requirements, especially for the Control Building battery chargers. This will ensure that the switchyard batteries remain available to supply switchyard control power. This same generator will also be used to provide power during the switchover of preferred transformer XSL7A to the new dual automatic throwover switch, as neither the standby nor preferred source of power will be available during this activity. The temporary generator will be removed from the switchyard upon completion of the permanent switchyard emergency generator installation.

The switchyard batteries provide power for protective relaying and metering, DC power for 345-kV breaker close and trip control as well as indication, and backup power for Supervisory Control & Data Acquisition. The batteries, located within the 345-kV Control Building, are normally charged with 208/120 volt AC power, with the switchyard design allowing the batteries to supply power for eight hours after all AC power is lost to the battery charger.

The temporary diesel generator (DG) in the switchyard will be used while performing modifications to the existing automatic throwover (ATO) switch. During work on the existing ATO compartment, the primary (preferred) power source is isolated and the temporary DG is placed in service to supply switchyard control power. The battery bank is a back-up source of power. The temporary DG is required because, during this work (converting the existing ATO compartment to a terminal box), the secondary (standby) source from the plant is required to be isolated. Therefore, the only available control power during this modification is the battery bank, which is an unwanted condition.

Upon completion of work in the old transfer switch compartment (now a terminal box), the temporary DG will be secured and the primary source re-energized. Because the secondary source from the plant is required to be isolated for this modification, the temporary DG is still in stand-by in the event that the primary source would be lost. If switchyard control power transferred to the batteries, the temporary DG would then be placed in service to minimize service time on the battery bank.

Safety Summary:

The proposed activity is a Western Resources modification of the switchyard for which the owner has design authority. Implementation of the proposed activity will not affect plant procedures, structures, systems, or components that are either summarized or described in the Updated Safety Analysis Report (USAR). In addition, the installation, operation, testing and maintenance of the temporary diesel generator will not require any tests or experiments that may adversely affect the adequacy of systems, structures, or components (SSCs) to prevent accidents or mitigate the consequences of an accident.

USAR Chapter 3 (Section 3.5) defines the scope of SSCs that are required to have adequate missile protection. The location of the proposed activity is in the 345 kV switchyard, which is well outside the analyzed areas.

The connection of the temporary diesel generator, along with any credible failure modes, was evaluated in regard to the switchyard battery charger. The protective circuitry for the battery charger would react to loss of voltage, undervoltage or overvoltage in the same manner as it would to loss of voltage, undervoltage or overvoltage from the station service power transformers. No new failure modes are introduced as any postulated failure of the diesel generator would have the same effect as failure of the station service power transformers.

USAR Chapter 6 (Section 6.2) assumes various accident conditions for containment systems. The proposed activity will not be performed in an area near the containment system. Any normal or abnormal operating, testing or maintenance activity involving the proposed change could not impact any containment system. There are no direct or semi-direct electrical, mechanical, physical, chemical, radiological, or thermal connections between the switchyard and areas of the plant closely monitored or analyzed for containment system accidents.

The same reasoning used for Section 6.2 above can also be applied to Section 6.3 "Emergency Core Cooling System", Section 6.4 "Habitability Systems", and Section 6.5 "Fission Product Removal and Control Systems". None of these systems could be impacted by the proposed modification. Accidents in Chapter 9 pertain to the plant auxiliary systems, an example of which is the accident analysis for the spent fuel storage rack. There is no connection with the temporary switchyard modification in these systems.

The accident analyses for WCGS contained within USAR Chapter 15 assume a loss of offsite power coincident with certain postulated events, such as a

LOCA, and with certain abnormal conditions, such as a turbine or reactor trip. The proposed activity will not increase the likelihood of a loss of offsite power. The activity only enhances the ability of the switchyard to cope with the loss of both sources of station service power during the installation process involved with DCP 07092. A loss of offsite power is analyzed in many ways and many places throughout the USAR. Although all references to loss of offsite power were reviewed, no situation was found where the meaning would be changed with the proposed activity.

The truck delivering and positioning the trailer-mounted diesel generator could run into or damage the Control Building, one of the bus supports or foundations. Any type of metal or conducting/grounding material that is allowed to get too close to overhead energized 345 kV busses could involve loss of life as well as cause a shut down of Wolf Creek Generating Station (WCGS). These activities will be closely monitored to ensure that none of these incidents occur. Should one of these incidents occur, there should be no impact on safety related equipment.

Accidents involving the diesel engine during installation, operations or testing include mechanical failure, loss of coolant, loss of lube oil, high engine temperature and diesel fuel spill. Accidents involving the generator include electrical failure, fire, low voltage and low/high frequency. Accidents involving the starting batteries include acid spill, incorrect specific gravity and shorted leads. All of the above conditions are considered equipment failures. None of these can create a credible accident.

Considering the extent, location and lack of interconnections with anything safety-related that is involved with the proposed activity, there will be no direct or indirect affects of the proposed temporary change on any equipment important to safety. All proposed activities are in the switchyard and are closely associated with the non-safety related SY system and the 120/208 volt Load Center in the Control Building.

Safety Evaluation: 59 1999-0078 **Revision:** 0

USAR Changes Related to Storage of Flamable Gases on the Hydrogen Analyzer Skid

Description:

During a walkdown of the Fire Areas in the plant, it was discovered that the Hydrogen Analyzers, SGS02A & 3A and SGS02B & 3B, have a bottle of hydrogen/nitrogen mixture stored on the analyzer skid. The Wolf Creek Generating Station (WCGS) Updated Safety Analysis Report (USAR) Section 9.5.1 states that all flammable gases used at WCGS are stored outside safety related areas so that a fire involving these gasses cannot cause the failure of any safety related equipment. The Auxiliary Building is a safety related structure and the Hydrogen Analyzers are safety related equipment. Also, Section 9.3.5 of the USAR states that gas bottles are located within the plant in non-safety related areas to provide small quantities of specialty gases for laboratory analysis or localized testing. Their location is shown in USAR Figure 9.3-10, and the statement is in 9.3.5.2.1, general description. This discrepancy was documented in Performance Improvement Request (PIR) 98-1288. The USAR is being revised to resolve the identified discrepancy.

Safety Summary:

The Hydrogen Analyzer skids are located on elevation 2047' in Fire Areas A-19 (Room 1506) and A-20 (Room 1505). Located on each skid are two small (C size) cylinders; one containing 100 percent oxygen for use as a reagent gas and one containing a mixture of 10 percent hydrogen and 90 percent nitrogen for use as a calibration gas. The hydrogen/nitrogen mixture is designated as a flammable gas since the percent of hydrogen is above the lower explosive limit (LEL) of four percent for hydrogen. Both gasses are reduced to 25 psig near the cylinders and further reduced to three psig on the skid. The calibration gas is only used during the periodic calibration of the analyzers, and the cylinder is left in the closed position when not in use.

While the calibration gas is classified as a flammable gas, the percentage of hydrogen is close to the LEL for hydrogen. If there was a leak, the hydrogen would quickly dissipate below the LEL. Additionally, due to the large volume of the room compared with the small amount of hydrogen in the cylinder, the LEL for hydrogen could not be reached, even if the entire tank were to leak.

The only source of ignition in the immediate area of the hydrogen cylinders is on the skid itself. The skid is specifically designed for use with hydrogen and therefore can be ruled out as an ignition source. Combustible loading in Fire Areas A-19 and A-20 is low and no combustibles present an exposure fire potential to the cylinders or skids.

It does not appear that the presence of hydrogen cylinders was originally considered in the Fire Hazards Analysis nor in Section 9.3 of the USAR regarding the use of compressed gasses in the plant. However, because of the small size of the cylinder, the low percentage of hydrogen in the cylinder, and the fact that the Hydrogen Analyzer skid was originally designed for use with hydrogen; the existence of the cylinders do not introduce a significant hazard and does not invalidate conclusions in Sections 9.3 or 9.5 of the USAR.

The Hydrogen Analyzers are original plant equipment. Procedures are in place for the use and testing of the analyzers and the hydrogen is included in the Combustible Loading Calculation XX-X-004. No additions or changes are necessary.

The design basis events considered include fire, explosion, and potential missile hazard. The evaluation discussed above concludes that the small cylinder with a low percentage of hydrogen does not introduce a significant fire or explosion hazard and does not affect previous conclusions in the USAR. Additionally, both the hydrogen and the oxygen cylinders are secured in place on the skid which is itself secured to the floor. Therefore, the small cylinders are not subject to becoming a missile hazard and damaging surrounding equipment.

Safety Evaluation: 59 1999-0079 **Revision:** 0

Change in the Operation of the Spray Additive System

Description:

In order to ensure that the required amount of Sodium Hydroxide (NaOH) solution is added to the containment sump, the spray additive eductor isolation valves in the NaOH solution supply headers, EN-HV-15 and EN-HV-16, are provided with an interlock from the tank level transmitters to preclude their closure prior to the addition of the required amount of NaOH solution. The proposed changes will allow an even more conservative NaOH addition to be accomplished by allowing valves EN-HV-15 and EN-HV-16 to remain open until the low-low isolation signal from the Spray Additive Tank automatically closes them. The current method of closure requires a manual operator action when the low Spray Additive Tank setpoint is reached. Plant analysis does not take credit for operator action and fully supports the Spray Additive Tank isolation valves remaining open until the low-low isolation signal is reached.

Specific Updated Safety Analysis Report (USAR) changes are as follows:

USAR Section 6.2.2.1.2.3:
Change the sentence from "...are automatically closed to prevent N2 from being drawn into the pump..." to "...are automatically closed to terminate the flow of spray additive solution and prevent N2 from being drawn into the pump...".

Table 6.2.2-3:
For Recirculation Phase Time 1.5 minutes, delete the sentences "If low level in the NaOH tank has been reached, manually initiate closing of NaOH tank outlet isolation valves. If low level in the NaOH tank has not been reached, manually initiate closing of the above valves upon that level being reached."

Section 6.5.2.2.3:
Change "...less than 5 percent..." to "...approximately 5 percent...". Also in the last paragraph change "...containment spray additive subsystem is remote-manually terminated..." to "...containment spray additive subsystem may be remote-manually terminated...".

Safety Summary:

The Containment Spray System in conjunction with the Containment Fan Cooler System and the Emergency Core Cooling System, is capable of removing sufficient heat and subsequent decay heat from the containment atmosphere following the hypothesized LOCA to maintain the containment pressure within design criteria. In addition, the Containment Spray System also reduces the iodine and particulate product inventories in the containment atmosphere post-LOCA. To enhance the iodine absorption capacity of the containment spray, the spray solution is adjusted to an alkaline pH to promote iodine hydrolysis. This is accomplished by adding 28 percent to 31 percent concentration by weight NaOH solution to the spray. Approximately 2960 gallons of NaOH solution is presently required to be added to the containment sump to achieve a final post-Loss of Coolant Accident (LOCA) containment sump pH of at least 8.5. A minimum pH of 8.5 in the containment sump is necessary to ensure long-term retention of iodine in the solution.

The proposed change deals with the injection of NaOH as a result of post-accident conditions. The change involves operational steps which are eliminating the need for remote-manual actions due to an automatic action that takes place later in the post-accident logic. The change is fully bounded by plant analysis and other USAR Sections. Therefore, no other credible accidents could be created.

Calculation GS-M-004, "Hydrogen Generation Analysis," assumes that the entire contents of the NaOH spray additive tank are injected into containment at a concentration of 31 percent with no adverse affects. The calculation also concluded the original design of the containment still bounds current plant configuration. Therefore, the proposed changes do not adversely affect the analysis for hydrogen generation due to corrosion and will aid operations by eliminating an unnecessary remote-manual action. In addition the increased NaOH volume being injected into the containment will bring about greater conservatism in mitigating the consequences of iodine during a LOCA. The proposed changes do not alter plant analysis and will not affect the safety or health of the public.

The proposed change eliminates the requirement for closing NaOH spray additive tank isolation valves at the low spray additive tank level. The change does not change the physical plant and does not change further automatic actions which close valves at low-low tank level. Current plant analysis assumes these valves do not close. Therefore, previously analyzed malfunctions are not affected by this change.

As noted previously the proposed change is already bounded by existing analysis. Therefore, the change does not affect the acceptance limits which are contained in the bases for the Technical Specifications.

Safety Evaluation: 59 1999-0080 **Revision:** 0

Change to Auxiliary Feedwater Data Table Resulting From the Updated Safety Analysis Report Fidelity Review

Description:

System Description M-00AL," Auxiliary Feedwater System Description SNUPPS," Section 3.3.2 states that the available Net Positive Suction Head (NPSHA) for the Turbine Driven Auxiliary Feedwater (TDAWF) pump is 28 ft. Calculation AL-16, "Auxiliary Feedwater System," indicates that the NPSHA for the TDAFW pump is 27 feet, based on the elevation of the centerline of the suction lines. Design Document Change Notice (DDCN) M-00AL-05-03 has been generated to revise M-00AL, Section 3.3.2 to indicate that the NPSHA for the TDAFW pump is 27 ft. The NPSHA of 28 ft. as shown in System Description M-00AL and USAR Table 10.4-12 is not consistent with the NPSHA of 27 ft. determined by Calculation AL-16. A USARCR has been generated to revise USAR Table 10.4-12 to show the NPSHA for the TDAFW pump as 27 ft.

Safety Summary:

The change to System Description M-00AL will provide an accurate representation of the TDAFW pump NPSHA. As determined in AL-16, the margin of minimum NPSH available over NPSH required remains greater than 10 ft. and the additional margin obtained from the Condensate Storage Tank low level setpoint remains unchanged. There is no change to the system function.

USAR Table 10.4-12, "Auxiliary Feedwater System Component Data," also indicates that the NPSHA for the TDAFW pump is 28 ft. The Record Supplemental Sheet to AL-16 which was generated by PIR 98-2602 determined that the NPSHA based on the elevation of the impeller centerline did not alter the minimum NPSHA of 27 ft. The minimum required suction head (NPSHR) as specified in M-021 is 17 ft.

The change being reviewed is a clarification of parameters that are established by pre-operational testing and calculated (in AL-16) to ensure that sufficient NPSH is available to allow the TDAFW pump to operate as it was designed. There is no additional impact on the performance of plant activities nor does the activity affect any system, structure, or component (SSC). Therefore, no design basis accident is identified for review.

Since the change is a clarification of one of the parameters calculated for the TDAFW pump, it makes no additional changes to the plant, does not affect the performance of plant activities and does not affect any SSC. Therefore, no credible accidents that could be created are identified.

This change makes no additional changes to the plant, does not affect the performance of plant activities and does not affect any SSC. Therefore, no new credible malfunctions of equipment important to safety are identified.

The change is a clarification of a parameter that has been established by calculation and, as such, makes no additional changes to the plant, does not affect the performance of plant activities and does not affect any SSC. Therefore, no acceptance limits are identified that could be affected.

Safety Evaluation: 59 1999-0081 **Revision:** 0

Inadequate Electrical Separation Within Switchboard NK003

Description:

This Unreviewed Safety Question Determination provides evaluation of a non-conforming condition as discussed below.

The routing of safety related field cables 3NNY01BC AND 3NNY01BD and the non-safety related annunciator ground wire inside 125 VDC distribution switchboard NK003 is such that it is not possible to maintain a six inch minimum separation. This configuration does not comply with the electrical separation criteria as stated in Updated Safety Analysis Report (USAR) Section 8.3.1.4.1.2, and engineering evaluation and technical justification have been documented in Reportability Evaluation Request 99-017 to accept this installation.

Safety Summary:

Analysis and Test results contained in Philadelphia Electric design verification report #48503 and Wyle Reports #46960-1 & 46960-3, prepared for Philadelphia Electric's Limerick Generating Station Units 1 & 2, were used as bases for this technical justification.

WCNOC has previously used the above reports as the technical basis for other exceptions from physical separation requirements as listed in USAR Section 8.3.1.4.1.4. These same three reports are currently referenced in the USAR and have been evaluated as being applicable to Wolf Creek due to being:

1. similar - compared to Wolf Creek Generating Station (WCGS) cables,
2. similar - compared to WCGS configurations,
3. conservative - compared to WCGS applications.

From these reports and the associated testing, it can be concluded that the proposed use-as-is activity will have no effect on an functions or failure modes of any systems, structures, or components (SSCs) at WCGS.

The 6 inch physical separation criteria prescribed in E-11013, Section 5.8, which is based on Regulatory Guide 1.75, September 1978, and IEEE 384-1974 (USAR Reference Section 8.1.4.3 & 8.3.1.4.1.2). USAR Section 8.3.1.4.1.4 lists exemptions to this criteria along with their justifying analyses.

There are no design basis accidents which can be affected by the exemption to the six inch minimum separation requirement.

There are no credible accidents which could be created by the exemption to the six inch minimum separation requirement.

There are no credible malfunctions of equipment important to safety which may be directly or indirectly affected by the exemption to the six inch minimum separation requirement.

The Technical Specifications do not contain information regarding minimum cable separation distances within panels. USAR section 8.3.1.4.1.2 lists the minimum separation criteria. Therefore, this condition does not form the bases for any licensing document.

Safety Evaluation: 59 1999-0082 **Revision:** 0

Reactor Coolant Pump Main Flange Bolt Cutoff

Description:

Configuration Change Package (CCP) 09008 allows the stuck #11 bolt on Reactor Coolant Pump (RCP) PBB01B to be cut off, the lower portion of the bolt left in place, and the pump operated with twenty three studs and nuts securing the pump cover. It has been evaluated as having no effect on the PBB01B's ability to perform its safety related functions.

Safety Summary:

The proposed change will result in a weight of approximately 205,567 pounds for PPB01B, which is between the two values shown in Updated Safety Analysis Report (USAR) Table 5.4-1, "Reactor Coolant Pump Design Parameters." A footnote will be added to the table stating that total pump weights between the values given are bound by existing analyses.

The change was reviewed for potential impact on the design basis accidents described in USAR Section 15.3, Decrease in Reactor Coolant System Flow Rate and Section 15.6.5, Loss of Coolant Accidents.

The only accident which operation of the RCP with twenty three main flange studs versus twenty four could cause would be failure of the RCP pressure boundary (Loss of Coolant) at the main flange, and this has been evaluated as not credible because adjacent bolt stresses remain within ASME Code limits with one missing bolt and stud. One missing bolt is not a problem as the ASME Code margin and rules allow for failed fasteners. The evaluation demonstrated that the required ASME Code limits remain satisfied.

The only potential malfunction of equipment important to safety which may be affected by this change would be failure of the RCP pressure boundary at the main flange, resulting in failure of the RCP to perform its safety related functions. This has been evaluated as not credible. The evaluation demonstrated that the ASME Code limits remain satisfied.

There are no acceptance limits which could be affected by the proposed change.

Safety Evaluation: 59 1999-0083 **Revision:** 0

Change to Room Drain Description Resulting From USAR Fidelity Review

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a change to the Updated Safety Analysis Report (USAR) which was generated as a corrective action (Performance Improvement Request-98-2639) resulting from the USAR Fidelity Review.

The change will revise USAR Section 9.5B.7 (A.26.4) to clarify a potentially misleading statement. The section currently reads "One 4-inch floor drain is provided in Room 1405 as well as in Room 1415." There are in fact 2 drains in Room 1415. The 3-drain configuration is correctly shown in USAR Figure 9.3-5, "Auxiliary Building Floor and Equipment Drain System," and design drawing M-0P1411. This change will make the USAR text consistent with the USAR figure. The text will be revised to read "One 4-inch floor drain is provided in Room 1405 and two in Room 1415."

Safety Summary:

This change is necessary to ensure the completeness and accuracy of the USAR. Specifically it will make the USAR text consistent with the associated USAR figure showing the actual plant configuration. The affected rooms contain no safety-related equipment and are not analyzed for flooding in the Auxiliary Building flooding calculations. No functional requirements are affected and no change to any design bases of the plant are represented by the change.

The affected rooms do not contain any safety related equipment. The affected drains are classified non-safety related. No SSCs or procedures are affected by this change. The change revises the text of the fire protection section of the USAR to make it consistent with the associated USAR figure and design drawings. The affected rooms and their associated drains are not associated with any test or experiment.

The configuration of the rooms are not included in any assumptions for any design basis accident (DBA), and would not affect any response to any DBA.

The change revises the text of the fire protection section of the USAR to make it consistent with the associated USAR figure and design drawings. There is no potential for the creation of any credible accident.

This USAR change does not affect any safety-related equipment, any setpoints or physical parameters in the plant. Therefore no acceptance limits are affected.

Safety Evaluation: 59 1999-0084 **Revision:** 0

Removal of Foreign Material From Reactor Vessel

Description:

During the reactor vessel foreign material object search and retrieval (FOSAR), a small bolt was observed at the bottom of the reactor vessel. Based upon the FOSAR video, it is estimated that the bolt is approximately .75 inches in length and roughly .25 inches in diameter at the head. It is assumed that the bolt material is stainless steel. The following evaluation will address the impact of operation of the plant with the loose bolt in the Reactor Coolant System (RCS). Operation of the plant with the bolt in the RCS can potentially impact numerous areas. These include the RCS components (i.e., reactor core, reactor vessel internals, steam generator tubes, tube sheet, divider plate and channel head, pressurizer, reactor coolant pumps, and piping). This condition also potentially impacts the auxiliary systems (e.g., the chemical volume and control system, emergency core cooling system, etc.), the accident analyses (i.e., LOCA, non-LOCA, etc.), the Technical Specifications, and Instrumentation and control systems.

