WOLF CREEK
NUCLEAR OPERATING CORPORATION

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MAR 9 2001

CO 01-0015

U. S. Nuclear Regulatory Commission
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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. Attachment IV identifies acronyms used in Attachment II.

This report covers the period from January 1, 2000, to December 31, 2000, and contains a summary of 50.59 evaluations performed during this period that were approved by the WCGS onsite review committee. If you have any questions concerning this report, please contact me at (620) 364-4034, or Mr. Karl A. (Tony) Harris at (620) 364-4038.

Very truly yours,

Clay C. Warren

Clay C. Warren

CCW/rlr

Attachments

cc: J. N. Donohew (NRC), w/a
W. D. Johnson (NRC), w/a
E. W. Merschoff (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.: 16

Reporting Period: January 1, 2000 through December 31, 2000

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 50.59 evaluations that were reviewed and found to be acceptable by the Plant Safety Review Committee (PSRC) for the period beginning January 1, 2000 and ending December 31, 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 1998-0098 **Revision:** 1

Zero Filtration System

Activity Description:

Revision 1 has been performed to further clarify the Zero Filtration System and its supporting equipment and components.

Activity Description:

It has been identified that certain insoluble materials in the liquid radwaste system can not be removed by the Demineralizer Skid and must be physically removed using filtration. The current filtration provided in the Liquid Radwaste System has the ability to remove particulate material of 0.45 microns and larger.

The Diversified Technology's ZERO Filtration System is being installed under Temporary Modification TMO 98-018-HB to determine the effectiveness and benefit to Wolf Creek Generating Station (WCGS). The ZERO Filtration System will be connected in series, upstream of the existing vendor supplied Demineralizer Skid. It will be physically located in the Radwaste Building.

Hoses will be routed through existing pipe penetrations to connect the ZERO Filtration System to the existing components of the Demineralizer Skid for further processing. The processed water can then be either released to the WCGS Lake or reused internally at WCGS.

The filter cartridge in filter housing FHB06 will be removed or modified for the duration of this TMO. The ZERO Filtration System will provide better filtration than any filter cartridge placed in FHB06. In addition, the filter removal or modification will greatly reduce the cost of labor, personnel radiation dosage and radwaste involved with the change out of the filters.

Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light-Water Nuclear Power Plants" is being followed for the equipment and installation of the ZERO Filtration System. Engineering has evaluated the unit's acceptability for use at WCNOC. The evaluation includes the floor loading, hypothesized shielding requirements and electrical power requirements. The installation of the ZERO Filtration System is acceptable.

50.59 Evaluation:

Any spill or leakage from hoses, temporary pumps, supporting equipment, or any other components of 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, documented in the Updated Safety Analysis Report (USAR), Sections 15.7.3 and 2.4.13.

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 installation of the ZERO Filtration System will not create any new credible accidents. The only credible accidents associated with this TMO is the potential for a spill or gaseous release. Both a spill and gaseous releases are bounded by existing analyses.

The installation of the ZERO Filtration System 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.

The equipment and hoses meet the same existing Regulatory Guide 1.143 requirements as the existing installed piping. Therefore, the hoses and fittings will not increase the probability of leakage or failure and the probability of occurrence of an accident is not affected by these changes. The previous USAR analyses bound any possible accidents that may occur with this equipment. Therefore, the radiological consequences of an accident are not affected by these changes.

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 Temporary Modification. However, the ZERO Filtration System can have a positive effect on the plant's radiological effluent releases and the associated release permit commitments.

Safety Evaluation: 59 1998-0112 **Revision:** 0

RHR and CCW Heat Exchanger and Containment Cooler Peak Duties

Activity Description:

This change to the Updated Safety Analysis Report (USAR) revises Tables 9.2-3, Essential Service Water (ESW) System Flow Requirements, and Table 9.2-11, Component Cooling Water (CCW) System Flow Requirements Post-LOCA, to clarify the peak post-LOCA duties for the Residual Heat Removal (RHR) heat exchangers, CCW heat exchangers and the containment air coolers (CACs). The peak duties currently presented in the tables are for a single train post-LOCA cooldown while the layout of the tables appears to list requirements for a limiting two train post-LOCA cooldown. The change adds notes to the tables to identify these as single train values and gives the corresponding two train cooldown values. These notes also explain that single train values can not be added together to estimate two train duties. In addition, a new note in Table 9.2-3 to indicate the ESW duties can not be added together since the peak duties for the CCW heat exchangers and CACs are not coincident. However, this change makes no physical changes to system flow rates.

This change also revises USAR Table 9.2-9, Component Cooling Water System Requirements Normal Operation, to change the CCW flow to the RHR pump seal coolers from 6 gpm to 4 gpm. In addition, Note 4 in Table 9.2-9 is revised to clarify that the flows and related duties for the Reactor Coolant Pump coolers and other non-essential components are nominal values. The revised Note 4 is consistent with the note added to Table 9.2-10 by a change evaluated previously.

This change revises USAR Table 9.2-20, "Heat Loads From Station Auxiliaries Normal Shutdown", to show an actual duty for the Reactor Coolant Pumps coolers of 2.1 x 1E 6 Btu/hr. The revised duty is consistent with Calculation EG-06-W, revision W-04, "Component Cooling Water System" and Table 9.2-10, "Component Cooling Water System Requirements Shutdown (@ 4 Hours) Operations."

This change clarifies the presentation of peak duty information for post-LOCA operation in USAR Table 9.2-3 and 9.2-11 to differentiate between a single train and a two train cooldown and clarify that peak duties are not additive because the peaks may occur at different times. Table 9.2-9 is updated to present current data for the RHR pump seal coolers and to clarify that flows and related duties for non-essential components are nominal values. Table 9.2-20 is updated to reflect the correct actual duty for the Reactor Coolant Pump coolers. The change will ensure the presentation of consistent and correct information in these USAR Tables.

50.59 Evaluation:

USAR Section 6.2.1.5 discusses a limiting analysis for minimum post-LOCA containment pressure (maximum Safety Injection with low lake temperatures). USAR Section 9.2.5 discusses Ultimate Heat Sink (UHS) sizing analyses. For the UHS, the limiting case is a

LOCA with maximum Safety Injection (using the full capacity of the CACs). This case corresponds to the conditions summarized in USAR Tables 9.2-3 and 9.2-11 and is the basis for USAR Figures 9.2-6A, 9.2-6B and 9.2-6C. USAR Table 9.2-9 deals with normal operation of the CCW system. Table 9.2-20 deals with heat loads from station auxiliaries during normal shutdown. This change clarifies the related flows and duties presented in the USAR Tables for the ESW and CCW systems. The design basis for the events described in Sections 6.2.1.5 and 9.2.5 is not changed. Therefore, there is no impact on the design basis accidents.

This change clarifies the presentation of peak duty information for post-LOCA operation in USAR Table 9.2-3 and 9.2-11 to differentiate between a single train and a two train cooldown and clarify that peak duties are not additive because the peaks may occur at different times. Table 9.2-9 is updated to present current data for the RHR pump seal coolers and to clarify that flows and related duties for non-essential components are nominal values. Table 9.2-20 is updated to reflect the correct Actual Duty for the Reactor Coolant Pump coolers. The change will ensure the presentation of consistent and correct information in these USAR Tables. No new type of credible accident is created as a result of this change.

No credible Malfunctions of Equipment Important To Safety are directly or indirectly affected by this change. Since the operation of the ESW and CCW systems is not changed as a result of this change, no acceptance limits are identified that could be affected. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1998-0113 **Revision:** 0

Revision to USAR to Reflect Radwaste Building Changes

Activity Description

USAR Section 11.4, "Solid Waste Management System", is being revised to reflect a number of changes. The USAR change clarifies that the Waste Bale Drumming Area is utilized as the Interim On-Site Storage (IOS) Facility as described in USAR Appendix 11.4A. In addition, it clarifies that waste other than solid or dewatered waste is stored in the IOS Facility. The mixed waste handling description is updated to reflect the current practice of shipping mixed waste to processors. Reference to a specific drum size for super compacted dry active waste (DAW) storage is deleted as a variety of drum/container sizes are used. The drums currently in use satisfy all applicable transportation and disposal requirements. This change is necessary because the predicted suspension of national disposal capability did not occur. This change is acceptable because it results in lower on-site storage than predicted and therefore does not create any regulatory or safety concern.

50.59 Evaluation

These USAR changes provide a consistent and clear description of the manner in which the waste storage facilities at WCGS are used. The USAR changes do not affect any system, structure or component (SSC) important to safety nor do they change the performance of activities that are important to the safe and reliable operation of WCGS. The radwaste system has no safety design basis and plays no part in any design basis accident.

These USAR changes make the description of the use of the waste storage facilities at WCGS in Section 11.4 consistent with current administrative procedures. No other procedures are affected.

There is no additional impact on the performance of plant activities nor affect on any SSC important to safety. No design basis accidents are identified as affected.

Since these USAR changes clarify the manner in which the waste storage facilities described in Section 11.4 are used and no additional changes are made to the plant, the performance of plant activities is not affected nor is any SSC affected. Therefore, no credible accidents that could be created are identified because the changes reflect a reduction in on-site storage of radioactive waste or reflect the use of of drums that satisfy all applicable transportation and disposal requirements.

No credible malfunctions of equipment important to safety are identified.

These USAR changes do not affect the performance of plant activities nor is any SSC affected, no acceptance limits are identified that could be affected. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0054 **Revision:** 0

Correction of Diesel Generator Sub-System Discrepancies in the USAR

Activity Description:

Updated Safety Analysis Report (USAR) discrepancies associated with Diesel Generator sub-systems in Section 9.5, "Other Auxiliary Systems" are being corrected.

The first change addresses the manner in which acceptable lube oil quality is maintained. Rather than replacing the lube oil at periodic intervals as the statement implies, the change allows the oil to be analyzed and appropriate actions taken based on analysis results. This change has no expected affects because analyzing the oil to determine if/when replacement is required is an acceptable alternative to periodic replacement. Either method assures that the lube oil quality is maintained. An added advantage is that equipment out of service time is minimized by eliminating unnecessary oil replacement evolutions.

The second change revises the point in the system where lube oil samples are taken. This change results in samples that are more representative of overall lube oil quality by allowing samples to be taken at places other than the drain connections and, therefore, has no expected adverse affects.

Both changes are consistent with existing plant design, operation, and maintenance and with the Emergency Diesel Generator Reliability Program. Therefore, the changes do not create a disagreement in these areas. The proposed changes do not constitute a test or experiment.

The first change is associated with maintaining lube oil quality. Analyzing the lube oil periodically and basing decisions for replacement on analysis results is an equivalent means of maintaining oil quality to that of periodic oil replacement. This method also minimizes equipment out of service time which is one of the parameters monitored by the Emergency Diesel Generator Reliability Program required by Technical Specification 5.0.

The second change is associated with periodic sampling of the lube oil for performance of oil analyses. This change allows samples to be taken from other points in the system instead of from drain connections. The change provides oil samples that are more representative of overall lube oil quality than taken from drain connections. The result is an improvement in the oil sampling program to insure oil quality and optimum equipment lubrication.

50.59 Evaluation

The affected equipment is not associated with the initiating events of any accident scenario. However, the diesel generators are required to operate during a Loss of 'Non-emergency AC Power to the Station Auxiliaries' accident as described in USAR Section 15.2.6, and the proposed changes are evaluated accordingly.

Both changes are associated with the Emergency Diesel Generator lube oil system with regard to analyzing and maintaining lube oil quality, and do not create any accident scenario initiating events.

The proposed changes do not affect the initiating events for accidents analyzed in the USAR. Therefore, the proposed changes will not increase the probability of occurrence of an accident previously evaluated.

The proposed changes have no adverse affect on the ability of the emergency diesel generators to functions as required during a loss of Non-emergency AC power to the station Auxiliaries. In fact these changes will help ensure a more reliable system by improving lube oil sampling techniques and minimizing equipment out of service times. Therefore, the proposed changes do not increase the radiological consequences of an accident in the USAR.

The changes represent methods of sampling lube oil and maintaining lube oil quality that are equal to or better than previous methods. Therefore, the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety. Similarly, since the changes represent an improvement in maintaining lube oil quality, proper function of components in fulfilling their safety functions is enhanced. Therefore, the proposed change does not increase the radiological consequences of a malfunction of equipment important to safety.

Both changes are consistent with requirements of the Emergency Diesel Generator Reliability Program addressed in Technical Specification 5.0. The proposed changes do not affect any acceptance limits in the Technical Specification bases or other licensing basis documents. Therefore, the change will not reduce the margin of safety.

Safety Evaluation: 59 1999-0056 **Revision:** 0

USAR Clarification of Plant Computer Outputs

Activity Description:

USAR Table 7.5-1 is being revised to indicate that the displayed parameter has computer display capability for the following: reactor coolant system pressure, containment pressure, steam generator pressure, reactor coolant system wide range temperature (hot), reactor coolant system wide range temperature (cold), refueling water storage tank level, steam generator water level, control room air intake gaseous radioactivity, containment gaseous radio-activity, containment hydrogen, containment normal sump level, containment normal sump level, containment purge gaseous radioactivity, fuel building gaseous radioactivity, containment air temperature, and containment post accident radiation. Although this change reflects the current plant as-built condition, this is an increase in the number of plant computer outputs described in the USAR.

The subject change reflects an increased number of computer outputs that are available to the operations staff. These computer outputs are available for display on demand. The computer outputs are non-safety related. Where computer outputs are available to monitor parameters important to safety, qualified indicators take precedence. In the event of a computer output malfunction, the presence of qualified indicators precludes equipment operation based upon unreliable information from the computer outputs. Therefore, the subject increase in the number of computer outputs does not create a challenge to safety.

The addition of these computer points requires no field work. The information supplied by the computer points is supplemental to other sources that are sufficient to monitor plant conditions. The proposed activities described do not impact any procedures, activities, administrative controls or sequence of plant operations nor are any other plant structures, systems, components or equipment affected. 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.

50.59 Evaluation

The computer points play no part in the design basis accident accidents discussed or referenced in the USAR. The information supplied by the computer points is not required to monitor plant conditions. Therefore, no design basis accidents are affected.

The presence of qualified indicators precludes equipment operation based upon unreliable information from the computer outputs. The addition of these computer points requires no fieldwork. Therefore, there are no credible accidents that could be created. In addition, no credible malfunctions of equipment important to safety are identified.

The computer outputs play no part in the acceptance limits contained in the basis for the technical specifications or other licensing documents. The information supplied by the computer points is supplemental to other sources that are sufficient to monitor plant conditions. No acceptance limits that could be affected are identified. Therefore, there is no decrease in the margin of safety.

Safety Evaluation: 59 1999-0066 **Revision:** 0

Change to Address Radwaste Discrepancies

Activity Description:

The proposed changes address and clarify minor discrepancies in the USAR for the radwaste building and the operation of the radwaste systems. The radwaste systems are non safety related or special scope and none of these documentation corrections affect the design basis function of the systems or its hazards and accident analysis in the USAR.

50.59 Evaluation

The changes address inconsistencies and typographical errors in the USAR. No new equipment is being added, removed or modified by this change that would affect the design basis, functions, failure modes, sampling requirements, release limits or consequences related to the radwaste systems. All components referenced in this change are either non-safety related or special scope (pressure boundary only). Therefore no safety-related equipment is affected by this change or any other equipment important to the safe and reliable operation of WCGS is affected by this change. No other USAR descriptions or conclusions would change or be untrue due to this change.

The USAR changes will not introduce or create a new release path to the environment.

Since all the liquid processing continues to be performed in accordance with approved procedures within the radwaste building, the probability of release of liquid or gases to the environment has not been increased. Isolation valves utilized in containing any leakage are not affected by this change.

The changes being described and any related consequences are bounded by the worst case accidents previously identified and evaluated for the radwaste processing equipment in the plant. Therefore, there are no accidents that are impacted or changed due to this change.

Since 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 no acceptance limits are included in the bases of the Technical Specifications or licensing basis documents, no acceptance limits are identified that could be affected. Therefore, there is no decrease in the margin of safety.

Safety Evaluation: 59 1999-0092 **Revision:** 0

Changes to Correct Refueling Water Storage Tank Alarm Logic and Accumulator Service

Activity Description

The USAR states that Refueling Water Storage Tank (RWST) level alarms initiate on two out of four logic. This does not agree with current plant configuration of one out of four logic for this level indication. The proposed changes will reflect actual plant conditions in the USAR. The logic and plant conditions are more conservative than those described in the USAR.

Also, the Emergency Core Cooling System (ECCS) accumulators are being placed into service at a lower Reactor Coolant System (RCS) pressure. This means they will be available to perform their safety function for a broader operating range that increases the availability of safety related equipment to mitigate the consequences of an accident. This is a more conservative method of operation.

The changes will provide for more conservatism and safety in the operation of the plant by providing greater sensitivity in the monitoring of tank levels and by placing safety related equipment into service at more conservative plant conditions. The changes do alter plant operation as described in the USAR while remaining fully bounded by original plant analysis. No plant system, structures or components (SSC's) have been modified or affected in any way.

50.59 Evaluations

The changes being reviewed are resultants of Loss Of Coolant Accident (LOCA) Analysis. Therefore, this accident was reviewed for this change. Based on this review the following conclusions were made:

The proposed changes deal with making the various alarms for the RWST level initiate with only one transmitter in an abnormal condition and the ECCS accumulators are being placed into service at a lower RCS pressure. The changes do not change the physical plant. The changes are fully bounded by plant analysis and other USAR Sections. Therefore, no other credible accidents could be created.

The current USAR descriptions assume less conservative conditions for equipment important to safety. Therefore, malfunctions are not affected by this change.

The proposed change is already bounded by analysis. Therefore, the change does not affect the acceptance limits which are contained in the bases for the Technical Specifications and the change does not decrease the margin of safety.

Safety Evaluation: 59 1999-0096 **Revision:** 0

Change Flows and Heat Duty Listings in the USAR Related to the Design and Performance of Water Systems

Activity Description

This USAR is being revised to change several USAR tables in chapter 9.2.

The following types of changes are being made to the USAR Tables:

- Editorial corrections to make the numbers in the Tables add up correctly and to make the Tables cross-reference correctly.
- Removal of component exit temperature information from Tables 9.2-2; 9.2-3; 9.2-4; 9.2-9; 9.2-10; and 9.2-11 and the modification of Notes associated with Tables accordingly.
- Replacement of design specification heat duties with calculated heat duties for the various room coolers in Tables 9.2-2; 9.2-3; 9.2-4; 9.2-19 and 9.2-20.
- Correction to the water flow rate for the Spent Fuel Pool Pump Room Coolers in Table 9.4-6 to reflect the design specification flow rate of 29 gpm. This had been previously changed to 32 gpm, which is the microbiologically induced corrosion (MIC) flow rate.
- Correction of Table 9.2-1 to reflect the replacement of the "C" air compressor with a larger unit that is cooled by the Central Chilled Water System (CeCWS).

The net affect of this change will be to make the Tables consistent with each other where information is referenced between Tables. It is also ensures that actual (calculated) heat duty information is used for the room coolers in the various Tables instead of the design specification heat duties. It should be noted that the component exit temperature information is not referenced in the text of USAR Section 9.2 which discusses the ESW and CCW systems.

These changes to USAR tables do not affect other sections of the USAR such that they would be made untrue. The operation of the ESW and CCW systems is not changed as a result of this change. As noted above, the component exit temperature information which is being removed from the Tables is not referenced in Section 9.2 of the USAR. This change does not make other information contained in the USAR untrue or inaccurate. There are no tests or experiments involved with this change.

50.59 Evaluation

USAR Section 6.2.1.5 discusses a limiting analysis for minimum post-LOCA containment pressure (maximum Safety Injection with low lake temperatures). USAR Section 9.2.5 discusses Ultimate Heat Sink (UHS) sizing analyses. For the UHS, the limiting case is a LOCA with maximum Safety Injection (using the full capacity of the CACs). This case corresponds to the conditions summarized in USAR Tables 9.2-3 and 9.2-11 and is the basis for USAR Figures 9.2-6A, 9.2-6B and 9.2-6C. USAR Table 9.2-9 deals with normal operation of the CCW system. Table 9.2-19 deals with heat loads from station auxiliaries during post LOCA conditions. Table 9.2-20 deals with heat loads from station auxiliaries during normal shutdown. This change clarifies the related flows and duties presented in the USAR Tables for the Service Water, ESW and CCW systems. The design basis for the events described in Sections 6.2.1.5 and 9.2.5 is not changed. Therefore, there is no impact on the design basis accidents.

The change will ensure the presentation of consistent and correct information in these USAR Tables. No new type of credible accident is created as a result of this change.

No credible malfunctions of equipment important To safety are directly or indirectly affected by this change.

Since the operation of the ESW and CCW systems is not changed as a result of this change, no acceptance limits are identified that could be affected. Therefore, there is no decrease in the margin of safety.

Safety Evaluation: 59 1999-0104 **Revision:** 0

Change to Class 1E Descriptions

Activity Description:

The reference to voltage instrumentation for the 480V supply system is being corrected to agree with plant design and the body of USAR Table 7A-3.

The Class 1E 480 volt system is fed from the Class 1E 4.16kV bus. In most cases the 480v system voltage will follow the voltage swings of the 4.16kV system. Current indication provides the status of the board demand for each 480V bus. Metering of voltage for these boards in the main control room is unnecessary. Undervoltage annunciation and current indication allow unit operators to perform appropriate actions when needed. The undervoltage circuit of these buses does not interface with Emergency Diesel Generator (EDG) operation or control. An additional eight volt meters in the control room are unnecessary and could cause unwarranted concern during minor transients such as load changes. Additionally, no action is required if voltages are above alarm conditions. Premature action would be inefficient and in most cases would prevent the assigned operator from performing more important duties. This change will also bring the Section statement and Table remarks into agreement with the information provided in Table 7A-3.

These changes to USAR Section 8.3.1.1.2 and Table 7A-3 do not affect other sections of the USAR such that they would be made untrue. The operation of the 480V supply systems is not changed. This change does not make other information contained in the USAR untrue or inaccurate.

50.59 Evaluation:

480V voltage indication is not used in the mitigation of design basis accidents. The 480V bus is fed from the 4160V bus. The control and instrumentation for 4160V system can cause a Emergency Diesel start but the 480V controls are independent and will not affect this related system. The 480V undervoltage will still alarm in the control room, which will result in the implementation of the appropriate alarm procedure. Design Basis accidents are not impacted by this change.

The proposed change deals with the need for voltage indicators for the Class 1E 480V supply buses in the control room. These indicators are not used to mitigate the consequences of a design basis accident. No credit is taken for these indicators in current plant analysis. Therefore, no other credible accidents could be created by the lack of voltage indication.

This change does not affect the physical plant. Therefore, no credible malfunctions of equipment import to safety are directly or indirectly affected by this change.

This change deals with monitoring instrumentation not used in the operation of the plant.

No limits have changed. The change does not affect the acceptance limits that are contained in the bases for the Technical Specifications. Therefore, the margin of safety has not been decreased.

Safety Evaluation: 59 1999-0106 **Revision:** 0

Update to Radiological Units

Activity Description:

This change to the USAR updates the radiation measurement units of mR/hr to mrem/hr in USAR. It also corrects a mathematical error for one point in USAR Table 11.4A-5B. The correct value for the point in USAR Table 11.4A-5B was shown in USAR Figure 11.4A-3, "Total Dose Rates Outside the IOS Up To The RCA Boundary."

Rem is the unit of dose equivalent in terms of biological effects and Roentgen is related to air dose of gamma or x-rays. An example of the use of Roentgen is the dose rate provided by a radiation protection instrument or monitor in a non-biological effect measurement application. A rem is equivalent in terms of its estimated biological effect relative to the dose of one roentgen for x-ray and gamma radiation. Therefore, this change is considered a change to procedures described in the USAR.

50.59 Evaluation:

Due to changes in the Code of Federal Regulations that governs radiation protection activities, two units of radiation dose, (rem and roentgen) used throughout the USAR need to be updated to current accepted applications. These two units of radiation measurement, rem (tissue equivalent) and roentgen (or R) (non-tissue equivalent) in some cases are stated incorrectly depending on the conditions of irradiation.

This USAR change makes the use the rem (or mrem) consistent throughout the USAR where the radiation doses being discussed are personnel radiation exposures. This use of the term mrem is consistent with the Code of Federal Regulations that covers Radiation Protection activities. No other sections of the USAR are made untrue by this USAR change. This USAR is an editorial change only; no physical changes are made to any plant structures, systems, or components and no procedure or accident analysis is affected.

Because this is a change to administrative values, no accidents are affected, and no equipment is affected. The changed does not create new equipment malfunctions or new accidents. Nor does the change affect acceptance limits that would cause a decrease in the margin of safety.

Safety Evaluation: 59 1999-0112 **Revision:** 0

Containment Humidity Clarification

Activity Description:

The USAR states the containment humidity measuring system has a range of 5 to 99 percent relative humidity at 80 degrees F. The actual range of the containment humidity measuring system in accordance with WCGS Total Plant Setpoint Document is 10 to 98 percent relative humidity at 80 degrees F. The containment humidity measuring system will continue to monitor containment humidity at normal levels. During accident conditions such as LOCA or Main Steam Line Break (MSLB) in containment, the containment relative humidity is assumed to be 100 percent and is outside the range of the containment humidity measuring system. The actual range of the containment humidity measuring system will continue to allow the system to serve as an aid in leak detection as described in the USAR.