Safety Summary:

It should be noted that the exact size of the loose bolt is not critical to the evaluation. It is assumed that the presence of the Debris Filter Bottom Nozzles (DFBN) on 192 of the 193 fuel assemblies will preclude the loose part in its current state from reaching the fuel rods. The fuel assembly located in H8 contains the standard bottom nozzle which has flow holes of 0.376 inches and 0.280 inches compared to the 0.190 inch holes of the DFBN. Assuming the object is .25 inches in diameter and aligns perfectly with the standard bottom nozzle flow hole, the loose bolt could pass through the bottom nozzle into the active fuel region H8.

Given the above, all the RCS components were evaluated for the presence of the loose part to ensure the safe operation of the plant. It is concluded that operation of the plant with the loose bolt described above in the RCS will not result in any adverse impact on the RCS components or systems noted above. In addition, it will not adversely affect the Technical Specifications or the accident analyses. Thus, the conclusions of the USAR will remain valid and bounding.

There are no procedures, structures, systems, or components outlined, summarized or described in the USAR that are impacted by the operation of the plant with the loose bolt in the RCS. Thus, the USAR remains valid and bounding.

All of the accidents were reviewed for the operation of plant with the above described bolt in the RCS. This includes the LOCA related events, the Main Steamline Break Mass and Energy Releases, the Containment Integrity analyses, the Chapter 15 non-LOCA analyses, and the Steam Generator Tube Rupture analyses.

The Operation of the plant with the above described loose bolt in the RCS with will not create any new credible types of accidents. For the single assembly out of the 193 assemblies with the standard bottom nozzle, the flow holes are slightly greater than the estimated diameter of the foreign object. However, the object would have to be located directly under the specific assembly, and be oriented exactly vertical to pass through a flow hole. In the unlikely event that this were to occur, or the fragmented object passes through a DFBN hole, the object would be postulated to travel no further than the lowest grid of that assembly; as such, any fuel failures would be limited to the very bottom of a few fuel rods, and would be within the one percent fuel failure assumed in the USAR. Therefore, the probability of malfunction (failure of fuel cladding) of equipment important to safety does not increase.

The operation of the plant with the loose bolt in the RCS does not result in a different response of safety related systems and components to accident scenarios currently postulated in the USAR such that radiological consequences would be impacted. The presence of the loose bolt will not result in any increases in the fuel failures currently assumed in the USAR for normal plant operation due to the malfunction of equipment important to safety, that is the fuel cladding, since the limit of one percent fuel failures will continue to be met. Also, Technical specifications limit RCS activity to less than 1 percent. Therefore, the presence of the loose bolt in the RCS will not increase to consequences of a malfunction of equipment important to safety previously evaluated in the USAR such that radiological consequences are impacted.

Potential impacts in the departure from nucleate boiling design basis, loss of coolant accident peak clad temperatures, and RCS coolant activity are considered to be negligible and existing acceptance limits continue to met. Therefore, the margin of safety is not reduced.

Safety Evaluation: 59 1999-0085 **Revision:** 0

Procedure Revised to Add Cooling Water Source to Safety Injection Pump

Description:

Revision 4 to procedure STS CV-210B, "ECCS SI and RHR Inservice Check Valve Test," provides a source of cooling water to the Safety Injection Pump oil cooler which is not the normal source of cooling water. When the plant is in Mode 6 or Defueled with the reactor head off and Component Cooling Water (CCW) not available to support running a Safety Injection Pump (SIP) the proposed change uses a small flow of demineralized water from the Reactor Makeup Water System (RMWS) by way of a hose to replace the CCW normally supplied to the oil cooler of the SIP. The SIP is not operable when being supplied with the RMWS cooling flow. The RMWS water will run through the SIP oil cooler and then be directed to a floor drain. This flow of RMWS water will provide cooling to the SIP oil cooler and thus support SIP operation for testing purposes only and will not be used to satisfy any Technical Specification requirement for the SIP. RMWS water flow, quality and temperature are compatible for performing this function.

Safety Summary:

The RMWS is summarized in Updated Safety Analysis Report (USAR) Section 9.2.7 and is reflected on USAR Figure 9.2-13. The uses of the RMWS as described in the USAR do not include cooling for the SIP lube oil. Off normal procedures used during plant Mode 6 or Defueled are not affected by the change.

A review of the accidents in USAR Chapters 2, 3, 5, 6, 9, and 15 concluded that no accidents are impacted by the change to procedure STS CV-210B, a procedure that can only be run when the plant is in Mode 6 or Defueled with the reactor head removed. The assumptions and conditions assumed prior to, during, and after these accidents are not changed by the proposed activity.

Leakage, hose burst, spray effects could occur during the time the RMWS is used to provide cooling to the SIP. If these events were to occur it could result in a housekeeping concern or a personnel safety incident. These event occurrences would be unlikely and not considered commiserate with an accident that challenges or degrades nuclear safety. The RMWS water is not considered as radiologically contaminated, thus no radiological concerns would exist as a result of the RMWS leakage. The RMWS flow through the SIP oil cooler is monitored by the procedure change as well as the SIP bearing temperature. If minor leakage were to occur that did not significantly change the monitored flow capacity or bearing temperatures, only a housekeeping concern would be created, a concern that eventually would be detected by sump level indications, test personnel, or operator rounds. The RMWS tank is provided with a low level alarm at the MCB and an automatic RMWS pump trip feature at low level.

If major leakage were to occur then personnel monitoring the RMWS flow and bearing temperature at the SIP would be aware of a loss of cooling flow and would safely secure the evolution. The personnel monitoring, low level alarm, and RMWS pump trips on low level are measures that provide assurance that in the unlikely event of leakage or hose burst these events would be detected in a timely manner to avoid serious equipment damage. Leakage spray effects from the RMWS hose would remain well within the analysis of the high energy pipe break or moderate energy crack analyses in the Emergency Core Cooling System (ECCS) pump rooms.

These pump rooms that the hose will pass through would be associated with the train of CCW that was not available. Thus, the equipment in these rooms would not be in an operable condition regardless of the evaluated spray effects. The hose run in the Auxiliary Building hallway does not present any spray hazard to any equipment. The doors breached by the hose run are controlled by the appropriate plant procedures. The flood hazard analysis of the 1974' level is not seriously challenged by a RMWS hose break should it occur because the maximum estimated flow from the small RMWS vent/drain valve is on the order of 100 gpm, well within the bounding flood flow from the fire protection header break flow that is an order of magnitude higher.

The proposed activity does not effect any evaluated accident or the analysis thereof. The changes made do not affect any previously evaluated operator accident mitigating actions or emergency procedures. Therefore, it is concluded that the proposed activity does not have the possibility to create any accident, nor create the possibility of an accident of a different type than previously evaluated.

This change lacks the potential to effect or challenge the plants design in an adverse or unique manner, let alone cause a new or different type of accident.

Safety Evaluation: 59 1999-0086 **Revision:** 0

Updated Safety Analysis Report Update In Support of Eighteen Month Fuel Cycle

Description:

The proposed changes to the Updated Safety Analysis Report (USAR) are as follows:

USAR Section 6.5.2.3:
Change "...maximum spray additive flow rate of 44 gpm." to "...maximum spray additive flow rate greater than 46 gpm."

USAR Table 6.5-2:
In Table 6.5-2 for Design Sodium Hydroxide (NaOH) flow rate per educator, add ± 2 behind 44.0.

Safety Summary:

One of the functions of the containment spray system is to reduce iodine and particulate product inventories in the containment atmosphere post-Loss of Coolant Accident (LOCA). To enhance the iodine adsorption capacity of the containment spray, the spray solution is adjusted to an alkaline pH to promote iodine hydrolysis. This is accomplished by adding 28 percent to 31 percent concentration by weight NaOH solution at a normal design flow of 44±2 gpm into the spray volume. The proposed changes will bring the USAR into agreement with actual NaOH flows used in support of the current 18 month fuel cycle. This change is consistent with Calculation EN-05-W, "Containment Spray Additive Eductor Parameter."

The changes do not impact any procedures, activities, administrative controls, or sequence of plant operation nor are any plant structures, systems, components or equipment affected. No other USAR conclusions will change or be made untrue as a result of these changes. No tests or experiments are involved with these changes.

The values being changed by this USAR change are based on actual pump test data and are bounded with margin by plant analysis. The values are not given in the Technical Specifications and do not affect NaOH concentrations values contained in the Technical Specifications.

Safety Evaluation: 59 1999-0088 **Revision:** 0

Change to System Description in the USAR

Description:

Sections 3.1.7 and 3.2.2 of System Description M-10AE, Feedwater System Description," and Updated Safety Analysis Report (USAR) Section 10.4.7.2.2, "Main Feedwater Control Valves and Control Bypass Valves," are revised to clarify when feedwater flow is transferred from the feedwater bypass control valves to the main feedwater control valves. These change will make these descriptions consistent with procedure GEN 00-003, "Hot Standby to Minimum Load."

Safety Summary:

During power ascension of a nuclear power plant many transitions must take place. One such transition involves the transfer from feedwater bypass control valves to the main feedwater control or regulating valves. Procedure GEN 00-003 "Hot Standby to Minimum Load" directs this operation to begin at approximately 20 percent power by unisolating the main feedwater control valves one at a time. Flow is regulated using the feedwater bypass control valves in manual until associated main feedwater control valve is unisolated. On reaching 25 percent power further adjustments are made bringing all feedwater bypass control valves to between 60 to 80 percent open. At approximately 30 percent power main feedwater control valves are placed in auto and the associated bypass valve jogged closed manually. During this feedwater control valve swap the main feedwater turbine speed controller is in manual preventing transients which could adversely effect the control valve operation. The System Description (M-10AE) and Updated Safety Analysis Report (USAR) have simplified discussions of this control valve swap that contain inaccuracies. The proposed changes to the System Description and USAR will correct these inaccuracies.

This change will bring the System Description and USAR into agreement with actual plant operation. The process being used by operations to swap the feedwater control valves is a very methodical approach which maintains the greatest amount of control of the plant. The one time transfer of all feedwater control as implied in the System Description and USAR would make it almost impossible to recover from a transient that could occur at that time. The proposed changes to the process discussed in the System Description and USAR are within the analyzed limits of the plant and will ensure there are no misunderstandings concerning actual plant operation.

The System Description and USAR discussions have over simplified the transfer process of the feedwater bypass control valves to the feedwater control valves. This process requires time for stabilization and assurance that all components are operating as designed. This time requirement was not implied or addressed in the System Description or USAR. These proposed changes will not affect any other USAR Sections or statements, and do not include any information written or implied dealing with test or experiments.

Loss of Normal Feedwater Flow is the only accident that could logically be associated with the proposed changes. The changes define normal plant activities and the order in which they are being performed. The proposed changes would in no way affect the occurrence or likelihood of a Loss of Normal Feedwater Flow.

Safety Evaluation: 59 1999-0089 **Revision:** 0

Procedure Revision for Filling Inoperable RHR Train

Description:

Revision 23 to procedure SYS EJ-110, "RHR System Fill and Vent Including Initial RCS Fill," will allow filling the inoperable train of the Residual Heat Removal (RHR) System from the operable train of RHR when the plant is in Mode 6 with at least 23' of water above the reactor flange (Technical Specification 3.9.4 high water level). The procedure change will allow opening both manual Chemical and Volume Control System (CVCS) low pressure letdown isolation valves at the same time when the plant is in the above configuration.

Safety Summary:

This configuration will allow a small portion of the Reactor Coolant System (RCS)/RHR flow downstream of the operable RHR heat exchanger to pass into the inoperable RHR train piping that is empty and fill the system. When the inoperable train is full, venting will take place and then the CVCS low pressure letdown isolation valve on the inoperable train will be closed. This evolution will be performed when the RCS is open, at a RCS pressure created by the static head of the Refueling Pool. Filling the inoperable RHR train under these conditions will not significantly challenge any pool level with or without the transfer tube open, violate any Technical Specification requirement, adversely affect any equipment, piping, or refueling operation, nor will it create a dilution or water hammer event.

The RHR Shutdown functions and features are summarized in Updated Safety Analysis Report (USAR) Section 5.4.7 and Appendix 5.4A. The RHR System Piping & Instrument Diagram (P&ID) is reflected on USAR Figure 5.4-7. This USAR Figure depicts the ECCS configuration of RHR and is not affected by the subject procedure change (which occurs in Mode 6). Reactor startup is discussed in Section 5.4.7.2.4 of the USAR and both CVCS low pressure letdown isolation valves are described as being open during this evolution. This condition would occur during Mode 4 or 5, and not during Mode 6 when the subject change can occur.

Off normal procedures used during Mode 6 are not affected by the change. It is concluded that this change, although not described in the USAR, does not invalidate any information in the USAR.

A review of the accidents in USAR Chapters 2, 3, 5, 6, 9, and 15 concluded that no accidents are impacted by the change to procedure SYS EJ-110. The assumptions and conditions assumed prior to, during, and after these accidents are not changed by the proposed change. Only the Fuel Handling Accident in USAR 15.7.4 and Spent Fuel Pool Drainage in USAR Section 9.1

are relevant, given the condition the plant is in when the change is allowed to take place (Mode 6).

Lifting the RHR suction relief on the inoperable (drained) RHR train (approximate setpoint of 450 psig) and losing RCS and Pool inventory to the Pressurizer Relief Tank (PRT) is not possible. This statement is based on the fact that the discharge pressure of the running operable RHR pump when taking suction from the open RCS will always be less than 250 psig, thus lacking the potential to lift the RHR suction relief valve on the inoperable RHR train.

There is no possibility of creating a water hammer, two phase flow or causing the RCS/RHR fluid to go to a saturated state in the operable or inoperable portion of the RHR or RCS Systems because of the system conditions (pressure/temperature) during this period of time. The amount of flow in the 10" RHR header (maximum possible of 5,500 gpm) diverted into the 2" CVCS low pressure letdown piping to the inoperable RHR train (maximum of ~250 gpm) will not compromise the ability to maintain a minimum of 1000 gpm recirculating through the operable RHR loop to maintain core cooling. Thus the potential to degrade core cooling is not created.

The inoperable RHR train equipment and piping being filled will be at ambient conditions and there is no heat source to cause a change of state of the fill water that could cause pipe whip, hanger or check valve damage to either RHR train. The RCS/RHR fluid, upon entering the inoperable RHR train, will not flash to steam as it is less than 140 °F, Mode 6.

An expected diminishment of Refueling Pool (RFP) level will occur during the fill process. Additionally Spent Fuel Pool (SFP) level, if the transfer tube is open and Fuel Transfer Gate removed, will also diminish. The diminishment of pool level will be slow, monitored and manually controlled by the CVCS low pressure letdown isolation valve. To fill an empty RHR train (RCS suction to discharge, operable crosstie closed) will take approximately 4,200 gallons, which correlates to approximately 4.3 inches of RFP level (967 gallons/inch, transfer tube closed) or about 2 inches of RFP/SFP level (2,100 gallons/inch, transfer tube open). Before filling the empty RHR train excess level in the pool(s) will exist such that the volume of water removed to fill the inoperable RHR train will not challenge the Technical Specification limits on pool levels. Pool levels are monitored to ensure Technical Specification compliance.

The failure modes and effects analysis of the RHR equipment listed in USAR Table 5.4-9 identifies a closed failure mode or unable to open condition of the CVCS low pressure letdown isolation valves. The inability to open is evaluated for when the plant is shutting down and the effect on the ability to adjust boron concentration. This malfunction has no effect on safety for RHR system operation and is not applicable during the time the procedure change will be utilized (Mode 6), thus it is not affected.

The failure modes and effects analysis of the safety related equipment used to achieve and maintain a cold shutdown list in USAR Table 5.4A-3 was also reviewed and found to be not applicable to mode 6 when the procedure change is used.

Loss of the ability to close the operable train manual CVCS low pressure letdown isolation valve (isolate the train) is unlikely, but would be acceptable. The isolation valve on the inoperable RHR train remains functional, and could be used to isolate the operable train. By normal design, credit is taken for the pressure boundary integrity of the closed, out of service or inoperable RHR train CVCS low pressure letdown isolation valve in the case when the operable RHR train is providing low pressure letdown. These valves are readily accessible, are not in a harsh environment during the times they are used by the procedure change, and crediting operator actions is acceptable per the constraints of procedure AP26C-004, "Technical Specification Operability"

Review of the Acceptance Limits contained Technical Specifications 3.9.8.1 & 2, 3.9.10.1, 3.9.11 and in other licensing basis documents, USAR, and Safety Evaluation Reports concluded that no limits are affected.

Safety Evaluation: 59 1999-0091 **Revision:** 0

Updated Safety Analysis Report Change -- Main Control Board

Description:

The entries for NK-IY-1 through NK-IY-8 will be deleted from the Updated Safety Analysis Report (USAR) Table 7.5-5, "Safety-Related Display Instrumentation Located on the Control Board - (BOP Scope of Supply). The purpose of this table is to list display instrumentation located on the Main Control Room BOP Control Board. NK-IY-1 through 8 are calibration devices (potentiometers) located internal to the control panels. These potentiometers are adjustable center tapped resistors used to calibrate 125V battery and battery charger ammeters. The potentiometers are not display instruments. Therefore these instrument should not be listed in the table.

Safety Summary:

The ammeters, not the associated calibration potentiometers, provide the information needed by operators to respond to design basis accidents discussed and referenced in the USAR. The location of the potentiometers has no affect on the information provided by the ammeters. Therefore, no accidents could be potentially impacted. Calibration potentiometers provide no information needed by operators to assess plant status for normal operation, anticipated operational occurrences or accident conditions. Therefore, the absence of these potentiometers as displays on the control board has no potential to create credible accidents.

The only equipment affected by this change are the calibration potentiomenters. The location of the potentiometers , whether on the control board or internal to the control board, has no bearing on the function or life of the potentiometers. Therefore, no malfunctions of equipment important to safety may be directly or indirectly affected by this change.

The potentiometers are only used for calibration purposes. The ability to utilize the potentiometers for calibration is unaffected by this change. Therefore, this change does not effect any acceptance limits.

Safety Evaluation: 59 1999-0093 **Revision:** 0

Revision to Procedure CKL BG-130

Description:

Procedure CKL BG-130, "Chemical and Volume Control System Switch and Breaker Lineup," is being revised to change the position of BG HIS-8146 from "open" to "closed" and the position of BG HIS-8147 from "closed" to "open". This takes valve BGHV-8146 (Regenerative HX to RCS Loop 1 Cold Leg) from being the normal charging flow path from the regenerative heat exchanger outlet to being the alternate (normally closed) charging flow path and it takes valve BGHV-8147 (Regenerative HX to RCS Loop 4 Cold Leg) from being the alternate charging flow path from the regenerative heat exchanger outlet to being the normal (normally open) charging flow path.

Safety Summary:

The purpose of swapping the flow paths is to balance out the thermal stress events between the two valves and associated piping. In the future, the normal flow path may be swapped each fuel cycle to continue the balancing of thermal stress between the two paths. The Updated Safety Analysis Report (USAR) Table 9.3-10 "Failure Modes and Effects" is also being revised to clarify and more accurately reflect the interchangability of the normal and alternate flow paths from the regenerative heat exchanger. Either path is acceptable per the design.

The normal valve position is not being updated on the system Piping and Instrument Diagram (P&ID), USAR Figure 9.3-8-01 because the Wolf Creek as-built criteria document WCNOC-3A, Revision 6, indicates that normal valve positions controlled by CKL or SYS procedures do not require updating.

Safety Evaluation: 59 1999-0094 **Revision:** 0

Updated Safety Analysis Report Change to Correct Various Tables and Figures Discrepancies

Description:

Performance Improvement Request (PIR) 98-2194 identified potential discrepancies with Updated Safety Analysis Report (USAR) Figure 6.2.4-1, "Typical Detail Sealing of Piping Penetration Through Containment Room, Floor or Wall," with respect to associated Piping and Instrument Diagrams (P&ID's) and USAR Figures. A review of the PIR findings determined that various changes to USAR Figure 6.2.4-1 are needed. In addition, other minor discrepancies were identified and are also being corrected. The following changes are needed to correct internal USAR discrepancies and to resolve discrepancies with approved P&ID's.

a) Revise USAR Figure 6.2.4-1 as follows:

· Add missing components and associated information to clarify diagrams.
· Correct or add missing component identifiers to diagrams
· Correct or add missing valve function annotations to diagrams (e.g., drain, vent, etc.).
· Correct or add missing piping class boundaries to diagrams.
· Clarify or correct piping origins and/or destinations on diagrams.
· Delete unnecessary components outside of containment boundaries on diagrams.
· Clarify Secondary Actuation Signal notations in tables.
· Revise valve positions in tables for various operating conditions.
· Correct power source notations in tables.
· Correct line/valve sizes in tables.
· Correct Applicable GDC No. in tables.
· Correct information in 'Fluid Contained' field of tables.
· Add missing USAR reference section notation.

b) Revise USAR Table 6.2.4-1 by relocating P-59 and P-91 from GDC-56 list to the GDC-55 list.

c) Revise USAR Table 6.2.4-1 by adding P-98 to the GDC-56 listing.

d) Revise USAR Table 6.2.4-1 by deleting "P-99 Containment pressure sensing monitor" and "P-101 Containment pressure sensing monitor".

e) Revise Section 6.2.4.2.1 (first paragraph on page 6.2-66, Rev. 0) by changing Table 6.2.4-1 to Figure 6.2.4-1.

f) Revise Section 6.2.6.3 to delete reference to penetrations P-99A and P-101A.

Safety summary:

Change a) results from a review of information in USAR Figure 6.2.4-1 versus related information provided on plant P&ID's and associated USAR figures. The corrections do not result in changes to penetration configuration but only correct or add missing piping class boundaries inside of penetration boundaries and add/or correct information outside of penetration boundaries. The change, therefore, has no expected affects.

The remainder of these changes are editorial and do not affect any plant equipment.

Safety Evaluation: 59 1999-0095 **Revision:** 0

Updated Safety Analysis Report Change to Reflect the Use of ASME Section III Appendix F

Description:

This change revises Updated Safety Analysis Report (USAR) Table 3.9(B)-6, "Stress Criteria for ASME Code Class 1, 2, and 3 Valves (Active and Inactive)," and USAR Table 3.9(B)-7, "Design Criteria for ASME Code Class 2 and 3 Piping." Section 3.9(B).1.4.2 states that the stress criteria of ASME Code Section III Appendix F is used for elastically analyzed code components for faulted conditions. USAR Tables 3.9(B)-6 and 3.9(B)-7 summarize the stress criteria used for valve and pipe design at Wolf Creek and have an allowable stress for faulted conditions that in some instances is more restrictive than allowed by Appendix F.

The Wolf Creek licensing basis allows use of Appendix F without restrictions as shown in USAR Section 3.9(B).1.4.2. Therefore, the tables are being revised to clarify that the stress criteria of Appendix F is used for the analysis of stresses created by the thermal overpressurization issue of Generic Letter 96-06. This change does not affect the USAR description for valve and pipe design other than for the issues identified in Generic Letter 96-06.

Safety Summary:

The NRC staff has stated that use of Appendix F is appropriate for evaluating the stresses developed due to the thermal overpressurization concern identified in NRC Generic Letter 96-06," Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," if Appendix F is a part of the plant's license basis. The staff did not indicate that restrictions needed to be placed on the use of Appendix F.

The code is written and approved to provide margin to pressure boundary failure. The NRC has stated that Appendix F is appropriate for evaluating the stresses developed due to the thermal overpressurization concern identified in G.L. 96-06. Use of Appendix F to analyze stresses in piping and valves is allowed by the Wolf Creek licensing bases. The USAR tables being revised have an allowable stress that in some conditions is more restrictive than allowed by Appendix F. This change clarifies that the allowable stresses of Appendix F will be used only when analyzing stresses caused by the thermal overpressurization concern of G.L. 96-06. All stresses will still meet the code requirements when analyzing for the concern of G.L. 96-06. The code is written and approved to provide margin to pressure boundary failure. As long as stress criteria are met, pressure boundary failure is not a credible malfunction. Since pressure boundary failure is the only malfunction identified, no malfunctions of equipment important to safety are affected by showing the use of Appendix F in the tables.

This change to the USAR shows the methods used to analyze stresses following an assumed accident(Faulted Condition). This change will not create the possibility of a new accident, nor will it increase the probability of occurrence of a previously analyzed accident. This change will not change the consequences of any accident as long as the stress criteria are met. This change will not create any new malfunctions of equipment important to safety, nor change the consequences of a malfunction of equipment important to safety as long as the stress criteria are met.

There are no plant procedures, structures, systems, or specific components affected by this change. Generically, class 2 and class 3 piping and valves are affected to the extent that stresses are evaluated to the criteria of Appendix F for the thermal overpressurization concern identified in NRC Generic Letter 96-06.

There are no design basis accidents which are impacted by this revision of Tables 3.9(B)-6 and 3.9(B)-7. This change only documents how stresses are evaluated for the thermal overpressurization concern of NRC Generic Letter 96-06 following an assumed loss of coolant accident or main steam line break.

Safety Evaluation: 59 1999-0097 **Revision:** 0

RM Panel Modification to Add an Isolation Valve

Description:

Design Change Package (DCP) 06544, Revision 4, provides for permanent installation of an isolation valve in the Process Liquid Sampling and Analysis System (RM) panel discharge line leading to line EB-095-HBD-1". This valve would facilitate isolation of the sample cooler relief valve discharge line from the closed cooling water system. The DCP 06544 Revision 4 installs a 1" carbon steel gate valve in the sample cooler relief valve discharge line. The valve shall be a normally open valve. The new valve will provide a means of isolating the line for maintenance and modifications to the RM panel.

Safety Summary"

Updated Safety Analysis Report (USAR) Figure 9.3-4-01, "Process Sampling System," provides the piping and instrumentation diagram of the process sampling system. The new valve will be shown on this diagram. The USAR figure will be updated by this change.

All of the accidents identified in the USAR were reviewed and it was determined that none of them would be affected by the proposed change.

Mis-positioning the isolation valve to the closed position during operation of the RM sample coolers could prevent the sample cooler thermal relief valves from performing their intended protection feature. However, the valve, as well as the sample system is non-safety related. There is no possibility of a safety related system being impacted by any credible accident due to this modification.

No malfunctions of equipment important to safety could be postulated as a result of this change.

There are no acceptance limits for any Technical Specification or licensing basis documents that could possibly be affected.

Safety Evaluation: 59 1999-0098 **Revision:** 0

Change to Reactor Coolant Drain Tank Heat Exchanger Flow in the USAR

Description:

Configuration Change Package (CCP) 09028 provides for a change to the Updated Safety Analysis Report (USAR) Section 11.2. "Liquid Waste Management Systems," and USAR Table 11.2-1, "Liquid Waste Processing System Equipment Principal Design Parameters." The USAR does not accurately reflect the design flow and developed head parameters for the Reactor Coolant Drain Tank (RCDT) pumps PHB02A and B. The pumps trip because the associated instruments measure the flow inaccurately as low flow.

Safety Summary:

Engineering was asked to determine the safe operating flow range of the pumps, to avoid the automatic nuisance of tripping of pumps PHB02A and PHB02B, and of the heat exchanger (EHB01) located down stream of the pump.

Engineering evaluation supports the increase in pump flow up to 114 GPM for the operation to overcome the inaccuracy of the associated instrument. Presently per the system flow diagram for the Liquid Radwaste System (M-01HB01), the flow is 100 gpm.

This change only corrects the inconsistency in the description for the heat exchanger in the Updated Safety Analysis Report (USAR) Section 11.2 and the design parameters of the pump listed in USAR Table 11.2-1 to reflect the current design conditions. There is no impact on the plant or the operation of the plant or change in fit, form, function, or material due to the subject changes. No new instrument/component has been added or no existing instrument/component has been modified for the heater by this change.

Safety Evaluation: 59 1999-0099 **Revision:** 0

Fire Protection Training Change

Description:

During the 1998 NRC Fire Protection Inspection, the NRC noted that procedure AP 10-05, "Fire Protection Training Program" allowed a 31 day grace period from the 2 year training cycle for fire brigade qualification. The NRC identified that Updated Safety Analysis Report (USAR) Section 9.5.1.7.5.2.1.5 did not allow the 31 day grace period. USAR section 9.5.1.7.5.2.1.5 and Table 9.5E-1 sheet 12 are being revised to reflect that a 31 day grace period is allowed for fire brigade re-qualification training.

Safety Summary:

The 31 day grace period has been in effect at WCGS some time, dating back to the ADM 13-100 series of procedures. During the transition into the AP format, the 31 day grace period was omitted. OTSC 98-060 was initiated to add the grace period back into the procedure. The grace period was established to allow training to be targeted on the two year cycle. After initial fire brigade qualification, fire brigade members attend a re-qualification training course each quarter. The content of the quarterly re-qualification classes is structured such that all of the content of the initial training is covered within the 8 classes. The 31 day grace period is provided to allow for ease of training scheduling due to the shift rotation. Given the short duration of the grace period, and the continuous nature of the fire brigade training the grace period does not affect the ability of any of the fire brigade members to carry out their assigned responsibilities.

As noted by the NRC in the inspection report, a WCGS License Condition allows the approved fire protection program to be changed provided it does not affect the ability to achieve and maintain safe shut down in the event of a fire. Because of the continuous nature of the fire brigade training process, and the short duration of the grace period, fire brigade response effectiveness is not changed. The ability to achieve and maintain safe shut down in the event of a fire is not affected by this change. Only the chapter 9.5 accidents has been reviewed for potential impact by the change.

Safety Evaluation: 59 1999-0100 **Revision:** 0

Updated Safety Analysis Report Change to 125 Volt DC Description

Description:

The USAR Fidelity Review Team initiated Performance Improvement Request (PIR) 98-1819 which identified a discrepancy in USAR Section 8.1.4.3. This PIR documented that the control room alarm (Non-Class 1E for 125V DC) had been identified as all-inclusive, which would make it much more than a summary alarm. This is not the case and will require the USAR change that follows:

From: "The non-Class IE dc system is provided with the following alarms in the control room:"

To: "The non-Class 1E dc system is provided with the following alarms on each main switchboard as applicable, which are grouped into a summary alarm in the control room."

Safety Summary:

This proposed change will bring the USAR into agreement with actual plant conditions. The inclusion of all Non-Class 1E 125 Volt DC System (PK) alarms is both unnecessary and unwanted in the main control room. These alarms would be a nuisance in accident conditions. The PK system is non-safety related and provides no safety function that mitigate the consequences of a design basis accident. During normal operation the summary alarm along with the plant computer will allow the unit operator to dispatch an operator to the affected PK panel to perform appropriate alarm response procedure. Remote troubleshooting of an electrical panel is not a recommended practice, and would require the control room operator to be away from normal responsibilities. Therefore, the proposed change does not adversely affect the safety functions or operations performed by the control operators. The change will eliminate unwanted nuisance alarms in the main control room, will not alter any plant accident analysis and will not affect the safety or health of the public.

Safety Evaluation: 59 1999-0101 **Revision:** 0

Abandonment of Boron Concentration Measurement System

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Configuration Change Package (CCP) 09061. CCP 09061 has been prepared to document that the Boron Concentration Measurement System (BCMS) has been abandon-in-place.

CCP 09061 adds notes to P&ID M-12BG02, "Chemical and Volume Control System," and System Description M-10BG Chemical and Volume Control System," stating that the BCMS is abandoned-in-place. The associated USAR change adds statements to those Updated Safety Analysis Report (USAR) Sections, Tables and Figures that describe or reference the BCMS and incorporates the revised P&ID M-12BG02 as USAR Figure 9.3-8-02, "Chemical and Volume Control System." The affected sections of the USAR are: Sections 7.7.1.10; 9.3.4.2.1.1; 9.3.4.2.1.3; 9.2.4.2.1.4; Table 7.7-2; Figures 7.7-10; 7.7-11; 7.7-12; 7.7-13; 9.3-8-02.

Revision 0 of CCP 09061 covered updating the BG System Description to reflect that the BCMS had been abandoned in place and also added the same information to the P&ID. As part of CCP 09061, the computer points associated with the BCMS were to be spared. The USAR change associated with this CCP was generated to make the appropriate notations in the USAR that the BCMS was abandoned in place.

In addition, Revision 0 of CCP 09061 also noted that procedure CKL BG-120, "CVCS Normal Valve Lineup,") and CKL BG-130, "CVCS Normal Switch & Breaker Lineup," needed to be updated to reflect that the BCMS had been abandoned in place, i.e., to isolate the BCMS. Both of these check lists were listed as affected documents. The CCP 09061 has been revised to Revision 1 to cover additional System Description as well as USAR clarifications. Revision 1 of CCP 090601 is evaluated by another USQD.

Safety Summary:

The BCMX was designed for use as an advisory system. The BCMS was not designed as a safety system or component of a safety system. The BCMS is not part of a control element or control system, nor is it designed for that use. No credit is taken for the BCMS in any safety analysis. There is no additional impact on the performance of plant activities nor affect on any system, structure or component (SSC).

The USAR change associated with CCP 09061 adds clarification to the Sections, Tables and Figures of the USAR that describe or reference the BCMS that the BCMS has been abandoned-in-place. No other portions of the USAR are made untrue by this change. This USAR change does not identify any test or experiment not described in the USAR.

Safety Evaluation: 59 1999-0102 **Revision:** 0

Alternate Location for Taking Tritium Samples

Description:

Configuration Change Package (CCP) 09066 provides new locations for taking tritium samples from the containment atmosphere by tapping the return lines of the containment atmosphere monitors GT-RE-31 and GT-RE-32. The return lines have redundant containment isolation valves powered from different electric separation groups. Therefore, one of the containment isolation valves in each containment penetration will close, if a single failure were to occur in the solid state protection system. The new sample points are being added to resolve a concern with the original sample point at valve GSV0030. The containment isolation valves associated with this valve are powered by the same electric separation group.

Safety Summary:

The addition of sample points in the return lines from the containment radiation monitors will not have any adverse impact on the function or performance of the radiation monitors because the sample valves will be normally closed. The tritium sample will be taken one train at a time. The inflow of air into the skid used for grabbing tritium samples is less than the normal air flow through the radiation monitor. Since the sample is taken after the radiation monitor, air quality being sensed by the radiation monitor is not affected.

This modification affects the intake line to the hydrogen analyzer and the containment atmospheric monitors return lines. Neither these monitors nor their inlet and outlet lines initiate any design basis accident or hazards. The inlet and outlet lines penetrate containment and have containment isolation valves. These valves receive an isolation signal during design basis accidents. This function is not affected by this modification.

This modification has no impact on any accident previously evaluated in the Updated Safety Analysis Report (USAR). Therefore, this modification will not increase the probability of occurrence of an accident.

Because his modification has no impact on any accident previously evaluated in the Updated Safety Analysis Report (USAR), the radiological consequences of an accident previously evaluated are not impacted.

This modification has no impact on the malfunction of equipment important to safety. Therefore, the probability of occurrence of a malfunction of equipment important to safety is not impacted.

Because this modification has no impact on malfunction of equipment, the radiological consequences of a malfunction of equipment important to safety is not impacted.

This modification does not create the possibility of a different type of accident than previously evaluated in the USAR because the modification will be installed using the existing design and installation specifications.

This modification has no adverse impact on the performance of any equipment important to safety. Therefore, the modification will not create the possibility of malfunction equipment important to safety than any previously evaluated in the USAR.

There is no margin of safety associated with the hydrogen analyzer, containment atmosphere monitor or their associated piping specified in any technical specifications.

Safety Evaluation: 59 1999-0103 **Revision:** 0

Computer Cable Installation Alternate Flame Test Requirements

Description:

Updated Safety Analysis Report (USAR) Sections 8.1.4.3, 8.3.1.4.1.4 and Table 9.5-1, Sheet 32 are being revised to allow electronic computer cables to meet either IEEE 383-1974, or UL 1666 (3rd Edition) flame test requirements.

Safety Summary:

Referenced in various sections of the Updated Safety Analysis Report (USAR) are statements and requirements that electrical cabling must meet the requirements of IEEE 383-1974 "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations". Included within this standard (section 2.5) is a flame test methodology which all cable must meet in order to be used within the power block at Wolf Creek Generating Station (WCGS). A specific need has arisen in which Category 5 computer cable must be installed to support the Nuclear Plant Information System (NPIS) Human Machine Interface (HMI) Y2K project (CCP 07870). Wolf Creek Nuclear Operating Corporation (WCNOC) has been unable to procure this cable type qualified to the flame test requirements of IEEE 383-1974.

However, the available vendors and manufacturers for this computer industry standard cable will supply this cable certified to UL 1666 (3rd Edition - February 27, 1997) "Test For Flame Propagation Height of Electrical and Fiber Optic Cables Installed Vertically in Shafts". A review and comparison of the two standards identified that even though the testing methodology is different, the evaluation and results required of the test will achieve the same end results. The primary difference in the testing methodologies is that the flame source and method for initiating the flame is different. The methodology in how the flame is applied to the cable(s), the manner in which the cable(s) are attached and the duration of the tests is nearly identical. The applicable sections within the USAR which specifically reference IEEE 383-1974 and require that cable meet the flame test requirements will be revised to allow either IEEE 383-1974 or UL 1666 (3rd Edition) for when a Flame Test is required. The only restriction for the use of UL 1666 for cable to be installed within the power block is that the cable installation shall be utilized for non-safety related applications only. The rationale behind this limitation is predicated on the knowledge that an ongoing and significant amount of discussion and work is being performed within the nuclear industry concerning cable qualifications and specifically fire hazards analyses.

By limiting the use of the UL code to only non-safety related applications, this will aid in any analysis and justification work required. The required separation of non-safety related and safety related cables into different cable trays (i.e. for safety related cables (separation groups 1-4) and for non-safety related applications (separation groups 5 and 6) will ensure that any cable procured to UL 1666 (3rd Edition) will not jeopardize in any manner safety related installations or functions.

The various fire analyses postulated throughout Section 9 of the USAR will not be affected by the use of cable procured to UL 1666 (3rd Edition) when the cable is installed in non-safety related applications. In the event that an error is made and cable is procured and installed in a safety related application, a comparison, evaluation and analysis of the two standards will additionally show that there will be no affect on any Design Basis Accident. The two standards will perform the same function and will arrive at the same evaluation results as far as flame testing is concerned.

There are no credible accidents which could be created by this revision to the USAR. This evaluation and analysis is based on the comparison of the two standards (IEEE 383-1974 and UL 1666 (3rd Edition) and the existing limitations on cable separation which will still exist. The limitation being placed on the installation of any cable procured to UL 1666 (3rd Edition) is viewed as a conservative measure when the standards are compared against each other.

There are no credible accidents which could be created by this revision to the USAR. This evaluation and analysis is based on the comparison of the two standards (IEEE 383-1974 and UL 1666 (3rd Edition) and the existing limitations on cable separation which will still exist. The limitation being placed on the installation of any cable procured to UL 1666 (3rd Edition) is viewed as a conservative measure when the standards are compared against each other. Since the NRC has not specifically evaluated UL 1666 (3rd Edition) against IEEE 383-1974, the affect on safety related cable which might be procured against this standard (UL 1666) might be questioned.

By limiting the installation of any cable procured to UL 1666 (3rd Edition) to non-safety related applications, there can be any affect, either directly or in directly, on any equipment which is important to safety. Therefore no malfunctions are identified.

There are no acceptance limits contained within the bases of either the Technical Specifications or the licensing basis which will be affected by this proposed revision to the USAR. Therefore, the margin of safety is not reduced.

Safety Evaluation: 59 1999-0105 **Revision:** 0

Updated Safety Analysis Report Clarification Relating to Liquid Radwaste

Description:

This activity incorporates clarifications and corrections to Updated Safety Analysis Report (USAR) Section 15.7.2, "Radioactive Liquid Waste System Leak or Failure," and 15.7.3, "Postulated Radioactive Release Due to Liquid Tank Failures," as part of the corrective action identified by the USAR Fidelity Review Team. These editorial changes include correcting USAR section references and correcting the name of the Solid Radwaste System to make it consistent with the rest of the USAR.

Also, text is added to USAR Section 15.7.2 to clarify the assumption of the Physical Model to include an explanation that the evaporator bottoms tank was selected vice the spent resin tank due to the increased likelihood of iodine released to the atmosphere.

Finally, the baseline calculations for the postulated liquid radwaste tank leak of rupture assumes that 10% of the iodine inventory is released as airborne activity. (ref. Bechtel FSAR Calculation 7.6.2-11-10466 and NSA Calculation SA-92-093). The USAR section for the accident, 15.7.2, incorrectly states the assumed iodine activity released to the atmosphere is 1%. The results and consequences for the accident presented in the USAR, Tables 15.7-5 and 15.7-6, correctly state the results as based on the assumed 10% inventory release. This change will correct Section 15.7.2.5.1.2d and Table 15.7-5 to indicate that 10% is the assumed amount of iodine activity released to the atmosphere, not 1% as currently stated.

Safety Summary:

The proposed changes provide minor text changes to add clarity to the USAR discussion in Section 15.7 for the postulated radwaste tank rupture. Also included, is a correction to make the USAR text and tables consistent with the basis calculations that support the analysis and consequences discussed in USAR Section 15.7. As a result of these changes the clarity and accuracy will be improved. There are no physical changes to any plant system, structure, or component and the information in the USAR will be true and accurate as a result of the changes. The changes do not affect any procedure, test or experiment, therefore none are identified. he postulated accident consequences remain unchanged.

Safety Evaluation: 59 1999-0107 **Revision:** 0

Updated Safety Analysis Report Changes

Description:

1. Updated Safety Analysis Report (USAR) Section 15.1.2.2, is revised to change the reference to the reactor trip function from "high-high" to "low-low" steam generator water level. Calculation AN-95-056, Revision 2, "Cycle 8 Core Follow Oct 1 to Nov 1, 1995," takes credit for the "low-low" steam generator water level. Additionally, on page 15.1-6 it states that the reactor is tripped when the "low-low" steam generator level is reached. The "high-high" level setpoint does not provide for a direct reactor trip, rather it provides an Engineered Safety Features (ESF)actuation function.

2. USAR Table 15.0-2 (sheet 2), for Incident 15.4, Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature, the change from indicating RETRAN to specifying the LOFTRAN code for the analysis performed. The RETRAN code was not used in the analysis of this incident, the LOFTRAN code was used to analyze the Startup of an Inactive Loop at the Incorrect Temperature.

3. USAR Section 15.0.9.2 is revised to explicitly indicate that the fuel clad gap activity, consistent with Table 15A-3 which is also referenced in the paragraph, is assumed to be 10 percent of the core activity for all isotopes except for Kr-85, where it is assumed to be 30 percent of the core activity.

4. USAR Table 15.0-6 and USAR Figures 15.0-7, 15.0-8, 15.0-11, 15.0-12, 15.0-16, 15.0-21, 15.0-22, 15.0-24, 15.0-25 and 15.0-27 are revised to relate them together and bring them into agreement with the major parameters that are discussed in the text for each of the respective accidents or listed in other tables or figures.

Safety Summary:

Changes proposed make the USAR consistent with Chapter 15 analyses of record. No procedures, activities, administrative controls, sequences of plant operations, plant structures, systems, components (SSCs) or equipment, or requirements are impacted by the change and thus the proposed activity would not invalidate USAR information or requirements. Since the proposed change ensures consistency between the USAR description and Chapter 15 analyses of record, the change would not adversely affect the mitigative capability of any SSCs, nor affect the ability of any SSC to prevent an accident.