50.59 Evaluation:

The containment humidity measuring system is outside the normal operating range during LOCA or MSLB inside Containment, and functions as designed during other accident conditions. The information obtained from this system is not used to mitigate the consequences of a design basis accident and provides no safety function. Therefore, no accidents are affected by this change.

The proposed change is associated to the operational range of the containment humidity measuring system which provides non-safety related information that is used in monitoring the containment environment during normal plant operation. Therefore, the proposed changes could not create any type of credible accident.

The proposed change alters the range of the containment humidity measuring system given in the USAR. The system still provides the unit operator relative humidity information for normal containment conditions. The system provides no output that could affect equipment important to safety and provides no information that will affect the operation of safety related equipment. Therefore, the change will not directly or indirectly cause a credible malfunction of equipment important to safety.

The range of the containment humidity measuring system is not a technical specification limit and the system provides no information that affects technical specification limits. Therefore, no acceptance limits found in the technical specification or licensing basis are affected by this change and the margin of safety has not been decreased.

Safety Evaluation: 59 1999-0116 **Revision:** 0

Elimination of the Mechanical Environmental Qualification Program.

Activity Description:

The Mechanical Environmental Qualification (EQ) program as listed in USAR Table 3.11(B)-3 has 2449 components. The deletion of the mechanical equipment from the EQ program is being evaluated.

Electrical EQ - forty-eight (48) of these mechanical components are qualified under their associated electrical EQ package. No separate mechanical EQ maintenance criteria were developed. Any EQ required maintenance is listed under the Electrical EQ requirements and will not be changed. These 48 mechanical listings will be deleted with no reduction in EQ monitoring since none has been previously identified or required by the mechanical EQ program.

No Organic parts - 135 of the mechanical components have been identified as having no organic parts or having no organic parts essential to the component's safety function, i.e., all metallic. The EQ program identifies the negative effects of the environment on non-metallic parts. The EQ program is not applicable to these components since there are no non-metallic parts essential to the components safety function. These 135 mechanical component listings will be deleted with no reduction in EQ monitoring since no monitoring has been previously identified or required by the mechanical EQ program.

Mild environment - 1409 of the mechanical components have been identified as located in a mild environment. Mild environment equipment need only be designed for service in the normal environment where it is located. This also includes NUREG-0588, Category C equipment [Cat. C equipment is equipment whose status post accident has no effect on plant safety and is treated in the same way as mild environment equipment]. These 1409 mechanical component listings will be deleted with no reduction in EQ monitoring since no monitoring has been required by the mechanical EQ program.

Appendix J - Seventy-seven (77) of the mechanical components are included in the 10 CFR 50, Appendix J, Containment Leak Rate Testing Program (Appendix J). The Appendix J program establishes a testing frequency for the identified components (valves - 1 - 3 cycles; penetrations - 5 years) based on the component's performance. The integrated leak rate test (ILRT) tests containment leakage as a whole, testing all Appendix J components every 10 years. [NOTE: Some of the mechanical components included in the Appendix J program have EQ maintenance requirements to replace non-metallic parts at a specified time to maintain a 40-year qualified life. These maintenance requirements have been in the preventative maintenance (PM) program as EQ requirements. The mechanical EQ maintenance requirements will remain in the PM program but will only be identified as "previous mechanical EQ requirements". Any changes to a component's preventive maintenance will be controlled by the PM program.] The Appendix J program monitors these mechanical components and will identify any performance problems and provide corrective action at a greater frequency than that required by the mechanical EQ program. Five (5) of these 77 components have already been included in the electrical EQ or mild environment and will not be counted as part of the Appendix J components. The remaining 72 mechanical component listings will be deleted with no reduction in EQ monitoring.

Qualified life >40 years - 884 of the mechanical components have been identified with a qualified life greater than 40 years, with some components requiring maintenance to achieve the 40 year qualified life. As described above, the mechanical EQ maintenance is part of the preventative maintenance (PM) program and will remain in the PM program with a reference to the maintenance previously being a mechanical EQ requirement. The 40 year qualified life is maintained by: 1) the WCGS procurement and material control program for safety related equipment ensuring that replacement parts meet the same requirements as the original parts as to form, fit and function; and 2) specifications require that the replacement equipment and parts be qualified to the environmental conditions. 106 of these 884 components have already been included under previous discussions for deletion (electrical EQ, no organics, or Appendix J). The remaining 778 mechanical components will be deleted based on their qualified life being greater than 40 years without any reduction in EQ monitoring.

American Society of Mechanical Engineers (ASME) In-service Testing (IST) program - 978 mechanical components are identified in the ASME IST program. 973 of the these 978 components have already been included under previous discussions for deletion (electrical EQ, no organics, mild environment, qualified life >40 years, and/or Appendix J). The 5 remaining components in the IST program have specific test procedures identified by the IST program for testing their performance which is more frequent than any EQ required monitoring. These 5 mechanical EQ components can be deleted based on their IST program monitoring without any reduction in EQ monitoring.

Two mechanical components which had been removed from the plant were still listed in the table. These two component listings are being deleted.

The mechanical EQ program has been reviewed and approved. The mechanical equipment either does not require EQ monitoring (no organic parts in safety function, mild environment, or greater than 40 year life) or has its operability periodically assessed (Appendix J program, ASME IST, or plant operation). Additionally, the procurement and refurbishment controls maintain replacement parts are equivalent in form, fit, function and material.

USAR Table 3.11(B)-3 is a listing of the electrical and mechanical components in the EQ program. USAR Table 3.11(B)-3 also serves as a comprehensive listing of hot and cold shutdown components. Because of this second function, mechanical components required for hot and/or cold shutdown will have their EQ information replaced with a footnote reference identifying their inclusion in the table for shutdown purposes only, and are not part of the EQ program.

Deletion of the mechanical EQ program will allow the resources to focus their efforts on the electrical EQ program without any reduction in equipment performance. All component and system functions will continue to be performed as designed.

50.59 Evaluation

The proposed change does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed activity will not alter the equipment, its function or the plant physical configuration. Therefore no accidents are identified as being affected by this change.

Elimination of the MEQ will not alter the equipment, its function or the plant physical configuration. Since the plant is not being changed, no accidents could be created.

Mechanical equipment is primarily fabricated from metallics which are unaffected by aging and radiation effects. The Mechanical EQ program analyzed the non-metallic parts that were used in the safety function of the mechanical components. The non-metallic parts with estimated qualified life < 40 years had replacement frequencies established and the required maintenance included in the PM program. The replacement requirements will remain in the PM program with a reference to being a previous MEQ requirement. The majority of the mechanical EQ components were either located in a mild environment, had no organic parts in a safety-related function, or had a qualified life > 40 years. The remaining mechanical equipment is monitored by an existing program that will identify any degradation in operation and provide corrective action. These conditions will remain when the MEQ program is eliminated. Therefore, no malfunctions of equipment important to safety are identified.

The proposed change will not alter the equipment, its function, or the plant physical configuration. Since all components and systems will continue to perform their original design intent; no acceptance limits that could affect the basis for any technical specification are identified. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0121 **Revision:** 0

Change to the Description of AMSAC Indicating Lights

Activity Description:

ATWS Mitigation System Activation Circuitry (AMSAC) provides control board annunciators and several indicating lights in the AMSAC logic cabinet. The USAR, Section 7.7.1.11 has descriptions of the indicating lights and the trouble conditions for the alarm. This change corrects the description of the indicating lights and the miscellaneous trouble conditions of the alarm to that described in the design bases documents. Also, the statement "AMSAC cannot be manually reset by the operators until after its mitigating actions are completed" implies that as if there is a manual reset function, but AMSAC does not have any manual reset function. This statement has also been revised in the proposed USAR change to delete the implication of a manual reset function.

50.59 Evaluation:

The form, fit, function, procedure or operation of equipment is not affected by the proposed revision to the USAR. This change makes the USAR information the same as the design basis documents and the AMSAC logic cabinet.

The proposed change in the description in the USAR, Section 7.7.1.11 will not impact directly or indirectly the design basis accidents. The proposed activities shall not create any credible accidents.

The proposed activity will not affect directly or indirectly any SSCs. The operation of the AMSAC will not be affected by the proposed activities. There are no acceptable limits in the technical specification or in the licensing basis documents that could be affected by correcting the USAR text to reflect the plant configuration. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0124 **Revision:** 0

Hydrogen in Battery Rooms

Activity Description:

USAR Section 9.5.B.7 and Table 9.5A-1 are being changed to reduce the hydrogen concentration that will actuate the hydrogen detector in the battery rooms that alarm in the control room. The setpoint is being lowered from the current 2 volume percent to a more appropriate and conservative setpoint of 1 volume percent.

50.59 Evaluation:

The proposed change will correct the inconsistencies between the more conservative battery room hydrogen detector setpoints and USAR descriptions. The change does not affect any SSC or change the performance of activities that are important to the safe and reliable operation of WCGS.

There is no additional impact on the performance of plant activities or effect on any SSC. Therefore, no Design Basis Accident is affected. No credible accidents that could be created are identified.

Because the proposed change reflects more conservative battery room hydrogen detector setpoints, the potential for equipment malfunction is reduced. The hydrogen detectors provide an alarm only. Lowering the setpoint does not increase the probability of spurious actuation. Therefore, no credible malfunction of equipment important to safety is created or increased.

Establishing lower setpoints is not relevant to any acceptance limits that are contained in the bases for the technical specifications or in licensing basis documents. Therefore, no acceptance limits are identified that could be affected and the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0128 **Revision:** 0

Revised Non-LOCA Radiological Consequence Analyses

Activity Description:

Radiological consequence analyses for several Chapter 15 design basis accidents have been performed using the updated source terms, transmitted by Westinghouse letters (SAP-98-154, dated 11/18/98 and SAP-99-113, dated 2/18/99). The design basis accidents analyzed include the steam line break outside containment, the loss of non-emergency alternating current (AC) power, the Chemical and Volume Control System (CVCS) letdown line rupture, the locked rotor accident, the radioactive gas/liquid waste system failure, and the rod cluster control assembly (RCCA) ejection accident.

The following proposed general USAR updates will incorporate the revised radiological consequence analysis results and assumptions used:

· Revise the tabular presentation of isotopic activities released to the environment, as calculated for the design basis accidents analyzed.
· Revise the tabular presentation of thyroid and whole body dose information, as calculated for the design basis accidents analyzed.
· Radiological consequence analysis assumptions include revising the SG water mass to 95,500 lbs, including the unaffected SG in the MSLB analyses, and revising the MSLB affected SG water mass and steam release to 164,500 lbs, to incorporate mass values based on the rerated conditions specified in WCAP-13447, Volume 1, of October 1992.
· A radiological consequence analysis assumption, for the waste gas decay tank rupture source determination, accounts for degassing of the reactor coolant at shutdown, consistent with the Westinghouse letter SAP-99-113, dated 2/18/99. The current licensing basis source determination and by extension the radiological consequence analysis did not consider degassing of the reactor coolant at shutdown.
· Revise Section 15.7.2.5.1.2.d and Table 15.7-5 to correctly indicate that 10 percent of the iodine activity is released as airborne activity, rather than 1%. Note: This is consistent with the current licensing basis.
· Revise the analysis conditions to indicate that ten percent of the fuel rod gap activity, except for Kr-85 and I-131, which are 30 percent and 12 percent respectively, is released to the reactor coolant. Note: The I-131 value of 12%, used in the revised radiological consequence analyses, reflects a more conservative value based on NUREG/CR-5009, February 1988, and I-131 dose equivalency is explicitly considered in the licensing basis analyses.

50.59 Evaluation:

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. 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 Chapter 15 radiological consequence analyses 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 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.

The Chapter 15 accidents for which radiological consequence calculations have been performed are shown by the following:

1) The Steam Line Break Outside Containment
2) The Loss of Non-emergency AC Power
3) The CVCS Letdown Line Rupture
4) The Locked Rotor Accident
5) The Radioactive Gas/Liquid Waste System Failure
6) The RCCA ejection accident

The proposed changes to the USAR text and tables, pertaining to input parameters, assumptions, computer codes, references, and results associated with accident analyses, are made to achieve consistency between USAR descriptions and the design and license basis analyses. The changes will not impact the overall system performance in a manner that could cause an accident previously evaluated to shift to a higher frequency category. As such, there will be no increase in the probability of occurrence of an accident previously evaluated in the USAR.

The proposed changes have no affect on mitigating capabilities of plant equipment, source terms or release paths. Therefore, there will be no increase in radiological consequences.

There are no physical modifications to the systems, components, and equipment or changes in methods of operation. Therefore, no credible accidents that could be created are identified.

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 identified.

The proposed changes do not affect the I-131 dose equivalent limits for the specific activities of the primary and secondary coolant, per the Technical Specification LCOs 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, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected. The margin of safety has not been reduced.

Safety Evaluation: 59 1999-0131 **Revision:** 0

USAR Update to Chapter 15.6 Text and Tables

Activity Description:

Revisions to the following USAR sections and tables, incorporating the clarifications or corrections, are proposed:

1. Sections and tables in chapter 15 will correct the erroneous unit expression for iodine activity from milli-curie per gram to micro-curie per gram.

2. A clarification statement to table 15.6-8 will indicate that the airborne iodine is conservatively assumed to be released immediately to the environment from the unit vent, via the emergency exhaust system without mixing or dilution in the surrounding auxiliary building atmosphere.

3. Table 15.6-9 will reference two newly added Tables 15.6-13 and 15.6-14 that express the pumped Safety Injection flow as a function of Reactor Coolant System pressures.

4. Table 7.7-4 will properly reference the "Inadvertent Opening of a Pressurizer Safety or Relief Valve" event in section 15.6.1.

5. Section 15.6.1.2 will replace the mismatched "turbine trip transients" with "inadvertent opening of a pressurizer safety valve transient". In addition, the appropriate reference cited in USAR Section 15.6.3.5 for RETRAN-02 and VIPRE-01 computer codes is made.

6. Section 15.6.2 revises the inappropriate statement "the release rate is within the capability of the reactor makeup system" to "makeup capability of the charging system".

7. Section 15.6.3.1.2 will change the statement in the last sentence from "With one of the required ARV's inoperable due to the single failure assumption, ..." to "With one of the required ARVs unavailable due to its association with the ruptured SG, ...".

8. Section 15.6.3.3.1.4 will revise the statement associated with leakage pathway to reflect the leakage release pathways considered in the calculated Steam Generator Tube Release dose consequences include the Steam Generator (SG) safety valves, Atmospheric Relief Valves, and the exhaust stack of the Turbine Driven Auxiliary Feedwater pump.

9. Section 15.6.3.3.1.2 changes the amount of reactor coolant discharged to the secondary side of the faulted SG from 161,000 lbm to 138,417 lbm to be consistent with the value shown in Table 15.6-4.

10. Section 15.6.3.3.3.2 changes the conclusion statement from "The resultant doses are well within the guideline values of 10 CFR 100 and Standard Review Plan 15.6.3" to statements that refer to a small fraction (10% of Part 100 limits) for the concurrent iodine

spike case and full Part 100 limits for the pre-accident iodine case.

11. Section 15.6.3.3 changes the unit expression for iodine activity to micro-curie per gram from milli-curie per gram.

12. Section 15.6 changes the computer code reference for the inadvertent opening of a pressurizer safety valve and SGTR events from "LOFTRAN" to "RETRAN-02/VIPRE-01" and deletes the LOFTRAN reference.

13. Section 15.6.5.3.3 changes the reference cited for ECCS Evaluation Model sensitivity studies from "20" to "17" .

14. Table 15.0-2 changes the computer codes listed in this table for the LOCA analyses to reflect the current licensing basis: NOTRUMP and LOCTA-IV for the small break LOCA, and SATAN-VI, WREFLOOD, COCO, BASH, and LOCBART for the large break LOCA.

50.59 Evaluation:

The proposed change to the USAR text and tables, pertain to input parameters, assumptions, computer codes, references, and results associated with accident analyses, are made to achieve consistency between USAR descriptions and the design and licensing basis analyses. The changes will not impact the overall system performance in a manner that could cause an accidents previously evaluated to shift to a higher frequency category. As such, there will be no increase in the probability of occurrence of an accident previously evaluated in the USAR nor will any accidents be created.

The proposed changes are made to achieve consistency between the USAR descriptions and analyses of record and have no affect on mitigating capabilities of plant equipment, source terms or release pathways. As such, there will be no increase in the radiological consequences.

No malfunctions of equipment important to safety have been identified either as being increased or created.

The proposed changes do not affect the I-131 dose equivalent limits for the specific activities of the primary and secondary coolant, per the T`echnical Specification LCOs 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, nor will there be any effect on the plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are affected and the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0140 **Revision:** 0

USAR Changes Associated to Boric Acid Requirements

Activity Description:

This change to the Updated Safety Analysis Report (USAR) incorporates clarifications and corrections to make the USAR reflect the existing safety analyses, Technical Resource Manual (TRM) bases, and the plant configuration. No change is proposed to the existing licensing basis analyses or to any plant structure, system, or component, or plant operating procedures. Revisions to the following USAR sections and tables, incorporating the clarifications or corrections, are proposed:

1) Revise Section 5.4A.3.2 to specify the volume requirement of water to be charged and letdown from full power conditions at the end of cycle to 83,754 gallons from 40,000 gallons, consistent with the TRM bases.

2) Clarify Section 9.3.4.3 to indicate that the boric acid transfer pump boration rate sufficient to take the reactor from full power to one percent shutdown in 90 minutes, at hot conditions without rods inserting, represents a design capability.

3) Revise Table 9.5B-1 to indicate the availability of the boric acid tanks and the boric acid transfer pumps, in addition to the Refueling Water Storage Tank and charging pump, for safe shutdown.

4) Insert a clarifying footnote identifying the 18,500 gallon value as the "design nominal" combined boric acid tank (BAT) boric solution volume. This is consistent with the terminology usage in Safety Evaluation 59 95-0155.

50.59 Evaluation:

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

Chapter 15 USAR Section Design Basis Accidents were reviewed and no potential impact due to the proposed activity was determined to exist. No credible accidents are created by the proposed changes. 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. 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. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0141 **Revision:** 0

Change Associated to Station Blackout

Activity Description:

There are several places in the USAR where the term "blackout" or "station blackout" is used referring to loss of non-emergency off-site power. The 10CFR50.2 definition applies specifically to a loss of all AC power (except buses fed by batteries through inverters). The purpose of this USAR change is to delete those references to the term station blackout where they are not appropriate. Also, Appendix 8.3A, Station Blackout, is proposed to be added to the USAR.

The Safety Evaluation (SE) and Supplemental Safety Evaluation (SSE) for the station blackout event are provided in the Letter 92-00148 and Letter 92-01202, respectively. However, the minimum water contained in the condensate storage tank (CST) for coping with station blackout has been changed from 151,000 gallons to 156,300 gallons. The plant Technical Specifications (TS) require a minimum of 281,000 gallons to be maintained in the condensate storage tank. This TS required capacity exceeds the amount of water necessary for coping with a station blackout event.

50.59 Evaluation:

Since the station blackout event has been reviewed by NRC, this Unreviewed Safety Question Determination is only applied to the change of the minimum water contained in the condensate storage tank and editorial clarification/corrections.

The proposed USAR changes add an appendix to address the station blackout event and the minimum water contained in the condensate storage tank for coping with station blackout is changed to 156,300 gallons from 151,000 gallons. This change reflects the increase in the maximum CST temperature from 100 degrees F to 130 degrees F assumed in the calculation of the CST inventory for decay heat removal following station blackout. This change has no effect on the event since the TS required capacity exceeds the amount of water necessary for coping with a station blackout event. The changes are not associated with any hardware and will not create any potential failure modes. Therefore, no accidents are affected or created.

This USAR change request is not a physical change to the plant and therefore will not directly affect malfunctions of equipment important to safety.

The plant Technical Specifications (TS) require a minimum of 281,000 gallons ``````be maintained in the condensate storage tank. This TS required capacity exceeds the amount of water necessary for coping with a station blackout event. No TS sections are affected by the proposed changes. Thus, no acceptance limits associated with any TS are affected by these modifications. The margin of safety has not been reduced.

Safety Evaluation: 59 1999-0146 **Revision:** 0

Change to Fire Protection Program to Address Fuel Oil Transfer Pump Function

Activity Description:

This change addresses updating USAR Appendix 9.5B sections D.1.7.1 and D.2.7.2. to be consistent with USAR section 9.5.4.2.3. The actual configuration is accurately described in section 9.5.4.2.3 as follows: "A fire detection signal from the diesel building stops the fuel oil transfer pump. However, the fuel oil pump will not be stopped if the diesel generator is running to preclude spurious trips from the fire protection system under accident conditions." A review of design and USAR documentation determined that the discrepancy arose when a change evaluated by USQD 59 93-0243 revised USAR section 9.5.4.2.3, but did not change associated USAR Appendix 9.5B sections D.1.7.1 and D.2.7.2. Therefore, an additional 50.59 evaluation has been performed to address changes to the Fire Protection Program.

50.59 Evaluation:

The function of the fuel oil transfer pump is to replenish the day tanks with fuel as it is consumed by the diesel engine. The pump runs whenever the diesel runs. The pump has a higher capacity than is needed to supply the engine. The excess fuel that is not used by the engine is recirculated back to the fuel oil storage tank via the day tank overflow. At all times, whether or not the pump is running, the quantity of fuel in the day tank remains unchanged (i.e., at the level of the overflow). Therefore, barring an equipment malfunction, which is evaluated below, bypassing the pump trip during engine operation does not result in an increase in the quantity of fuel in the diesel room and does not result in an addition to the combustible fire load in the diesel generator rooms. Under emergency operation mode (i.e., design basis accident condition), a fire is not postulated to occur based on the fire hazards analysis. During a diesel generator test, a fire in the diesel generator building will be detected by the operations personnel or the fire detection system, at which time immediate action will take place to shut down the diesel generator and transfer pump thereby, cutting off the fuel supply. Therefore, under all modes of diesel generator operation, the continuous operation of the fuel oil transfer pump will not result in a greater hazard than that which is already analyzed.

PMR 04253 provided an evaluation of the potential for fire hazard due to fuel oil spillage resulting from equipment malfunction. The evaluation concluded that spillage of fuel oil into the diesel generator room would be no greater than that which would occur with a ruptured fuel oil supply line, so no new unanalyzed hazard exists.

The bypass of the fire detection trip of the fuel oil transfer pump is required to ensure that a spurious trip of the fire detection circuit would not adversely impact the operation of the diesel. The fire detection trip of the fuel oil transfer pump has no accident related function. As stated in USAR Section 9.5B.2.b: "A design basis accident occurring simultaneously with a fire hazard is not assumed." Therefore, no design basis accident is identified affected.

Bypassing the pump trip during engine operation does not result in an increase in the quantity of fuel in the diesel room. Further, as stated in USAR Section 9.5B.2.b: "A design basis accident occurring simultaneously with a fire hazard is not assumed." Therefore, no design basis accident is affected.

Bypassing the fire detection fuel oil pump trip when the diesel generator is running precludes spurious trips from the fire protection system under accident conditions. No additional credible malfunctions of equipment important to safety are directly or indirectly affected by this change.

There are no relevant acceptance limits contained in the bases for the technical specifications or in licensing basis documents. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0147 **Revision:** 0

Change to Fire Protection Information

Activity Description:

This change to the USAR incorporates clarifications and corrections to USAR section 9.5B.7 to be consistent with the design provisions for the fire protection system contained in section 9.5.1 and appendix 9.5A. Revisions incorporating the clarification or corrections proposed are:

§ Table 9.5A-1, D.4.f (Sheet 37) will be revised to indicate that the Communications Corridor stairway, which is described in Fire Pre-Plans as an access route for fire-fighting, is enclosed in a 2-hour-rated fire barrier instead of 3-hour. The communications corridor is not a safety-related structure, contains no safe shutdown equipment or circuits, and is a non-seismic Category I structure. The WCGS response complies with APCSB guidelines by stating: "In non-seismic Category I buildings, stairwell doors have a 1½ hour rating. Fire preplans and drills are performed to provide escape and access routes for all areas."

§ Section 9.5B.7, C.13.7.1 and C.14.7.1 will be revised to indicate that if a fire occurs in the charcoal adsorber unit, it will be extinguished manually using hose station and or portable extinguishers.

§ Section 9.5B.7, C.27.7. will more clearly describe the locations of hose stations and portable fire extinguishers which may be used for fire suppression.

§ Figure 9.5.1-2-03 indicates that there is a hose station located in a fire area that is not addressed in the description in section 9.5B.7.F.1.7.1. The proposed change will add a hose station to the description of fire suppression equipment.

§ Figure 9.5.1-2-03 indicates that there is a fire extinguisher located in a fire area that is not addressed in the description in section 9.5B.7, F.5.4. The proposed change will add a portable extinguisher to the description of fire suppression equipment contained in the section.