Safety Evaluation: 59 1999-0108 **Revision:** 0

Reanalysis of Radiation Source Terms

Description:

The radiation source terms have been reanalyzed by Westinghouse with assumptions and parameters commensurate with the current plant operation and the foreseeable changes on fuel management program. The major assumptions and parameters used in the analysis include:

· Effective Full Power Days (EFPD) - 510 EFPD (18-month cycle with a 30 day refueling and 98% capacity factor),
· Enrichment equal to or less than 5.0 w/o -consistent with Spent Fuel Pool (SFP) rerack modification,
· Core average cumulative burnup at end-of-cycle - 38,400 MWD/MTU, and
· Discharge burnup equal to or less than 56,200 MWD/MTU (average assembly).

As a result of the reanalysis of the radiation source terms, the following tables in the Updated Safety Analysis Report (USAR) are revised based on the revised information provided by Westinghouse.

1. USAR Table 11.1A-1 lists the plant parameters and assumptions used in the radiation source term calculation.
2. USAR Tables 11.1-1, 11.1-4 and 11.1-5 list the reactor coolant and secondary coolant specific activities for assumed fuel defects of 0.12%,0.25%, and 1% respectively.
3. USAR Table 11.1-6 (sheets 13 and 16) list the liquid waste sources contained in the boron recycle holdup tank and the evaporator bottoms tank (primary).

Safety Summary:

The proposed updating of the radiation source information would only affect certain dose consequences evaluations because these source terms were explicitly utilized in the dose consequences analyses. However, the proposed activity does not change any administrative control which would reduce the effectiveness of existing programs, reduce the qualification of Wolf Creek Nuclear Operating Corporation (WCNOC) personnel, nor does it affect the performance of activities that are important to the safe and reliable operation of WCGS.

The proposed change is to ensure the parameters and assumptions used in the radiation source terms analysis and the subsequent analysis results reflect the current plant operations and the fuel management strategy. The proposed activity will affect the dose consequences analyses for Chapter 15 accidents. However, no procedures, activities, administrative controls, or sequences of plant operations are impacted by the proposed change.

There are no physical modifications to the systems, structures, or components (SSCs). Therefore, no credible accidents that could be created are identified.

The proposed updating of USAR Chapter 11 table, pertaining to the radiological source term information, to reflect the increased cycle length, higher enrichments and higher burnup levels used in the reanalysis and its results, would not increase the radiological consequences of the Chapter 15 accidents.

Since the proposed changes do not involve any design changes nor are there any changes in the method by which any safety related plant system performs its safety function, no credible malfunctions of equipment important to safety are affected.

The effect of a longer operating cycle decreases as the cycle length is increased and does not usually have a large effect on the total fission product energy released. An increased cycle length, higher enrichments and higher burnup levels will affect individual fission product nuclide concentrations. That is, certain isotopes will be present in increased amounts, while concentrations of other isotopes will remain the same or decrease slightly because of isotopic half-life differences. In addition, the specific activities of the primary and secondary coolant remain unchanged. Therefore, the resulting radiological dose rates will be about the same and the calculated accident dose values are expected to be close to that presented in the current USAR and remain below the 10 CFR 100 guideline limits.

The proposed changes would not affect the I-131 dose equivalent limits for the specific activities of the primary and secondary coolant, which are limited by Technical Specification Limiting Conditions for Operation (LCO) 3.4.8 and 3.7.1.4. The proposed changes do not affect the manner regarding how safety limits or limiting safety system settings are determined, not will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are affected by this change and the margin of safety is not affected.

Safety Evaluation: 59 1999-0109 **Revision:** 0

Clarification of Compressed Air System

Description:

Configuration Change Package (CCP) 09094 addresses discrepancies in the Compressed Air System air receivers associated to air compressors (CKA01A, CKA01B, and CKA01C). These discrepancies involve the actual volume of the tanks and the design pressure information in the field different than the design documents identify.

The proposed changes involve document changes to incorporate the correct volume and design pressure for air receiver tanks (TKA01A, TKA01B, and TKA01C). The compressors and the tanks are located in the Turbine Building. The changes are as follows:

1) the volume of the tank has been changed from 57 cu. ft. to 52 cu. ft.

2) the change in the volume above and the design pressure of the tank have affected the associated engineering calculations, which required them to be revised,

3) the change in volume and design pressure have also affected the associated Drawings M-050-00036, M-050-00037, M-050-00038, system description M-10KA, and specifications M-050, M-050A, M-050B, which required them to reflect the correct volume of the tanks.

The calculations have been revised to reflect the design pressure of 125 psig and volume 52 cu. ft. for the tanks. The calculations involved are KA-M-002, "Compressed Air Receiver Sizing Verification," KA-286, "Compressed Air System," and KA-338, "Instrument Air Requirements and Flow Parameters at Nodal points of Compressed Air System and Air Receiver Size." The calculations were performed to calculate : (1) the time duration of the tank air volume that would be available until the stand by compressors come on line in the event the lead compressor fails, (2) calculate the internal energy of the tank, and (3) the required tank size (volume).

Safety Summary:

The calculated time and the air tank volume are within the acceptable limit. The internal energy of the tank is below the previously calculated internal energy and identified in the USAR. The reduction in the energy should not be a concern as there is no adverse impact on the plant performance. There is no safety related system that is affected due to this change. There is no impact on the plant or the operation of the plant or change in fit, form, function, or material due to the subject design documents changes. The changes only corrects the documents. No new instrument/component has been added or no existing instrument component has been modified for the system. The changes are non-safety related.

Safety Evaluation: 59 1999-0110 **Revision:** 0

Essential Service Water Piping Change

Description:

The Essential Service Water (ESW) System pipe connection between the 8 inch main supply line and the 4 inch branch line which supplies SGF02B (auxiliary feedwater pump room cooler), will be reconfigured. The present configuration has the 4 inch branch line connecting to the bottom side of the 8 inch line. The new configuration will have the 4" branch line come out the top of the 8 inch line. The existing vent valve (EFV0193) will not be reinstalled when the existing section of 8 inch line is cut out and replaced. The existing chemical addition valve (EFV0362) will serve as a vent and chemical addition point. The vent connection will change location from upstream to downstream of the branch connection for SGF02B supply line.

The proposed modification will reduce the amount of silt, debris and clam shells entering safety related room cooler SGF02B.

When the 6 inch supply line to PAL01B (auxiliary feedwater pump) is flushed through valve ALV0157 (B ESW to B motor driven auxiliary feedwater pump suction flush valve) the flow velocity in the 8" line is less than 2 ft/s. The slow moving debris, silt and clam shells fall into the 4 inch branch supply line due to gravity. By relocating the suction of the branch line from the bottom to the top of this 8 inch line, the slow moving debris should be flushed out.

Safety Summary:

The proposed modification will add a pressure drop of less that 0.3 psi to the supply piping for SGF02B (Reference calculation supplement to EF-11-W, Revision 3, Essential Service Water Differential Pressure Normal Operation [Service Water Supplied] With 90°F and 95°F Inlet Temperature). The outlet valve (GFV008) of SGF02B is throttled per procedures STN PE-037B, "ESW Train B Heat Exchanger Flow and DP Trending," and STN OQT-001B, "Operations B Train Quarterly Tasks," to meet the minimum flow requirements of USAR Tables 9.2-2, 9.2-3, 9.2-4, and 9.4.8 (sheet 7), which is 100 gpm. This minor pressure drop will not have an adverse effect on the ESW System flow balance. By throttling SGF02B outlet valve GFV008 to maintain minimum required flow rates, the heat removal capacity is unchanged for this plant modification. There is no design basis change in heat removal capacity of SGF02B.

There is no change in venting requirement due to relocation from down stream to upstream of vent valve position relative to supply branch line connection. The existing configuration had vent valve EFV0193 and chemical addition isolation valve EFV0362 less than 2 feet apart on the same horizontal run approximately 2 feet upstream of the 4" branch connection. By relocating the tee connection approximately 6' upstream and deleting valve EFV0193, the vent valve EFV0362 is now located down stream of this branch connection. This will not create any venting problems for PAL01A (motor driven auxiliary feedwater pump A) or SGF02B due to being on the same horizontal run.

The modified ESW System piping configuration will meet all ASME pipe stress requirements and maintain system pressure boundary.

The new piping configuration is designed to seismic, ASME Section III Code, CL-3 requirements per USAR Table 3.2.1. Pipe stress and pipe support loads are below allowable stress limits for the new pipe configuration. Therefore the new pipe configuration and existing pipe supports for this section of ESW pipe will maintain pressure boundary during normal plant conditions and all design basis accidents.

The fire protection analysis for Room 1206 will be unaffected by this plant modification.

A Fire Protection Water System was installed in room 1206 by Design Change Package 04585. The new configuration of 4" piping installed was moved upstream to avoid interference with junction box 1UJ029 (which is covered with Darmat for added fire protection) and to stay out of the spray pattern for the closest sprinkler head which provides coverage for this junction box. Also by having the new vertical pipe configuration on the back side of the 8" line, the spray pattern is unaffected for this junction box and motor-operated-valve ALHV0031.

The accidents to be reviewed will involve the ESW system and the motor driven auxiliary feed-water pumps (MDAFWP). The design basis function of safety related room cooler SGF02B is remove heat from room 1325 to allow PAL01B to perform its design basis function. The design basis function of the ESW supply line to PAL01B and SGF02B is to maintain system pressure boundary.

The design basis accidents reviewed rely on the AFW System.

1) 15.1.4 Inadvertent opening of a steam generator relief or safety valve.
2) 15.1.5 Steam system piping failure.
3) 15.2.6 Loss of non-emergency AC power to the station auxiliaries.
4) 15.2.7 Loss of normal feed-water flow.
5) 15.2.8 Feed-water system pipe break.
6) 15.6.3 Steam generator tube rupture.
7) 15.6.5 Loss of coolant accidents resulting from a spectrum of postulated piping breaks with the reactor coolant pressure boundary.

The new pipe configuration of the ESW supply line to safety related room cooler SGF02B and PAL01B will maintain pressure boundary for all plant conditions. Therefore none of the above design basis accidents are affected by this plant modification.

Safety Evaluation: 59 1999-0111 **Revision:** 0

New Procedure for Secondary Radiation Monitor Setpoint Calculations

Description:

This change implements procedure AI 21D-004, "Secondary Radiation Monitor Setpoint Calculations," which contains instruction on methods to calculate setpoints for appropriate secondary process radiation monitors. The procedure provides the methodology to adjust radiation monitor alarm setpoints to provide prompt indication of a primary to secondary leakage under varying plant conditions. As communicated through Electric Power Research Institute (EPRI) TR 104788-R1, "PWR Primary-to-Secondary Leak Guidelines-Revision 1," the industry standards have changed. Monitor setpoints are calculated to provide detection of leakage in the range of the first operating condition at which action is required (5 gpd). Listed in Updated Safety Analysis Report (USAR) Tables 11.5-1 and 11.5-3 are the process radiation monitors, GE-RE-92 (condenser air discharge monitor), BM-RE-25 (steam generator blowdown process radiation monitor) and SJ-RE-02 (steam generator liquid radioactivity monitor), requiring adjustment of the alarm setpoints based on plant conditions.

Safety Summary:

Changing the setpoints based on Reactor Coolant System (RCS) activity levels, condenser inleakage flowrate, steam generator blowdown rate and radiation monitor background levels may result in monitor alarm setpoints for GE-RE-92, SJ-RE-02, and BM-RE-25 different from those listed in USAR Table 11.5-1 and 11.5-3. Although the monitor alarm setpoints may differ from those listed in USAR Tables, the change provides for effective use of Radiation Monitoring System (RMS) setpoints to provide prompt indication of primary to secondary leakage. The proposed activity does not increase the likelihood of exceeding effluent radioactivity release limits of 10CFR20 and 10CFR50, Appendix I, but enhances the identification of leakage that requires compensatory actions.

USAR Table 11.5-1, footnotes (3) & (4), and Table 11.5-3, footnotes (9) & (10), states "Setpoint may be changed during a monitored S/G tube leak to provide prompt indication of increased leakage". The footnote in the Tables should give guidance to adjust the setpoints based upon varying plant conditions as recommended in EPRI TR 104788-R1.

The design basis accident in Chapter 15.6, Decrease in Reactor Coolant Inventory, caused by a Steam Generator Tube Rupture, was reviewed for possible impact created by the proposed activity.

No credible accidents could be created by changing the monitor setpoints. The operation, testing, design function and maintenance of the radiation monitors remains the same and the 10 CFR 20 and 10 CFR 50, Appendix I, limits for gaseous/liquid radiological releases are still applicable.

Changing the setpoints on radiation monitors BM-RE-25, SJ-RE-02, and GE-RE-92 provides for process control and alerts the operators of prompt indication of a primary to secondary leak. No credible malfunctions of equipment may be induced by the proposed activity. Adjusting the setpoints based upon plant conditions provides for a more aggressive approach to identifying primary to secondary leakage and does not affect the isolation function of any effluent liquid or gaseous release monitor.

Technical Specification Limiting Condition For Operation 3.4.6.2.c, states that a limit of "1 gpm total reactor-to-secondary leakage through all steam generators not isolated from the Reactor Coolant System and 500 gallons per day through any one steam generator" is required. The leakage limit of 1 gpm for all steam generators ensures the dosage contribution from the tube leak will be limited to small fraction of 10 CFR 100 dose guidelines in the event of a steam generator tube rupture or steam line break. The 500 gpd leak limit per steam generator assures steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. The acceptance limits contained in the bases of Technical Specification 3/4.4.6 are not affected by adjusting process radiation monitor setpoints to better identify a primary to secondary leak with sensitivity of 5 gpd, which is one percent of limit. Compensatory actions are presently in place in procedure AP 21D-001, "Primary to Secondary S/G Leakage," to provide guidance to stay within the acceptance limits of Technical Specification 3.4.6.2.

Safety Evaluation: 59 1999-0113 **Revision:** 0

Clarification of Separation Criteria

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes to the Updated Safety Analysis Report (USAR) Section 9.5B.7 A.27.

Change 1. This change provides the basis for why the reactor trip switchgear (SB102A/B) circuitry is fail safe. 10 CFR 50 Appendix R, Section III.G.3 allows the use of an alternate method in lieu of 10 CFR 50 Appendix R, Section III.G.2 separation requirements. The use of an alternate method will be added to USAR Section 9.5B.7, Power Block Fire Hazards Analysis, Fire Area A.27.

USAR Section 9.5B A.27.6 currently identifies specific raceways in the fire area. However, USAR Section 9.5B.6 states that listing specific raceways is inappropriate since the safe shutdown equipment is listed in Table 9.5B-2. Additionally, the section currently and incorrectly states that no equipment required for safe shutdown is located in the fire area. However, USAR Table 9.5B-2 identifies the sage shutdown equipment in the fire area. This change is made to provide consistency.

Change 2. This change removes the raceway list and adds a reference to USAR Table 9.5B-2 for safe shutdown equipment in the Fire Area A-27. This change makes the Fire Area A-27 discussion consistent with the remainder of the fire areas identified in the Fire Hazards Analysis. A second change involves revising USAR Section 9.5B to identify the means used to achieve control rod insertion. The existing analysis had omitted that discussion.

During a review of Fire Area A-27, it was determined tht the reactor trip switchgear (SB102A?B) and their related raceway in Fire Area A-27, Room 1403 do not meet the separation requirements as identified in 10 CFR 50 Appendix R, Section III.G.2. Both trains of the reactor trip switchgear are located adjacent to each other with no physical separation between the two cabinets in Fire Area A-27. An analysis of the failure modes for the circuits and the breakers determined that a fire that damaged both switch gear could result in failure of both reactor trip breakers to open.

USAR Section 9.5B is being revise to identify the alternate means of performing the control rod insertion function if the reactor trip switchgear breakers are damaged. The current safe shutdown analysis does not address the control rod insertion capability in the event of a fire in Fire Area A-27, Room 1403.

This USAR change and associated Unreviewed Safety Question Determination is performed to document that a diverse means exists to achieve a plant trip and rod drop using equipment that is outside Fire Area A-27 and that for a fire in Fire Area A-27, the plant can achieve and maintain safe shutdown.

Safety Summary

This change evaluates an existing condition and provides clarification and justification for meeting the regulatory requirements.

The credible malfunction of the reactor trip switchgear is that both breakers would be damaged and fail in the closed position preventing control rod insertion. The changes will provide an alternate means of achieving control rod insertion. No other credible malfunctions of safety-related equipment can occur due to this change.

The worst case postulated failure for the reactor trip switchgear is a fire in the switchgear cabinets causing both breakers to become damaged and remain n their closed position. Under this condition, initiation of a reactor trip using the normal reactor trip switchgear circuitry will not be possible.

However, control rod insertion can be accomplished by using hand switches PGHIS16 and PGHIS3, located in the Control Room to de-energize load centers PG19 and PG20 respectively. The circuits for PGHIS16 are located outside of Fire Area A-27. PGHIS3 trips PA0207 to de-energize PG20 to initiate control rod insertion for reactor trip switchgear PG19 and PA0207 are non-safety related and serve no safe shutdown function.

A design basis fire for Fire Area a-27 assumes failure of all circuits and equipment in the fire area due to a worst case fire. No new equipment malfunctions are created by this change. This change does not create any new failure modes or accidents. All fire induced failures have been evaluated and it has been determined that safe shutdown can be achieved.

USAR Section 9.5B2 b. states that "A design basis accident occurring simultaneously with a fire hazard is not assumed." Therefore the only design basis accident assumption to be made is a fire in Fire Area A-27, Room 1403. Conservatively, no loss of off site power is assumed and the motor generator sets continue to operate which prevents control rod insertion.

The changes provide clarification and justification for meeting the regulatory requirements. Therefore the changes will not create any new credible accidents and will not affect any acceptance limits contained in the bases for the technical specifications.

Safety Evaluation: 59 1999-0114 **Revision:** 0

Correction of Fire Protection Terms

Description:

This change to the Updated Safety Analysis Report (USAR) is initiated to correctly reflect the use of the terms fire zone or zone within USAR Section 9.5. In many places, the term zone or fire zone has been used instead of the correct terminology, fire area. Listed below are the affected sections and a description of the change:

1. Section 9.5.1.2.2.3, page 9.5-21, revised zone to fire area and fire zone to fire area.
2. Section 9.5.1.1.1, revised combustion zone to fire area.
3. Section 9.5.1.2.2.5, revised zone 1405 to Fire Area A-26, zone to fire area, zone 1408 to Fire Area A-16.
4. Table 9.5A-1, sheet 5, revised fire zone to fire area.
5. Section 9.5B.6, revised fire zone to fire area.
6. Section 9.5B.7, revised fire zone to fire area.
7. Section 9.5B.7, revised fire zone to fire area.
8. Table 9.5.1-3, added the term "detection" in front of the word zone to clarify that the discussion applied to fire detection zones out of service.
9. Table 9.5B-2, revised fire zone to fire area.
10. Table 9.5B-3, revised fire zone to fire area.
11. Table 9.5B-4 revised fire zone to fire area.
12. Table of Contents for Table 9.5B-3, 9.5B-4 descriptions revised fire zone to fire area.

USAR Appendix 9.5B, "Fire Hazards Analysis," states in Section 9.5B-1, "As shown on Figure 9.5.1-2, the safety related areas of Wolf Creek Generating Station (WCGS) have been divided into numbered fire areas. This appendix is arranged by the fire area numbers shown on USAR Figure 9.5.1-2." Throughout USAR Section 9.5.1 and its tables and appendices, there are references and discussions regarding the WCGS fire areas. The whole basis for WCGS compliance with NRC fire protection regulations is that for a fire in any single fire area, the plant can achieve safe shutdown. In a few instances, the term fire zone was inappropriately used in place of the term fire area.

Changes 1, 2,4-7, and 9-12 all involve changing the phrase fire zone to fire area. Change 3 is made to the section that discusses WCGS design and administrative controls for handling ion exchange resin. In two locations in the discussion, reference was incorrectly made to zone 1405 as the area where fresh resin are loaded into the vessels. The resin loading chute is located in Room 1405, which is in Fire Area A-26 as identified on USAR Figure 9.5.1-2. In both locations, the reference to zone 1405 is being corrected to state Fire Area A-26. Additionally in the discussion, there is an incorrect reference to hose stations in adjacent zone 1408. The hose station being referenced is shown on USAR figure 9.5.1-2 and it is located in Room 1408, which is Fire Area A-16. The reference to zone 1408 is being corrected to Fire Area A-16.

Change 8 is made to Table 9.5.1-3, "Fire Protection System Technical Requirements." Item 1 of the table discusses the operability requirements for fire detection instrumentation. In the discussion, the term zone is used when referring to a zone of detection. To clarify the terminology, the USAR is being changed to use the phrase "detection zone."

Safety Summary:

These changes do not affect the inputs or assumptions used in the Fire Hazards Analysis. No equipment failure modes are affected by this change and no new equipment failures can be created by this change. This change is made solely to provide consistency in USAR Section 9.5.1, its tables and appendices with regard to the plant use of "fire areas". All existing analysis and bases for fire safe shut as described in the USAR is not affected by this change. Only the USAR Section 9.5-1 accident relating to plant fires have been reviewed for potential impact by this change.

Based on the above evaluation, this change does not effect any of the inputs, assumptions, or conclusions used to evaluate the impact of fire for any plant fire areas. This change does not create any new failure modes or accidents. This change is made solely to provide consistency in the USAR in discussions regarding

Safety Evaluation: 59 1999-0115 **Revision:** 0

Containment Flood Depth and Recirculation Sump Velocities

Description:

Configuration Change Package (CCP) 09102 revises the containment flood depths and the containment recirculation sump velocities following a loss of coolant accident (LOCA)/main steam line break (MSLB) listed in the system description of M-10EN, "System Description Containment Spray System." This change revises Updated Safety Analysis Report (USAR) Table 6.2.2-9 to reflect the new recirculation sump velocities. The containment flood depths changed because the containment flood levels following a LOCA or MSLB were revised recently by CCP 08037 and approved by USQD 59 99-0075. The average approach velocities to containment recirculation sump trash racks remain within 0.01 and 0.08 fps as previously described in USAR Section 6.2.2.1.3. NPSH was evaluated in DCP 08037 and hence it is not discussed in this evaluation.

Safety Summary:

Wolf Creek Generating Station (WCGS) is committed to Regulatory Guide 1.82, which specifies that sufficient screen area be provided to keep the coolant flow velocity at the screen approximately 0.20 ft/sec. This velocity ensures that debris with a specific gravity of 1.05 or more is settled before reaching the screen.

The velocities of the coolant approaching the containment recirculation sump trash rack remain within 0.01 and 0.08 fps as stated in the USAR. The velocities of the coolant through the trash rack (50 percent clogged) and the inner screen (50 percent clogged) have increased in some cases by approximately 0.01 fps. The new velocities are 0.20 fps or less except for two cases. At the containment spray pump switchover following a LOCA, they increase from 0.29 fps to 0.30 fps through the trash rack and from 0. 0.28 fps to 0.29 fps at the inner screen.

This increase in the velocities is not significant enough to adversely impact any safety related pump performance drawing water from the sump. The switchover phase lasts for a short duration only. The fluid velocity approaching the sump screen for the long term cooling continues to be less than 0.20 fps. Thus, the intent of Regulatory Guide 1.82 is met.

The safety function of the residual heat removal (RHR) pumps and the containment spray system (CSS) pumps depend upon their ability to draw adequate amounts of water from the containment sump. The revised recirculation sump velocities have no adverse impact on RHR or CSS pump performance. Therefore, the proposed change has no impact on any Chapter 15 accidents previously analyzed in the USAR.

The proposed change cannot create a new type of accident as there is no adverse impact on the performance of safety related pumps that draw water from the sumps following an accident. The sumps have no other function. The revised sump velocities are not significantly different from the previous values described in the USAR.

The changes in the containment recirculation sump velocities and the flood depths have no adverse affect on the performance of CSS or RHR pumps as there is sufficient margin in the net positive suction head available for the RHR and CSS pumps. The recirculation sumps do not provide suction water to any other critical equipment important to safety. Therefore, there are no new credible malfunctions of equipment important to safety.

There are no acceptance limits contained in the bases for the technical specifications for the containment recirculation sump velocities following an accident.

Safety Evaluation: 59 1999-0117 **Revision:** 0

Installation of New Chlorine Monitor

Description:

This modification provides for replacement of the chlorine monitor, 1WM028E, "Chlorine Detector/Monitor," which is installed in the raw water house.

The scope of the proposed change includes complete removal of the existing monitor 1WM028E, installation of the new remote sensor unit, chlorine transmitter, 24VDC power supply and associated wiring. These will be installed in the same area utilizing the existing cable, conduit and power source.

Safety Summary:

The form, fit, and the design of the proposed new monitor are different from the present monitor. The proposed modification using an monitor of different design is functionally equivalent to the existing monitor. This will indicate as well as initiate an alarm function when the atmospheric chlorine in the raw water house exceeds or equals to the alarm setpoint.

The design features of the new monitor will improve upon the reliability of Chlorine Monitoring System in the raw water house. This monitor has no safety related function.

The above proposed change will affect the Updated Safety Analysis Report (USAR) Figure 9.2-5, "Demineralized Water Makeup System."

The chlorine monitor, 1IWM028E has no safety related function. The activities for the removal of the existing monitor and installation of the new monitor will not impact directly or indirectly the design basis accidents as discussed or referenced in the USAR Sections 2, 3, 6, 9, or 15.

The proposed activities will not create any credible accidents. The proposed activity is non-safety related. The proposed activity will not affect directly or indirectly any safety related structures, systems or components (SSCs). The operation of the Makeup Demineralizer System will not be affected by the changes.

The operation of the Makeup Demineralizer System will not be affected by the proposed activities. There are no acceptance limits in the bases to the technical specification nor in the licensing basis documents that could be affected by replacement of the chlorine monitor.

Safety Evaluation: 59 1999-0118 **Revision:** 0

Condenser Hot Well Sample Changes

Description:

This modification provides for the addition of instrumentation and an acceptable level of electrical isolation capabilities for the condenser hotwell sample pump skid (RM-185).

There are two problems being addressed by this proposed change. The first problem addresses the premature failure of the condenser hotwell sample pumps. The proposed modification will provide a time-delayed interlock circuit to stop the pumps when the discharge pressure falls below a preset value. This will require the installation of a pressure switch, along with tubing, connections, support hardware, four time delay relays, and any other necessary appurtenances, at the RM-185 skid. At present, whenever a sample pump loses its prime, it will continue to run until failure due to overheating.

The second problem involves the capability to electrically isolate the sample pumps. This modification will provide an auxiliary enclosure which will contain a three-phase disconnect switch, two pole fuse block and fuses, and control transformer at the RM-185 skid. Electrically isolating the sample pumps through the addition of the disconnect switch and control transformer will also require some minor wiring changes within the RM-171 recorder panel. At present, the single disconnect switch for the entire skid requires de-energization of both pumps and the solenoid valves whenever one pump requires maintenance activities. The single disconnect switch also allows the spare pump motor to be inadvertently energized even though its pump is mechanically isolated.

The addition of a pressure switch to the condenser in-leak detection return line affects Updated Safety Analysis Report (USAR) Figure 9.3-4-03, "Process Sampling System," which reflects the piping & instrumentation diagram (P&ID) for the Process Sampling System. This P&ID and USAR Figure will need to be revised to indicate the addition of pressure switch RM-PS-0589.

No design basis accidents were identified that needed to be reviewed for potential impact due to adding a pressure switch, a second disconnect switch, four time delay relays, associated fittings, and hardware.

The proposed change provides additional instrumentation and an acceptable level of electrical isolation capabilities for the RM-185 skid. There is no safety related equipment in the vicinity of the condenser in-leak detection skid. No new credible accidents would be created by this change.

There is no affect on any structures, systems, or components (SSCs).

Therefore no credible malfunctions of equipment important to safety are identified.

Neither the condenser in-leak detection skid nor the Process Sampling System are in the technical specification or any of the technical specification bases. No acceptance limits were identified that could be affected. Therefore, the margin of safety is not affected by this change.

Safety Evaluation: 59 1999-0120 **Revision:** 0

Control Room Building Ventilation System

Description:

Design Change Package (DCP) 06207, Revision 0 which was evaluated by USQD 59 96-0079, approved removal of thermal relief valves, GKV0769, GKV0770, GKV0771, and GKV0772 from the Control Room air conditioning units (SGK04A/B) and the Class 1E equipment room air conditioning units (SGK05A/B). This DCP allowed capping the associated piping tee with blind flange.

Configuration Change Package (CCP) 07960, Revision 0, approved replacement of an inlet nozzle from a tee assembly to a straight piping piece. The relief valve and the tee assembly were parts of the air conditioning skid and not shown on the piping isometric drawings. During replacement of a tee assembly with a straight piping piece in the field, it was identified that though the tee connection was not shown on the piping isometric drawing; the tee connection is shown on the piping and instrument diagram (P&ID).

Revision 1 of CCP 07960 revises the affected P&IDs and the associated USAR Figure 9.4-1-03, "Control Building HVAC."

Safety Summary:

The function of the Control Building Ventilation System is to provide conditioned outside air for ventilation and cooling to different levels of the control building for personnel and equipment. The system is required to function following a design basis accident and to achieve and maintain the plant in a safe shutdown condition. Inlet nozzle and outlet nozzle are the transient pieces of the condensing unit which connect the piping system through which the cooling media, water flows.

CCP 07960, Revision 0, approved the change from a tee connection to a straight pipe connection. This revision is issued only to incorporate this change into the P&ID drawings which was missed at that time. There is an administrative change package issued to correct the design basis drawings.

This modification changes a tee assembly with a straight piping assembly. It does not change any design basis function of the structure, system or components. This change does not initiate any design basis accident or create any hazards. Therefore, there are no design basis accidents affected by this modification.

By replacing a tee assembly with a straight piping assembly, there is no compromise of leakage of the pressure boundary. Both fittings have the passive function of maintaining the pressure boundary. This modification will not create any other types of credible accidents.

This is only a change in the inlet nozzle pipe fitting of the air conditioning units. This modification will not affect any equipment important to safety.

There are no acceptance limits impacted by this change.

Safety Evaluation: 59 1999-0122 **Revision:** 0

Corrections to the Fire Protection USAR Figures

Description:

Configuration Change Package (CCP) 09110 proposes changes to Updated Safety Analysis Report (USAR) Figures 9.5.1-01, "Fire Protection System (power block)," 9.5.1-02, "Fire Protection System (power block)" 9.5.1-03," Fire Protection System (power block)" 9.5.1-04, "Fire Protection (Halon) System," and USAR Section 9.5B.7, Fire Area C.6. These changes are initiated based on the recommendation of the USAR Fidelity Review Team.

A. USAR Figures 9.5.1-01, 9.5.1-02, 9.5.1-03, and 9.5.1-04 indicate that Room 3201 (elevations 2000' through 2087') has a two hour fire resistive rating. It has been determined from drawings A-0343, A-0904, and A-0905 that the interior walls for the stairway enclosure at these elevations were designed to have a three hour rating. Since the enclosure walls, barrier penetration seals, and heating ventilation and air conditioning (HVAC) fire dampers have a minimum three hour fire resistive rating, the stairway at elevations 2000' through 2087' has been re-evaluated to have a three hour fire resistive rating. The proposed change will revise USAR Figures 9.5.1-01, 9.5.1-02, 9.5.1-03, and 9.5.1-04 to indicate that Room 3201 has a three hour fire resistive rating.

B. The re-evaluation of the fire restrictive rating for Room 3201 at elevations 2000' through 2087' (Change A above) has made the first sentence of USAR Section 9.5B.7,C.6.2 unclear. Section 9.5B.7,C.6.2 will be clarified by stating that the two hour fire resistive rating applies only to the stairway enclosure at the 1984' elevation.

Safety Summary:

The proposed changes to the USAR are for consistency only and do not affect any system, structure or component (SSC) nor do they change the performance of activities that are important to the safe and reliable operation of WCGS. They reflect actual plant design.

The proposed changes do not impact any procedures, activities, administrative controls, or sequences of plant operations nor are any plant structures, systems, components or equipment impacted. No requirements outlined in the USAR are revised by these changes. Therefore, the changes do not affect any design basis accidents or any credible accidents. The changes will not create any new malfunctions of equipment important to safety and do not impact any margins of safety as defined in the bases of the technical specifications.

Safety Evaluation: 59 1999-0125 **Revision:** 0

Security Plan Change

Description:

This Unreviewed Safety Question Determination (USQD) evaluates changes incorporated into Revision 31 of the Wolf Creek Generating Station (WCGS) Physical Security Plan, Safeguards Contingency Plan, and Security Training and Qualification Plan.

This revision to the Security Plan provides for revising the definition of the secondary alarm station (SAS) and central alarm station (CAS) operators to allow the other operator to complete functions assigned to the primary operator. This will allow for timely responses when the one operator is busy, the other one can perform the function.

This revision to the Security Plan provides for revising terminology used in the Security Plan to accurately describe the functions of the new Security Computer system. (i.e. valid ACAD to active ACAD). All required functions still exists with the new Security system.

This revision to the Security Plan provides for increasing the committed number of armed responders by one.

In two paragraphs when talking about alarm indication on certain devices the Security Plan stated that it would be an intrusion alarm. These devices are more consistent with what we currently use as a tamper switch on cabinets. This revision of the Security Plan will allow the use of either type of alarm (intrusion or tamper) to alarm these cabinets.

This revision added compensatory measures in Security Plan to give detailed description of compensatory measures for loss of a multiplexer or a complete computer system loss. This will define exact measures to be taken rather than left up to interpretation.

This revision removed detailed information from Safeguards Contingency Plan on Local Law Enforcement Agencies (LLEA) response capabilities. Wolf Creek is still required to have letters on file identifying LLEA response capabilities. This change reduces Security Plan changes every time a new letter is received from a LLEA.

Safety Summary:

Changes being made do not effect any safety related system that could cause a design basis accident. No credible accident can be created by the non-safety related changes due to their lack of interface with safety related equipment.

The changes identified do not interface with any safety related equipment that could cause a malfunction of equipment important to safety. None of the changes are tied to the WCGS technical specifications and do not affect plant operability.

Safety Evaluation: 59 1999-0126 **Revision:** 0

Temporary Procedure to Inject Hydrogen Peroxide into the Floor Drain Tank

Description:

This temporary procedure, TMP 99-005, "Floor Drain Tank Chemical Treatment," provides instructions to inject hydrogen peroxide into the Floor Drain Tank (THB01A) to remove organics (oil) from the fluid. The hydrogen peroxide is injected via a temporary pump and tygon tubing through 'A' Floor Drain Tank's suction flush valve. Thirty percent hydrogen peroxide is an acid (2.5 to 3.5 pH). The addition of hydrogen peroxide to the floor drain tank to treat the contents will not change the function of the system as described in the Updated Safety Analysis Report (USAR) as the system is designed to receive acid or caustic as necessary to treat the contents for pH adjustments. The use of the temporary pump and injection tubing will be a change to USAR Figure 11.2-1-02, "Liquid Radwaste System."

Safety Summary:

USAR Section 3.4.1.1.2, "Internal Flooding Protection," states: "All safety-related equipment rooms located below grade are protected from back-flooding by the remote location of waste-processing components in the radwaste building. The floor and equipment drains in powerblock seismic Category I buildings drain to sumps in the lowest level of the building in which they are located. These sumps are pumped to the floor drain tank or the waste hold-up tank located in the radwaste building. Should these tanks rupture or leak, flow into safety-related areas will not occur since these tanks are located below radwaste building flood level." Since this temporary procedure is associated with the floor drain tank, the USAR Chapter 3 analysis of internal flooding protection remains valid.

In USAR Section 15.7.2 the rupture of the boron recycle holdup tank (RHUT) and primary evaporator bottom tanks (EBT) are evaluated as the worst liquid accident in the power block. In USAR Section 15.7.3, the rupture of the refueling water storage tank is evaluated as the worst accident outside the power block.

Since the volume of the floor drain tank is less than the volumes analyzed in Chapter 15 of the USAR, this procedure is bounded by the existing USAR analyses and no design basis accidents are affected.

Hydrogen Peroxide is a strong oxidizing agent/acid that reacts slowly with the organics in the floor drain tank to generate carbon dioxide gas. The carbon dioxide gas that has entered the floor drain tank will be vented to the radwaste HVAC system via the normal floor drain tank vent. Therefore, no unmonitored liquid or gases will be released to the environment.

There is no safety related equipment within the vicinity of the temporary pump or hoses. Therefore, any water from a ruptured hose will not damage any safety related equipment. Since the system's functions are not changed, no credible accidents that could be created are identified.

The floor drain tank and temporary injection components in the radwaste building are not safety related nor are they located near safety related equipment. Therefore, no malfunctions of equipment important to safety will be affected by the proposed activities.

Since there are no acceptance limits associated with the floor drain tank, no acceptance limits could be affected.

Safety Evaluation: 59 1999-0127 **Revision:** 0

Temporary Modification to Provide Cooling to Battery Room

Description:

SGE17, "Battery Room. No. 1 HVAC Unit Communications Corridor," provides cooling to PK03 and PK04 battery rooms. SGE17 has failed and no alternate cooling is provided. To preserve the life of the equipment located in these rooms, an alternate cooling method is provided.

This temporary modification will provide cooling to the battery rooms. A supply duct for the computer room (Work Controls Center) is in close proximity to the battery room. A fan will be placed next to an open access hatch and the air will be directed to the battery room using an elephant trunk. The trunk will be routed through one of the doors to the room and the other door will be opened to provide an exhaust path.

Ventilation is supplied to the battery rooms for equipment life issues and for removal of any accumulation of hydrogen from the batteries. This temporary modification will provide adequate cooling and venting of the rooms.

This temporary modification will allow the cooling for the Work Controls Center units to provide cooling to both the PK03 and PK04 battery room and the computer room while SGE17 is out of service.

Safety Summary:

Hydrogen build up in the battery room is the only credible accident identified. Normally the room is closed and a small volume of air is constantly exhausted to prevent the build up of hydrogen to an explosive limit in the room. Local indication of hydrogen concentration is provided to alert the operators of an explosive level of hydrogen in the room. This temporary modification will still allow air to be exhausted from the room. If the ventilation is lost, the doors are opened which will prevent the build up of hydrogen to an explosive limit under the temporary modification. The configuration under the temporary modification is an improved configuration compared to the normal configuration for preventing this type of accident. This temporary modification will still adequately prevent this type of accident

Loss of cooling to the PK03 and PK04 battery room will shorten the life of equipment located in the room, but will not cause a malfunction of equipment in the room. The equipment located in this room is non-class 1E and is not important to safety. Loss of cooling to the computer room (Work Controls Center) will not cause equipment important to safety to malfunction since this area is currently being used as office space.

The equipment affected by this temporary modification is not important to safety and is not governed by technical specifications.

Safety Evaluation: 59 1999-0129 **Revision:** 0

Installation of Temperature Indicating Controller for Valve EATV0007

Description:

Configuration Change Package (CCP) 09146 provides for the installation of a temperature indicating controller (EATIC0007) for valve EATV0007. The function of the valve is to control hydrogen temperature in the turbine generator hydrogen coolers. This function is accomplished by controlling service water flow through the hydrogen coolers. This controller will allow Operations personnel to have either automatic control or manual control of valve EATV0007. This controller will also allow the transfer between manual control and automatic control without impacting system operation. The proposed effects will be to allow Operations personnel to swap from manual control to automatic control of the valve with no problems or impacts to system operation. This is a non-safety related CCP involving non-safety related systems and components.

The only impact that this change will have on the Updated Safety Analysis Report (USAR) is the change to the USAR Figure 9.2-1-02, "Service Water System." The only change is the addition of the controller to the control lines of the valve (EATV0007). There are no tests or experiments not described in the USAR that are impacted by this change. This change involve a non-safety related system and non-safety related components.

Safety Summary:

This CCP installs a reliable temperature indicating controller which will improve the operation of the Hydrogen Cooling System. This CCP affects only non-safety related components (EATV0007 and EATIC0007) and a non-safety related system (Service Water System). This CCP is not directly or indirectly associated with any safety related components, systems or structures. This change cannot initiate or impact any design basis accidents. Therefore there can be no design basis accidents discussed or referenced in the USAR that need to be reviewed for impact by the proposed CCP.

These changes will not affect in any manner any failure mechanism or mode for any safety related components of any plant system. There are no credible malfunctions of equipment important to safety which may be directly or indirectly affected by the proposed activity.

No acceptance limits are specified in the bases of technical specifications (or licensing basis documents) for this system. As such, the margin of safety is not impacted.

Safety Evaluation: 59 1999-0130 **Revision:** 0

Changes to the USAR for Consistency with the Fuel Handling Accident

Description:

The proposed Updated Safety Analysis Report (USAR) changes, identified during the USAR Fidelity Review and constituting a portion of the corrective actions to disposition of Performance Improvement Requests (PIRs) 98-2546, 98-2591, and 98-1693, are provided by the following:

1) The description in USAR Section 15.7.4.5.2c is being revised, consistent with the current licensing basis, to indicate during a postulated fuel handling accident that radioactive material is released immediately from the reactor building containment to the environment and the auxiliary building and also that subsequent releases from the auxiliary building and the fuel building releases occur over a two hour period. No mixing of radioactive material released in the containment is assumed.

2) Clarify the general descriptions USAR Sections 15.7.4.2 and 15.7.4.3, to explicitly indicate the applicability of the postulated fuel handling accident in containment and explicitly indicate applicability only to the postulated fuel handling accident from the fuel building USAR[Section 15.7.4.5.2d.

3) Clarify the description to indicate that the "postulated fuel handling accident" is being discussed [Section 15.7.4.5.1.1] and that the statement that "only one assembly can be handled at a time" refers to the refueling machine, transfer system, and spent fuel pool bridge crane during the offload in the context used.

4) Clarify the description by removing a redundant phrase pertaining to the benefit of the normal purge exhaust filters, not credited in the licensing basis analysis USAR Section 15.7.4.5.2e.

Safety Summary:

In summary, the proposed activity does not change any administrative controls which would reduce the effectiveness of existing programs, reduce the qualification of WCNOC personnel, nor does it affect any systems, structures, and components (SSCs). The proposed activity does not change the performance of activities that are important to the safe and reliable operation of WCGS. The proposed USAR changes make the USAR consistent with the revised USAR Chapter 15 postulated fuel handling accident radiological consequence analysis of record. No procedures, activities, administrative controls, sequences of plant operations, plant structures, systems, components or equipment, or requirements are impacted by the change and thus the proposed activity would not invalidate USAR information or requirements. Since the proposed change ensures consistency between the USAR description and Chapter 15 postulated fuel handling accident radiological consequence analyses of record, the change would not adversely affect the mitigative capability of any SSCs, nor affect the ability of any SSC to prevent an accident.