§ Section 9.5B.7, HMS.1.4 will be revised to discuss that only one hose station is contained in this fire area.

§ Section 9.5B.7, AB.1.2 will be revised to delete the statement: "Ducts penetrating the fire barriers are fitted with 3-hour-rated fire dampers." No ventilation ductwork penetrates the fire barriers of this fire area and therefore, no fire dampers have been installed.

§ Section 9.5B.7, AB.1.5 will be revised to eliminate the reference to fire dampers since the auxiliary boiler room roof is not equipped with vents with fusible links. Section 9.5B.7, AB.1.5 will be revised to indicate that smoke will be removed using portable fans and flexible ducting.

50.59 Evaluation:

The clarification of WCGS compliance with Auxiliary and Power Conversion Systems Branch (APCSB) guidelines and the accurate representation of fire protection systems and equipment does not affect any other SSC, nor does it change the performance of activities that are important to the safe and reliable operation of WCGS. The fire protection system and associated procedures are not affected.

The proposed changes will provide an accurate representation of the fire protection system consistent with the design provisions for the fire protection system contained in Section 9.5.1 and Appendix 9.5A. 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 is identified for review. In addition, no credible accidents that could be created are identified.

Since the changes do not affect the fire ratings and Fire Hazards Analyses for the affected areas, no credible Malfunctions of Equipment Important to Safety are identified.

Fire protection systems and equipment 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 and the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0148 **Revision:** 0

USAR Change to Radiological Consequences of a LOCA

Activity Description:

This change will revise the USAR sections regarding radiological consequences of a LOCA. The radiological consequences analysis of a postulated LOCA were reanalyzed using the new source activities to account for higher fuel enrichment and burnup. In addition to the containment leakage and ECCS recirculation leakage presented in the current USAR, the following leakage pathways are also considered in this revised LOCA dose analysis:
1. Operation of the containment mini-purge system prior to a LOCA event.
2. Leakage of post-LOCA containment recirculation sump water from ECCS boundary valves to the atmosphere via RWST.
3. 10 cfm control room direct unfiltered inleakage to account for the possible increase in air exchange due to ingress or egress.

It was determined that the containment mini-purge system can be operated at power. Therefore, the flow via the mini-purge should be included in the LOCA radiological consequences analysis to account for radioactive releases due to the operation of containment mini-purge system. This flow will exist until the containment isolation signal is received and the valves can be closed.

An alternate leakage path from the containment sump through ECCS boundary valves back to the RWST, which is vented to the atmosphere, was identified. Section 15.6.5.4.1.4 was added to include this alternate leakage path for the LOCA dose calculation.

Table 15.6-6 is updated to revise the activity released to containment and sump inventory. Table 15.6-8 is updated to reflect the results of radiological consequences of a loss of coolant accident. Also, the fuel and rod gap inventories presented in Table 15A-3 are replaced with new source activities provided by Westinghouse.

LOCADC computer program was developed to perform the radiological consequences of LOCA. Version 3 of LOCADC added an option to model the unfiltered infiltration to the control room to account for the possible increase in air exchange due to ingress and egress. Version 4 added an option to model the alternate leakage path of containment sump water from ECCS boundary valves to the atmosphere via Refueling Water Storage Tank. To be consistent, it is proposed to modify the equations for two-region spray model and control room radiological consequences calculational model in Sections 15A.2.3 and 15A.3.

The control room flow path model and description in Section 15A.3 as well as the operator action time presented in Table 15A-1 will be updated. The assumed operator action time to ensure no bypass pathways exist for unfiltered air to enter the control room following the worst single failure of the failure of the filtration fan in one of the two filtration system has been changed from 5 hours to 1.5 hours. Procedures have been revised to reflect this change.

50.59 Evaluation:

The proposed change incorporates the results of the reanalysis of LOCA radiological consequences. This proposed change does not involve any hardware changes. There will be no change to normal plant operating parameters. Therefore, there will be no increase in the probability of any accident occurring due to these changes. No new accidents will be created.

The results of the offsite doses remain below the guideline values of NUREG-0800 and 10 CFR Part 100. The results of the control room doses remain below the guideline values of 10 CFR 50 Appendix A Criterion 19. Also, the incremental increases in offsite doses from the proposed change are within a minimal increase in consequences. Note that 'minimal increase' is defined to be 10 % of the difference between the previously calculated value and the regulatory limit.

The proposed changes do not involve any design change nor are there any changes in the method by which any safety-related plant system performs its safety function. Therefore, there will be no increase in the probability of occurrence of a malfunction of equipment important to safety due to these changes. Nor will there be an increase in the consequences of a malfunction of equipment important to safety. No new types of malfunctions are created.

The acceptance limits affected by the proposed activity are the radiological dose results of the LOCA and FHA analyses contained in the USAR. However, there will be no effect on the manner in which safety limits or limiting safety system setting are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. Therefore, the proposed changes will have no impact on the margin of safety as defined in the basis for any technical specifications.

Safety Evaluation: 59 1999-0163 **Revision:** 0

Fire Safe Shutdown For Fire Area A-8

Activity Description:

In evaluating an event at another nuclear plant for applicability to WCGS, it was determined that the separation requirements of Appendix R section III.G were not met for the Volume Control Tank (VCT) level transmitters. Additionally, the level of detail provided in section 9.5B of the USAR to describe how fire safe shutdown is achieved for a fire in the area of the equipment required enhancement to establish that an equivalent level of protection to Appendix R section III.G is provided.

This change to the USAR clarifies the assumption of a loss of off-site power (LOSP) occurring simultaneously with the fire, and documents the fire safe shutdown analysis for a fire in the fire area affecting the VCT Room, the VCT Valve Room and adjacent corridor. Also changes are made to USAR Section 9.5B to enhance the description of the ability to achieve and maintain safe shutdown based on the above and on relocation of one of the VCT level transmitters from the corridor to the VCT valve room. The level transmitters automatically open their associated RWST discharge valves to provide adequate suction pressure for the charging pumps in the event of low VCT level. Relocation of the transmitter assures survivability of one of the redundant transmitters in the event of a fire. Additionally, Table 9.5B-2 is corrected to properly indicate which VCT level transmitters are required for cold shutdown and separation groups 1 and 4.

The field work associated with the proposed relocation of one of the VCT level transmitters is re-routing of the conduit and cable, installation of instrument mounting supports, installation of instrument capillary supports, installation of a junction box, and breaching and re-sealing of an existing penetration.

50.59 Evaluation:

By analysis, evaluation and the fire area's physical configuration after modification, it has been established that an equivalent level of protection to Appendix R Section III.G is provided and a fire in this area will not prevent safe shutdown of the plant.

The form, fit, function, span, and the setpoint, instrument support, and material are all equivalent to the existing installation. No other safety related equipment will be impacted by the proposed activity. No new equipment malfunctions are created by this change. This change does not create any new failure modes or accidents. The design basis fire for the fire area has assumed the worst case fire damage to important equipment. 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 shut down can be achieved with the changes encompassed by this 50.59 Evaluation.

This change invokes a safety-related design. All materials, installation, design, and the standards are appropriately considered for safety related application. No other equipment, nor the operation of the plant will be adversely impacted by the proposed activity of relocating the transmitter. A design basis fire for this fire area assumes failure of all circuits and equipment in the fire area and a simultaneous LOSP where appropriate 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.

No acceptance limits are identified for the level transmitter. The level transmitters must automatically open their associated RWST discharge valves to provide adequate suction pressure for the charging pumps in the event of low VCT level. Relocation of the transmitter assures survivability of one of the redundant transmitters in the event of a fire and therefore no acceptance limits are affected and the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0164 **Revision:** 0

Revisied the Discussion of WCGS Commitment to Regulatory Guide 1.36, Rev. 0

Activity Description:

This change will revise the discussion of Wolf Creek's commitment to Regulatory Guide (RG) 1.36, Rev. 0, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel" to clarify the use of repeated Production Testing versus repeated Qualification Testing. A reading of the current R.G. 1.36 finds there is a typographical error leading to confusion about what is required in the testing of insulating materials. This confusion comes from the fact that the Regulatory Guide incorrectly refers in Section C.3 back to Section C.2.a versus Section C.2.b. The only portion of the USAR affected is the discussion in Table 6.1-6. Also being added, is a discussion that the Regulatory Guide is considered to be in error with respect to the previous reference.

50.59 Evaluation:

Section C.3 of Reg. Guide 1.36, 1973, states:
"Production Test: A representative sample from each production lot of insulation material to be used adjacent to, or in contact with, austenitic stainless steels used in fluid systems important to safety should be chemically analyzed to determine leachable chloride, fluoride, sodium, and silicate ion concentrations as in C.2.a. above."

Regulatory Guide 1.36, Section C.2.a. refers to two acceptable tests to assure that the insulation formulation does not induce stress corrosion. Neither test speaks to chemically analyzing to determine leachable fluoride, sodium, and silicate ion concentrations. It does speak to determining the chloride ion concentration, pertaining to the distilled/deionized water to be used in the testing.

Regulatory Guide 1.36, Section C.2.b. does refer to the various chemical analyses, as stated below:
"Chemical analysis to determine the ion concentrations of leachable chloride, fluoride, sodium, and silicate: Insulating material that is not demonstrated by the analysis to be within the acceptable region of Figure 1 of this guide should be rejected. This analysis should also be used as a comparison basis for the production test specified in C.3 below."

There will be no affect on the Design Basis Accidents associated with any of these chapters as a result of this clarification to the Regulatory Guide 1.36 commitments. The process of approving the use of insulating materials for use on Austenitic Stainless Steel is not significantly changed. In accordance with the Regulatory Guide a qualifying test and a production test would have been performed on the first batch of insulating materials. Subsequent testing was confused as to whether it required repeated qualifying and production testing or just production testing. With this clarification, subsequent batches of insulating material will receive only the production testing unless the insulating material is reformulated or altered chemically, at which time another qualifying test shall be performed in addition to the production testing.

This change will not create any credible accidents or equipment malfunctions. The insulating materials will continue to receive adequate testing to assure that no unacceptable levels of leachable chloride or fluoride will come into contact with the austenitic sinless steel.

The acceptance limits of the technical specification or other licensing basis documents remain unchanged. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0165 Revision: 0

Revise SYS GT-300 for New Instructions for Securing and Restoring the Unit Vent

Activity Description:

A procedure is being revised to provide instructions for securing and restoring the Unit Vent. The procedure will secure all ventilation that discharges to the Unit Vent. The emergency exhaust will be stopped, but it will still start on a valid manual or automatic signal. All other ventilation discharging to the Unit Vent will be secured to prevent starting. The Radwaste tunnel doors will be open so that the Radwaste building exhaust system will pull a negative pressure on both the Auxiliary Building and the Radwaste Building to prevent having an unmonitored release path. The implementation of this procedure will secure the Auxiliary Building normal ventilation systems and redirect exhaust from the Auxiliary Building through the Radwaste Building exhaust system. This lineup will be different from the normal exhaust flow path as shown in USAR figures 9.4-3-04 and 9.4-5.

50.59 Evaluation:

The Unit Vent is typically secured for maintenance activities and administrative controls will be provided to restore the Emergency Exhaust System upon the receipt of an actuation signal. TS 3.7.13 allows the pressure boundary to be breached, provided a dedicated individual is located at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for auxiliary building or fuel building isolation is indicated.

The safety function of the emergency exhaust system is to pull a negative pressure on the Fuel Building or Auxiliary Building when a Fuel Building Isolation Signal (FBIS) or a Safety Injection Signal (SIS) is received. This reduces off site dose by filtering the exhaust air from the building during various accident scenarios. The implementation of this procedure does not affect the safety functions of these systems since the emergency exhaust will start and maintain a 1/4" negative pressure on the buildings when the pressure boundary openings are restored as allowed by Technical Specifications.

Securing the nonsafety related fans that discharge into the unit vent will not affect the ability of the safety related systems to perform their design function. If a FBIS or SIS were initiated, these nonsafety related fans would be automatically secured and the emergency exhaust system would start.

One of the design functions of the emergency exhaust system is to maintain a negative pressure in the Auxiliary or Fuel Buildings greater than or equal to 1/4" water gauge. In the event an FBIS or SIS is received, administrative controls require the pressure boundary breaches to be immediately closed, thus the ability to maintain the required negative pressure will not be affected as allowed by Technical Specifications.

No credible malfunction of equipment important to safety has been identified nor have any of the single failure assumptions in the USAR was affected.

The emergency exhaust ventilation is covered under TS 3.7.13. A note is placed in the applicability of the LCO to allow breaching the pressure boundary served by the emergency exhaust system. The administrative controls described in the bases will be followed during the performance of this procedure. Since the specification will be followed, no acceptance limits will be affected by this activity and the margin of safety has not been reduced.

Safety Evaluation: 59 1999-0168 **Revision:** 0

Change to Radiological Consequences of a Fuel Handling Accident

Activity Description:

Revise the USAR sections regarding radiological consequences of a fuel handling accident (FHA). For FHA dose calculation, the I-131 gap activity is assumed to be 12% of the core activity according to NUREG/CR-5009 instead of 10% assumed in the current analysis. Analysis was also performed to demonstrate that the increase in doses are mainly from the revised conservative assumption of 12% I-131 gap activity.

50.59 Evaluation:

There are no USAR described plant procedures, structures, systems, or components in the above change that would adversely affect information in the USAR.

Radiological consequences of a fuel handling accident are described in USAR 15.7.4.5. The proposed change incorporate the results of the reanalysis of FHA radiological consequences. Results of the offsite doses remain below the guideline values of NUREG-0800 and 10 CFR 100. Also, the incremental increase offsite doses from the proposed change are less than a minimal increase in consequences. Note that "minimal increase" is defined to be 10% of the difference between the previously calculated value and the regulatory limit.

The proposed USAR change request replaces some parameters used in the current radiological consequences analysis for FHA. This change is not associated with any hardware and will not create any potential failure modes.

This change is not a physical change to the plant and therefore will not directly affect malfunctions of equipment important to safety.

The acceptance limits affected by the proposed activity are the radiological dose results of the FHA analyses contained in the USAR. However, there will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions Therefore, the proposed change will have no impact on the margin of safety as defined in the basis for any technical specifications.

Safety Evaluation: 59 1999-0169 **Revision:** 0

USAR Clarification to Plant Evolution Information

Activity Description:

USAR Sections 5.4.A.3.1 and 2, and Table 5.4A-3, 5.4-9, and 6.3-5 are being revised to clarify inaccurate or unclear material in the Residual Heat Removal System description.

50.59 Evaluation:

A review of the accidents in USAR concluded that no accidents are impacted by the subject USAR changes. The assumptions and conditions assumed prior to, during, and after these accidents are not revised by the proposed changes.

Making the subject USAR changes does not change any plant equipment, setpoints or emergency procedures. Essentially the USAR changes are wording changes and corrections to unclear or inaccurate statements in the USAR. No new or credible accidents are created by the subject changes to the USAR.

The subject USAR changes do not directly or indirectly affect any equipment important to safety. The changes to the USAR text and tables cause no physical changes to the operation of any plant equipment. Additionally the changes and clarifications to the results of the single failure analysis of USAR Tables 5.4-9, 5.4A-3 and 6.3-5 only correct and clarify certain information in these tables and do not alter the failure modes or the potential for failure of any equipment. No credible malfunctions to any equipment have been created by the subject clarification.

Review of the Acceptance Limits contained in the licensing basis documents, ITS, TRM, USAR, SERs has concluded that no limits are affected. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0001 **Revision:** 0

Hydrazine Supply Tank Modification

Activity Description:

Due to personnel safety concerns with the toxic and carcinogenic hazards associated with handling hydrazine, a better method of filling Hydrazine Storage Tanks (TAQ09 and TFE02) will be installed.

The present method of filling tank TFE02 is done by manually filling and emptying 5 gallon containers of hydrazine into a funnel located on top of the inlet line for this tank. The present method of filling tank TAQ09 is to transfer hydrazine from 55 gallon drums via an air driven pump to the inlet line for this tank. These fill methods require full personnel protection to prevent exposure to hydrazine.

A new 1" stainless steel line will be installed from the recirculation line of TAQ01A and TAQ01B and run to the inlet lines of TAQ09 and TFE02. Valve line ups will direct the process fluid to the desired tank. The existing manual fill method of TAQ09 and TFE02 will remain in place and available if needed.

The function of the new hydrazine supply line is to maintain system pressure boundary and direct the process fluid to the desired tank by using valve line-ups.

Hydrazine is presently stored in Room 4326 along with ammonia hydroxide. Design drawings 10466-A-1103 and 10466-A-1008 both designate Room 4326 for ammonia hydroxide storage only. These drawings required that door 43261 be labeled "ammonium hydroxide storage only".

The change made to the above design drawings and USAR locations will replace the phrase "ammonium hydroxide" with "Oxygen control and pH control chemical" for Room 4326 storage designation and re-label door 43261. These changes will allow both ammonium hydroxide and hydrazine to be stored in Room 4326.

USAR Section 2.2.1.2.4 will require revision to add hydrazine to this section and clarify which chemicals are used for oxygen control and pH control.

Section 9.4.4 will require all references made to Room 4326 as the "ammonium hydroxide" storage room be change to " Oxygen control and pH control chemical" storage room.

Table 9.4-10 sheet-7 is revised to change the Room 4326 label from "ammonium hydroxide" storage room to " Oxygen control and pH control chemical" storage room.

The following USAR figures are revised to add new valves and piping configuration.

USAR Figure 10.4-7-01 (P&ID M-12AQ01)

USAR Figure 10.4-7-02 (P&ID M-12AQ02)
USAR Figure 9.5.9-1-04 (P&ID M-12FE01)

50.59 Evaluation:

The Condensate and Feedwater Chemical Addition (AQ) System and the Auxiliary Steam Chemical Addition (FE) System are non-safety related, non-special scope, non-seismic systems located in the Turbine Building. The proposed activity location is in the Turbine Building for installation of the new alternate supply line.

There are no credible accidents applicable in USAR Chapters 3, 5, or 6 for this proposed activity.

USAR Chapter 2 discusses chemical spills in relation to Control Room habitability. A spill of 2000 gallons of hydrazine in the Turbine Building will not affect Control Room habitability as analyzed per calculation AQ-M-001 R/0. A quantity of 1200 gallons of 35% solution of hydrazine may be stored in any combination of 55 gallon drums or totes in Room 4326 or in the bermed area for tanks TAQ01A/B and TAQ02A/B. Hydrazine and ammonium hydroxide may be stored together in Room 4326.

A new Calculation, AQ-M-001 R/0 analyzes a hydrazine spill in the Turbine Building with regards to Control Room habitability per Regulatory Guide 1.78. Calculation AQ-M-001 R/0 assumes 2000 gallons of a 35% solution of hydrazine spilling in the bermed area in front of tanks TAQ01A/B and TAQ02A/B and Room 4326. The results of calculation AQ-M-001 R/0 show that a 2000 gallon spill of hydrazine in the Turbine Building will not affect Control Room habitability as outlined per Regulatory Guide 1.78.

USAR Chapter 9.5 contains the fire protection system. The spray pattern of one sprinkler head in the Turbine Building would be slightly altered due to the new 1" pipe line running near the sprinkler head. An evaluation determined the new line is acceptable in regard to the sprinkler head.

Hydrazine is purchased in a 35% solution. A hydrazine solution of less than 50% is defined as a non-flammable liquid. The new steel piping will not add any combustible loading to the Turbine Building. Therefore the fire hazards analysis in Section 9.5B for zone T-2 is not affected due to this modification. Therefore, no fire protection changes are required due to this modification.

USAR Chapter 15.2.3 analyzes the turbine trip. The routing of this new line does not cross any rotating or moving equipment on the 2000' elevation of the Turbine Building that could cause a turbine trip. The new piping will be socket welded pipe from tank skid to tank skid. Design and installation will be per ANSI B31.1 standards. If a line break were to occur, no turbine trip would occur due to this new pipe interfering with pumps or valves.

The new pipe line is located in the Turbine Building and is routed along existing concrete walls and close to existing Turbine Building structural steel beams. The new pipe routing avoids crossing or being located near high energy lines or rotating and moving equipment in the Turbine Building that could cause a turbine trip. The new valves, pipe and pipe supports will be installed per ANSI B31.1 standards. No credible accidents could occur due to this new pipe line being installed.

If all the hydrazine located in the Turbine Building (total of 2000 gallons per table 1) were to spill at once, Control Room habitability will meet Regulatory Guide 1.78 criteria. Therefore the storage of hydrazine barrels in Room 4326 or hydrazine totes in the bermed area for tanks TAQ01A/B and TAQ02A/b is acceptable. Isolation valves AQV1000 and AQV1002 will be closed during normal plant operation except when filling either TAQ09 or TFE02.

If a combined spill of 30% ammonium hydroxide solution and 35% hydrazine solution occurs, no reaction between these chemical solutions will take place.

The AQ and FE systems do not interface with safety related plant systems. Both the FE and AQ systems are located in the Turbine Building. Any changes made to either the AQ or FE systems will not effect nuclear safety.

The AQ and FE systems do not have any Technical specifications associated to them. Therefore, the changes do not reduce the margin of safety.

Safety Evaluation: 59 2000-0002 **Revision:** 0

Temporary Procedure for Diagnostic Testing of Letdown Temperature Control Valve

Activity Description:

Temporary Procedure (TMP) 99-010 provides instructions for diagnostic testing of Letdown Temperature Control Valve to determine the optimum replacement valve size. This procedure will test the valve under conditions of 75 and 120 gpm letdown while monitoring and recording several system variables. Two temporary pressure gauges will be installed to measure valve differential pressure and a Controlotron 1010 unit will be used to measure CCW flow.

Expected Effects

1. Potential Effects To 7300 Process Control System:
USAR Table 7.1-7, "Conformance to Reg. Guide 1.118, Rev. 2, 6/78, "Periodic Testing of Electric Power and Protection Systems", Item 6a allows use of temporary jumper wires from portable test equipment when the safety system equipment to be tested is provided with facilities specifically designed for connection of this test equipment. The parameters to be tested in this TMP are considered a part of control systems not required for safety as defined in USAR Section 7.7, but do meet this particular guideline of this Regulatory Guide.

2. Potential Effects to Seismic Analysis:
The vent point at CCW system valve EG-V327 will be used to monitor system pressure. Since this is not designated as a design pressure test point, the following engineering basis is provided to justify that the installation will not affect the seismic analysis. The gauge will be installed using flexible tubing at the proper pressure and temperature rating and will be securely attached to building steel, heavy hanger components or seismic scaffold members. Cognizant engineering personnel will approve installation. The design pressure test point at BG-V235 will also have a pressure gauge installed. The procedure requires flexible tubing and engineering approval following installation. The procedure requires both gauges to be isolated when readings are not being taken. These requirements will ensure that no significant system effects would be experienced during a seismic event.

The battery powered Portable Stem Travel Transducer (PSTT) potentiometer will be attached to the body of the tested valve at a convenient point. Due to the small weight of the potentiometer as compared to the total weight of the valve no significant effects would be experienced during a seismic event.

3. Pressure / Temperature Effects
The performance of this TMP will cause letdown pressure and temperature to be cycled around existing control points. The procedure specified upper and lower control limits for letdown pressure are acceptable for normal operation. This lower limit provides sufficient margin to prevent flashing of the letdown coolant before it enters the letdown heat exchanger.

The upper limit, although outside the normal range described in USAR for operation with a steam bubble in the pressurizer, produces no negative effects on plant operation. The pressurizer level control system will adjust charging flow as required to maintain the program level.

The TMP specified upper limit for letdown temperature is also the alarm point where the alarm response would be entered. Operation between the initial starting temperature and the upper limit produces no negative effects on operation. Some increase in RCP seal leakoff should be expected as VCT temperature rises, however approach to the high seal injection temperature is not expected even at maximum VCT temperatures

4. Reactivity Effects
Prior to shifting to 120 gpm letdown, the Chemical & Volume Control System (CVCS) demineralizers will be bypassed to avoid any reactivity effects from boron being absorbed or released at the resin ion exchange sites due to intentional temperature perturbations; therefore, no reactivity effects would be expected by performance of this test.

5. Operator Control
This procedure allows operator action to prevent alarms and return letdown pressure or temperature to the normal control point if directed by the reactor operator or recommended by the test director through the Control Room Supervisor. This enables the plant to be returned to normal control points in the automatic mode if a plant transient or off-normal condition occurs that may or may not be related to this test.

50.59 Evaluation:

The TMP is attaching a pressure gauge at a vent normally used for venting during system filling or draining. The TMP is also attaching vendor test equipment to a seismically qualified valve without previous analysis.

The design basis accidents, effects of natural phenomena, and other hazards were reviewed for potential impact by TMP 99-010. In particular, the following accidents were reviewed:
· CVCS Malfunction That Results in a Decrease in the Boron Concentration in the RCS
· Inadvertent Operation of the ECCS During Power Operation
· CVCS Malfunction That Increases RCS Inventory
· Break in Instrument Line or Other Lines From RCS Pressure Boundary That Penetrate the Containment
· LOCAs Resulting From a Spectrum of Postulated Piping Breaks Within the RCS Pressure Boundary

Performance of this test procedure produces no circumstance which would affect the existing accident analysis or create a new kind of accident. The test is conducted in the reactor control room under complete control of Operations personnel and does not place the plant in an unanalyzed condition. Operations personnel may terminate the test at any time and return temperature or pressure to the normal control point.

The following failure mode analyses were reviewed for potential impact by TMP 99-010:
· RHR and Safety Related Cold Shutdown Operations-Failure Modes and Effects Analysis
· Loss of Any Single Instrument
· Loss of Power to a Protection Separation Group
· Loss of Power to a Protection Separation Group
· CCW System Single Active Failure Analysis
· Failure Mode and Effects Analysis-CVCS Active Components, Normal Plant Operation and Safe Shutdown

Performance of this test procedure produces no circumstance which would affect the existing single active failure or passive failure (structural failures) analyses. No new kinds of malfunctions are created by this test. If either of the controllers adjusted in this test or any other plant equipment experiences an unexpected malfunction, operator action can immediately be taken to terminate the test and enter the appropriate alarm or off normal procedure.

The seismic response of the CVCS and CCW systems is not affected by this procedure due to the controls on temporary gauges and other test equipment. In the event of a safe shutdown earthquake, these systems will respond as required.

No acceptance limits are impacted since no safety analyses are being affected by this test. Normal letdown is not required for accident mitigation or safe shutdown. Operation of letdown as described in this test is within current plant analysis. Therefore, there is no impact on the Margin of Safety.

Safety Evaluation: 59 2000-0003 **Revision:** 0

Reactor Vessel Lift Rig Storage

Activity Description:

The physical storage location during mode 1-4 for the lift rig and the load cell is being changed to Elevation 2068'-8" of containment. They will be located between the A & B containment coolers. The lift rig is to be secured to the structural steel floor beams with wire rope and Crosby clips, precluding any sliding of the lift rig. This is implemented by administrative control processes to ensure the lift rig is stored for operation in accordance with the design change. Handling and storage of the lift rig and load cell during mode 5 & 6 is in accordance with existing administrative controls.

50.59 Evaluation:

The reactor vessel head lift rig is allowed to be stored on 2068'-8" of containment. It will be secured over the structural steel on 2068'-8". Other than relocation, the effect of relocating the lift rig is additional floor loading to the area, which remains within the allowable.

The reactor vessel head lift rig storage location is shown on 2047'-6" in figure 1.2-13 of the USAR. The new, approved location is 2068'-8" in figure 1.2-14.

There are no design basis accidents affected or impacted by this proposed change. Relocation of the equipment to 2068'-8" elevation of containment will not create any credible accident scenarios. The floor loading is adequate for storage in these locations and proximity to safety related equipment is adequate to preclude this equipment from striking safety related equipment. Furthermore, the lift rig is secured to the permanent structure to preclude movement.

This change does not alter or affect equipment important to safety. The relocation of the lift rig is to a secure location that can handle the additional floor loading and proximity to safety related equipment is not a concern since the equipment is secured to the permanent structure.

There are no acceptance limits affected by this modification. Therefore, there is no change to the Margin of Safety.

Safety Evaluation: 59 2000-0004 **Revision:** 0

Steam Generator Tube Rupture Analysis Change

Activity Description:

A Steam Generator Tube Rupture (SGTR) overfill analysis, with revised operator action times, and an associated SGTR radiological consequence analysis, has been performed using updated source terms transmitted by a letter from Westinghouse.

As a result of the revised SGTR overfill analysis and the associated SGTR radiological consequence analysis, the following general proposed USAR updates, incorporating the analysis results and assumptions, with the associated sections, tables, and figures are proposed:

1) Revise the description of the method of analysis assumptions employed in the SGTR overfill analysis with revised operator action times
2) Update the time sequence of events from the time of tube rupture for the SGTR with failed-open auxiliary feedwater control valve and safety valve.
3) Revise the tabular presentation of the parameters used in evaluating the SGTR with forced overfill radiological consequences.
4) Update the tabular presentation of isotopic activities released to the environment, as calculated in the updated SGTR radiological consequence analysis.
5) Revise the tabular presentation of thyroid and whole body dose information, as calculated in the updated SGTR radiological consequence analysis.
6) Update the figures to present the revised transient parameter output information from the SGTR overfill analysis with revised operator action.

50.59 Evaluation:

The proposed USAR changes make the USAR consistent with the revised SGTR transient analysis, with forced overfill and stuck-open safety valve, and the associated SGTR 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 the SGTR transient analysis and the SGTR radiological consequence analysis 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.

There are no physical modifications to the systems, components, and equipment or changes in methods of operation. Therefore, no credible accidents that could be created are identified.

The changes will not impact the overall system performance in a manner that could cause an accident previously evaluated to shift to a higher frequency category. As such, there will be no increase in the probability of occurrence of an accident previously evaluated in the USAR

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 identified.

The calculated accident radiological personnel dose values in both analyzed cases, pertaining to the pre-accident and concurrent accident radioiodine spiking, remain far below the 10 CFR 100 guideline limits. Note that the proposed change results in a minimal increase in dose consequences, as the new value is greater than the previously calculated value. However, the incremental increase in dose consequences does not exceed ten percent of the difference between the previously calculated value and the regulatory limit.

The proposed changes do not affect the I-131 dose equivalent limits for the specific activities of the primary and secondary. The proposed changes do not affect the manner regarding how safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected, and the margin of safety is not affected by this change.

Safety Evaluation: 59 2000-0006 Revision: 0

Temporary Modification Restoring Level Indication to the Primary Spent Resin Storage Tank

Activity Description:

This temporary modification provides a temporary connection to restore the level indication for the Primary Spent Resin Storage Tank (PSRST). The temporary line is routed from valve at the suction of the PSRST Sluice Pump to the valve which is the lower calibration and drain connection for the level transmitter on the PSRST. An additional valve will be installed to permit the draining of the temporary line during the restoration of the temporary modification.

50.59 Evaluation

The temporary line will bypass the blockage in the level sensing line and restore the level indication to the original design function.

Any spill or leakage from the temporary line that connect the suction of the PSRST Sluice Pump will be contained within the existing design of the radwaste drainage system. Any possible spills inside the Radwaste Building are bounded by the analysis in USAR. 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.

Since this temporary line change is associated with the Secondary Liquid Radwaste System, the USAR 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. Therefore, a potential gaseous release is bound by previously evaluated tank ruptures analyzed in the USAR.

The proposed temporary line 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 temporary line 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 temporary modification. Therefore, there is no change in the margin of safety.

Safety Evaluation: 59 2000-0008 **Revision:** 0

Demineralizer Water Makeup System As-Built Configuration

Activity Description:

Updated Safety Analysis Report (USAR) figure 9.2-5-02 for the Demineralizer Water Makeup System (DWMS) is being revised to reflect the current condition of the portion of the Shop Service Air system in the Chlorine Building. This change will not affect any analyses or procedures.

The affected components are two valves and two air lines, all of which are non-safety related. One valve and one air line are being deleted from the figure. The second valve was shown on the incorrect side of a tee connection. The second air line is being revised to show its actual diameter of 3/8 inches instead of 3/4 inches.

Safety Evaluation:

The DWMS is used to supply demineralized water for use in the plant during normal operations and has no function important to safety. There is no equipment malfunctions important to safety identified that will be impacted by this design change, therefore the probability of an occurrence is not affected by this change.

There are no credible accidents identified that will be impacted by this change, therefore the probability of an occurrence is not affected by this change.

There are no radiological consequences identified by this change, therefore the probability of an occurrence is not affected by this change.

There are no acceptance limits impacted by this design change, therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0009 **Revision:** 0

Temporary Modification to the Service Water System

Activity Description:

This temporary modification installs temporary pump(s), temporary tank, piping, valves, and supports to pump the water leaking through holes drilled in the walls of the southwest corner of the condenser pit. Water will be pumped to the service water system through valve EAV0143 and line EA-123-HBD-2". Valve EAV0143 is upstream from EAHV006 and is normally closed and the pipe nipple capped. Valve EAHV006 is a Service Water (EA) return isolation valve to the Circulating Water System (DA).

50.59 Evaluation:

The proposed change is to the valve line up of the EA system as depicted on Fig 9.2-1-02 of the USAR. The valve will be open and connected to a temporary pipe and pump with check valves to prevent reverse flow.

The conditions of increase and decrease in heat removal by the secondary system were reviewed for potential impact. The loss of normal feedwater flow was evaluated further with consideration of a loss of normal feedwater from pump failures.

The service water supply to the closed cooling water is ahead of the proposed point of connection. The loss of the flow out of the service water system (380 GPM) would have an insignificant effect on the service water system to supply the required flow. Therefore, this temporary modification will not have any impact on the accidents and conditions identified above as discussed in chapter 15.2.7 of the USAR.

No new types of accidents could be created by this modification. The failure of the piping with the corresponding leakage into the condenser pit, if left unattended, is the same result as failure of a circulating water system expansion joint discussed in USAR Appendix 3B Hazards Analysis.

The temporary modification will be installed in the condenser pit in the Turbine Building. There is no equipment important to safety in this area of the turbine building. The effect on the Service Water System or ESW System would be negligible. The Service Water pumps are capable of delivering the additional flow and maintaining sufficient system pressure for both safety related and non-safety related equipment.

There are no acceptance limits contained in the technical specifications that this temporary modification could affect.

The loss of the small quantity of service water to beneath the turbine building floor or on the pit floor will not increase the probability of occurrence in the decrease in heat removal by the secondary system caused by the loss of the pump.

The severity or details of the accident will not change therefore, the radiological consequences of the accident will not change.

Flow to equipment important to safety will be maintained and the remote location away from the equipment precludes increasing the malfunction of equipment important to safety.

The radiological consequences remain the same since there is not any change to the accidents nor to the equipment as evaluated in the USAR.

The type of failure of the temporary modification will not be any different than the circulating water pipe break or the consequences of losing a Feedwater pump due to some other type of failure.

The proposed modification can not create any different type of malfunction of equipment important to safety due to the negligible effects on the ESW system and the remote location from safety related equipment.

No margins of safety as defined in the basis for any technical specifications will be reduced since the required flow and pressures will be maintained.

Safety Evaluation: 59 2000-0010 **Revision:** 0

Clairification of Chemical & Volume Control System Description

Activity Description:

Changes are being made to the Chemical & Volume Control System (CVCS) System Description M-10BG (Q) and USAR text to ensure this material will not conflict with existing approved plant procedures. In addition, clarifications are being made to USAR material that is unclear or inaccurate.

50.59 Evaluation:

A review of the accidents in USAR Chapters concluded that no accidents are impacted by the subject USAR changes. The assumptions and conditions assumed prior to, during, and after these accidents are not changed by the proposed changes.

Making the subject USAR changes does not change any plant equipment, setpoints or emergency procedures. Essentially the USAR changes are wording changes and corrections to unclear or inaccurate statements in the USAR. No new or credible accidents are created by the subject changes to the USAR.

The subject USAR changes do not directly or indirectly affect any equipment important to safety. The changes to the USAR text and CVCS System Description cause no physical changes or procedural changes to the operation of any plant equipment. Moving of the text from USAR section 9.3.4.2.3.3 to section 9.1.4.2.3.1 will better reflect the actual process that has been in use. No credible malfunctions to any equipment have been created by the subject changes.

Acceptance limits contained in the licensing basis documents, Technical Specifications, USAR, and Safety Evaluation Reports were reviewed. This review has concluded that no limits are affected. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0012 **Revision:** 0

Changes to the Fire Protection Program

Activity Description:

A USAR inconsistency regarding the minimum 3 hour rated wall which separates the Turbine Building from the Auxiliary Building Area 5 was identified. The Fire Areas in question are T-2 and A-23 respectively. In the Fire Hazards Analysis, USAR Section 9.5B, the description of Fire Area A-23 states that "Fire Area A-23 is separated from all adjoining areas and buildings by a minimum 3 hour rated fire barrier". Contrary to this statement, at approximately the 2026' elevation there is a personnel emergency escape hatch installed which separates T-2 from A-23 which does not meet the requirements of a 3 hour rated fire barrier assembly. A USAR change is being initiated to revise the description of the A-23 and T-2 fire areas in section 9.5B to include a discussion of the escape hatch. Additionally this evaluation will document that the level of fire protection provided by the hatch is commensurate with the fire hazards in the area of the hatch.

The description of fire areas T-2 and A-15 are also being revised to identify the presence of an installed blow out panel. In the Fire Hazards Analysis, USAR Section 9.5B, the description of Fire Area A-15 states that "Fire Area A-15 is separated from all adjoining areas and buildings by a 3 hour rated fire barrier". Contrary to this statement, at approximately the 2018' elevation there is a blow out panel installed which separates T-2 from A-15 which does not meet the requirements of a 3 hour rated fire barrier assembly. The blow out panel, P135W2346, is installed between the Turbine Driven Auxiliary Feedwater pump room and the Turbine Building.

50.59 Evaluation:

Based on the results of an engineering analysis, the blow out panel in the current configuration does not affect the ability of the plant to achieve and maintain safe shutdown. The blow out panel located at the 2018' elevation is a non-rated feature which is commensurate with the postulated fire hazards in the adjoining areas.

USAR Section 9.5B provides a summary of the fire hazards analyses which were performed to ensure that the plant could be maintained in a safe condition following a major fire in any safety-related area of the plant. Based on the engineering evaluation, the presence of the non-rated equipment hatch and blow out panel does not effect the ability to achieve and maintain safe shut down in the event of a fire in either fire area A-23, A-15 or T-2. This change does not affect any of the inputs, assumptions, or conclusions used to evaluate the impact of fire for the three fire areas. This change enhances the description of fire areas T-2, A-23 and A-15 to include a discussion of the equipment hatch and the blow out panel. Therefore, this change does not affect any design basis accidents nor does it create any new failure modes or accidents.

Original design drawings indicate that they were identified as a non rated feature but USAR

9.5B did not accurately reflect the design drawings. Therefore, this change does not create any new failure modes or malfunctions. The escape hatch and blow out panel will act as fire barriers for the postulated fires in the three affected fire areas.

No acceptance limits in the technical specifications or licensing basis documents apply. The engineering evaluation establishes that the fire barriers discussed in the USAR are maintained. Since no acceptance limits were identified that could be affected the margin of safety is not affected by this change.

Safety Evaluation: 59 2000-0013 **Revision:** 0

Containment Leakrate Testing Changes

Activity Description:

The change updates containment integrated leakage rate testing (ILRT) design documents to the requirements of the current containment leakage rate testing program. The general testing methods and leakage rate acceptance criteria are not changed. The containment leakage rate testing program as described in Technical Specification section 5.5.16 is not affected.

This change voids specifications M-665 for pressurization equipment and J-801 for instrumentation. All equipment described in these specifications is test equipment, not needed to support plant operation. The equipment is in storage except when used for an ILRT. The containment leakage rate testing program provides specifications for this test equipment which will be leased when needed for future ILRTs.

MS-01 will be revised to reflect the change in component status. The ILRT system description will be revised. Calculation GP-M-001 will be revised, and two related calculations, GP-2 and GP-3, are no longer applicable and are made void.

USAR figure 6.2.6-1 currently shows components as described by the specifications J-801 and M-665. The revision to this figure will reflect the points of connection for ILRT equipment. The revision will show the normal configuration of the ILRT system. The piping and electrical cabling used to support future ILRTs and permanently installed in the plant is not changed.

USAR section 6.2.2.2.2.3 describes operation of the containment air coolers during an ILRT. The wording is changed to clearly show that operating the containment air coolers is a permitted action, not a required action.

USAR section 6.2.6 describes containment leakage testing, section 6.2.6.1 describes containment integrated leakage rate testing. The current revision discusses the test in terms of plant preoperational startup testing. Regulatory documents and testing methods from 1972 are the reference material for the discussion. With the significant regulatory changes which occurred in 1995, and subsequent adoption of 10 CFR 50, Appendix J, Option B by WCGS, the current ILRT description in the USAR is not accurate. This change rewrites this section describing the test in current terms, referencing the approved containment leakage rate testing program (procedure AP 29E-001) and associated regulatory documents.

50.59 Evaluation:

The Design Basis Accident (DBA) most closely related to containment integrated leakage rate testing would be fuel handling accidents as discussed in section 15.7.4. The proposed activity will have no impact on these accidents because containment isolation barriers are not changed and fuel handling is a totally separate activity from containment integrated leakage rate testing.

The proposed activity does not change or affect any components which are involved in any way with initiators of credible accidents. Performing an ILRT proves that the containment is leak tight within the limits of stated acceptance criteria. Performing periodic ILRTs is required. The proposed activity does not change the testing requirements.

Neither equipment important to safety nor their credible malfunctions are affected by the proposed activity. The design equipment of the two specifications being voided is non-safety related equipment. It is test equipment, normally in storage except when used for an ILRT. Revising the related design documents does not affect any plant equipment. There are no component modifications associated with this change.

Acceptance limits for containment leakage rate testing are discussed in the Technical Specification bases B 3.6.1. The containment leakage rate testing program is outlined in Section 5.5.16. The acceptance limits are not affected by the proposed activity. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0014 **Revision:** 0

USAR Changes to the Residual Heat Removal System Design Parameters

Activity Description:

The proposed USAR changes revise the following:

USAR Chapter 5, "Reactor Coolant System and Connected Systems," Table 5.4-7, "Design Bases for Residual Heat Removal System Operation," to present the decay heat generation at 20 hours after reactor shutdown as 75.2E6 British Thermal Units/hour (BTU/hr), in accordance with Westinghouse Calculation Note SAE/FSE-C-SAP-0171 (dated 2/24/98), consistent with use of the ANSI/ANS-5.1-1979, "Decay Heat Standard" as described in USAR Section 5.4.7.2.1 "Residual Heat Removal, Design Description," and

USAR Table 5.4-7 title to explicitly indicate that the table presents the design parameters for Residual Heat Removal (RHR) operation, consistent with the table reference in USAR Section 5.4.7.2.1,.

The current decay heat generation value at 20 hours after reactor shutdown of 78.2E6 BTU/hr (Table 5.4-7) is based on the 1971 Decay Heat standard while the revised value of 75.2E6 BTU/hr is based on the ANSI/ANS-5.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors."

50.59 Evaluation:

The proposed activity does not change any administrative controls that 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 WCGS.

The proposed USAR changes make the USAR consistent with the use of decay heat generation values based on ANSI/ANS-5.1-1979 and make the associated table title consistent with the reference to the table. No procedures, activities, administrative controls, sequences of plant operations, plant structures, systems, components or equipment, or requirements are impacted by these changes and thus the proposed activity would not invalidate USAR information or requirements. Since the proposed changes ensure consistency between the USAR description presented in Section 5.4.7.2.1 and information presented in USAR Table 5.4-7, the change would not adversely affect the mitigative capability of any SSCs, nor affect the ability of any SSC to prevent an accident.

The proposed changes ensure consistency between the USAR description presented in Section 5.4.7.2.1 and the information presented in USAR Table 5.4-7 and would not affect the design bases for RHR operation. Note: At the time of RHR operation, the limiting portion of the design basis accidents have been completed.

There are no physical modifications to the systems, components, and equipment or changes in methods of operation. Therefore, no credible accidents that could be created are identified.

The proposed changes to USAR Chapter 5 Table 5.4-7 are made to achieve consistency between the USAR description and the table, based on use of the ANSI/ANS-5.1-1979 Decay Heat Standard. These changes will not impact the overall system performance in a manner that could cause an accident previously evaluated to shift to a higher frequency category. As such, there will be no increase in the probability of occurrence of an accident previously evaluated in the USAR. The change will not increase the radiological consequences of an accident previously evaluated in the USAR.

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 identified.

The proposed changes do not affect the manner regarding how safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected. Since no acceptance limits were identified that could be affected, the margin of safety is not affected by this change.

Safety Evaluation: 59 2000-0015 **Revision:** 0

Hot Water Heater Replacement

Activity Description:

The proposed change replaces the hot water heater (1WD02T) in the Chrysler Building water treatment area with a new smaller unit. The change to the non safety related Domestic Potable Water (WD) System does not affect the design basis function of the system or any hazards and accident analysis in the USAR.

The WD System is non-safety related. The changing of the existing hot water heater, to a smaller unit, would not affect the design basis, functions or failure modes of the WD System or affect any safety-related equipment important to the safe and reliable operation of Wolf Creek Generating Station (WCGS).

50.59 Evaluation:

No other USAR descriptions or conclusions will be affected as a result of this change.

There are no design basis accidents discussed or referenced in the USAR that are impacted by the proposed equipment modification. The WD System serves no safety function and has no safety design basis.

Since no design basis functions are 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 no acceptance limits are included in the bases of the Technical Specifications or licensing basis documents, no acceptance limits are identified that could be affected and the margin of safety is not affected.

Safety Evaluation: 59 2000-0016 **Revision:** 0

USAR Spent Fuel Pool Leak Detection Description Clarification

Activity Description:

Currently the USAR states that the vertical chases and horizontal chases of the spent fuel pool structures are leak tight and isolatable by zone. This is not an accurate interpretation of this paragraph as only the vertical chases are described as seal welded and hence can be considered leak tight and isolatable. The horizontal channels/chases are not described as seal welded and thus should not be considered as leak tight. The horizontal chase zones can be isolated by zone but this isolation is limited and not leak tight. In the construction of the pools, the top of the horizontal steel channels including the boundaries between horizontal zones are not welded to the bottom of the floor liner.

The USAR will be revised to state that the horizontal channels are not seal welded at the top of the channel to the bottom of the pool liner and only the vertical chases have leak tight capability.

50.59 Evaluation:

A review of the accidents in the USAR concluded that no accidents are impacted by the subject USAR clarification. The assumptions and conditions assumed prior to, during, and after these accidents are not changed by the proposed clarification changes.

Making the subject USAR clarification does not change any plant equipment, setpoints or emergency procedures. Essentially, the USAR change involves wording changes that clarify statements in the USAR. No new or credible accidents are created by the subject clarification to the USAR.

The USAR clarification does not directly or indirectly affect any equipment important to safety. The clarification to the USAR text causes no physical changes to the pool structures, liners or associated equipment. The single failure analysis of the Fuel Pool Cooling System and the Floor and Equipment Drainage System equipment listed in USAR Tables 9.1-6 and 9.3-7 are not affected by the clarification of the USAR. No credible malfunctions to any equipment has been created by the subject clarification.

Acceptance limits contained in the licensing basis documents were reviewed. This review has concluded that no limits are affected. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0017 **Revision:** 0

USAR Change to Component Cooling Water Description

Activity Description:

A sentence in section 9.2.2 of the USAR document is being revised. The sentence when read implies CCW flow is required to flow through the heat exchangers at all times when spent fuel is in the pool. This read is too rigid and does not align itself with other sections of the USAR. The sentence currently reads as, "Cooling water flow through the fuel pool cooling heat exchangers is required during periods when spent fuel is stored in the pool." The sentence will be revised to read as, "Cooling water flow through the fuel pool cooling heat exchangers normally exists as needed during periods when spent fuel is stored in the pool." This deletion and agreed upon wording of the new proposed change will make this sentence in USAR section 9.2.2 agree with other sections of the USAR.

50.59 Evaluation:

A review of the accidents in the USAR concluded that no accidents are impacted by the USAR change. The assumptions and conditions assumed prior to, during, and after these accidents are not changed by the proposed wording changes.

This sentence does not change any plant equipment, setpoints or emergency procedures. This change involves wording changes that make the USAR document agree with itself. No new or credible accidents are created by the subject sentence change to the USAR.

The USAR sentence change does not directly or indirectly affect any equipment important to safety. The change causes no physical changes to the operation of any plant equipment. The single failure analysis of the Fuel Pool Cooling System and the CCW System equipment listed in USAR Tables 9.1-6 and 9.2-13 are not affected. No credible malfunctions to any equipment has been created by the subject change.

The acceptance limits contained in the licensing basis documents, ITS, USAR, and SERs were reviewed. This review has concluded that no limits are affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0019 **Revision:** 0

Title Change in the Maintenance Organization

Activity Description:

This USAR revision addresses a title change as represented in USAR Figure 13.1-2b. "Superintendent Of Maintenance Planning" is changing to "Superintendent Maintenance Shift Crews". All required job functions and qualifications under the Maintenance department continue to be maintained. The only portion of the USAR that is affected is the organization of Maintenance as represented in USAR Figure 13.1-2b.

50.59 Evaluation:

There are no design bases accidents that take credit for the position title change. Also, a change in title has no impact on design bases accidents.

A title change will not create any new or different accident because all functions continue to be maintained. Qualification requirements will remain the same.

There are no credible malfunctions of equipment important to safety that may be directly or indirectly affected by this change because the change is administrative in nature and all functions performed by the Maintenance department continue to be performed. Qualification requirements will not be affected.

This position title does not constitute an acceptance limit and was not specifically accepted in the granting of the Operating License. Therefore, this change does not impact the acceptance limits. Therefore the margin of safety is unaffected.