Safety Evaluation: 59 1999-0132 **Revision:** 0

Test of Prototype Automatic Voltage Adjust Card in the "A" Emergency Diesel Generator Static Exciter Voltage Regulator

Description:

This Unreviewed Safety Question Determination (USQD evaluates Temporary Modification 99-011NE As part of continuing efforts to correct the volts active reactive (VAR) surge phenomena on "A" Emergency Diesel Generator(EDG). The problem has been traced to the automatic voltage adjust card in the static exciter voltage regulator. This card's design is susceptible to noise which may occur simultaneous with an actuation of the control switch. This noise can cause the card to change its output, in the desired direction, by significantly more than the amount desired by the operator. Troubleshooting in the lab has resulted in a prototype of a re-designed card which corrects these problems. Lab testing confirms that the previous problems with operation of this card are no longer present. To confirm the problems encountered in the field on A EDG have been corrected, the prototype card must be installed in the A EDG and tested. This will be performed using a temporary modification, the plant work order process and presently approved plant procedures for EDG operation. During the evolution the "A" EDG will be considered inoperable and the appropriate Technical Specification actions taken. This USQD evaluates this evolution as a potential "test or experiment" as discussed in 10 CFR 50.59. This card is described in Technical Manual M-018-00905, in the voltage regulator section.

The proposed changes consist of:

1) Changing resistance values to match the signal impedance to the logic type being used. The card uses a mixture of CMOS and diode logic. The CMOS should have high impedance, while the diode logic works better with low impedance. The card as built uses essentially all high impedance connections.

2) Installing "power supply decoupling capacitors" at the IC chips to increase their ability to withstand noise.· Connecting the counter reset to ground, as recommended by the vendor in drawing notes.·

A3) adding a small capacitive load to the input gate from the debounce circuitry to slow down the gate.

Safety Summary:

The failure modes of the Voltage regulator Adjuster card which is the subject of the Temporary Modification will have the failure mode of Fail High(+10V output)/Fail Low(0V output)/Fail as-is/Fail power supply shorted. The adjuster card output is limited by design to prevent extremes which might place the EDG or equipment supplied by the EDG in immediate jeopardy.

The changes made to the prototype card were designed to shield the card from adverse external influences and not such that would change the original operation and design function. The failure modes of the prototype card are the same failure modes of a qualified card.

Fail power supply shorted will result in shutdown of excitation and the output breaker will trip open on loss of field, separating the EDG from the emergency bus with no loss of power to the emergency bus. Other protective relays may also actuate, providing the same results.

Fail as-is will maintain a constant level of excitation, and the VAR load of the diesel will vary in response to the voltage changes from the switchyard. Over a relatively short period of time, these changes will be minor, and will not result in any safety challenges. However, should a drastic change in yard voltage occur, the generator overcurrent relay may trip, and open the output breaker without affecting the normal power supply. Should a LOP subsequently occur, resulting from the transient that originally caused the yard voltage change, the DG will remain available to supply the emergency bus although the output voltage will be fixed at its failure point.

Fail Low will result in the generator becoming a reactive load with a commensurate increase in current. The loss of field relay may act to open the output breaker, separating the EDG from the emergency bus. Should the EDG remain on-line, the increased current resulting from the increased reactive power may trip the overcurrent relay, with a subsequent output breaker trip, separating the EDG from the emergency bus. No loss of the normal power supply to the emergency bus will result.

Fail High will result in the generator supplying greater reactive (VAR) power. Again, the increased current resulting from the increased reactive power may trip the overcurrent relay, with a subsequent output breaker trip, separating the EDG from the emergency bus. No loss of the normal power supply to the emergency bus will result.

Should the output breaker not be tripped because it did not reach the setpoints of the protective relaying, the increased reactive power will be reflected in the switchyard. However, it will remain within the capability of the emergency bus and normal power supply to that bus, allowing the operators to take compensatory action to correct the condition or open the output breaker.

The greatest consequences to the voltage regulator adjuster card failing high or low is to the generator. Depending on the real load at the time, it may be found to be operating outside the generator capabilities curve. This is a degradation over time consequence, which is evaluated and tracked by engineering. This would not result in immediate concern regarding the ability of the EDG to perform its function, and has no effect on the emergency bus.

The prototype card, prior to modification had been installed in the field, and its ability to perform its function was established. With the modification subsequently installed, the prototype card was extensively test in the laboratory both for its function, and for its ability to correct the noted VAR problem. The ability of the card to perform its function was not in question. The purpose of the test was to provide field evidence and experience to verify that it did indeed correct the noted misoperation of the unmodified adjuster card. The intent of the test was not to prove that it would perform its function.

Again, the failure modes resulting from the prototype card are not changed. No protection is bypassed or disabled by this temporary change or by the procedures which accomplished the testing. Any of these failures, with the qualified card or with the prototype card are fully expected to result in all protective action necessary to maintain the emergency bus powered from the normal power supply.

The description in the USQD details the design features that will prevent loss of offsite power resulting from a failures of the voltage regulator adjuster card. A statement to this effect would not add further depth to the USQD.

This change does not affect the USAR, as this change will only be installed for a short time under these documents. Based on the results of this evolution, similar changes may be permanently installed under separate modification documents.

The overall results of these changes to the design of the voltage regulator electronic adjuster card are that the card will function essentially the same as before, except that the susceptibility of electrical noise causing an undesirable change in output signal is greatly reduced.

The equipment of concern in this evolution is used for accident mitigation. It is not associated with accident initiation sequences and the likelihood of an accident occurring is not increased by this change.

Radiological consequences of an accident are dependent on proper functioning of accident mitigation equipment. Based on one train of accident mitigation equipment being available, as required by Technical Specifications, the proposed change will not increase the consequences of an accident previously evaluated in the USAR.

The EDG will be considered inoperable by Technical Specification for the entire time this card is installed. Because the EDG is considered "failed" and the appropriate actions taken, the probability of a malfunction of this equipment is unchanged by this evolution.

The worst case failure resulting from this evolution could result in the loss of one essential bus and its associated train of equipment temporarily, if a loss of offsite power occurred concurrent with this evolution because the EDG is considered inoperable in this case. This malfunction is enveloped by the existing accident discussion in the USAR. There is no increase to the probability of a malfunction of plant equipment important to safety. The radiological consequences of a malfunction do not increase.

This temporary modification affects accident mitigation equipment and has no effect on the possibility of occurrence of accidents. Therefore, this evolution does not create a new type of accident.

No acceptance limits are identified that might be challenged by this change. Therefore, the margin of safety is not affected by this temporary modification.

Safety Evaluation: 59 1999-0133 **Revision:** 0

Change in Motor-Operated-Valve Stroke Times

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a change to the Updated Safety Analysis Report (USAR). The stroke times for the motor-operated valves (MOVs) listed below will be changed as follows:

EMHV8801A/B & EMHV8803A/B will increase from 10 seconds to 20 seconds.
BNHV8812A/B will increase from 17 seconds to 45 seconds.
EJHV8804A/B will increase from 15 seconds to 30 seconds.

This change in stroke time results from an increased overall gear ratio for the affected valves. The affected valves have functions associated with the Emergency Core Cooling System (ECCS). USAR Table 6.3-1 describes the ECCS MOVs as having a 15 second maximum opening/closing time for valve 8-inches and under, and a stroke time based on size for larger valves. For BNHV8812A/B, the only affected valve larger than 8 inches, this calculates to a maximum opening/closing time of approximately 17 seconds. Therefore a USAR change is required to make exceptions for these valves on the affected USAR table.

DISCUSSION

As a result of Wolf Creek Nuclear Operating Corporations' (WCNOCs') response to NRC Generic Letter 95-07, "Pressure Locking and Thermal Binding (PL/TB) of Safety-Related Power Operated Gate Valves," the Nuclear Regulatory Commission (NRC) has requested further information in the form of a Request for Additional Information (RAI) dated April 1, 1999 (WCNOC Letter 99-00482). Per the RAI and discussions with the Commission, WCNOC has utilized a more conservative criteria to evaluate MOV gate valves for pressure locking and thermal binding than originally used in the Dominion Report DEI-411 (E-025-00038, Revision 1).

This criteria is based on a combination of the methodology and increased margins used by Commonwealth Edison and adopted by WCNOC, and by conservative positions adopted by other utilities throughout the industry when determining susceptibility to pressure locking and thermal binding.

The new criteria necessitated a revision to the original susceptibility report developed for WCNOC by Dominion Engineering, Inc. The revision increased the number of valves now considered susceptible to pressure locking and thermal binding. As a result, it has been determined by using the Commonwealth Edision methodology and the Commonwealth Edision recommended margins that several of the MOVs in the PL/TB scope, included those added in response to the RAI, do not have sufficient thrust margin to meet the acceptance criteria established for resolution of the RAI.

In response to the RAI, the identified valves have been reanalyzed by Engineering using Preslok, a computer analysis developed by Commonwealth Edison and endorsed by the Westinghouse Owners Group (WOG) and the NRC. Engineering has also applied this analysis with regard to the minimum margins applied between the calculated pressure-locking thrust and actuator capability. Each Maximum Expected Differential Pressure (MEDP) calculation and Torque/Thrust (T/T) Calculation has been revised as applicable for the respective valves. In cases where the affected valves cannot meet the T/T margin acceptance criteria, gear ratios will be changed as required.

The proposed change will alter the stroke time values listed for the ECCS in USAR Table 6.3-1. The affected table will be revised to identify the new calculated values. Additionally, stroke time surveillance procedures for the affected valves will be revised to identify the new opening and closing limits as part of the design change process. No tests or experiments are identified for the affected valves.

EJHV8804A/B (Ref: USAR Table 6.3-8)

EJHV8804 A & B are the isolation valves between the centrifugal charging pump (CCP) and safety injection pump, respectively, suctions and the residual heat removal (RHR) heat exchanger header. These valves are closed during normal power operation. They are opened during the cold leg recirculation phase of emergency safety injection.

Several limiting scenarios were examined. Fast operation limits the switchover time from injection to the recirculation mode of operation. Manual actions are performed from the Control Room to complete the changeover operation from the injection mode to the recirculation mode. These actions during switchover encompass a sufficient time period to allow a relaxation in the valve stroke time from 15 seconds to 30 seconds. USAR Tables 6.3-8, 12, and 11 identify an allowable opening time of 40 seconds. The modification is therefore acceptable.

Additionally, Westinghouse passive failure requires leaks be isolated in 30 minutes. A relaxation in valve stroke time to 30 seconds appears acceptable relative to the requirements associated with this function.

BNHV8812A/B

These valves provide isolation for the refueling water storage tank (RWST) from the RHR pumps, so that the suction of the pumps can be changed during safety injection. The subject valves have to be opened to provide suction for the RHR pumps from the RWST in place of the Reactor Coolant System (RCS) in the event of a loss of RHR.

The limiting scenario of a RWST outflow during a Large Break LOCA (valve closes on a safety injection signal (SIS) coincident with RWST Lo-Lo level) with or without a single failure assumption was evaluated. Fast operation limits the switchover time from injection to the recirculation mode of operation. In addition, procedure EMG ES-12 "Transfer to Cold Leg Recirculation" and associated responses were examined. The relaxation in stroke time to 45 seconds, for the valve to automatically close during RHR pump switchover from injection to recirculation, is not anticipated to have a significant impact on the overall operator response time for the manual actions.

EMHV8801A/B

EMHV8801 A & B are part of the RCS boundary and isolate the boron injection tank (BIT) from the RCS. They open on a safety injection signal and are manually opened during RCS leakage and loss of shutdown cooling events as part of operations to add water to the RCS. These valves are closed during normal power operation. They are opened during safety injection and other emergency procedures as part of operations to add water to the RCS. The valves close when needed by emergency procedures.

Examination of current BIT operation including the BIT analyzed boron concentration, BIT path usage including use as a redundant safety related flowpath (inlet valve), and charging pump/safety injection pump crossover use for the outlet valve, indicates that a relaxation in valve stroke time to 20 seconds is acceptable relative to the requirements.

EMHV8803A/B

EMHV8803 A & B isolate the BIT from the CCPs. They open on a safety injection signal and are manually opened during RCS leakage and loss of shutdown cooling events as part of operations to add water to the RCS. These valves are closed during normal power operation. They are opened during safety injection and other emergency procedures as part of operations to add water to the RCS. The valves close when needed by emergency procedures.

Examination of current BIT operation including the BIT analyzed boron concentration, BIT path usage including use as a redundant safety related flowpath (inlet valve), and charging pump/SI pump crossover use for the outlet valve, indicates that a relaxation in valve stroke time to 20 seconds is acceptable relative to the requirements.

Safety Summary:

The proposed changes will provide added assurance that the valves will perform their designed safety function. The increased stroke times still fall well within the response times required to perform those functions. Since the valves are still capable of performing their design basis function within the time required for accident mitigation, and there are no new failure modes introduced, there is no potential for the creation of any credible accident.

The proposed modifications replace the motor pinion and worm shaft gears. The replacement will increase the overall gear ratio of the actuators with gears identical to those currently installed, except for the number of teeth. The replacement gears are functionally identical to the existing gears. Therefore, there is no potential for introducing new failure modes or increasing the likelihood of existing credible failure modes. The indirect effects of the proposed change to valve stroke times have also been evaluated.

Since the valves still operate within the designed response times required to perform their safety function there is no acceptance limit that is affected by the change.

Safety Evaluation: 59 1999-0133 **Revision:** 1

Change in Motor-Operated-Valve Stroke Times

Description:

Revision 1 of this Unreviewed Safety Question Determination (USQD) was approved for incorporation in to the Updated Safety Analysis Report (USAR) by the Plant Safety Review Committee on January 26, 2000. However, it is being reported for the 1999 reporting period. Revision 0 to USQD 59 1999-0133 is also reported for the 1999 reporting period. Revision 1 of provides an enhanced evaluation of this change to the USAR. The stroke times for the motor-operated valves listed below will be changed as follows:

The USAR Table 6.3-1 stroke times for the motor-operated valves (MOVs) listed below will be changed as follows:

1. Revise USAR Table 6.3-1 (Sheet 4) by adding to the MOV opening/closing time exceptions the following valves:

· EMHV8801A/B and EMHV8803A/B - the maximum opening/closing time will increase from 15 to 20 seconds (CCP 09116);

· BNHV8812A/B - the maximum opening/closing time will increase from 17 seconds to 25 seconds (CCP 09120);

· EJHV8804A/B - the maximum opening/closing time will increase from 15 seconds to 30 seconds (CCP 09118).

The change in stroke time results from an increased overall gear ratio for the affected valves. The affected valves have functions associated with the Emergency Core Cooling System (ECCS). USAR Table 6.3-1 describes the ECCS MOVs as having a 15 second maximum opening/closing time for valves 8-inches and under, and a stroke time based on size for larger valves. For BNHV8812A/B, the only affected valves listed above larger than 8 inches, the current maximum opening/closing time is approximately 17 seconds. Therefore, a USAR Change Request is required to make exceptions for these valves on the affected USAR table.

Note: These changes are a result of the corrective actions which resolve the NRC's Request for Additional Information to NRC Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding (PL/TB) of Safety-Related Power Operated Gate Valves," identifying non-conservative calculations of the Maximum Expected Differential Pressure (MEDP) of these MOVs. The reconfigured MOVs, with their upgraded/modified motor pinion and worm shaft gear sets (optimized MOV valve actuators), will yield enhanced component performance and will improve reliability via the gained additional safety function margin.

2) Revise the minimum time available for the operator to accomplish the switchover of the ECCS pumps from the injection to the recirculation mode

to 9.39 minutes from 9.6 minutes in USAR Section 6.3.2.2 and Table 6.3-11. This revision is necessary to allow the maximum opening/closing time of MOVs BNHV8812A/B to be increased from 17 seconds to 25 seconds.

3) Revise the Case 1 and Case 2 time length of injections, at flow conditions of two trains of ECCS/two trains spray without and with a worst postulated single failure respectively, to 22.50 and 21.44 minutes respectively in USAR Table 6.2.2-4. This revision is necessary to allow the maximum opening/closing time of MOVs BNHV8812A/B to be increased from 17 seconds to 25 seconds.

ADDITIONAL DISCUSSION

As a result of Wolf Creek Nuclear Operating Corporation's (WCNOC's) response to NRC Generic Letter 95-07 "Pressure Locking and Thermal Binding (PL/TB) of Safety-Related Power Operated Gate Valves" the Commission has requested further information in the form of a Request for Additional Information (RAI) dated April 1, 1999 (WCNOC Letter 99-00482). Per the RAI and discussions with the NRC, WCNOC has utilized more conservative criteria to evaluate MOV gate valves for pressure locking and thermal binding than that originally used in the Dominion Report DEI-411 (E-025-00038, Revision 1).

This criteria is based explicitly on a combination of the methodology and increased margins used by Commonwealth Edison and conservative positions adopted by other licensees for the determination of susceptibility to pressure locking and thermal binding.

The criteria necessitated a revision to the original susceptibility report developed for WCNOC by Dominion Engineering, Inc. The revised report indicated an increase in the number of valves considered susceptible to pressure locking and thermal binding. Consequently, application of the criteria resulted in identifying several MOVs in the PL/TB scope, included those added in response to the RAI, not having sufficient thrust margin to meet the acceptance criteria established for resolution of the RAI.

To disposition the RAI, the identified valves have been reanalyzed by Engineering using Preslok, a computer analysis method developed by Commonwealth Edison and endorsed by the Westinghouse Owners Group (WOG) and the NRC. Engineering has also applied this analysis with regard to the minimum margins applied between the calculated pressure-locking thrust and actuator capability. Each MEDP calculation and Torque/Thrust (T/T) Calculation has been revised as applicable, for the respective valves. In cases where the affected valves cannot meet the T/T margin acceptance criteria, gear ratios will be changed as required.

The proposed change will alter the stroke time values listed for the ECCS in USAR Table 6.3-1. The affected table will be revised to identify the new calculated values. Additionally stroke time surveillance procedures for the affected valves will be revised to identify the new opening and closing limits as part of the design change process. No tests or experiments are identified in the USAR for the affected valves.

Assuming a BNHV8812A/B valve stroke time of 25 seconds, indicated a reduction in available operator action time to accomplish the ECCS pump switchover from the current 9.6 to 9.39 minutes, with associated Case 1 and Case 2 flow conditions, with or without a postulated single failure, spray injection phase duration revisions, while the operator action time for a Large Break LOCA with the worst postulated single failure remained at 8.3 minutes. Accordingly, changes to USAR Tables 6.3-11 and 6.2.2-4 and Section 6.3.2.2 are proposed. No procedure revisions are necessary.

EJHV8804A/B

EJHV8804 A & B are the isolation valves between the centrifugal charging pump (CCP) and safety injection pump, respectively, suctions and the residual heat removal (RHR) heat exchanger header. These valves are closed during normal power operation. They are opened during the cold leg recirculation phase of emergency safety injection.

Several limiting scenarios were examined. Fast operation limits the switchover time from injection to the recirculation mode of operation. Manual actions are performed from the Control Room to complete the changeover operation from the injection mode to the recirculation mode. These actions during switchover encompass a sufficient time period to allow a relaxation in the valve stroke time to 30 seconds and this conclusion is supported by Calculation CCN BN-M-013-001-CN002, "RWST Volume Requirements for Injection ECCS Containment Spray Pumps Transfer Time Available for Operator Actions." The modification is therefore acceptable.

Additionally, Westinghouse passive failure requires leaks be isolated in 30 minutes. A relaxation in valve stroke time to 30 seconds is acceptable relative to the requirements associated with this function.

BNHV8812A/B

These valves provide isolation for the refueling water storage tank (RWST) from the RHR pumps, so that the suction of the pumps can be changed during safety injection. The subject valves have to be opened to provide suction for the RHR pumps from the RWST in place of the reactor coolant system (RCS) in the event of a loss of RHR.

Various limiting scenarios were examined including the limiting scenario of a RWST outflow during a Large Break LOCA (valve closes on a safety injection signal coincident with RWST Lo-Lo level) with or without a single failure assumption. The increase in valve stroke time affects the minimum time available for operator manual actions to accomplish the switchover from the injection to the recirculation mode of operation.

CCN BN-M-013-001-CN002, assuming a BNHV8812A/B valve stroke time of 25 seconds, indicates a reduction in available operator action time to accomplish the ECCS pump switchover from the current 9.6 to 9.39 minutes, while the operator action time for a large break LOCA with the worst postulated single failure remained at 8.3 minutes. As the slight reduction in operator action time has been deemed acceptable, the relaxation in valve stroke time to 25 seconds is acceptable.

EMHV8801A/B and EMHV8803A/B

EMHV8801 A & B are part of the RCS boundary and isolate the boron injection tank (BIT) from the RCS. They open on a safety injection signal and are manually opened during RCS leakage and loss of shutdown cooling events as part of operations to add water to the RCS. These valves are closed during normal power operation. They are opened during safety injection and other emergency procedures as part of operations to add water to the RCS. The valves close when needed by emergency procedures.

EMHV8803 A & B isolate the BIT from the CCPs. They open on a safety injection signal and are manually opened during RCS leakage and loss of shutdown cooling events as part of operations to add water to the RCS. These valves are closed during normal power operation. They are opened during safety injection and other emergency procedures as part of operations to add water to the RCS. The valves close when needed by emergency procedures.

Examination of current BIT operation including the BIT analyzed boron concentration, BIT path usage including use as a redundant safety related flowpath, and charging pump/safety injection pump crossover use, indicates that a relaxation in valve stroke time to 20 seconds is acceptable relative to the requirements.

Safety Summary:

Prior to the implementation of the increased MOV stroke times affecting this ECCS realignment to recirculation mode, a quantitative analysis, documented in calculation BN-M-13 and updated in CCN# BN-M-013-001-CN001 and CCN# BN-M-013-001-CN002, was performed.

Calculation BN-M-013 provided the basis for the proposed USAR change specifying 9.6 minutes as the minimum time available for the operator to accomplish the switchover of the ECCS pumps from the injection to the recirculation mode. The two CCNs provided the basis for the proposed USAR change specifying 9.39 minutes as the minimum time available for the operator to accomplish the switchover of the ECCS pumps from the injection to the recirculation mode and the statement that the minimum time available for the operator to accomplish the switchover of the ECCS pumps for a large break LOCA with the worst postulated single failure remains unchanged at 8.3 minutes.