Safety Evaluation: 59 2000-0020 **Revision:** 0

USAR Changes to Containment Penetration Descriptions

Activity Description:

Two similar parenthetical remarks in USAR section 7.3.8.1.1 that describes the Nuclear Steam Supply System (NSSS) Engineered Safety Feature Actuation System (ESFAS) are being deleted. One parenthetical remark identifies the safety injection and containment spray containment penetration lines as the only lines not isolated when a Containment Isolation Signal Phase A and B (CIS-A & B) are taken together. The other parenthetical remark lists the lines for this same condition as the safety injection and residual heat removal lines.

50.59 Evaluation:

These two parenthetical remark listings are not consistent nor completely correct. USAR section 6.2.4 describes the Containment Isolation System and all lines penetrating containment are identified in USAR Figure 6.2.4-1. This figure also identifies the actuation signals that isolate the penetrations that provide containment isolation. A review of penetrations identified in this figure has concluded that the ESW for the containment air coolers, Reactor Coolant Pump (RCP) seal water supply, safety injection, RHR and spray supply penetration lines are the only lines not isolated by a CIS-A & B when taken together. A review of the USAR figures for these systems also substantiates this conclusion because the associated containment penetration valves on these lines do not receive a CIS-A or B. The CVCS charging line is a nonessential penetration and is isolated by a SIS signal, a SIS generates a CIS-A. The containment spray suction lines from the recirculation sump receive a CIS-A to verify closure. A Containment Spray Actuation Signal (CSAS) causes a CIS-B. A CSAS/CIS-B along with a RWST Lo Lo-2 alarm informs the operator to manually open the containment spray recirculation sump suction valves for accident mitigation.

The above conclusion is further collaborated within USAR section 18.2.11 that discusses containment isolation dependability. USAR Table 18.2-2 lists all the containment penetrations. In this containment penetration list, the ESW for the containment air coolers, RCP seal water supply, safety injection including boron Injection tank (BIT) injection, RHR and containment spray recirculation sump suction and supply penetrations are designated as essential; the CVCS charging penetration is not. The essential designation means that the containment penetration isolation valve(s) on the respective lines are required to be open for safe shutdown or mitigation of the consequences of an accident. The CVCS charging line is not essential for safe shutdown or accident mitigation. The component cooling water lines to and from containment are listed as essential but are isolated by a CIS-B when containment high pressure 3 setpoint is reached.

A review of the accidents in the USAR concluded that no accidents are impacted by the subject change. The assumptions and conditions assumed prior to, during, and after these accidents have not been changed.

Deletion of inaccurate information in the USAR to make it consistent and true will only help the reader of this document to understand the plant and does not change any plant equipment, setpoints or emergency procedures. No new or credible accidents are created by deleting the subject words in the USAR especially when the deleted words are misleading and inaccurate and are of minor importance in their existing context.

The subject USAR change does not directly or indirectly affect any equipment important to safety. There are no physical changes to the operation of any plant equipment. The single failure analysis of the ECCS, Containment Spray and ESW equipment listed in USAR Tables 6.3-5 & 6, 6.5-4 and 9.2-6 are not affected. No credible malfunctions to any equipment have been created by the subject change.

The acceptance limits contained in the licensing basis documents were reviewed. This review has concluded that no limits are affected. Therefore, the margin of safety has not been affected.

Safety Evaluation: 59 2000-0022 **Revision:** 0

Safety Injection Pump Discharge Relief Valve Setpoint

Activity Description:

The Safety Injection (SI) pump discharge relief valve (EM8851, EM8853A/B) setpoints are being increased from the current 1750 psig to 1825. USAR Table 6.3-2 will be revised to reflect the increase.

50.59 Evaluation:

The relief valves are installed to protect lines which have a lower design pressure than the RCS. SI pump discharge relief valves, EM8851, and EM8853A/B, protect the piping associated with the SI pumps' discharge line from overpressurization due to back leakage through the check valves from the RCS. Leakage past the check valves pressurizes the safety injection pump discharge header, resulting in control room indication of increasing pressure and eventually lifting of relief valve EM8851 or EM8853A/B. Relief valve lifting is detected by increasing levels of boron recycle holdup tanks which indicate and alarm in the radwaste control room and provide a general system alarm in the main control room. The relief valves are currently set to relieve at 1750 psig, which is above the shutoff head of the pumps, and must accommodate only minimal flow (20 gpm).

These types of unintended lifting have been seen across the industry, when the relief valve setpoint is close to that of the system pressure. When the SI pump is started, a pressure surge is created, and that causes the setpoint to be exceeded and the valve lifts. Increasing the setpoint will prevent this undesired opening of the valves.

50.59 Evaluation:

Valve manufacturer, Crosby, has evaluated the effect of increasing the pressure on EM8851 and EM8853A/B. They indicate that the increased pressure is well within the valve's design capability, as allowed by the ASME Code. Their Seismic/Stress Report for the valves indicates the calculations were done using a pressure of 2735 psig. Thus it is concluded that nothing further need be done to demonstrate the Crosby valve's qualification with a set pressure of 1825 psig.

Valve manufacturer, Westinghouse, has evaluated the effect on their valves that will see the increased pressure (EM8821A/B, EM8921A/B, and EM8922A/B). They indicate that the increased pressure is well within the valve's design capability, as allowed by the ASME Code. The valves were built to ASME Section III, 900 pound, Class 1 requirements, which allows up to 1825 psig at 300 °F. Thus it is concluded that nothing further need be done to demonstrate the Westinghouse valve's qualification with a system pressure of 1825 psig.

WCNOC Design Engineering has evaluated the effect of increasing the system pressure to 1825 psig on the system piping. It was shown that all stresses would remain within Code allowable values.

WCNOC Nuclear Engineering evaluated the proposed increase for any effect on the accident analyses, and concluded that there would be no detrimental effect.

The proposed setpoint revisions provide added assurance that the valves will be capable of performing their designed safety function, as it will eliminate inappropriate cycling (and therefore premature failure). The increased setpoints do not result in pressure increase beyond the design capability of the affected valves and piping. Since the valves are still capable of performing their design basis function, which is to protect the system from overpressurization, 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 setpoints, result in valves which are functionally identical to the current configuration. Therefore, there is no potential for introducing new failure modes or increasing the likelihood of existing credible failure modes.

There is no acceptance limit affected by the change. Therefore, the margin of safety has not been decreased.

Safety Evaluation: 59 2000-0023 **Revision:** 0

Organization Change to Manager Licensing & Corrective Action

Activity Description:

The organization reporting to the Vice President Operation Support is being changed due to the resignation of the Manager Licensing and Corrective Action (L&CA). The new organization structure will divide the current L&CA organization into two with a Manager Regulatory Affairs, and a Manager Organization Performance in place of the single manager. In addition, Document Control and Records Management will now report to the Manager Administrative Services. The title Manager Document Services will no longer exist. Reporting to the Manager Regulatory Affairs will be the Licensing/Compliance group and Nuclear Safety Engineering group. Reporting to the Manager Organization Performance will be the Human Performance group and Corrective Action group. This change in organization is not currently described in the USAR. The functions will continue to be met and qualifications have not been changed. There will be no negative effects of this change.

The functioning of the organization is described in chapter 13 and 17, and 18 of the USAR. In addition, the resume of Manager L&CA was located in chapter 13. This description will change with the new organization. Chapter 17 discusses trending functions by organization title. This title will no longer be correct (although the function will continue).

50.59 Evaluation:

This is an organization change. All functions will continue and qualifications remain the same. Therefore, there will be no impact on the plant. Since there is no impact on the plant and no accident analysis takes credit for organization, there are no design basis accidents created or changed. In addition, no equipment will be affected.

Organization change was not considered an acceptance limit during licensing. Organization is understood to change and adjust throughout the life of a plant. Functions important to the safe operation of the plant continue and qualifications of personnel have not been diminished. Therefore, the original intent of the organization has not been altered and the margin of safety has not been affected.

Safety Evaluation: 59 2000-0026 **Revision:** 0

Revised Motor Operated Valve Evaluation Criteria

Activity Description:

As a result of WCNOC's response to NRC Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves," the NRC requested further information through a Request for Additional Information (RAI). In response to the RAI and discussions with the NRC, WCNOC now utilizes more conservative criteria to evaluate MOV gate valves for pressure locking than that originally used.

As a result of Wolf Creek changing to more conservative criteria than originally used to evaluate Motor Operated Valves (MOVs) for susceptibility to pressure locking, the number of MOVs considered susceptible to pressure locking increased. To address the additional MOVs considered susceptible to pressure locking, a modification was developed to prevent pressurization of water inside the valve bonnet. The subject modification provides a bypass/pathway for the water inside the valve bonnet to escape back into the system piping. This prevents entrapment of water inside the valve bonnet and limits the bonnet pressure to that of the system. As a result of this modification the bonnet pressure will not exceed the opening capability of the valve actuator, and therefore the valve's design basis functions will be met.

This activity did not change any administrative controls which would reduce the effectiveness of existing programs, reduce the qualification of WCNOC personnel, nor did it affect any licensing basis analyses and systems, structures, and components. The proposed activity did not change the performance of activities that are important to the safe and reliable operation of WCGS.

50.59 Evaluation:

This change required revision to USAR figures 5.4-7, 6.2.2-1, and 6.3-1 to ensure the plant configuration is properly depicted. No administrative controls, sequences of plant operations, plant structures, or requirements were impacted by the change. The licensing basis analyses were not affected and the change would not adversely affect the mitigative capability of any SSCs, nor affect the ability of any SSC to prevent an accident. Additionally, no tests or experiments were identified in the USAR for the affected valves.

This change is not anticipated to have a detrimental impact on the integrity of any plant structure, system or component. There are no changes in function or methods of operation, and no changes to the accident analyses, including the radiological consequences. The bypass/pathway provides added assurance that the valves will be capable of performing their designed safety function, as it eliminates the potential for pressure locking. The change does not alter the operation of any plant equipment, otherwise increase their failure probability, or impact the overall system performance in a manner that could cause an accident previously evaluated to shift to a higher frequency category. Therefore, this change does not affect any design basis accidents (DBAs) discussed or referenced in the USAR. Likewise, since no DBAs are affected, there is no increase in the probability of occurrence or the radiological consequences of an accident previously evaluated in the USAR.

This change will not impact the initial conditions of a design basis accident (DBA) or transient 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 noted above, this change helps to assure the affected valves remain capable of performing their design basis function and there are no new failure modes introduced. Therefore, this change does not affect a DBA or create the possibility of an accident of a different type than any previously evaluated in the USAR.

These changes do not alter the operation of any plant equipment or otherwise increase their failure probability, no installed equipment is being operated in a new or different manner, nor are there any changes in the methods by which any safety-related plant system performs its safety function. These changes do not impose or eliminate any requirements, and adequate control of information will be maintained. No changes were made 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 the plant equipment operates or responds to an actuation signal. The probability that equipment failures resulting in an unanalyzed event will occur is reduced as these changes will eliminate the potential for pressure locking, assuring the valves remain functionally identical to the current configuration. Although leakage past the valve seat is not expected to occur, its effect on downstream systems and containment boundary would be negligible. Therefore, no new failure modes were introduced and no credible malfunctions of equipment important to safety that may be directly or indirectly affected by this change were identified.

Likewise, since no malfunction of equipment important to safety was identified, there is no increase in the probability of occurrence or the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

Also, since no malfunctions of equipment important to safety were identified, no different type of malfunctions of equipment could be created. Therefore, this change does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

There are no design changes or equipment performance parameter changes associated with these changes. No setpoints are adversely affected and no operational limits are impacted by these changes. As the accident analyses were not revised, there is no impact on the analysis results, acceptance limits, or margin to the existing limits. Since no acceptance limits were affected, the margin of safety as defined in the basis for any technical specification would not be reduced.

Safety Evaluation: 59 2000-0027 **Revision:** 0

Change to Fire Protection Reviews

Activity Description:

USAR Change Request 91-049 inserted wording in USAR Table 9.5A, section A.1, to provide qualification criteria for individuals who assist in the review of ongoing revisions to the fire protection system. However, the wording used was incomplete and implies that the individuals may perform the reviews without direction from someone who meets the originally required qualifications. The added wording is being deleted from the USAR. This will remove any misconception about the required qualifications for the individuals performing reviews of ongoing revisions to the fire protection system, and will essentially increase the qualifications for those individuals.

50.59 Evaluation:

The change will require fire protection reviews to be performed by individuals meeting the qualifications as originally stated in the USAR (member of the Society of Fire Protection Engineers (SFPE) or meets qualifications for membership in the SFPE). The change removes the implied option for these reviews to be performed by individuals not meeting the SFPE eligibility requirements.

There are no procedures, structures, systems, or components outlined, summarized or described in the USAR which, when implementing this change, would make information in the USAR no longer true or accurate. The definition of the Fire Protection Engineer exists in USAR Table 9/5A, section A.1, and is not being revised. The section of USAR being revised is not a definition of the Fire Protection Engineer, but a description of the qualifications for an individual who assists the Fire Protection Engineer in the reviews. This subtle difference caused confusion, so it is being removed from the USAR. The new wording will be that additional assistance has been provided by Licensed Fire Protection and Graduate Fire Protection Engineers. This essentially increases the qualification requirements for individuals assisting the Fire Protection Engineer.

There are no Design Basis Accidents that are impacted by this change.

There are no credible accidents that this administrative change to the USAR could create. The qualification requirements of the individual assisting in fire protection reviews have increased. Therefore, there is no new type of credible accident that this change can cause.

There is no equipment important to safety affected by this administrative change to the USAR. The qualification requirements of the individual assisting in the reviews have increased. There is no new credible malfunction of equipment important to safety that can be affected by removing this statement about the reviewer's qualification.

There are no acceptance limits that discuss the qualifications for an individual who assists the Fire Protection Engineer. Therefore, there are no acceptance limits affected and the margin of safety is unchanged.

Safety Evaluation: 59 2000-0028 **Revision:** 0

Containment Purge System Test Valves

Activity Description:

A new 1" test valve connection pipe with pipe cap will be installed in each of the two containment penetrations for the Containment Purge System. The new containment purge test valves will be designed and fabricated per ASME section III, class 2 requirements (minimum). The test valves will be classified as containment isolation valves. Because the valves are only 1" in size, administratively secured closed, and consist of a double barrier (the pipe cap provides the second barrier), they are exempt from the testing requirements of 10CFR50, Appendix J.

The expected result of the installation of these new test valves at each penetration will be the ability to pressurize the piping between all four isolation valves without installing a test flange or cycling the 36" valves during testing. Also, valve wear is expected to be reduced by not requiring the cycling of the 36" isolation valves during leak testing. Overall critical outage time will be saved in the testing of the containment isolation valves. No significant reduction in margins of safety is expected due to the installation of the valves. The Technical Specification Bases section 6.3.6 will also be revised to to clarify the acceptability of opening the test valves in the containment purge system under administrative controls. Any time the test valves are opened the Technical specification 3.6.3 limiting condition operation will be entered. The effect is no different than the opening of one of the 18" valves and the same LCO will apply.

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USAR Figure 6.2.4-1, "Containment Penetrations" and USAR Figure 9.4-6, "Containment Purge System" will be updated to depict the new test connection valves. The description of the containment purge isolation valves in USAR section 9.4.6.2.2 will also be revised to describe the existence of the test valves. USAR section 9.4.6.2.3 is also revised to describe the test valves function within the system.

50.59 Evaluation:

The LOCA section 15.6.5 and rod ejection accidents section 15.4.8 and other events which generate a containment purge isolation signal (CPIS) such as decrease in heat removal accidents section 15.2 and fuel handling accident inside containment section 15.7.4 were reviewed.

The new test valves are normally locked closed valves. They are opened only during testing while under administrative control. Anytime these valves are open Technical Specification LCO 3.6.3 will be entered. USAR section 6.2.4.4 states that manual valves serving as vents, drains, and test connections within the isolation valve envelope are subject to administrative procedures to ensure that they are in the proper position. If a design basis accident were to occur while the test valves are in the open position, no single active failure needs to be assumed for the inside containment purge isolation valve because the valve will already be in its fail safe position. The active function of the containment purge isolation valves is to close. Since the test valves are already closed, the only safety function is a passive function to maintain the pressure boundary. Per the single failure criteria discussed in USAR section 3.1.2 a single passive failure is only assumed for components associated with long term cooling capability. If an accident were to occur while the test valve is open it would be immediately closed by the dedicated operator to reestablish containment isolation. Although the closing time may be longer than the 3 second closing time prescribed for the containment purge isolation valves, the potential leakage opening is much smaller and the leakage path would have to travel a more extended rout through the Auxiliary Building to reach the unit vent. Thus, the accident is bounded by existing analysis. No new accident is postulated.

If the new 1" containment isolation test valve pressure boundary were assumed to fail this could prevent the valve from performing its passive safety related function to maintain the pressure boundary. The consequences of such a valve failure are mitigated by the pipe cap that forms the secondary boundary. Also, the consequences of such a failure are greatly less and bounded by the consequences of the postulated failure of one of the much larger existing 36" or 18" containment isolation valves.

Technical Specification 3.6.3 addresses Containment Isolation Valves. Under TS ACTIONS, a modifying note states that all penetration flow paths except for containment shutdown purge valve flow paths may be unisolated intermittently under administrative controls. The reason for this is stated in the BASES section. Because of the large size of the containment purge line penetration and the fact that the penetration exhausts directly from the containment atmosphere to the environment via the unit vent, the penetration flow path is not allowed to be opened under administrative controls. For all other penetrations a single valve in a flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1. The administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way the penetration can be rapidly isolated when a need for containment isolation is indicated.

Since the new 1" test valve is many times smaller than the existing purge system isolation valves (18"& 36") and since it provides a more indirect flow path to the unit vent via the Auxiliary Building, it has been concluded that it would be acceptable to have this test valve open under administrative controls similar to other containment test valves.

The initiators of the accidents noted above are completely independent of the containment penetration being evaluated in this change. There is no possibility of increasing the probability of occurrence of these accidents.

The radiological consequences of the accidents noted above would not be increased based on the fact that the 1" test valve is much smaller than the existing isolation valves located in the same penetration and is thus bounded by any postulated failure of the larger valves. The new manual valve is also administratively controlled as a locked closed valve by plant procedures ensuring that there is no mispositioning of the valve.

The postulated malfunction noted above, does not increase the probability of the malfunction previously evaluated and the consequences are bounded.

As noted previously, any possible malfunction is bounded by previous analysis. There is no increase in consequences due to the small valve size and strict administrative control.

There are no new or different types of accidents that can be postulated.

Any possible malfunction of the new valves is bounded by evaluation of malfunctions of the larger existing isolation valves.

There will be no reduction in the margin of safety and the same limiting condition of operation will be applied to the new test valves.

Safety Evaluation: 59 2000-0029 **Revision:** 0

Reactor Vessel Closure Head Bolting Material Properties

Activity Description:

Updated Safety Analysis Report (USAR) Table 5.3.6, Reactor Vessel Closure Head Bolting Material Properties, will be reformatted. The table currently includes test data for each installed closure head stud, nut and washer. The table will be reformatted to condense the test data into data ranges, i.e., minimums and maximums.

50.59 Evaluation:

USAR paragraph 5.3.1.7 discusses reactor vessel closure head bolting material properties. Bolting materials fracture toughness data is provided in Table 5.3-6. USAR Appendix 3A addresses compliance with Regulatory Guide 1.65, Materials and Inspections for Reactor Vessel Closure Studs. The purpose of Table 5.3-6 can be found in Regulatory Guide 1.70 and the Standard Review Plan. These documents clearly indicate the NRC expected licensees to include reactor vessel fastener test results in the FSAR. Further, it is clear the NRC used this data to confirm tensile and fracture toughness properties of the original fasteners were acceptable.

NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports" defines excessively detailed textual information as descriptive information that is not important to providing an understanding of the plant's design and operation from either a general or system functional perspective, e.g., component model numbers.

Replacement reactor vessel closure head bolting material properties must comply with technical requirements described in USAR Section 5.3.1.7 and Appendix 3A, and also be consistent with (but not identical to) the data included in Table 5.3-6. The current format of Table 5.3-6 would result in a USAR change each time a stud, nut or washer would be replaced, even if the material properties of the replacement would be enveloped by previously accepted test results. This is an unnecessary administrative burden. Individual test data for each currently installed stud, nut and washer is not important to providing an understanding of the plant's design and operation from either a general or system functional perspective. Rather, it is the range of the data that is important. Therefore, the individual data meets the NEI 98-03 definition of "excessively detailed textual information." Reformatting the individual data in terms of minimums and maximums will make the table more meaningful and usable. The data to be listed in Table 5.3-6 will continue to allow confirmation that material properties of replacement studs, nuts or washers are enveloped by previously accepted values.

These sections of the USAR will continue to provide true and accurate reflections of bolting material fracture toughness data, the range of previously accepted values, and the licensing basis for acceptability of the reactor vessel closure bolting. The information in 5.3.1.7, Appendix 3A and Table 5.3.6 will continue to be adequate to allow comparison of future replacement materials with the bounds of previously accepted values.

Since the range of previously accepted values will be retained, this activity does not approve or create the potential for variance from reactor vessel closure head bolting material properties previously accepted by the NRC. Therefore no design basis accidents are affected.

In addition, no malfunctions of equipment important to safety are identified or affected.

No acceptance limits contained in the bases for the technical specifications, or in the licensing basis are affected. Therefore, the margin of safety is not decreased.

Safety Evaluation: 59 2000-0033 **Revision:** 0

Updated Safety Analysis Report Corrections

This Unreviewed Safety Question Determination has been performed to support changes to USAR Figure 9.3-5, Section 9.5, and Appendix 9.5B.

Activity Description:
Figure 9.3-5 Sheet 11, Auxiliary Building Floor and Equipment Drain System is being revised to correct the designations of portions of Floor and Equipment Drain system piping, located in the Auxiliary Building.

Section 9.5.9.2.1 is being revised to correct the building location shown for the Auxiliary Steam Condensate Recovery and Storage Tank, from the Turbine Building to the Auxiliary Building.

Five fire area descriptions in Appendix 9.5B are being revised to correct the configurations listed for plant equipment located in those fire areas.

These USAR changes are being made to resolve discrepancies identified during Wolf Creek's USAR Fidelity Review.

50.59 Evaluation:
The clarification/correction of the USAR text and figure has no effect on any procedure, activity, administrative control, or sequence of plant operations. This is because the subject changes clarifying the USAR as described above are consistent with the current plant design conditions. The USAR description of the plant configuration would be incorrect if left unchanged. The changes merely reflect the true plant configuration, and are textual only.

This change will not make any information in the USAR no longer true or accurate. Also, this change does not involve any TESTS or EXPERIMENTS not described in the USAR which may adversely affect the adequacy of plant structures, systems, or components to prevent accidents or mitigate the consequences of an accident.

The proposed activity does not alter the physical condition of the plant. The changes are textual and the associated documents are being corrected to reflect existing design bases. There is no impact on any design basis accidents described or referenced in USAR.

The proposed activity does not alter the physical condition of the plant. Only the affected documents have been revised to reflect the actual design basis of the plant. Therefore, no credible accidents could be created by this change.

The proposed activity does not alter the existing physical condition of the plant. It simply revises the subject documents to reflect the correct design basis. No safety related equipment is adversely affected by the proposed change that could alter any of the existing credible malfunctions important to safety.

There are no acceptance limits in the Technical Specifications which will be affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0035 **Revision:** 0

Installation of Tag Numbers on Differential Pressure Switches

Activity Description:

Figure 10.4-1-01 is being revised to show two differential pressure switches located inside the Condenser Tube Cleaning System control panel, within the Circulating Water (DA) System. The switch tag numbers are being corrected to correspond with tag numbers for the associated differential pressure transmitters and their respective indicators in the control panel. The discrepancy occurred due to a prior configuration change made to this loop of the Condenser Tube Cleaning System.

No setpoint changes are associated with the proposed change. There was inconsistent data in the vendor documents which led to incorrect tag numbers for the differential pressure switches.

50.59 Evaluation:

This change does not involve a physical modification of the plant. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures relied upon to mitigate a design basis event.

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 change does not affect any design basis accidents.

The proposed change will 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 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 change will not affect a DBA or create a credible accident.

The change will not alter the operation of any plant equipment differently than it has previously been operated, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. This change 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 SSCs.

There are no acceptance limits in the Technical Specifications which will be affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0036 **Revision:** 0

Revision to Fire Protection Information to Allow Painted Surfaces Outide Containment

Activity Description:

USAR Table 9.5A-1, Section D.1(d) is being revised to update WCGS conformance with NRC Branch Technical Position (BTP) APCSB 9.5-1, Appendix A, regarding General Guidelines for Plant Protection / Building Design. The subject USAR table provides a comparison of WCGS's conformance with requirements of the BTP, that deals with the overall requirements of a nuclear plant fire protection program.

Many of the walls and floors outside Containment have been painted since initial plant startup. The associated statements in Table 9.5A-1 are being revised to state "Interior wall surfaces are generally painted CMU or concrete. Floor areas have also been coated." This change also reflects a revision to WCGS design specification for protective coating systems outside the containment, to allow the Sigma Coating System as an alternate to the Keeler & Long self-leveling floor coating.