In addition to increasing the stroke time for the EJHV8804 and BNHV8812 valves, the two CCNs also incorporated additional changes based on updated operator action time information with respect to the revised procedures. These changes included 1) Assuming a reduction from 30 to 25 seconds in the increased times conservatively added to the maximum valve stroke time for each step that requires manual valve positioning, for calculating the maximum RWST outflow, and 2) Changing the period from 30 to 25 seconds of the operator action time assumed for each verification process, such as verification of the establishment of CCW to the RHR heat exchangers. The reduction of these assumed operator action times offset the effect of the assumed increased MOV stroke times so that the statement that the minimum time available for the operator to accomplish the switchover of the ECCS pumps for a large break LOCA with the worst postulated single failure remains unchanged at 8.3 minutes.

The proposed stroke time revisions provide added assurance that the valves will perform their designed safety function. The increased stroke times still fall well within the response times required to perform those functions. Since the valves are still capable of performing their design basis function within the time required for accident mitigation, and there are no new failure modes introduced, there is no potential for the creation of any credible accident.

The proposed modifications, resulting in the increased stroke times, replace the motor pinion and worm shaft gears. The replacement will increase the overall gear ratio of the actuators with gears identical to those currently installed, except for the number of teeth. The replacement gears are functionally identical to the existing gears. Therefore, there is no potential for introducing new failure modes or increasing the likelihood of existing credible failure modes. The indirect effects of the proposed valve stroke time revisions have also been evaluated.

Since the valves are anticipated to continue operating within the designed response times required to perform their safety function, there is no acceptance limit affected by the change.

Safety Evaluation: 59 1999-0135 **Revision:** 0

USAR Revision to Reflect Compliance with 10 CFR 50.68

Description:

This Updated Safety Analysis Report (USAR) Change Request adds compliance to 10 CFR 50.68 in USAR Table 1.3-4 and revises references to the exemption to 10 CFR 70.24 granted by the NRC on June 24, 1997, in USAR Sections 9.1.1.1.1, 9.1.2.1.1, and 12.3.4.1.1.2; and USAR Tables 1.3-4 and 12.1-1.

10 CF 70.24 provides criticality accident requirements from which Wolf Creek has been granted an exemption by the NRC. Replacement of the spent fuel storage racks and an associated increase in maximum allowed enrichment of the fuel will result in information provided as basis for the exemption to be inaccurate. Subsequent to obtaining the exemption, 10 CFR 50.68 was issued providing licensees the option of compliance with this regulation or 10 CFR 70.24. The requirements of 10 CFR 50.68 are generally the requirements that the NRC used to grant specific exemption from the requirements of 10 CFR 70.24. Compliance with 10 CFR 50.68 is preferred to a revised exemption request.

Safety Summary:

This change is limited to the scope necessary to meet the new requirements of 10 CFR 50.68. Therefore, no unreviewed safety question exists.

Safety Evaluation: 59 1999-0136 **Revision:** 0

Procedure Revision to Update Requirements for Camera Verification of Fuel Assembly Identification Numbers due to Spent Fuel Pool Rerack

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Revision 2 to procedure AP 19C-002, "Special Nuclear Material Safeguards and Accountability," which requires camera verification of fuel assembly identification numbers immediately prior to transfer from Region 1 to Region 2. The current spent fuel pool (SFP) racks consists of two storage regions. This verification is required by Updated Safety Analysis Report (USAR) Section 9.1A.7, "Administrative Control of Fuel Movement and Storage in Region 2."

Design Change Package (DCP) 07484, "Spent Fuel Pool Storage Expansion," replaces the current Spent Fuel Pool (SFP) racks with new racks consisting of three storage regions. Proposed Revision 2 to AP 19C-002 requires camera verification of fuel assembly IDs immediately prior to transfer into a region which requires more burnup. This change would generalize the current verification requirements to address fuel assembly transfers between the current and new racks, and fuel assembly transfers between regions within the new racks. The proposed change would make information contained in USAR Section 9.1A.7 incomplete. License Amendment No. 120, which incorporates the new SFP racks, has been approved by the NRC.

USAR Section 9.1A.7, which only addresses the current SFP racks, requires camera verification of fuel assembly IDs immediately prior to transfer from Region 1 to Region 2. Replacement of the current SFP racks with the new SFP racks, which adds three different storage regions, will make this information in the USAR incomplete.

Safety Summary:

Changes in administrative controls of fuel storage in the SFP can potentially impact SFP criticality analysis contained in USAR Section 9.1A.

The proposed change could only affect the amount of shutdown margin in the SFP. No other accidents could be created by this change.

The proposed change concerns administrative controls over ensuring that fuel assemblies are placed in proper SFP storage regions.

The proposed change would require identical controls for fuel assembly identification number verification prior to moving a fuel assembly to a more restrictive region (i.e., a region with higher burnup requirements) as that currently required in USAR Section 9.1A.7. A higher degree of controls for the new SFP racks is not required, nor considered necessary.

The probability of a SFP criticality accident would not be increased as a result of the proposed change.

The proposed change could not affect the amount of radioactivity released resulting from an accident, thus would not increase the radiological consequences of an accident previously evaluated in the USAR.

The proposed change could have no direct or indirect affect on plant equipment, thus would not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change could have no direct or indirect affect on plant equipment, thus would not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

The proposed change could have no direct or indirect affect on plant equipment, thus would not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

The proposed change does not affect any acceptance limits in the bases for the technical specifications nor in the licensing basis documents, thus would not reduce the margin of safety as defined in the basis for any technical specifications.

Safety Evaluation: 59 1999-0137 **Revision:** 0

Changes to Fire Barrier Figures Resulting From the USAR Fidelity Review

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Configuration Change Package (CCP) 09170 which proposes changes to Updated Safety Analysis Report (USAR) Figures 9.5.1-2, "Fire Area Delineation," Sheets 02 and 04. The proposed changes are based on comments documented in Performance Improvement Request (PIR) 98-2360 which was initiated to perform a detailed USAR Fidelity Review of USAR Section 9.5B.7, "Power Block Fire Hazards Analysis."

Safety Summary:

Figure 9.5.1-2-02 indicates that the duct chases at the 2000′ and 2016′ elevations in the Control Building for Fire Areas C-9 and C-14 are enclosed on the south and east sides by 3-hour fire barriers. Drawings A-1325, M-1G051 (USAR Fig. 1.2-24-00) and M-1H3411 indicate that there are no fire barrier enclosures at these locations. A walkdown performed with the Fire Protection (FP) Engineer indicated that the ducts have no enclosure around them. Figure 9.5.1-2, Sheet 02 incorrectly depicts fire barriers at these locations.

There are 3-hour-rated fire dampers installed in all four ducts at the floor penetrations at the 2016′ elevation to isolate Room 3416 (Fire Area C-14) from Room 3301 (Fire Area C-9). There are also fire dampers installed in the two ducts which penetrate the floor at the 2000′ elevation to isolate Room 3301 (Fire Area C-9) from Room 3220 (Fire Area C-5). Since the adjacent fire areas are isolated from each other by 3-hour-rated barriers in the form of fire dampers, fire-rated enclosures at the 2000′ and 2016′ elevations would provide unnecessary isolation of the ductwork. The 3-hour fire barriers are not needed.

Figure 9.5.1-2, Sheet 04 indicates a cable chase with 3-hour rated walls and an access door at the 2047′ elevation, east wall of the control room at column line C3. Drawings A-1326, M-1G052-4 (USAR Fig. 1.2-25) and M-1H3611 indicate that there is no fire barrier enclosure at this location. A walkdown performed 8/19/99 with FP Engineer indicates that there is no enclosed cable chase at this elevation.

The corresponding cable chase at the 2032′ elevation is designated as Fire Area C-26 and is contained in the lower cable spreading room. Section 9.5B.7.C.26 indicates that the circuits contained in the cable chase are designated as Separation Group 3. Section 9.5B.7.C.27.7.1 indicates that Separation Group 3 circuits from the lower cable spreading room feed directly into the control panels and cabinets. Since none of the Separation Group 3 circuits extend into the 2073′-6" elevation, a cable chase at the 2047'-6" elevation would serve no design function.

The proposed changes to the USAR are for accuracy and consistency with design documents and the Fire Hazards Analyses as described in USAR Section 9.5B. The correction of USAR Figures to eliminate nonexistent chases does not affect any other system, structure, or component (SSC) nor does it change the performance of activities that are important to the safe and reliable operation of WCGS. The existing plant configuration is consistent with the design basis for the fire protection system and does not affect either the existing fire protection system, associated procedures, or the Fire Hazards Analyses as described in USAR Section 9.5B. No other sections of the USAR are impacted by these changes.

As noted, the proposed changes will provide an accurate representation of the Control building fire barriers to be consistent with the Fire Hazards Analyses contained in Section 9.5B. Since, as stated in USAR Section 9.5B.2.b: "A design basis accident occurring simultaneously with a fire hazard is not assumed.", no design basis accident would be affected. Since the changes maintain the fire rating for the affected areas, no credible malfunctions of equipment important to safety are introduced. Fire barriers are not relevant to any acceptance limits contained in the technical specifications or licensing basis documents. Therefore, no acceptance limits are identified that could be affected.

In summary, the changes are being made to ensure consistency between the plant and the USAR and the Fire hazards Analysis.

Safety Evaluation: 59 1999-0138 **Revision:** 0

Changes to Plant Public Address System

Description:

Configuration Change Package (CCP) 07718 revises Updated Safety Analysis Report (USAR) Figure 9.5.2-2, "PA System Riser Diagram," which is inaccurate in representing the plant Public Address (PA) system equipment. A field walk down of the PA system equipment, located outside the containment, was performed. This walk down indicated that USAR Section 9.5.2, Figure 9.5.2-2, does not accurately reflect the type of equipment installed in the plant. The proposed change will revise Figure 9.5.2-2 to accurately reflect the plant as-built equipment. Also, some of the equipment (Amplifier and/or Housing for the PA hand-sets) will be replaced to make the plant configuration in accordance with the design standards for the PA system.

Safety Summary:

The proposed activities of revising the drawing, installing a different type of amplifier or housing for the existing hand-sets, and adding a provision for installing a PA system in the Radwaste Storage Facility are all associated with non-safety SSCs. The PA system has no safety related function. The PA system is a support system and provides audible and visual evacuation alarm under an accident condition. This function is not altered in any form by the proposed change. Therefore, the proposed activity will have no impact on any design basis accidents discussed or referenced in USAR Sections 2, 3, 6, 9, or 15.

The proposed activities are all non-safety related. The modification of the existing equipment will be done to make the plant configuration in accordance with the design standards for the PA system. The function and operation of the PA system will not be affected by the proposed changes. The proposed extension of the PA system, as shown in the USAR Figure will be done in the Radwaste Storage Facility that is non-safety, out of engineering design scope, and will require routing of new cables from the Radwaste Building to the new Radwaste Storage Building. The new handsets, stations and loudspeaker assemblies will require a new electrical power circuit to be supplied from two separate non-vital instrument busses. All communication systems circuits are enclosed in conduit or site designated raceway to provide protection for cables. The PA system is completely independent of other communication systems in the plant. No credible accidents will be created by the proposed change.

The proposed activity will not affect directly or indirectly any safety related SSCs. The PA system contains no equipment important to safety, nor is the system connected, directly or indirectly, to any equipment or plant SSC important to safety. No new equipment is being installed inside any safety related structures of the plant under this CCP. No credible malfunction of equipment important to safety will be directly or indirectly affected by the proposed change.

As stated in USAR Section 9.5.2.1.1, there is no safety design basis for the PA system. The function of the PA system will not be affected by the proposed change. There are no acceptance limits or bases contained in the Technical Specifications associated with the PA system, nor there is any basis that could be affected by the proposed activities.

Safety Evaluation: 59 1999-0139 **Revision:** 0

Modification of Condensate System Piping to Mitigate Pipe Wall Thinning Due to Flow Accelerated Corrosion

Description:

Configuration Change Package (CCP) 09070 is issued to provide guidelines for the modification/replacement of the Condensate System line AD-149-HBD-16 with a low alloy steel (2 1/4 Cr - 1 Moly, piping class HAD) to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report (Figure 10.4-6, Sheet 3, "Feedwater Heater Extraction Drains & Vents," and Figure 10.4-2, Sheet 3, "Condensate System,") due to change the Piping Class from HBD to HAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0142 **Revision:** 0

Main Steam Line Modification to Mitigate Flow Accelerated Corrosion

Description:

Configuration Change Package (CCP0 09067 is issued to provide guidelines for the modification/replacement of Main Steam lines AB-257-HBD-4, AB-255-HBD-2, AB-256-HBD-2 and AB-257-HBD-2, with a low alloy steel (2 1/4 Cr - 1 Moly, piping class HAD) to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report (Figure 10.3-1, Sheet 3, "Main Steam System," and Figure 10.4-2, Sheet 6, "Condensate System,") due to changing the Piping Class from HBD to HAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0143 **Revision:** 0

Modification to Mitigate Flow Accelerated Corrosion in the Feedwater Heater Extraction Piping

Description:

Configuration Change Package (CCP) 09069 is issued to provide guidelines for the modification/replacement of Feedwater Heater Extraction piping lines AF-455-GBD-20 and AF-206-GBD-12 with a low alloy steel (2 1/4 Cr - 1 Moly, piping class GAD)to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report (Figure 10.4-6, Sheet 3, "Feedwater Heater Extraction Drains & Vents,") due to changing the Piping Class from GBD to GAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0145 **Revision:** 0

Modification to Mitigate Flow Accelerated Corrosion in the Feedwater Heater Extraction Piping

Description:

Configuration Change Package (CCP) 09072 is issued to provide guidelines for the modification/replacement of lines AF-269/-277/-296/-322/-324-HBD-3; AD-101/-103/-106/-110/-112-HBD-3; AF-266/-274/-322/-330-HBD-1.5; AF-293/-321/-329-HBD-1; and AF-267/-275-HBD-4 with a low alloy steel (2 1/4 Cr - 1 Moly, piping class HAD)to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report (Figure 10.4-6, Sheet 5, "Feedwater Heater Extraction Drains & Vents," and Figure 10.4-2, Sheet 3, "Condensate System,") due to change the Piping Class from HBD to HAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0149 **Revision:** 0

Spent Fuel Pool Liner Repair

Description:

The Spent Fuel Pool (SFP) floor liner plate was damaged during work activities on the SFP, re-rack project. Non-Conformance Report (NCR) 99-212970-000 documents the damage to the floor liner plate. An underwater cutting torch accidentally touched the floor liner plate and caused a through hole and some minor linear indications with the hole. The SFP liner plate is non-safety related and serves no safety function.

The damaged floor liner plate area will be ground smooth (if needed) , then a patch plate will be installed over the damaged area. Nominal patch plate size is ¼" thick x 4.125 inches square.

The area of the SFP associated with the leak has been cleared out. The old storage racks have been removed (fuel was shuffled to other areas of the pool) from this area. Repair work will not be performed near any spent fuel bundles.

The repair acceptance criteria are:

1. The patch plate will stop leakage caused by the damage to the floor liner plate.
2. The patch plate will not interfere with the new storage rack legs or flow holes.
3. The patch plate will not interfere with the SFP cooling inlet and outlet pipe penetrations.
4. The damaged floor liner plate area will not corrode and migrate past the patch plate area.
5. The seismic analysis of the SFP will not be affected by the addition of this patch plate.

Safety Summary:

As a result of this repair, USAR Figure 9.3-7, "Fuel Building-Area 1, Stainless Steel Liner Plate Plan,(Sheet 2) Spent Fuel Pool," will be revised to show the location of this new patch plate. USAR Section 9.1.2.2, "Facilities Description," identifies the material used for the SFP liner plates (and also defined in specification C-171), which is grade 304L, hot rolled, annealed, pickled and cold rolled. The repair patch plate procured per Purchase Order 0705310 R/0 is not cold rolled. This doesnot meet USAR Section 9.1.2.2 or specification C-171. The purpose for performing the additional process of cold rolling the steel plate is to provide a smoother plate surface. There is no structural concern for not having the plate cold rolled. The cold rolling requirement is a commercial issue which is used to reduce the cost of decontaminating the liner plates by having a smoother surface

USAR Section 9.1.2.2 identifies the weld repair criteria for fuel pool liner plates. All criteria is met with the exception of the pressure test of 30 PSIG with a leak detection solution and ASME Code Section VIII welding requirements. Under water welding ASME Code Case N-516-1 will be used with Weld Procedure Specification (C-172-0004) and Procedure Qualification Records (C-172-00003). This Code Case for ASME underwater welding is more stringent and conservative for work being performed than the original installation requirements. Weld acceptance criteria is defined per Vendor Procedure C-172-00008. This acceptance criteria is acceptable for ASME welding being performed. Therefore weld integrity is acceptable and a pressure test is not needed or required to be performed. The weld repair area will be vacuum tested which will further ensure no leakage exist for the repaired area.
In addition, Procedure STN EC-001, "Spent Fuel Pool/Refuel Pool Leak Detection," will be performed to determine whether there is any leakage past the fuel pool liner plates.

The SFP liner plate is not considered in any design basis accidents. This liner plate is non-safety related. The repair patch plate will be installed in an open area, away from any spent fuel bundles/storage racks. The patch plate will not interfere with the new rack support legs, flow holes or the spent fuel cooling inlet and outlet pipe penetrations. In addition, the patch plate will be located under a new spent fuel rack and will not interfere with the support legs on the bottom of the new rack or the flow holes in the new racks.

Safety Evaluation: 59 1999-0150 **Revision:** 0

Containment Cooler Modification

Description:

The Essential Service Water (ESW) Return Isolation Valves from the Containment Coolers, EFHV0049 (Train A) and EFHV0050 (Train B) are utilized as throttle valves to maintain required Containment Cooler ESW flow. These valves are currently throttled at approximately 30 degrees open. These valves are also containment isolation valves which must close and seal to maintain Containment Integrity. The throttling of these valves has resulted in accelerated seat wear, which has resulted in Local Leak Rate Testing (LLRT) problems.

Configuration Change Package (CCP) 09194 will shift the primary flow control location for the Containment Cooler ESW return from Valves EFHV0049, (ESW A From Containment Coolers Outside Containment Isolation Valve) and EFHV0050 (ESW B From Containment Air Coolers Butterfly Valve) to valves GNV0001, GNV0002, GNV0003, and GNV0004 (Containment Cooler ESW Manual Outlet Valves). Flow balancing will determine the optimum position for all six valves to protect the sealing function of EFHV0049 and EFHV0050 to the maximum possible extent. The primary objective is to maximize the OPEN position on EFHV0049 and EFHV0050, while balancing the erosion effects of the restricted flow through all six valves. The bulk of the ESW flow control will be shifted to the non-containment isolation valves to reduce the seat wear on EFHV0049 and EFHV0050. ESW flow to the Containment Coolers will not change. Since the only ESW service load in containment is the Containment Coolers, changing the throttle location does not impact any other equipment.

Safety Summary:

The Containment Cooling System, (CCS) in conjunction with the containment heating, ventilation, and air conditioning (HVAC) systems functions during normal plant operation to maintain a suitable atmosphere for equipment located within the containment. During a design basis accident (DBA), the Containment Cooling System provides a means of cooling the containment atmosphere to reduce pressure and thus reduce the potential for containment leakage of airborne and gaseous radioactivity to the environment.

The CCS provides cooling by recirculation of the containment air across air-to-water heat exchangers. The containment coolers supply air to the lower portions of the steam generator compartments. The air is exhausted from these compartments by means of the hydrogen mixing fans, which have a high discharge velocity, directing the air-stream upward. This action in conjunction with the operation of the Control Rod Drive Mechanism (CRDM) cooling system and the cavity cooling systems, which take suction from the lower area of the containment and discharge it upwards, produces a normal containment air flow circulation path from the bottom to the top of the containment.

The ESW System removes heat from plant components which require cooling for safe shutdown of the reactor or following a DBA. The ESW consists of two redundant cooling water trains. The ESW does not directly interface with radioactive systems.

There are no design basis accidents impacted with this change. The ESW flow to the containment coolers will not change just the location were the flow is throttled changes. Since no changes occur in the ESW flow rate to the containment coolers or any other equipment serviced by ESW, there is no affect on any SSC. No credible accidents that could be created are identified.

Valves EFHV0049 and 50 currently are throttled approximately 20 degrees open. This modification's primary objective is to maximize the valves in the open position. With the valve open further the stroke close time for these valves increases. This is not a problem because USAR Figure 6.2.4-1, "Containment Penetrations," Pages 25 and 51 of 74 show these valves having no maximum stroke close time. The valves are only required to close and meet Local Leak Rate Test (LLRT) requirements. Also, these valves do not receive any of the Containment Isolation Signals; however, they receive a Safety Injection Signal (SIS) to open. This provides the means of cooling the containment atmosphere to reduce pressure and thus reduce the potential for containment leakage.

The relocation of the Containment Cooler (SGN01A, SGN01B, SGN01C, and SGN01D) return water flow throttling from the Containment isolation valves (EFHV0049, and EFHV0050) to valves GNV0001, GNV0002, GNV0003, and GNV0004 will involve performance of a flow balance. Preliminary estimates reveal that throttling the GN valves to approximately 30 degrees will allow the ESW valves to be FULL OPEN (approximately 80 degrees). Support and System Engineering shall determine if the Containment Cooler valves exhibit acceptable flow characteristics around the 30 degree setting as well as flow characteristics in the required range (at least 1000 gpm for each Containment Cooler). If the flow can be totally controlled by the Containment Cooler valves, then the associated ESW valves can be set at the FULL OPEN position (approximately 80 degrees).