50.59 Evaluation:

In Table 9.5A-1, the phrase "Interior wall surfaces are bare CMU or concrete" is inappropriate, since many wall surfaces have been painted since startup. Therefore, "Interior wall surfaces are generally painted CMU or concrete" is more appropriate.

The only Design Basis Accident described in the USAR - which has the potential to be impacted by additional paint coatings on walls and floors - is a postulated fire as described in the plant Fire Hazards Analysis, presented in Section 9.5B.2 of Appendix 9.5B. However, the proposed replacement coating and previously approved coatings have been reviewed for their combustibility, and can be classified as non combustible. Therefore, the addition of the paint coatings on the walls and floors does not affect the Fire Hazards Analysis.

Since the new materials being approved for use in the buildings are non combustible, this change could not create any credible accident.

The coating materials are non combustible and therefore cannot affect the fire hazards analysis. The Sigma coating is 100% solids by volume and has no volatile organic content (VOC), thus it would not adversely affect any charcoal filters that are relied upon to mitigate the consequences of a radiological accident. Therefore, this change would not result in any credible malfunctions of equipment important to safety.

There are no acceptance limits in the Technical Specifications which will be affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0037 **Revision:** 1

USAR Clarification of Procedures Associated to Reactor Vessel Flange

Activity Description:

Revision 1 has been performed to support a change to the discussion of Reg. Guide 1.65 in USAR Appendix 3A and to USAR Section 5.3.1.7.

USAR Appendix 3A and Section 5.3.1.7 are being revised. Currently, the last statements on the discussion of Reg. Guide 1.65 are: "The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure. Thus, the bolting materials and stud holes are never exposed to the borated refueling cavity water." Also, the last statement in section 5.3.1.7 currently reads "The stud holes in the reactor vessel flange are sealed with special plugs before removing the reactor closure, thus preventing leakage of the borated refueling water into the stud holes."

The proposed change will replace "before removing the reactor closure" in 3A and 5.3.1.7 with "prior to flooding the reactor cavity." It will also replace the word "never" in 3A with "not expected to be."

This change will allow the reactor vessel closure head to be removed prior to installing the stud hole plugs. The plugs will be required to be installed before flooding the reactor cavity. This change will not prevent the plugs from being installed before the head is removed. It will give the option to do so after the head is removed, (prior to flood-up), in order to allow ultrasonic examination of the ligaments of the stud holes. The stud holes will still be protected from the borated refueling cavity water and will not be exposed to unnecessary contamination.

50.59 Evaluation:

In Appendix 3A, the use of the word "never" when describing how the stud holes are protected from borated water is inappropriate. There is always a slight possibility that the plugs will leak. Therefore, "not expected to be exposed to borated refueling cavity water" is more appropriate than "never exposed to borated refueling cavity water."

This change will also make the USAR more consistent. The USAR states that the studs, nuts and washers must be removed and put in storage racks or suspended from the reactor vessel head while the head is removed to its storage stand. If desired to have the studs suspended from the head while it is being removed, the closure head would need to be removed prior to installing the stud hole plugs.

This change will not make any information in the USAR no longer true or accurate. Also, this change does not involve any TESTS or EXPERIMENTS not described in the USAR which may adversely affect the adequacy of plant structures, systems, or components to prevent accidents or mitigate the consequences of an accident.

The design basis accidents were reviewed and are not affected. The proposed change will take place only when the plant is in Mode 6 (one or more reactor head closure bolts less than fully tensioned), and prior to any fuel movement. The postulated head drop accident in USAR 9.1.4.3 occurs at this time. This accident was reviewed and it was determined that the conditions or outcome are not affected.

No new accidents could be created. The stud holes will still be protected from borated water.

No malfunctions of equipment important to safety will be affected. The stud holes will still be protected from borated water.

There are no acceptance limits in the Technical Specifications which will be affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0038 **Revision:** 0

Organization Change to Training

Activity Description:

The organization change will replace the person currently assigned to the position of Manager Training with the person currently assigned to the position of Superintendent Operations Training.

This change will affect USAR Section 13.1.3.2 which describes the qualifications of plant personnel. This section contains the resume of Manager Training which will change. No other sections of the USAR are affected.

50.59 Evaluation:

Following implementation of the change, the Manager Training will continue to meet or exceed the minimum qualifications of ANSI/ANS 3.1-1978 in accordance with TRM requirement TR 5.8.1. There will be no direct effect on the normal day to day activities associated with these positions. No changes are being made to the duties and responsibilities of either of these positions. All functions currently associated with these positions will continue to be performed.

This is an organization change. Since no changes are being made to the duties and responsibilities of the positions affected by this proposed change, no design basis accidents are identified that could be impacted by this proposed change.

There are no credible accidents that could be created by replacing the Manager Training with another qualified individual.

There are no malfunctions of equipment important to safety that could be affected by replacing the Manager Training with another qualified individual.

Since no changes are being made to the duties and responsibilities of the positions affected by this proposed change, no acceptance limits are identified that could be impacted by this proposed change. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0039 **Revision:** 0

Addition of Class 1E Electrical Equipment A/C to Technical Requirements Manual

Activity Description:

A new Technical Requirement (TR 3.7.23) is being added to the Technical Requirements Manual (TRM) which verifies OPERABILITY of the Class 1E Electrical Equipment air conditioning (A/C) trains in Plant Modes 1, 2, 3, and 4. Also, a new surveillance requirement, TSR 3.7.23.1, will verify each Class 1E Electrical Equipment A/C train has the capability to remove the assumed heat load. The surveillance will be performed on an 18-month frequency. In addition, TR 3.7.22, "Area Temperature Monitoring," Table TR 3.7.22-1, is revised to include the DC switchgear rooms and the NK battery rooms. The maximum temperature specified in TR 3.7.22 for the ESF switchgear rooms is revised from less than or equal to 117°F to less than or equal to 101°F based on the associated engineering calculation. The values used in the TRM include an allowance for instrument error of ± 3 degrees F.

This activity is to implement WCGS commitment, given in its updated response to NRC Generic Letter 89-13, for Class 1E Air Conditioning Units SGK05A and SGK05B. The commitment involves making programmatic adjustments to the testing program that will verify the heat transfer capability of all safety-related heat exchangers cooled by service water.

The purpose of the Class 1E Electrical Equipment A/C is to maintain the area temperature of the Class 1E equipment rooms to ensure OPERABILITY of the Class 1E Electrical Distribution System and the Class 1E DC Electrical Power System.

50.59 Evaluation:

Operation of the Class 1E Electrical Equipment A/C was evaluated in the WCGS plant-specific probabilistic safety assessment (PSA) and assumed two failure modes: failure on demand and failure during operation. Operator actions to restore Class 1E room cooling was also included in the plant-specific PSA model. The PSA indicated that removing SGK05A or SGK05B from service for 168 hours resulted in a slight increase in core damage frequency (CDF). The increase is less than the acceptance criteria in Section 2.2.4 of Regulatory Guide 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." Establishing a Technical Requirement-allowed outage time of 7 days for the Class 1E Electrical Equipment A/C trains is, therefore, justified, based on a CDF increase which is within the Regulatory Guide 1.174 Region III Acceptance Guidelines for CDF. Further, the Class 1E Electrical Equipment A/C is therefore not considered significant to the health and safety of the public.

The proposed change to the TRM describes applicability of existing safety analyses to the current plant design, and does not make or allow any changes to either the facility or procedures as described in the USAR. The proposed change would not make information in the USAR no longer true or accurate, and it 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 proposed change does not have a detrimental impact on the integrity of any plant structure, system or component. It will not alter the operation of any plant equipment, or otherwise increase its failure probability. Therefore, the proposed change does not affect any design basis accidents.

The proposed change will 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 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 change will not affect a DBA or create a credible accident.

The proposed change will not alter the operation of any plant equipment from the manner in which it has been previously operated, or otherwise increase its failure probability. The manner in which the Class 1E Electrical Equipment A/C trains is operated is consistent with previous operation of the equipment. Any changes in the manner in which these SSCs are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the change. This change will not increase the probability of occurrence of a malfunction of equipment important to safety because it will not involve any physical changes to plant SSCs.

The proposed change does not involve a physical modification of the plant. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures relied upon to mitigate a design basis event. The proposed change does 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 change to the TRM does 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 the proposed change. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of this change. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0040 **Revision:** 0

Revision to TRM 3.7.19 Applicability

Activity Description:

The Applicability of TRM 3.7.19 is being modified to state "Whenever the temperature of primary or secondary coolant in associated Steam Generator (SG) is less than or equal to 70 °F and the primary or secondary systems are capable of being pressurized.". This change is being made to allow suspension of the requirement to hourly monitor SG pressure and temperature when the SG Pressure/Temperature (P/T) limits cannot be exceeded. In addition, TRM 3.7.19 Applicability Bases is being modified to describe when the primary and secondary systems are considered to no longer be capable of being pressurized (e.g., reactor vessel head removed and secondary system manways removed).

50.59 Evaluation:

The TRM Bases revision has been added to clarify the TRM requirements. The purpose of TR 3.7.19 is to establish operating limits that provide a margin to brittle failure of the carbon steel ASME Class 1 portion of the steam generators (SGs), which comprise part of the reactor coolant pressure boundary (RCPB). TR 3.7.19 requires that the pressure of the primary and the secondary coolant in the SGs be maintained less than or equal to 200 psig when the primary or secondary coolant temperature of the associated SG is less than or equal to 70 °F. However, when the primary and secondary systems are not capable of being pressurized, exceeding the SG operating limit of 200 psig when SG temperature is less than or equal to 70 °F is not possible. The margin to brittle failure of the SGs is not impacted by this change and is considered acceptable. This TRM revision is further justified in that the Technical Specifications requirements associated with leakage and Reactor Coolant System pressure/temperature limits continue to ensure that RCPB degradation is detected.

The proposed changes to the Technical Requirements Manual describes applicability of existing safety analyses to the current plant design and does not make or allow any changes to either the facility or procedures as described in the USAR. 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 change does not affect any design basis accidents.

The proposed change will 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 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 change will not affect a DBA or create a credible accident.

The change will not alter the operation of any plant equipment differently than it has previously been operated, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. This change 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 SSCs.

The change does not involve a physical modification of the plant. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the 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 change to the TRM does 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 mitigating 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 this change. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0041 **Revision:** 0

Integrated Plant Scheduling Manager Change

Activity Description:

A new Manager Integrated Plant Scheduling (IPS) has been appointed. There are no reporting changes associated to this personnel change. There should be no adverse impact due to this change in management.

The USAR, chapter 13.1 contains the resume of the IPS Manager. This resume will need to be updated to reflect the individual assigned to the position.

50.59 Evaluation:

This is a personnel change. All organization functions will continue and competency expectations remain the same. Therefore, there will be no adverse impact on the plant. Since there is no impact on the plant and no accident analysis takes credit for the IPS organization there are no design bases accidents created or changed. This logic also applies to the creation of new accidents. In addition, there will be no adverse impact on the plant. Therefore, no equipment will be effected.

There are no acceptance limits associated to the organization. Organization is understood to change and adjust throughout the life of a plant. Functions important to the safe operation of the plant continue and competencies of personnel have not been diminished. Therefore, the original intent of the organization has not been altered and the margin of safety has not been decreased.

Safety Evaluation: 59 2000-0043 **Revision:** 0

Additon of Hoses to Allow Direct Drainage of the Chemical Addition Tank

Activity Description:

Figure 9.2-15 is being revised to reflect how the CCW Chemical Addition Tank (TEG02) will be drained. Note 1 to Figure 9.2-15 indicates that the CCW Chemical Addition Tank is to be drained to portable containers or pumped back into the CCW system using a temporary pump. This note is being deleted, since it's now known that the corrosion inhibitor in the tank can exhaust the resin in the Radwaste Demineralizer system. This is a commercial concern and not a safety issue.

This change will allow a revision to the Operations procedure which controls draining of this tank, while maintaining consistency between the procedure and the USAR. The procedure is being revised concurrently to direct the Operator to drain the tank to an Auxiliary Building sump, the contents of which are processed normally through the Secondary Waste system in the Turbine Building. All discharges from the sump will continue to be monitored for chemical content and radiation prior to release, in accordance with existing plant procedures.

This modification is a document change only and does not involve any changes to permanent plant equipment.

50.59 Evaluation:

Note 1 to Figure 9.2-15, indicating that the CCW Chemical Addition Tank is to be drained to portable containers or pumped back into the CCW system using a temporary pump, is no longer appropriate. The note is being deleted to allow draining the tank to an Auxiliary Building sump using temporary drain hoses. The hoses will be installed on a temporary basis, and removed in accordance with the Operations procedure when drain down of the tank is complete. The hoses are not shown on Figure 9.2-15.

This change does not involve a physical modification of the plant. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures relied upon to mitigate a design basis event.

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 change does not affect any design basis accidents.

The proposed change will 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 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 change will not affect a DBA nor create a credible accident.

The change will not alter the operation of any plant equipment differently than it has previously been operated, nor otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. This change 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 SSCs.

There are no acceptance limits in the Technical Specifications which will be affected. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0044 Revision: 0

Installation of Flexible Hose in the CCW Supply and Return Cooling Lines of Heater Drain Pump Motor

Activity Description:

Figure 10.4-6-03 is the Piping & instrumentation Diagram of the Feedwater Heater Extraction Drains & Vents (AF) system. This figure includes a detail of the existing configuration of the cooling lines, which are presently hard-piped to the heater drain pump motor cooling coil nipples. Since the 3/8" piping connecting the nipples to the 3/4" cooling lines is being replaced with flexible hoses, the associated detail in Figure 10.4-6-03 is no longer appropriate. The figure is therefore being revised to show the new configuration of the flex hoses being installed. This plant will minimize vibration-induced failure, which has occurred twice in the nipple connecting the cooling water supply line to the DPAF01A pump motor cooling coil.

50.59 Evaluation:

Although this change involves a physical modification of the plant, there are no alterations to the parameters within which the plant is normally operated. The heater drain pump motor cooling coil return and supply lines are all non-safety related. Installing a same size stainless steel braided hose that meets or exceeds the original design pressure and temperature requirements for the pipe will not affect pump/motor performance. No changes are being proposed to the procedures relied upon to mitigate a design basis event.

The decrease in feed water temperature due to a spurious heater drain pump trip is discussed in USAR section 15.1.1. The modification being made has no impact on the probability of the heater drain pump trip or its consequences as discussed in the USAR. Further, the proposed change does not have a detrimental impact on the integrity of any plant structure, system or component. The change will not alter the operation of any plant equipment, or otherwise increase their failure probability. Therefore, the proposed change does not affect any design basis accidents.

The proposed change will 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 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 pump, motor, and cooling water lines are non-safety related. The new flexible hoses have the same pressure and temperature rating as the hard pipe. Installation of the hoses will enhance the existing design by reducing vibration-induced failure. As such, installation of the flexible hoses will not affect a DBA or create a credible accident.

The proposed change will not alter the operation of any plant equipment differently than it has previously been operated, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the change. Therefore, there are no credible malfunctions of equipment important to safety which will be directly or indirectly affected by this change.

There are no acceptance limits associated with the Turbine Building Closed Cooling Water lines to the heater drain pump, or associated with the pump itself, discussed in the Technical Specifications. Therefore the margin of safety is not affected.

Safety Evaluation: 59 2000-0046 **Revision:** 0

Instrument Tunnel Sump Level Indication

Activity Description:

The level indication on the Instrument Tunnel Sump (LFLT0079) will be upgraded so that measurements can be taken starting from 1/2" above the bottom of the sump. The instrument span is increased from 14" to 24" and a longer hanger extension is added to achieve this coverage. This will allow the detection of a 1 gpm leak within 1 hour as required by Reg. Guide 1.45 and ITS LCO 3.4.15.

A conservative estimate of the minimum detectable change in the instrument tunnel sump level will be approximately 15 gallons. Also, a conservative estimate of the minimum detectable change in the containment normal sump level is approximately 5 gallons. If the instrument sump is completely dry the minimum detectable level change will be 10 gallons for the 1/2" space below the displacer plus 15 gallons for the first 0.5" detectable or a total of 25 gallons. The USAR is updated to reflect these new values. These recalculated detectable minimums will slightly increase the detection time of a leak in the instrument tunnel sump but detection time will still meet the Technical Specification Bases requirement of detecting a 1 gpm leak within 1 hour. This is reflected in revised Figure 5.2-2.

The text discussion in USAR section 5.2.5.2.3, "Containment Sump Level And Flow Monitoring System," is revised to accurately describe the system detection capabilities. USAR Figure 5.2-2, "Primary Coolant Leak Detection Response Time", is also revised to reflect the revised detection time for the instrument tunnel sump.

50.59 Evaluation:

The instrument tunnel sump level (LFLT0079) is a non-safety related component and it is not relied upon to mitigate the consequences of any design basis accidents. The change will not affect any accident previously evaluated in the USAR.

The proposed change will not impact the initial conditions of any DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. It is not part of the primary success path which functions or actuates to mitigate a DBA or transient. This equipment is installed instrumentation that could be used to detect and indicate a significant abnormal leak from the reactor coolant pressure boundary. However, the proposed change will not affect a DBA or create a credible accident.

The change in sensor range will not alter the operation of any other plant equipment differently than it has previously been operated, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. This change will not increase the probability of occurrence of a malfunction of equipment important to safety because it will not involve any changes to a safety related plant SSC. The manner in which the containment sump level and flow monitoring system is operated is consistent with previous operation of the equipment. Any changes in the manner in which these SSC are operated, maintained, or inspected continue to ensure equipment important to safety is maintained OPERABLE.

There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the 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 change to the instrument tunnel sump level range does not adversely affect any acceptance limits contained in the bases of the Technical Specifications. In fact this new instrument sensing range assures the ability to meet the Technical Specification requirements. 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 equipment performance parameter changes associated with these changes. No safety setpoints are affected, and no changes are being proposed in the plant operational limits as a result of this change. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0047 Revision: 0

Temporary Modification to Install a Temporary Chemical Metering Pump

Activity Description:

Difficulties with the chemical injection pump operation have been noted resulting in an inability to add chemicals to the auxiliary steam system. Historically, the installed pumps have exhibited problems pumping chemicals into the auxiliary steam system due to such occurrences as loss of prime, oil side problems, and internal check valve difficulties.

A temporary chemical metering pump will be installed in place of the hydrazine addition pump PFE02 using the current suction and discharge piping. This modification will enable the chemicals to be kept at an acceptable concentration in the Auxiliary Steam System and the Auxiliary Boiler until a permanent modification has been installed.

50.59 Evaluation:

This temporary modification does not alter the design function of the FE system.

No design basis accidents listed in the USAR 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 chemical addition system is non-safety related, has no safety design basis, and is not referenced or controlled by the technical specifications, no acceptance limits were identified as being affected by this temporary modification. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0048 **Revision:** 0

Change to Dynamic Pressure Loads for SSE

Activity Description:

This change revised the lateral earth pressure information for the site specific and SNUPPS Civil Structural Design Criteria to be consistent with the plant design bases. The lateral earth pressure loading tables have been revised for the site specific criteria to reflect the proper dynamic loads for the Safe Shutdown Earthquake (SSE) based on 0.15g instead of the original 0.12g. This change increases the dynamic earth pressure loads for the SSE case. The basis for this change is identified in Sect. 3.4 of the Site Specific Structural Design Criteria. The NRC's review and acceptance of our evaluation for the higher seismic load was documented in NUREG-0881. The site specific design criteria was also annotated to indicate that any analytical and design work involving free field seismic loads on the ESW buried pipe and duct bank must be based on the SNUPPS envelope to be consistent with USAR Section 2.5.4 and Appendix 3C. The lateral earth pressure tables have been revised for the SNUPPS Civil Structural Design Criteria to reflect the dynamic lateral loads for seismic based on the 3 site envelope consisting of an SSE of 0.2g and OBE of 0.12g rather than the original 4 site envelope which consisted of an SSE of 0.25g and OBE of 0.13g. These changes result in the design criteria being consistent with the USAR (Section 2.5.4.9 and Appendix 3C) as originally reviewed and approved by the NRC during the licensing phase. Therefore these changes do not affect the USAR and by themselves would not require further evaluation. Design bases calculations completed for the site specific ESW structures, buried pipe and duct bank were reviewed to ensure that the higher SSE values were considered. Two Calculation Change Notice's (CCN's) were included to identify later calculations that have supplemented the earlier ones. The CCN's do not affect the USAR.

In addition, this review included numerous USAR changes that were identified as a result of the USAR Fidelity Review as a compilation of potential discrepancies from USAR Sections 2.5.4 (Stability of Subsurface Materials) and 2.5.6 (Embankments And Dams). The resulting USAR changes were summarized in the Regulatory Screening and also included changes to USAR Sections 1.2.1.6 (General Plant Site Description - Seismology) , 2.5.2 (Seismology - Vibratory Ground Motion) and 3.7(S) Seismic Design. These changes were corrections, typos, clarifications, and changes for consistency between different USAR sections or the design basis. Most of these changes were in sections of the USAR that would be considered historical information. None of these changes represented a change to the design basis or licensing basis.

50.59 Evaluation:

The changes to the Civil Structural Design Criteria make information in these documents consistent with statements in the USAR. The changes to the USAR are corrections, clarifications or changes to make statements in one location consistent with other locations in the USAR. Therefore, this change does not make any information in the USAR no longer true or accurate or would violate a statement in the USAR.

The changes to the Civil Structural Design Criteria do not result in any new criteria or criteria that is less severe than that which has been previously reviewed and approved during the licensing phase of the plant. Therefore, this change does not impact any of the design basis accidents discussed or referenced in the USAR. Likewise, this change would not increase the probability of occurrence or the consequences of an accident previously evaluated in the USAR.

The changes to the seismic criteria in the Civil Structural Design Criteria and USAR would not create any type of credible accident since the change does not result in any seismic design criteria less conservative than that which has already been reviewed and approved during the licensing phase of WCGS. The USAR changes are corrections, clarifications or changes to make statements consistent with other information in the USAR. They do not change the content or meaning of any information in the USAR. Most of these changes are in sections where the information is historical in nature. Therefore, there are no credible accidents that this change could create. Likewise, this change could not create the possibility of an accident of a different type than any previously evaluated in the USAR.

This activity does not directly change anything that could result in a credible malfunction of equipment important to safety. This change did not allow in any new seismic design or evaluation for any equipment less conservative than that which had already been reviewed and approved during the licensing phase for WCGS. Therefore, there were no credible malfunctions of equipment important to safety that this change could affect. Likewise, this change could not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the USAR, nor could it create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

The seismic design limits are the only acceptance limits contained in the licensing basis documents that could be affected by this change. However, the changes do not allow any seismic design criteria less conservative than that which had already been reviewed and approved by the NRC during the licensing phase. Therefore, this change did not affect any acceptance limits contained in the bases for the technical specifications or licensing basis documents, so the margin of safety was not reduced.

Safety Evaluation: 59 2000-0050 **Revision:** 0

Revise USAR to Reflect Commitment to Regulatory Guide 1.181

Activity Description:

The discussion of Wolf Creeks commitment to Regulatory Guide 1.70, Revision 3, was revised to indicate that guidance document NEI 98-03 will be used to determine what information in the USAR is to be updated, what level of detail the update needs to reflect, and what type of information may be removed from the USAR. In addition, a commitment to Regulatory Guide 1.181 will be added in which the NRC endorses, without exception, the Nuclear Energy Institute (NEI) guidance. This will effectively change how Wolf Creek is committed to revising the USAR in accordance with 10 CFR 50.71(e).

50.59 Evaluation:

This change reflects how Wolf Creek is committed to revising the USAR in accordance with 10 CFR 50.71(e). It is administrative in nature describing how the USAR document is revised. No plant procedures, structures, systems, or components outlined, summarized or described in the USAR will be affected. Because this is an administrative change, no design basis accidents discussed in the USAR will be affected, no credible accidents will be created, and no credible malfunctions of equipment important to safety will be affected. The NRC did not base licensing of the plant on the USAR update process, therefore, no acceptance limits will be affected and the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0051 **Revision:** 0

Elimination of Redundant ISEG Functions and Assignment of Responsibility to Engineering

Activity Description:

This change reassigned Independent Safety Engineering Group (ISEG) functions prescribed by NUREG 0737, with the exception of the Industry Technical Information Program (ITIP)/Operating Experience Program, from Nuclear Safety Engineering (NSE) to the Engineering Department. The change also eliminated the requirement for five dedicated engineers and the two year experience requirement for ISEG personnel. Based on the evaluation and justification (included in the procedure package) of the change, there should be little or no effect other than improved efficiency since the reassigned ISEG functions were also already being performed by Engineering. The ITIP/Operating Experience program will remain with NSE.

50.59 Evaluation:

This change affected discussion of the ISEG functions in both Chapters 17 and 18 of the USAR. It also removed the experience requirement associated to the ISEG function. These sections of the USAR reflected information to either satisfy NUREG -0737 items or commitments made due to engineering conditions at the time of licensing (i.e., a fairly young engineering team located at the corporate offices in Wichita). Responsibilities for the functions were assigned to the NSE group. However, equivalent functions were also being performed by Engineering (now a mature organization located on site).