If the Containment Cooler valves do not appear capable of handling the total flow, then the associated EF valve will be throttled to maintain the desired flow. This will result in a sharing of the flow restriction between the Containment Cooler and the ESW valves. The intent will be to minimize the throttling of the ESW valves so that the wear conditions associated with throttled flow will be minimized. The Containment Cooler Valves are manually operated and serve no safety function other than ESW system pressure boundary.

The ESW flow to the containment coolers will not require change to any current Technical Specifications or any of the Technical Specifications bases and acceptance limits. Since no acceptance limits were identified that could be affected, the margin of safety is not affected by this change.

Safety Evaluation: 59 1999-0152 **Revision:** 0

Change to Separation Criteria in the Updated Safety Analysis Report, Chapter 15

Description:

Configuration Change Package (CCP) 07503, Revision 1, re-numbers existing Item 20 in Updated Safety Analysis Report (USAR) Section 8.3.1.4.1.4, "Independence of Redundant Systems," to Item 21. This is an editorial change. The proposed activity also revises USAR Section 8.3.1.4.1.4 for an additional exception to physical separation requirements by adding a new item 20 for a Non-Class 1E festooned power cable for hoist HKF23 and Class 1E conduits 1U3A1C and 4U1150.

The Class 1E conduits contain circuits for dampers GEHZ101 (Condenser Air Removal Filtration Damper), GFHZ30B (Main Steam Enclosure Building Damper) and GKHZ184C (Control Room Ventilation Isolation Damper). The event of concern is that the Non-Class 1E festooned cable would fault and damage the Class 1E cables such that they would not perform their intended functions. The Non-Class 1E festooned cable provides 480 VAC power to hoist HKF23 and has a local disconnect switch which is opened whenever the hoist is not in service. Consequently the Non-Class 1E festooned cable is only energized intermittently and in the presence of the hoist operator.

Safety Summary:

Regulatory Guide 1.75, "Physical Independence of Electric Systems," (Regulatory position c.4) and IEEE 384-1974 (Section 4.5(3)) provides the option to demonstrate by tests that the absence of physical separation could not significantly reduce the availability of Class 1E circuits. Wyle Laboratories Test Reports 46960-1 and 46960-3 and Philadelphia Electric Company's Test Report #48503 evaluated the impact of physical separation less than those specified in IEEE 384-1974 and Regulatory Guide 1.75, Rev. 1 and determined the acceptability of reduced separation for cable sizes up to #2 AWG. The Non-Class 1E festooned power cable is #6 AWG which is much smaller than the tested cables. The Non-Class 1E festooned cable is within the bounding conditions of the above test reports. Therefore, it is technically justified to accept the separation which is less than the standard distance in accordance with the USAR.

The proposed change will affect USAR Section 8.3.1.4.1.4 by including an additional exception from the physical separation requirements for Non-Class 1E and Class 1E cables and/or raceways. Class 1E conduits 1U3A1C and 4U1150 and the Non-Class 1E festooned power cable for hoist HKF23 do not meet the separation requirements of IEEE 384-1974 and Regulatory Guide 1.75, Revision 1, due to practical limitations. This change will not affect any procedures, structures, systems, or components that are outlined, summarized or described in the USAR. There are no tests or experiments identified by the proposed change. The USAR requirements to maintain the minimum separation distance will be violated without the proposed USAR change.

The proposed USAR change will not impact any procedures. No plant structures systems or components are directly affected by the proposed change. The plant configuration remains unchanged and no field work is generated by the proposed change. The proposed change for the minimum separation criteria is within the design basis, codes and regulatory requirements. The validity of the acceptance criteria for less than minimum separation distance is supported by Wyle Laboratories test reports and Philadelphia Electric Company's test report. Based on these reports, a fault on the Non-Class 1E festooned power cable would not degrade the ability of the Class 1E circuits to function. Therefore, the proposed change does not have the potential to impact any design basis accidents discussed or referenced in USAR Chapters 2, 3, 5, 6, 9 or 15.

The proposed change will not adversely affect any safety related equipment and will not introduce any new failure mechanism for safety related SSCs. Operation of plant equipment is not being changed and no plant operating procedures are affected. No equipment of a new or different type is installed or relocated. No field work is required by the proposed change. The proposed change will not affect the safety related functions of the dampers. There are no possible failure modes created by this change which could cause the dampers to fail and create the possibility of a new accident. Therefore, no credible accidents including anticipated operational transients or design basis accidents will be created by the proposed change.

The Technical Specification bases were then reviewed to determine any possible impacts of the proposed activity to safety-related dampers. It is concluded that no acceptance limits, including limits for temperature, humidity and radiation exposure to Control Room personnel, will be affected by this additional exception to the

Safety Evaluation: 59 1999-0153 **Revision:** 0

Evaluation of Procedure SYS HF-203

Description:

This Unreviewed Safety Question Determination (USQD) evaluates Revision 17 to Procedure SYS HF-203, "Radwaste Secondary Liquid Waste Monitor Tank Operations." SYS HF-203 is being revised as follows:

1. Add step 2.1.7 to identify the new section 6.8. "SLWMT Processing through the ZERO System".

2. Add new steps 4.3 through 4.15 (plus attached notes and cautions) to be consistent with the other processing procedures.

3. Add new steps 5.3 through 5.9 to be consistent with other processing procedures.

4. Add new section 6.8 to provide instructions on how to process the SLWMT through the ZERO System.

5. Add Attachments C, D, and E to provide effluent alignment, Effluent Realignment, and Effluent Restoration to be consistent with the other processing procedures.

Safety Summary:

Change number 4 listed above changes the flow path for the Secondary Liquid Waste Monitor Tank (SLWMT) effluent as described in Updated Safety Analysis Report (USAR) Figure 10.4-12-02, "Secondary Liquid Waste System," and Section 10.4.10.2.3, "System Description." USAR Section 10.4.10.2.3 states; "If, for any reason, the SLW monitor tank water does not meet the necessary chemical requirements for discharge or recycle, the water may be processed through any combination of the SLW evaporator feed, the SLW evaporator, the SLW charcoal adsorber, the SLW demineralizer, or can be drained to the DRW sumps and processed through the liquid radwaste demineralizer skid." In the proposed change Secondary Liquid Waste (SLW) monitor tank effluent will bypass the Dirty Radwaste (DRW) sumps and be routed through the ZERO Skid and the SLW demineralizers. The ZERO skid is an enhanced filtering system that was previously evaluated under Temporary Modification (TMO) 98-018-HB and Unreviewed Safety Question Determination (USQD) 59-98-0098. The expected effect of the proposed change is an enhancement of SLW processing and reduction of radiological effluent to the environment.

Any spill or leakage from the hoses that connect the SLW monitor tank effluent to the ZERO Filtration System will be contained within the existing design of the radwaste drainage system. Any possible spills inside the Radwaste Building are bound by the analysis in USAR Section 15.7.2. If any liquid radwaste does escape the Radwaste Building, this scenario is bounded by the analysis of the rupture of the worst case liquid radwaste storage tank, see USAR Sections 15.7.3 & 2.4.13.

USAR 3.4.1.1.2 Internal Flooding Protection: This section states "All safety related equipment rooms located below grade are protected from back-flooding by the remote location of waste-processing components in the radwaste building. The floor and equipment drains in power block seismic Category I buildings drain to sumps in the lowest level of the building in which they are located. These sumps are pumped to the floor drain tank or the waste hold-up tank located in the radwaste building. Should these tanks rupture or leak, flow into safety related areas will not occur since these tanks are located below radwaste building flood level." Since this procedure change is associated with the Secondary Liquid Radwaste System, USAR Chapter 3 analysis of Internal Flooding Protection remains valid.

The gases in the radwaste tanks are vented to the Radwaste Building HVAC, where the gases are extracted or vented prior to reaching the ZERO Filtration System. Therefore, a potential gaseous release is bound by previously evaluated tank ruptures analyzed in the USAR Section 15.7.1.

The proposed procedure change will not create any new credible accidents. The only credible accidents associated with this procedure change is the potential for a spill or gaseous release. Both a spill and gaseous releases are bounded by existing analyses.

The proposed procedure change will not cause any systems, structures or components important to safety to malfunction, or to malfunction in a way not already analyzed. All of the proposed equipment is either Special Scope D-Augmented or non-safety related and will be located in the Radwaste Building that physically separates it from any interaction with safety related equipment.

There are no acceptance limits contained in the bases for the technical specifications or other licensing basis documents that could be negatively affected by this TMO. However, the proposed procedure change can have a positive affect on the plants radiological effluent releases and the associated release permit commitments.

Safety Evaluation: 59 1999-0154 **Revision:** 0

Clarification to the Updated Safety Analysis Report, Chapter 15

Description:

This Updated Safety Analysis Report (USAR) change provides clarifying changes to the USAR in response to Performance Improvement Request (PIR)98-3113. These changes are proposed to eliminate various discrepancies identified in Chapter 15 of the USAR.

1) USAR Section 15.1.2.1, "Identification of Causes and Accident Description," is misleading. This clarifying change will explicitly indicate that the steam generator high-high level trip initiates feedwater isolation and trips the turbine and main feedwater pumps. The current wording implies that the steam generator high-high trip isolates the feedwater isolation valves. The feedwater isolation valves close following the feedwater pump trip.

2) USAR Section 15.1.2.2, "Analysis of Effects and Consequences," Subsection, Method of Analysis, is misleading because the reactor trip on turbine trip no longer being credited in the licensing basis analysis. This clarifying change will indicate that subsequent to feedwater isolation the reactor continues to operate until the low-low- steam generator level setpoint is reached.

3) USAR Section 15.1.2.2, Subsections, Results, and Method of Analysis, Item e, is changed to clarify that the steam generator high-high level trip initiates feedwater isolation and a feedwater pump trip and once the main feedwater is isolated, the reactor continues to operate until the low-low steam generator level trip setpoint is reached. This is due to the turbine trip and reactor trip on turbine trip not being credited in the analysis.

4) This change revises the USAR Table 15.1-1, "Time Sequence of Events for Incidents that Result in an Increase in Heat Removal by the Secondary System," event description for the Feedwater System Malfunction transient to state "rod motion" instead of the nonsense phrase "Rod Motion Close Automatically", at the current transient time of 59.9 seconds. The "close automatically" is a pre-revision 10 vestigial information not correctly implemented as part of USARCR 96-046. Therefore, this change is considered an editorial revision.

5) This change revises USAR Table 15.1-1 and USAR Figure 15.1-2A "Core Average Temperature Transients for Feedwater Control Valve Malfunction," to incorporate the current licensing basis departure from nucleate boiling ratio (DNBR) information. The DNBR information presented in the USAR was not updated in Revision 10 and is therefore not consistent with the feedwater malfunction accident transient parameters presented. Therefore,

this change is considered an editorial revision.

Each of these proposed changes will correct existing discrepancies in the USAR by revising the analysis or plant description and results to be consistent with existing safety analyses. No change is proposed to the existing licensing basis analyses or to any plant structure, system, or component, or plant operating procedures.

USARCR 99-084, and the associated USQD 99-0107, included changes/clarifications affecting Table 15.0-6 in which it is indicated that the SG Hi-Hi level trip does not provide for a direct reactor trip, rather it provides an ESF function. Note: The SG HI-Hi level trip, described as a ESF function, is indicated as producing a feedwater isolation and a turbine trip. Both USARCRs are correct.

This USAR change regarding the HI-Hi trip on Steam Generator level are primarily to achieve consistency and clarity between USAR statements presented in the USAR Chapter 15 text and the current licensing basis calculation of the Feedwater Malfunction (AN-95-006 Revision 2) information, and USAR Table 15.1-1.

Safety Summary:

There are no plant procedures, activities, administrative controls, or sequences of plant operations, etc., that are impacted by the proposed changes. The proposed changes will correct existing discrepancies in the USAR, which were identified by the Fidelity Review Team.

The following USAR Sections were reviewed for potential impact:

USAR Section 15.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

USAR Table 15.1-1 Time Sequence of Events for Incidents that Result in an Increase in Heat Removal By the Secondary System

USAR Figure 15.1-2A Core Average Temperature Transients for Feedwater Control Valve Malfunction

Based on the review, no potential impact due to the proposed activity was determined to exist.

No credible accidents are created by the proposed changes. The proposed USAR changes are intended to clarify statements and remove discrepancies between the USAR and existing safety analyses. No changes to the plant or to the accident analyses are proposed.

No malfunction of equipment important to safety is impacted by the proposed USAR changes. The proposed USAR changes are intended to clarify statements and remove discrepancies within the USAR and between the USAR and existing safety analyses. No changes to the plant or to the accident analyses are proposed.

No acceptance limits are impacted by the proposed changes. Since the accident analyses are not being revised, there is no impact on the analysis results, acceptance limits, or margin to the existing acceptance limits.

Safety Evaluation: 59 1999-0158 **Revision:** 0

Temporary Modification to Chemical Injection in the Auxiliary Steam System

Description:

Injection of chemicals into the Auxiliary Steam system has been problematic for the past several years. Difficulties with the chemical injection pump operation and the feed water line relief valve (FBV0161) lifting have been noted. As the normal system pressure is close to the relief valve nominal set point the pulsation of the fluid in the chemical feed line due to the positive displacement chemical injection pumps raises the pressure at the relief valve to the set point and opens the relief valve.

Temporary Modification 99-014-FB will reroute the chemical injection point to the suction of the Auxiliary Steam Feedwater Pumps to provide a low pressure injection site, promote good chemical mixing, and remove the pulsating pressure from the inlet of FBV0161 (Auxiliary Steam Chemical Addition Header Relief Valve). Chemical injection can then occur without lifting the relief valve.

Safety Summary:

This temporary modification does not alter the design function of the Auxiliary Steam Chemical Addition (FE) System or the Auxiliary Steam (FB) System.

Updated Safety Analysis Report (USAR) Figure 9.5.9-1-2, "Auxiliary Steam System," and USAR Figure 9.5.9-1-04, "Auxiliary Steam Chemical Addition System," show the "as installed" configuration of the chemical injection point and pump connections. Since a temporary hydrazine pump will be installed and the chemical injection point will be changed, installation of this temporary modification will render these USAR figures temporarily inaccurate.

This temporary modification does not make any changes to the hydrazine tank (TFE02). The amount of hydrazine involved is small - maximum of 30 gal of 18 percent hydrazine. The temporary pump will to be installed on the existing pump skid which includes a basin which drains to a turbine building sump. Calculation, AQ-M-001 R/0, postulates a spill of 2000 gal of 35 percent hydrazine in the north end of the turbine building and concludes that there is no affect on control room habitability due to the spill. Hydrazine has low volatility and is not considered to be flamable at concentrations less than 50 percent. Therefore, the worst possible mishap involving the Fe system hydraxine (i.e. complete release of a full tank of chemical - 30 gallons of 18 percent solution) would not degrade the safety of the plant.

No design basis accidents listed in USAR chapters 2, 3, 5, 6, 9, and 15 were identified as being affected. No credible accidents created by this temporary modification were identified. No credible malfunctions of equipment important to safety affected by this temporary modification were identified. As the auxiliary steam system and the auxiliary steam chemical addition system are non-safety related, have no safety design basis, and are not referenced or controlled by the technical specifications, no acceptance limits were identified as being affected by this temporary modification.

Safety Evaluation: 59 1999-0159 **Revision:** 0

Piiping Replacement Because of Flow Accelerated Corrision on the Main Turbine System

Description:

Configuration Change Package (CCP0 09234 is issued to provide guidelines for the modification/replacement of lines AC-090-GBD-2, AC-092-GBD-2,AC-093-GBD-2, AC-094-GBD-2, AC-095-GBD-2, AC-281-GBD-2 and AC-282-GBD-2 with a low alloy steel (2 1/4 Cr - 1 Moly, piping class GAD)to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report Figure 10.2-1-02, "Main Turbine," due to change the Piping Class from GBD to GAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0160 **Revision:** 0

Piiping Replacement Because of Flow Accelerated Corrision on the Main Turbine System

Description:

Configuration Change Package (CCP) 09236 is issued to provide guidelines for the modification/replacement of lines AC-096-GBD-2, AC-097-GBD-2,AC-098-GBD-2, AC-099-GBD-2, AC-100-GBD-2, AC-101-GBD-2, AC-285-GBD-2 and AC-286-GBD-2 with a low alloy steel (2 1/4 Cr - 1 Moly, piping class GAD) to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC).

Safety Summary:

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. Therefore, there are no malfunctions of equipment important to safety identified.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed change will affect the Essential drawings in the Updated Safety Analysis Report Figure 10.2-1-02, "Main Turbine," due to change the Piping Class from GBD to GAD on the replacement line. However, the proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being affected by this change.

Since the proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings), no new accidents could be created.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (yield or tensile strength, and/or code allowable stresses), or the geometric configuration; therefore, no new malfunction of equipment important to safety is introduced.

Since the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are affected.

Safety Evaluation: 59 1999-0166 **Revision:** 0

Operations Organization Change

Description:

This Unreviewed Safety Question Determination (USQD) evaluates a change in the Operations organization. The proposed change promotes an individual to Shift Supervisor (Shift Manager) based on closely observed exemplary performance of his licensed duties, his demonstrated leadership ability and the successful completion of the Shift Supervisor qualification program. Due to his promotion and change in duties, the individual's qualifications will be added to the Updated Safety Analysis Report (USAR). In addition, updates need to be made to other Shift Supervisor (Shift Manager) - qualified individuals to indicate that their qualifications are current even if they are not assigned to one of the Operating crews.

Safety Summary:

The above changes do not alter operating practices of the Wolf Creek Generating Station (WCGS) Operating Organization. The changes ensure that the USAR correctly reflects the Organization Personnel utilized at WCGS.

This USQD is being performed in support of the organizational change which requires a USAR change to ensure USAR accuracy. USAR Section 13.1.3.2 is revised to update licensed personnel qualified for the position of Shift Supervisor. No other USAR descriptions or conclusions will change or be untrue due to this change. These changes do not result in any test or experiments not described in the USAR which could adversely affect the adequacy of SSCs to prevent accidents or mitigate the consequences of an accident.

There are no design basis accidents identified as being affected because of these organizational changes. These changes do not change any administrative control which would reduce the level of qualification of Wolf Creek Nuclear Operating Corporation (WCNOC) personnel, nor do they affect performance of activities or any SSC.

Since there are no proposed changes which would reduce the level of qualification of WCNOC personnel, and there is no affect on any SSC, no new credible accidents could be created.

Since the proposed changes do not affect controls for activity performance that reduce the level of personnel qualification and there is no affect on any SSC, no new credible malfunctions of equipment important to safety are identified.

Since the proposed changes to the USAR do not affect any administrative controls on activities that lower the level of personnel qualification, or affect any SSC, no acceptance limits are affected.

Safety Evaluation: 59 1999-0167 **Revision:** 0

Emergency Plan Revision

Description:

The Radiological Emergency Response Plan (RERP) is being changed to ensure the plan content conforms to regulatory requirements in 10 CFR 50, Appendix E and 10 CFR 47, Part b. Some of the changes have been made because there was nothing in the RERP to cover parts of the content requirements. Other changes were made to provide more detail for meeting the content requirements. Several editorial changes were made to update names, position titles, and locations, and some minor changes have also been made to cover comments made by the State and County during their annual review of the RERP. There are no associated changes to Emergency Preparedness procedures or practices, but only a more thorough description in the RERP.

One addition was made to add the Security Coordinator position to the Technical Support Center to provide a communication link between the TSC and Security. The intent of this position is to improve the communication between the TSC and the Security group. This does not decrease the effectiveness of the RERP.

Several changes were made to Attachment E, EPA/KANSAS PROTECTIVE ACTION GUIDES. This attachment is included in the RERP to inform Wolf Creek Generating Station (WCGS) of what actions the State will take and when the State will take them.

Safety Summary:

The only part of the USAR that is impacted by these changes is the RERP, which is referenced in USAR Section 13.3. The changes described above do not impact E-plan procedures or processes, nor do they decrease the effectiveness of the RERP. Therefore there are no USAR design basis accidents which could be impacted by these changes.

Since no E-plan procedures or processes are changed and there is no reduction in the effectiveness of the RERP, no accidents are identified that could be created by this change.

Since no E-plan procedures or processes are changed and there is no reduction in the effectiveness of the RERP, no malfunctions of equipment important to safety are identified that could be created by this change.

These changes do not impact the plant, the way the plant responds to accidents, or the operator response to accidents, so there is no impact on acceptance limits.

Safety Evaluation: 59 1999-0171 **Revision:** 0

Updated Safety Analysis Report Change to Reflect Organization Change

Description:

Due to the retirement of the Vice President and Chief Administrative Officer, a restructuring of the organization has been proposed. The proposed organization will eliminate the position of Vice President and Chief Operating Officer. The title Vice President and Chief Administrative Officer will be re-titled Vice President Operation Support. Reporting functions will be aligned with the appropriate vice president. Since all functions will continue to be performed with the appropriate oversight and all personnel are qualified, there will be no effects to the operation of Wolf Creek Generating Station.

Safety Summary:

The Updated Safety Analysis Report (USAR) will require revision to reflect the organization and reporting change (Chapters 13 and 12) .

This change is administrative in nature. It affects the organization structure alone, no functions will be deleted. Therefore, no accidents have been affected. As described above, this change is administrative. No new accidents could be created.

As discussed above, this change is administrative in nature. Functions have not been deleted and qualifications are met. Therefore, equipment will not be affected.

There are no acceptance limits associated to the organization structure that would be invalidated by this change. All functions continue to be performed and qualifications met.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Michael J. Angus, Manager Licensing and Corrective Action, at Wolf Creek Generating Station, (316) 364-4077.

COMMITMENT	Due Date/Event
None	N/A