The ISEG functions do not figure into the accident analysis for Wolf Creek Generating Station. The duplicated oversight functions do not directly or indirectly impact plant operation. The experience requirement is no longer relevant due to the experience level within the Engineering organization. Therefore, no design basis accidents discussed or referenced in the USAR would be potentially impacted by this change, no credible accidents would be created, and no credible malfunctions of equipment important to safety would be directly or indirectly affected.

Although addressing NUREG 0737 guidance was a condition of licensing and was also originally located in the Technical Specifications, by definition, these functions would not have constituted an issue that would have been an acceptance limit nor factored into the bases for any technical specification. These functions were administrative in nature and were oversight activities that were duplicated by the Engineering department. The current review and audit processes in place are the equivalent or superior to those whose acceptance was documented in NUREG-0881. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0052 **Revision:** 1

Use of Startup Transformer XMR01 for Continuous Operation

Activity Description:

Unit Auxiliary Transformer (UAT) XMA02 failed on 9/4/00. The failure was a result of a short circuit caused by a squirrel on a nearby capacitor bank. The short circuit caused substantial external damage in the area of the transformer secondary bushings. As a result, the transformer was left unable to perform its function of supplying 13.8 kV to the plant PA (13.8 kV) buses.

Revision 0 of Temporary Modification (TMO) 00-14-MA and USQD 59-00-0052, Rev. 0 were issued on 9/6/00 to allow the plant to return to 100% power using Start-Up Transformer (SUT) XMR01 to supply both PA buses. The technical justification for this configuration was provided in Calculation XX-E-006 and in the TMO evaluation. The SUT normally supplies power to the PA buses until the plant reaches 30% power. At that point, the configuration is changed to allow the UAT to supply the PA buses, and the SUT is left supplying only the PA stub bus and safety related transformer XNB02.
After the plant returned to power on 9/6/00, the UAT was repaired to the point that it could be tested for internal damage. Testing revealed that the transformer had been damaged internally. As a result, the transformer had to be removed and sent off site for repairs. The original TMO and USQD allowed operation at 100% power using the SUT to supply the PA buses until the end of Refueling Outage 11. To allow the internal repairs to the UAT to be completed it was necessary to extend the duration of the TMO.
Revision 2 of the TMO and Revision 1of the USQD were issued to extend the duration of the TMO through 4/21/01 to allow internal repairs to be made to the UAT.

50.59 Evaluation:

USAR Section 8.3 states that during startup, the electrical load is transferred from the Start-up Transformer (SUT) to the Unit Auxiliary Transformer (UAT) "by a manually initiated transfer". With this temporary modification installed, that transfer of electrical load will not occur. Section 15.3 states that normal power for the reactor coolant pumps "is supplied through individual busses connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to operate." This will no longer be true when the temporary modification is implemented, because the reactor coolant pumps will already be receiving power from the transformer supplied from external lines. Drawing KD-7496 (USAR Figure 8.2-4-00) was modified as part of the TMO to document this.

No tests or experiments not described in the USAR will be completed during this TMO.

The Design Basis Accidents in chapters 2,3,5,6,9 and 15 were reviewed for potential impact by this TMO. It was concluded that only the partial and complete loss of forced reactor coolant flow scenarios provided in Sections 15.3.1 and 15.3.2 would be affected by this

TMO. Calculation XX-E-006 and USAR section 8.3.1.1.1.2 confirm that the SUT has the capacity to power all of the equipment that would normally be powered by the UAT in addition to the normal SUT loads. It was determined that the probability of a loss of forced reactor coolant flow while operating at 100 % power on the SUT would be increased, but that the increase would be "minimal". The increase would be less than 10%. The percent increase was determined by comparing the probability of a transformer failure to the probability of a failure to fast bus transfer in response to a transient initiating event. In addition, the increase in Core Damage Frequency (CDF) was determined to be "not risk significant" (less than 1E-6) for the duration of the TMO.

The only impact this TMO has is a minimal increase in the probability of losing power to the PA buses. Because the consequences of losing power to the PA buses would be the same regardless of how the electrical power is lost, the radiological consequences of an accident previously evaluated in the USAR would not be changed. Likewise, the radiological consequences of a malfunction of equipment important to safety would not be changed as a result of this TMO.

This TMO does not create the possibility of an accident or event of a different type than any previously evaluated in the USAR. The only impact this TMO has is a minimal increase in the probability of losing power to the PA buses. The SUT is designed to supply the PA buses in the event of an electrical fault in the generation system. The use of the SUT has been previously analyzed, and all accidents that could be caused by losing power to the PA buses have been previously evaluated because the possibility of losing the PA buses exists in the normal configuration. Therefore, there are no new accidents that could be created, because this is an acceptable configuration. A loss of offsite power would ultimately have the same impact on the plant regardless of which transformer is supplying the PA buses. Likewise, all malfunctions of equipment important to safety that could be caused by losing power to the PA buses have been previously evaluated because the possibility of losing power to the PA buses exists in the normal configuration. Therefore, this TMO does not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

There are no credible Malfunctions of Equipment Important to Safety previously evaluated in the USAR that are directly or indirectly affected by this TMO. Therefore, this change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. During normal operations, the SUT supplies power to the ESF No. 2 transformer XNB02 only. During a plant start up, (up to 30% Power) and during plant refueling, the Start-Up transformer supplies XNB02 and both PA buses. Calculation XX-E-006, and USAR section 8.3.1.1.1.2 confirm that the startup transformer has the capacity to power all of this equipment. With the Start-Up transformer supplying both the PA buses and the associated loads, the additional power going through the main transformers (XMA01A, B &C) will not cause damage provided the transformers stay within the temperature limits outlined within Operating Procedures.

There are no Acceptance Limits in the bases for the Technical Specifications that are affected by the installation of this TMO. The TMO allows the Start-up Transformer to continuously supply power to the PA buses instead of just during refueling and startup.

Calculation XX-E-006 and USAR section 8.3.1.1.1.2 confirm that the Start-up Transformer has the capacity to power all of the equipment that would normally be powered by the Unit Auxiliary Transformer in addition to the normal Start-up Transformer loads. This modification does not impact any Acceptance Limits in the bases for the Technical Specifications or in the licensing basis documents. Therefore, the margin of safety as defined in the basis for any technical specification is not reduced.

Safety Evaluation: 59 2000-0053 **Revision:** 0

Replacement of Explosive Detectors

Activity Description:

This modification replaced six ITI Model 75 explosive detectors with three newer Model 85 explosive detectors. The manufacturer (ITI) will no longer support the older Model 75 explosive detectors as they consider them obsolete. These explosive detectors are located within the Security Building outside of the Protected Area Barrier (PAB). There are no expected effects of this change on the plant.

50.59 Evaluation:

The only impact to the USAR was to Figure 3.1-4 of USAR Section 13.6 (Security Plan), which shows the equipment layout in the WCGS Security Building. This Figure (3.1-4) was modified to show the new layout with the six older explosive detectors removed and the three new ones installed.

Security System Explosive Detectors (SSEDs) located in the Security Building cannot have any impact on any accident discussed or referenced in the USAR. Therefore, replacement of obsolete SSEDs with newer SSEDs cannot have any impact on, or increase the probability of occurrence of, any accident previously evaluated in the USAR. Likewise, this change cannot increase the radiological consequences of any accident previously evaluated in the USAR.

Replacement of obsolete SSEDs with newer ones cannot affect, either directly or indirectly, any credible malfunctions of equipment important to safety. Therefore, this change cannot increase the probability of occurrence of, or the radiological consequences of, a malfunction of equipment important to safety previously evaluated in the USAR.

There is no change that can be done to SSEDs located in the Security Building that could create the possibility of an accident of a different type than any previously evaluated in the USAR. Likewise, there is no change to the SSEDs located in the Security Building that could create the possibility of different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

This change deals with Security System Equipment. The replacement of obsolete SSEDs with new SSEDs cannot affect any acceptance limits which are contained in the bases for the technical specifications (or if not in the bases for the technical specifications, in the licensing basis documents). Therefore, this change does not reduce the margin of safety as defined in the basis for any technical specifications.

Safety Evaluation: 59 2000-0054 **Revision:** 0

Temporary Modification to the Fire Protection System

Activity Description:

The turbine outage office trailer was placed on the turbine deck during Refuel 11. The office trailer was a 10' by 40' (approximately) wooden structure. It was required by WCNOC's insurance carrier to be protected with a sprinkler fire suppression system. This TMO documented the connection for this sprinkler system to a fire protection system hose rack station. Standard 1 ½ " fire hose was routed from the hose rack to the office trailer.

The anticipated duration of this TMO was about 1 week following completion of RF11 or 11/10/00.

50.59 Evaluation:

This modification did not impact the ability of the fire protection system to suppress fires elsewhere in the turbine building, and the operability of the affected hose rack was maintained. This change altered the fire protection system as shown in USAR Figure 9.5.1-1-01.

This activity did not impact any previously analyzed accidents in the USAR.

This change had no impact on systems, structures, or components that could potentially initiate an accident.

This change had no potential impact on equipment important to safety.

Fire protection is not addressed in the Technical Specifications (TS) or the Technical Requirements Manual (TRM). Therefore, this change did not reduce the margin of safety as defined in the basis for any TS.

Safety Evaluation: 59 2000-0055 **Revision:** 0

Temporary Modification to Provide Cooling to Battery Room

Activity Description:

SGE17 provides cooling to the Non-Safety Related PK13 and PK14 battery room. SGE17 needed to be taken out of service for repair and no alternate cooling method was available. To preserve the life of the equipment located in this room, an alternate cooling method was needed.

This TMO provided an alternate means to adequately cool and ventilate the PK13/14 battery room. It was put in place for both equipment life issues and to prevent accumulation of hydrogen from the batteries. A supply duct for the computer room (Work Controls Center) is in close proximity to the battery room. A fan was placed next to an open access hatch in the supply duct and the air was directed to the battery room using an elephant trunk (flexible ductwork). The elephant trunk was routed through one of the doors to the battery room and the other battery room door was opened to provide an exhaust path.

50.59 Evaluation:

SGE10A and SGE10B provide cooling to the computer room (Work Controls Center). This TMO allowed these units to provide cooling to the PK13/14 battery room and the computer room while SGE17 was removed from service.

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 TMO allowed air to be exhausted from the room by the doors being left open. If the ventilation was lost, the doors were open which still would have prevented the build up of hydrogen to an explosive limit. The configuration under this TMO was an improved configuration compared to the normal configuration for preventing hydrogen accumulation. Therefore, this change did not increase the probability of occurrence of an accident previously evaluated in the USAR since hydrogen accumulation was still controlled. Likewise, this activity did not increase the radiological consequences of an accident previously evaluated in the USAR.

No credible accidents of a different type than any previously evaluated in the USAR were created.

Loss of cooling to the PK13 and PK14 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 was being used as office space. Therefore, this change will not increase the probability of occurrence of, or the radiological consequences of, a malfunction of equipment important to safety previously evaluated in the USAR. Likewise, this change did not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR since it does not affect any equipment important to safety.

The equipment affected by this temporary modification is not important to safety and is not governed by technical specifications. Therefore, the margin of safety as defined in the basis for any Technical Specifications was not reduced.

Safety Evaluation: 59 2000-0056 **Revision:** 1

Cycle 12 Reload Design Changes

Activity Description:

Revision 0:
Revision 0 of 10CFR 50.59 Unreviewed Safety Question Determination (USQD) 2000-0056 was completed to address changes to the USAR associated with the Cycle 12 Reload Design as documented in Configuration Change Package 09063, Rev. 0.
One fuel related item which could potentially impact the reload, which was not a design feature, was also evaluated in the framework of the standard reload design process. At several reactor sites, twice-burned fuel assemblies have exhibited broken or evidence of broken spring screws. WCGS had twice-burned fuel assemblies that could have been susceptible to the postulated spring screw failure mechanism that were scheduled for reinsertion into the Cycle 12 core. This potential spring screw failure mechanism and impacts to the reload design were evaluated, and compensatory measures to address that issue were developed and included in the USQD evaluation.

Revision 1:
After the issuance of Revision 0 of this USQD, another nuclear power plant found broken spring screw pieces in their Steam Generator. To provide further assurance against a similar occurrence at Wolf Creek, more conservative compensatory measures were developed. Revision 1 of USQD 2000-0056 was completed to address these more conservative compensatory measures. The rest of the USQD remained the same and still addressed CCP-09063, Rev. 0.

The reload design was performed utilizing USNRC 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 12 Reload Design were reviewed and approved by the NRC.

The specific changes to the USAR associated with the Cycle 12 Reload Design are listed in the approved 10CFR 50.59 Unreviewed Safety Question Determination (USQD).

The Wolf Creek Generating Station Cycle 12 reactor core is comprised of 193 fuel assemblies. The Cycle 12 core inventory consists of 20 Region 12 assemblies, 84 Region 13 assemblies, and 89 Region 14 assemblies. Twenty Region 12 assemblies were reused in Cycle 12. Three assemblies are the Robust Fuel Assembly (RFA) design. The RFA design was previously evaluated and found to be acceptable for use in reload core designs at WCGS. The remaining Region 12 assemblies and all the Region 13 assemblies are the Westinghouse V5H with Performance+ features (V5H P+) design with rotated mid-grids.

The RFA and V5H P+ design are hydraulically identical. The Region 14 assemblies are the Westinghouse V5H with Performance + features and Zirlo+2 (V5H P+,Z+2) design with rotated mid-grids. All the assemblies in Cycle 12 include Intermediate Flow Mixing (IFM) grids with rotated mid-grids so no transition core DNB penalties are applied to the Cycle 12 analyses.

All the design features used in the Cycle 12 Reload (i.e., Fully Enriched Annular Axial Blankets, ZIRLO IFM grids, +2 Features, 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 12 Reload Design may be found in the Reload Safety Evaluation.

The evaluations of the fuel modifications and analyses to support the Cycle 12 reload design confirmed that they would not result in a potential unreviewed safety question, as defined in 10CFR50.59.

As discussed above, one fuel related item which could potentially impact the reload, which is not a design feature, was also evaluated in the framework of the standard reload design process. At several reactor sites, irradiated fuel assemblies have exhibited broken or evidence of broken top nozzle spring screws. WCGS has irradiated fuel assemblies that may be susceptible to the postulated spring screw failure mechanism which are scheduled for reinsertion into the Cycle 12 core. The potential spring screw failure mechanism and impacts to the reload design were evaluated. With respect to the potential for top nozzle spring screw failure, evaluations confirmed that no potential unreviewed safety question exists when certain compensatory measures (delineated in the approved USQD) are enacted.

50.59 Evaluation:

The Cycle 12 Reload Design affects and initiates appropriate changes to Chapter 4 of the USAR. These changes incorporate discussion of the Westinghouse V5H with Performance + features and Zirlo+2 (V5H P+,Z+2) fuel design which was introduced at WCGS for the first time in Cycle 12. All cycle specific design values are bounded by licensing analyses reviewed and approved by the NRC.

Documentation for the Cycle 12 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) were performed.

Impacts to Design Basis Accidents for all of the fuel types featured in the Cycle 12 Reload Design, including the potential for spring screw failure, were evaluated and found to be acceptable. There are no new fuel features incorporated as part of this design that would adversely impact any Design Basis Accident. Use of approved analysis methodologies and demonstrated equivalency of identified design parameters insured that the loading pattern developed for Cycle 12 had no impact to any Design Basis Accident. The Cycle 12 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 12 RSE and Cycle 12 COLR demonstrate that the Cycle 12 Reload Design, including potential spring screw failure, 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 were introduced for any system, component, or piece of equipment as a result of the design. In addition, the safety functions of safety related systems and components related to accident mitigation were not altered. The implementation of the Cycle 12 Reload Design does not impact either the normal plant operation or the response to accident conditions. Therefore, the Cycle 12 Reload Design does not create the possibility of any new type of accident different from those already evaluated in the USAR.

The safety evaluation presented in the RSE and the Cycle 12 Reload Safety Analysis Checklist (RSAC) demonstrate that the probability of an accident previously evaluated in the USAR is not increased. Operation of WCGS in Cycle 12 with the introduction of the Region 14 fuel has been analyzed in accordance with methodologies reviewed and approved by the NRC. Demonstrated adherence to applicable standards and acceptance criteria preclude any new challenges to components and systems that could increase the probability of any previously evaluated accident or malfunction of equipment.

The primary accident of concern associated with the fractured top nozzle holddown spring screws is the Fuel Handling Accident (FHA). The improper latching of a fuel assembly by the fuel handling equipment due to the fractured top nozzle holddown spring screw issue has not occurred and by design, cannot occur with proper operation of the fuel handling equipment. Consequently, the probability of an FHA as described in the USAR does not increase. The potential effects of plant operation with fractured fuel assembly holddown spring screws has been evaluated including the potential for loose parts, potential effects on reactor internals, potential effects on fuel design and core, as well as associated systems and components as noted herein. In no case will this situation lead toward increasing the probability of an accident previously evaluated in the USAR.

The Cycle 12 Reload Design does not create any new release pathway nor does it affect any other system or component which provide accident mitigation functions required to limit a radiological release. There are no new accident scenarios different than those previously evaluated in the USAR which result from implementation of the Cycle 12 Reload Design and therefore, results of the current USAR analysis on radiological releases remain bounding.

Since the probability an FHA has not increased and the radiological consequences of an FHA have already been addressed in the USAR, there is no increase in consequences associated with the FHA associated with fractured holddown spring screws. The current radiological analysis for the FHA as described in the USAR remains valid and bounding. Additionally, the potential effects of fractured holddown springs on plant operation has been evaluated, and in no case will this situation lead toward increasing the consequences of an accident previously evaluated in the USAR.

The Cycle 12 Reload Design, including potential spring screw failure, does not result in the alteration of any original design specification nor does it result in a different response of safety related systems and components to accident scenarios postulated in the USAR. All design and performance criteria continue to be met and there are no new failure modes. No new equipment malfunctions have been introduced that will affect fission product barrier integrity. In addition, no new release pathway is created and there is no affect on the mitigation of the radiological consequences of an accident described in the USAR. Therefore, since the Cycle 12 Reload Design has no impact on either normal operation or accident response of the plant, it will not increase the consequences or the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

Likewise, the Cycle 12 Reload Design, including potential spring screw failure, does not create any failure mode that could adversely impact safety related equipment or cause the initiation of any accident. The design does not create any new 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 have not been altered. Therefore, the Cycle 12 Reload Design will not create the possibility of a malfunction of equipment important to safety different than those previously evaluated in the USAR.

The fracture of the top nozzle holddown spring screws will not have an impact on equipment important to safety as previously evaluated in the USAR. The key concern is with the generation of loose parts interfering with RCCA insertion. The likelihood of generating a loose part which causes a control rod to stick is extremely low. The failure of two or more rods to insert due to loose parts is not considered a credible event. The potential effects of plant operation with fractured holddown spring screws has been completely evaluated including the potential for loose parts, potential effects on reactor internals, potential effects on fuel design and core, as well as associated systems and components. This situation will not have any adverse safety impact on fuel handling operations either in the core or the spent fuel pool. Although latching difficulties may occur, no increase in the probability of a malfunction of equipment important to safety is created since once the fuel is properly latched, the latching tools will continue to perform their intended function. Therefore, the probability of a malfunction of equipment important to safety as evaluated in the USAR is not increased.

The holddown spring screw fractures have been evaluated for their effects on other components of the fuel assembly and interfacing components of the reactor internals. The evaluation concluded that potential upward displacement of the fuel assemblies would have negligible effects on core reactivity and power distribution. Further, the response of the fuel with potentially fractured spring screws to worst-case design mechanical loads (seismic and LOCA) would not result in unacceptable structural behavior. Hence, the fuel's nuclear and mechanical performance would not be affected.

Equipment failures that may affect fuel were also evaluated. The potential for loose parts would not have any direct impact on fuel integrity, nor are loose parts expected to indirectly affect the fuel by interfering with control rod insertion. The potential for other components to affect fuel integrity, such as damage to the fuel assembly alignment pins, was also evaluated. In all cases the results demonstrated acceptable performance of fuel assembly and internals components such that fuel failures would not be expected to occur as a result of spring screw failures. There is no effect on RCS pressure boundary integrity or on containment integrity, nor would the presence of fractured spring screws prevent any system or component from performing a mitigative function in the event of a postulated accident.

There is no effect on fission product barriers, the assumptions in the radiological consequences analyses, or the mitigation of the radiological consequences of an accident described in the USAR. Therefore, the consequences of a malfunction of equipment important to safety as evaluated in the USAR are not increased.

The Cycle 12 Reload Design, including potential spring screw failure, 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 12 Reload Design has no impact on inspections or surveillance required by the Technical Specifications.

The Cycle 12 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 12 Reload Design accounts for both normal operation and postulated accident conditions for WCGS. The LOCA evaluation demonstrated 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 2000-0057 **Revision:** 0

Revision to Site Specific Civil Structrual Design Criteria

Activity Description:

The site specific Civil Structural Design Criteria will be revised to be consistent with the latest revision of the design basis calculations that are also the bases for statements in the USAR. This change by itself would not affect any existing information in the USAR.

In addition, changes to the USAR were made that were identified as a compilation of potential discrepancies from section 2.4 and 3.4. These are corrections and changes for consistency. None of these changes represent a change to the design basis or licensing basis.

50.59 Evaluation:

The changes to the Civil Structural Design Criteria will make information in these design input documents consistent with statements in the USAR. The proposed changes to the USAR are corrections, clarifications or changes to make statements in one location consistent with other locations in the USAR. Therefore, this change does not make any information in the USAR no longer true or less accurate or would violate a statement in the USAR.

The site flood analysis for safety related structures, systems, and components was reviewed for potential impact by this proposed change. The proposed change to the Civil Structural Design Criteria does not result in any new criteria or criteria that is less severe than that which has been previously reviewed and approved by the NRC during the licensing phase of WCGS. The changes to the USAR tables also do not result in any new or less severe information regarding the site flood analysis. Instead these USAR changes make information in one table consistent with other tables or USAR text. This change does not impact the site flood analysis or any of the other design basis accidents discussed or referenced in the USAR. Therefore, this change would not increase the probability of occurrence of an accident, nor would it increase the radiological consequences of an accident previously evaluated in the USAR.

The changes to the site flooding criteria in the Civil Structural Design Criteria and USAR would not create any type of credible accident since the change does not result in any flood design criteria less conservative than that which has already been reviewed and approved during the licensing phase of WCGS. The USAR changes are corrections, clarifications or changes to make statements consistent with other information in the USAR. They do not change the content or meaning of any information in the USAR. Therefore, this change could not create the possibility of an accident of a different type than any previously evaluated in the USAR.

This activity does not directly change anything that could result in a credible malfunction of equipment important to safety. This change does not result in any new site flood design criteria or evaluation for any equipment less conservative than that which has already been previously reviewed and approved. Therefore, there are no credible malfunctions of equipment important to safety that this change could affect. This change could not increase the probability of occurrence of, or the consequences of, a malfunction of equipment important to safety previously evaluated in the USAR, nor could it create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

The site flood design limits are the only acceptance limits contained in the licensing basis documents that could be affected by this change. However, the changes do not allow any site flood design criteria less conservative than that which has already been reviewed and approved by the NRC during the licensing phase. Therefore, this change does not affect any acceptance limits contained in the licensing basis documents, and does not reduce the margin of safety as defined in the basis for any technical specification.

Safety Evaluation: 59 2000-0059 **Revision:** 0

USAR Figure Correction to Reflect Hand Indicating Switch Logic

Activity Description:

USAR Figure 7.6-3 reflected an incorrect Hand Indicating Switch (HIS) configuration for HIS EJHIS8811A, EJHIS8811B and BNHIS8812A and BNHIS8812B. The HIS configuration was shown as having an "AUTO" push button as part of the HIS, which was not correct. The figure also contained logic wiring associated with the "AUTO" push button, which was not in conformance with plant configuration. Field walkdown confirmed that the corresponding plant electrical drawings displayed the correct HIS configuration and wiring. The incorrect configuration shown in USAR figure 7.6-3 never existed at WCGS. This change corrected the HIS configuration and also removed the associated logic wiring shown in USAR Figure 7.6-3 that was not in conformance with plant configuration.

50.59 Evaluation:

USAR Section 6.3.2.8 describes the functions associated with these hand switches. All functions described in USAR Section 6.3.2.8 remain unchanged. The HIS installed on the Main Control Board are momentary contact type switches and function equivalent to the design described in the USAR. All automatic functions occur.

This change was made to correct the USAR so that it reflects the configuration of the plant. This was a document change only. No procedures, structures, systems, or components were impacted by this change.

This change did not modify, add or remove any SSCs from the plant or from procedures. This change did not affect any design function, controlling functions, nor any evaluation. This was a document change only that corrected the USAR and brought it into agreement with the plant configuration. This change did not affect any design basis accidents. Therefore, this change would not increase the probability of occurrence of an accident, nor would it increase the radiological consequences of an accident previously evaluated in the USAR.

The proposed change required no field work, tests, experiments or procedure changes. This change corrected typographical and drafting errors, and did not affect plant configuration or operation. Therefore, this change could not have created the possibility of an accident of a different type than any previously evaluated in the USAR.

Additionally, since no SSCs (Safety Related or Non Safety Related) or procedures were affected directly or indirectly by this change, no credible malfunctions of equipment important to safety were affected. Therefore, this change could not increase the probability of occurrence of, or the radiological consequences of, a malfunction of equipment important to safety previously evaluated in the USAR. Likewise, since no SSCs or procedures were affected, this change could not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

The change did not alter any design basis limits, nor did it impact, directly or indirectly, any design functions or operations of the plant. Therefore, no acceptance limits contained in the bases for the technical specifications are affected by this change and the margin of safety is not reduced by this change.

Safety Evaluation: 59 2000-0060 **Revision:** 0

Procedure Change to Allow for Nitrogen Purge of Reactor Coolant Drain Tank

Activity Description:

Procedure SYS HB-203, Revision 13, was issued to allow for a nitrogen purge of the Reactor Coolant Drain Tank (RCDT) prior to the chemical degassing of the RCS. The nitrogen purge will dilute the hydrogen contained in the gas space in the RCDT to a safe level to prevent a hazardous mixture from being created after chemical degassing.

Chemical degassing of the RCS is an alternative method to mechanical degassing, both of which reduce the RCS hydrogen concentration to a specified level. Chemical degassing of the RCS is performed in accordance with other plant procedures to support plant cool down during refueling outages.

USAR Section 11.3.2.3 (Shutdown and Degassing of the Reactor Coolant System) only described the mechanical degassing method. Therefore the USAR is being revised to include a description of the chemical degassing process to provide a more complete description and to provide consistency with plant procedures.

50.59 Evaluation:

Engineering Evaluation Request (EER) 92-BB-08 identified chemical degassing as an alternate or augmenting method of degassing the RCS to specified levels during plant cool down. This method was included in approved plant procedures. Procedure SYS HB-203, Revision 13 provides a detailed method of purging the RCDT with nitrogen to support the chemical RCS degassing process. The changes do not impact any procedures, activities, administrative controls, or sequences of plant operations, nor are any plant structures, systems, components or equipment (SSCs) 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.

This change corrected the inconsistencies between the USAR Gaseous Waste Management System Description and the plant operating procedures. The addition of detailed instructions to SYS HB-203 to perform the nitrogen purge of the RCDT is consistent with the requirements of EER 92-BB-08 for chemical degassing. There is no additional impact on the performance of plant activities or any affect on any SSC. The degassing process is not associated with Design Basis Accidents relating to hydrogen as described in USAR Chapter 6, since the dissolved hydrogen in the RCS does not contribute to hydrogen levels in the containment. The degassing process is designed to prevent hazardous gas accumulations within the RCS itself while the plant is shutdown. No Design Basis Accidents were impacted. Therefore, this change would not increase the probability of occurrence of an accident, nor would it increase the radiological consequences of an accident previously evaluated in the USAR.

Chemical degassing is designed to achieve the same hydrogen levels on cool down as the mechanical degassing method, so it is bounded by the same analysis. Therefore, no new variables or parameters were introduced as a result of this change that could create the possibility of a different type of accident than previously evaluated in the USAR.

The RCDT has no safety design basis. The back pressure for the Reactor Coolant Pump (RCP) seals is not adversely affected because RCDT normal pressure ranges are maintained. No new missile potential has been created by the use of nitrogen bottles. Overpressure protection of the RCDT has not been affected. These changes make no additional changes to the plant, do not affect the performance of plant activities, and do not affect any SSC. No credible malfunctions of equipment important to safety are directly or indirectly affected by this change. Therefore, this change could not increase the probability of occurrence of, or the radiological consequences of, a malfunction of equipment important to safety previously evaluated in the USAR, nor could it create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

No acceptance limits contained in the bases for the technical specifications are affected by this activity and the margin of safety is not

Safety Evaluation: 59 2000-0061 **Revision:** 0

Revision to Procedure Controlling Bus Interlocks

Activity Description:

Procedure SYS SL-200 transfers the SL-2 bus to the switchyard SL-8 alternate source. Revision 4 adds detail to strengthen the configuration, which prevents crosstie of PA02 with switchyard sources and to remove/restore an additional fuse block to allow closing the alternate source breaker. The procedure continues to temporarily disable breaker interlocks and provide the necessary administrative controls to prevent the simultaneous energization of the SL-2 bus from both the switchyard SL-8 and PA02 buses.
The normal power supply for the SL-2 (1SL002E) bus is the PA02 bus (Breaker PA00210). When PA02 is de-energized for maintenance activities, the SL-2 bus needs to be transferred to the switchyard SL-8 bus by closing the 13-42 breaker. Performing this action using the control room handswitch is prevented by protection relays. A dead bus transfer is performed by first de-energizing the SL-2 bus, placing the PA00210 breaker handswitch in the pull to lock position, racking down the PA0210 breaker, and affixing an information tag alerting personnel to the abnormal alignment. The fuses in the coil circuits of the 227BSL2X and 259BSL2A/B relays are then pulled, disabling the relays, until the 1SL002E13-42 breaker is closed. Once this breaker is closed and bus voltage is normal the fuses to both relays are reinstalled.

50.59 Evaluation:

USAR section 8.3.1.1.1.1 states that electrical interlocks exist to prevent both the PA-02 and SL-8 buses from being closed on the SL-2 bus at the same time. During normal operation, electrical interlocks between these two breakers do exist. However, when performing the transfer, these interlocks have to be removed for a short period of time. Once, the bus is powered from SL-8 no electrical isolation will exist to prevent the PA00210 breaker from closing. The isolation will be maintained by proceduralized administrative controls. By performing this procedure, the electrical isolation device used during normal operation will be removed (and reinstalled).

The USAR describes "electrical interlocks prevent interconnection of the onsite auxiliary power system with the Burlington normal source or the switchyard station power normal source." The Design Basis Accidents in chapter 2, 3, 5, 6, 9, and 15 have been reviewed for potential impact by this procedure. None of the accidents reviewed are impacted by this procedure. There are no credible accidents that the implementation of this procedure could create. The safety related buses are not affected by changing the power source of the SL-2 bus from PA02 to SL-8. The administrative controls placed on the PA0210 breaker and handswitch will prevent the two sources from being tied together. There are no credible malfunctions of equipment important to safety that are directly or indirectly affected by this procedure. Procedure SYS SL-200 will prevent the SL-2 bus from being powered from both the PA02 and SL-8 buses at the same time. There are no Acceptance Limits in the bases of the Technical Specifications that could be affected by the implementation of this procedure.

The procedure allows the use of an administrative control instead of an electrical interlock to prevent tying the PA02 and SL-8 buses together. This does not have any impact on the existing acceptance limits. Therefore, the margin of safety has not been affected.

Safety Evaluation: 59 2000-0063 **Revision:** 0

Valve and Piping Replacement in the Makeup Demineralizer Acid Feed System

Activity Description:

Temporary modification 00-018-WM allows replacement of some piping and fittings with 316L SS material in the sulfuric acid system. The existing piping and valves are extensively corroded even though they are lined to give enhanced corrosion resistance. Like-for-like parts are not available for replacement. Air operated valves 1WM-952 and 1WM-953 will be replaced with manual valves. Valves 1WM-952, 1WM-953, and associated piping are part of the Makeup Demineralizer Acid Feed System. Valves 1WM-952 and 1WM-953 function to direct concentrated sulfuric acid from the acid pumps to the process water stream for dilution to proper working strength or to a drain. The valves originally had air operators because they were installed inside an acid berm and the remote operation capability precluded operators having to enter the berm to operate them.

50.59 Evaluation:

The impact of installing manual valves is that operators will have to enter the acid berm area to open and shut the valves. Satisfactory operator safety is maintained. There is a master timer/sequencer which can run a regeneration cycle "automatically" and remote operation is useful for this. The "automatic" cycle will still work with the intervention of the operator in steps that require manipulation of the valves. The "automatic" cycle is not the preferred method of regenerating the beds and is not used often. The resin regeneration procedures are revised to change the method of opening and shutting the valves. This temporary modification does not alter the design function of the Makeup Demineralizer Acid Feed System.

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 Makeup Demineralizer Acid Feed System is non-safety related, has no safety design basis, and is not referenced or controlled by the technical specifications, no acceptance limits are affected by this temporary modification. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0064 Revision: 0

Minor Revisions to the Analysis of the Inadvertent Boron Dilution Transient

Activity Description:

A USAR error related to the analysis assumption associated with the inadvertent boron dilution transient in Mode 2 was identified during Margin Management review. The current USAR description in Section 15.4.6.3, under headline "Dilution During Startup" indicates that the reactor is assumed to trip when the flux level reaches the power range neutron flux high setpoint for Mode 2 startup operation. This description is inconsistent with the current licensing basis analysis as documented in Calculation No. AN-00-019, which shows that the reactor is assumed to trip once the flux level reaches the source range high flux trip setpoint.

50.59 Evaluation:

The proposed change makes the related USAR text description in Section 15.4.6.3 consistent with the design and licensing basis calculations that form the basis of the USAR description. The proposed change would not affect how SSC design functions are performed or controlled as described in the USAR, because no procedures, activities, administrative controls, sequences of plant operations, plant structures, systems, components or equipment are impacted by the proposed change.
There are no credible accidents which could be created by incorporating clarification or correction.
No credible malfunctions of equipment important to safety could be identified which may be affected by incorporating this correction.
The proposed change do not affect the manner regarding how safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected. The Margin of safety has not been reduced.

Safety Evaluation: 59 2000-0065 **Revision:** 0

Replace Temperature Control Butterfly Valve for the Letdown Heat Exchanger with a Globe Valve

Activity Description:

BGTV130, Letdown Heat Exchanger Outlet Temperature Control Valve, will be replaced to improve automatic control of letdown temperature so as to minimize associated reactivity changes. The current 6" butterfly valve and control components will be replaced with a 6" globe valve with restricted trim and with associated enhanced control components. It is anticipated that the replacement globe valve will allow effective control of the letdown temperature within a more limited operational band without resort to operator intervention to control letdown temperature and minimize.

During operational system transients, the temperature control valve for the letdown heat exchanger (BGTV130), has a very slow response time. For example, starting the ESW system typically results in a CCW temperature reduction. Subsequently, the letdown temperature is reduced until BGTV130 responds after a sufficient deviation has existed for a period of time. By the time of the BGTV130 response, a substantial error has accumulated such that the temperature control valve will overshoot making it not uncommon to receive a letdown high temperature alarm. Given sufficient time (30-45 minutes), it is anticipated that the system would correct itself, however, variations in the letdown temperature result in reactivity changes due to boron absorption effects in the Mixed Bed Demineralizers. In transient situations, the reactor operator may bypass the demineralizers, take manual control of BGTV130, or both, to control the letdown temperature and minimize reactivity changes.

50.59 Evaluation:

The temperature control valve for the letdown heat exchanger does not have an active safety function. The replacement 6" globe valve will have the same passive safety function (pressure boundary) as the original 6" butterfly valve.

The change will not change any administrative controls that would reduce the effectiveness of existing programs, reduce the qualification of WCNOC personnel, nor does it affect any systems, structures, and components other than that described. The change will not change the performance of activities that are important to the safe and reliable operation of WCGS.

The design function will not be altered by the valve replacement and no procedures, activities, administrative controls, sequences of plant operations, or requirements are impacted by the change and thus other USAR information and requirements is not invalidated.

It is not anticipated that the change will adversely affect the mitigative capability of any

SSCs, nor affect the ability of any SSC to prevent an accident. Examination of design basis accidents discussed or referenced in the USAR, including an examination of USAR Chapters 2, 3, 5, 6, 9, and 15, indicate that no design basis accidents credit the operation of this valve. Therefore, there is no potential impact due to the replacement of the 6" butterfly valve by a 6" globe valve.

There is no design function, methods of operation change, or credible accident scenarios created by replacing the 6" butterfly valve by a 6" globe valve. Since the changes do not involve a change to a design function nor are there changes in the method by which any safety-related plant system performs its safety function, no credible malfunctions of equipment important to safety are identified.

The changes do not affect the I-131 dose equivalent limits for the specific activities of the primary and secondary coolant. The changes do not affect the manner regarding how safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected and the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0066 **Revision:** 0

Addition and Deletion of Site Support Buildings

Activity Description:

The proposed change includes the following:

· Construction of a 100' X 200' concrete pad and metal building north of the Paint Shop. This is located in the southwest corner of the Site Plan, USAR Figure 1.2-44.
· The existing temporary Kelly buildings (identified as pole barns in the north yard area on USAR Figure 1.2-44) are to be removed after the construction of the 100' X 200' building is complete.
· Construction of a small office/restroom building, approximately 24' X 20', just east of the Paint Shop. The existing building will be removed and a new replacement concrete pad and building will be installed. This facility will be equipped with a sewage holding tank and will not be tied into the site sanitary sewer system.
· Addition of several existing miscellaneous structures, e.g., vehicle search building, radiography building, to USAR Figure 1.2-44. These are all support structures and do not affect power production.

50.59 Evaluation:

The proposed activity does not change any administrative controls that would reduce the effectiveness of existing programs, reduce the qualification of WCNOC personnel, nor does it affect any systems, structures, and components other than that described. The proposed activity does not change the performance of activities that are important to the safe and reliable operation of WCGS.

No procedures, activities, administrative controls, sequences of plant operations, or requirements are impacted by the change and thus the proposed activity would not invalidate other USAR information and requirements. It is not anticipated that the proposed activity would adversely affect the mitigative capability of any SSCs, nor affect the ability of any SSC to prevent an accident.

Examination of design basis accidents discussed or referenced in the USAR, including an examination of USAR Chapters 2, 3, 5, 6, 9, and 15, indicate that no design basis accidents could be affected by the construction of the new buildings or removal of the Kelly buildings. The locations are separated from all safety related SSC's, and none of the buildings contain or will contain safety related equipment. Since rainwater will run directly off the roofs and onto the ground and the site flood analysis regarding local intense Probable Maximum Precipitation is based on 100% runoff, this change cannot increase the maximum runoff. The location of the buildings will not adversely affect the previously analyzed flow paths for runoff from the power block area. Thus this change cannot affect the existing site flood analysis in any way.

There is no design function or methods of operation change, credible accident scenarios created, nor credible malfunctions of equipment important to safety created by the proposed activity.

The proposed changes do not affect the manner regarding how safety limits or limiting safety system settings are determined, nor will there be any effect on those plant systems necessary to assure the accomplishment of control and protection functions. Therefore, no acceptance limits are identified that could be affected and the margin of safety is not decreased.

Safety Evaluation: 59 2000-0067 **Revision:** 0

Removal of Abandon Pipe During CCW Pipe Repair

Activity Description:

A pipe weld crack was found during RF-11 on the cooling water outlet nozzle of heat exchanger EBG01, the let down heat exchanger. Subsequent inspections showed that there are weld cracks in down stream piping also. These sections of CCW piping will be replaced. During this corrective action rework, two existing abandoned pipe nipolets will be permanently deleted from line BG034HBC-6. One of these Nipolets is labeled line BG033HBC-3/4 and located on both P&ID's M-12BG02 and P&ID M-12EG03. The second is a 1" nipolet located 180 degrees from temperature indicator BGTI0005. This second nipolet does not appear on either P&ID. All work will be performed to ASME Section XI. All replacement piping is purchased as Safety Related Material.

50.59 Evaluation:

The pipe fluid flow will be improved and local corrosion will be eliminated where these short dead end sections of pipe will be permanently removed. Outage time will be saved by not having to replace these abandoned nipolets when the main pipeline is replaced.

No design basis accidents are affected. For emergency condition following a LOCA, CCW removes heat from the RHR heat exchangers, RHR pump seal coolers, centrifugal charging pump oil coolers, safety injection pump oil coolers, fuel pool cooling heat exchangers, and the post accident sampling system. Appendix 5.4A discusses the systems needed to go from hot shutdown to cold shutdown. The CCW heat removal function is required both short term and long term. This heat removal function will not be affected.

The sections of CCW piping will be replaced with the plant in mode 6 and the reactor de-fueled. The replacement piping will be like for like material and pipe class designation, HBC. The deletion of a capped 1" and ¾" pipe nipolet will not compromise pressure boundary integrity nor create any credible accident scenario.

No malfunctions of equipment important to safety are affected. The deletion of capped pipe nipolets will improve pipe fluid flow in the immediate area and reduce local corrosion inside the nipolets due to the dead legs. The new replacement piping will maintain pipe pressure boundary per original design.

Removing or maintaining abandoned pipe nipolets is not discussed in the technical specifications and does not affect any acceptance limits. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0068 **Revision:** 0

Replacement of Shell Side Outlet Nozzle on Letdown Heat Exchanger

Activity Description:

The existing 8" diameter shell side cooling water outlet nozzle on the letdown heat exchanger (EBG01) will be replaced with a new 12" diameter outlet nozzle. The new nozzle is fabricated from 12" diameter schedule 40 carbon steel pipe ASME SA-106 grade B. The length of the new nozzle will extend 8" from the shell wall instead of the existing 6": One 12"x 8" concentric pipe reducer and one 8" x 6" concentric pipe reducer conforming to HBC pipe class shall be installed between the new nozzle and the existing piping to achieve proper fit-up. The expected effect of this modification will be the removal of defects that were identified in the heat exchanger shell by radiography. The new nozzle will be attached to sound material without requiring additional base metal repair. The new nozzle will be designed in accordance with the ASME Section III Code.

50.59 Evaluation:

The heat exchanger and system are expected to perform their design basis function in the same manner as before.

Chapter 15 accidents were reviewed but none are found to be affected. Pipe break hazards were reviewed but none are affected. The change will not affect any accident previously evaluated in the USAR.

The proposed change will not impact any DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The replacement of the letdown heat exchanger cooling water outlet with a larger size and installation of an additional piping reducer will not cause any credible accident or malfunction of equipment important to safety.

The manner in which the CVCS system is operated will be consistent with previous operation of the equipment. The change does 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 change to the letdown heat exchanger does not adversely affect any acceptance limits contained in the bases of the Technical Specifications. There are no safety equipment performance parameter changes associated with these changes. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0069 **Revision:** 0

NRC Emergency Telecommunications System

Activity Description:

The NRC is implementing an upgrade to its Emergency Telecommunications System (ETS). The existing Federal Telecommunications System, the FTS-2000 network, will be replaced with an FTS-2001 network. The NRC has concluded that conversion to the FTS-2001 network will provide a potential efficiency enhancement for communications between the NRC and a nuclear power reactor site both during an emergency and for normal operations. Licensees have been requested to provide all necessary actions to support installation and operation of the NRC emergency telecommunications system upgrade.

FTS-2000 designations appear on a number of WCGS drawings, including one that is also USAR Figure 9.5.2-1. All references to an FTS-2000 NRC phone system will be replaced with a generic reference to an "NRC ETS LINE".

This change is a document change only. No WCGS communications equipment is affected by this change. This switchover is scheduled to occur on October 27, 2000.

50.59 Evaluation:

The proposed activity described above involves changes to USAR Figure 9.5.2-1. References to "NRC ENS LINE" and "NRC ENS FTS2000" will be replaced with "NRC ETS LINE". The proposed drawing revisions change the designation of the incoming NRC emergency telecommunications lines, and do not affect any WCGS communications equipment. Consequently, this proposed change does not have the potential to impact any design basis accidents that are discussed or referenced in USAR Chapters 2, 3, 5, 6, 9 or 15.

The proposed change does not affect any safety related equipment and does 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. The proposed change will not affect any safety related functions. There are no possible failure modes created by this change which 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.

There is no affect on any SSCs. Therefore there can be no credible malfunctions of equipment important to safety.

Components of the NRC Emergency Telecommunication System are not in the technical specifications, technical specification bases, or technical requirements manual. Consequently, no acceptance limits are affected and the margin of safety is not decreased.

Safety Evaluation: 59 2000-0070 **Revision:** 0

Valve Monitoring Change

Activity Description:

The proposed modification alters the method of monitoring the position of a single valve in the fire protection system. Two methods of monitoring isolation valve positions are allowed by the USAR in paragraph 9.5.1.2.2.2. Isolation valve 1FP037 is currently monitored electrically by a position switch. This method has failed because of a cable fault. This modification will change the method of monitoring to administrative control and visual monitoring. The current alarm output will be deactivated and the valve locked in its normal open position. The valve will be added to the Locked Valve Log and monitored monthly.

USAR Figure 9.5-1 shows that the status of 1FP037 position switch is shown on a control room fire protection display window. That information will be removed from the drawing. There is no text description of the valve 1FP037 status monitoring so no additional changes are required for the USAR.

50.59 Evaluation:

Valve 1FP037 is not Safety Related, It is not considered in and can have no impact on any Design Basis Accidents.

Valve 1FP037 cannot create any credible accidents. Its function is to allow water flow in the fire protection system external to the Power Block.

The design function and normal OPEN position of 1FP037 will not be changed by this modification.

There are no acceptance limits affected by this change to the method of monitoring the position of valve 1FP037. Therefore, the margin of safety is not affected.

Safety Evaluation: 59 2000-0072 Revision: 0

USAR Revisions to Correct Errors Made During the Initial Update

Activity Description:

The following USAR changes will correct errors, clarify meanings, and improve accuracy. These are document changes only. Neither plant equipment nor operation is affected.

· USAR Table 3.2-1 will be revised to correct errors that occurred during the preparation of the Draft USAR and for consistency with the information contained in USAR Figures 10.4-8 and 11.2-1. The classification information for piping/valves for the Liquid Radwaste System derived from USAR Figure 11.2-1 was inserted in the incorrect system during the preparation of the Draft USAR. In addition, classification information for penetration piping was omitted during the same process and a typographical error was made in the Quality Assurance classification for Class 3 piping. The proposed change to Table 3.2-1 corrects these errors to accurately reflect the design of the Liquid Radwaste System and the Steam Generator Blowdown System as shown in USAR Figures 10.4-8 and 11.2-1.
· USAR Section 10.4.8.2.2 will be revised to replace the expression "as an installed spare" with "as a backup". The proposed change in terminology more accurately describes the functions of the two Steam Generator Blowdown Discharge Pumps without affecting the intended meaning.
· USAR Section 10.4.8.2.3 will be revised to read "< 190°F" to indicate that the outlet temperature from the regenerative heat exchanger is maintained at a maximum temperature of 190°F during normal operations.

50.59 Evaluation:

The change will clarify the description of the normal operation of the Steam Generator Blowdown System and has no effect on the design or function of the Steam Generator Blowdown System or any of its components.

The changes do not impact any other document, activity or SSC described in the USAR. There are no changes to equipment or the way in which equipment is operated. There is no impact on the performance of plant activities or affect on any SSC. No Design Basis Accident is identified as affected.

No credible accidents or equipment malfunctions will be created or affected.

No acceptance limits are identified that could be affected. Therefore, the margin of safety in not affected.

Safety Evaluation: 59 2000-0073 Revision: 0

Temporary Procedure in the Event the Start-up Transformer (XMR01) Fails While the Unit Aux Transformer (XMA02) is Out of Service.

Activity Description:

The activity being evaluated is the implementation of procedure SYS SL-336, "Energizing Bus SL2 and PA02 from Alternate Power Source". This is a temporary procedure to be implemented in the event the Start-up Transformer (XMR01) fails while the Unit Aux Transformer (XMA02) is out of service. This condition is not the same as a Loss of Offsite Power since power is available in the switchyard. This procedure provides the necessary administrative controls to allow energization of the PA02 bus from bus SL-8 via bus SL-2. Due to the size of installed cabling, the power feeding back to PA02 from SL-2 will be limited so that only a selected group of loads will be powered.

The selected load group will assist in dissipating heat from the secondary side by re-establishing an operational condenser. This heat dissipation is beyond that which is required and provided by the ESFAS on the primary side. This additional heat rejection capacity is desirable as a means of preserving Condensate Storage Tank inventory by reducing the amount of steam released through the Atmospheric Relief Valves during plant cooldown. Although it is not required from a nuclear safety aspect (the plant can be safely shutdown with either safety bus NB01 or NB02), this additional capacity for cooldown provides more operational margin and a means to reduce the loss of secondary side process fluids. The selected loads are:

1 Circulating Water Pump (D1CW001PC)
1 Condensate Pump (DPAD01B)
2 Service Water Pumps (D1WS001PA & B)
1 Closed Cooling Water Pump (DPEB01B)
1 Start-up feedwater Pump (DPAE02)
1 PG Battery Charger (PJ-21)
1 Circulating Water Screen House Air Compressor (SL4A3RB)
2 Condenser Vacuum Pumps (CCG01A & B)
1 Demineralized Water Transfer Pump (AN Tank) (DPAN01B)
Water Treatment System Power
Aux. Boiler (Steam Seals)
Miscellaneous 480 volt loads required to support operation of the above equipment

Procedure SYS SL-336 temporarily disables breaker interlocks and provides the necessary administrative controls to provide energization of the PA02 bus from the switchyard SL-8 bus through the SL-2 bus. The normal power supply for the SL-2 bus is the PA02 bus (Breaker PA00210). When PA02 is de-energized due to a loss of the start-up transformer XMR01, the SL-2 bus can be transferred to the switchyard source and then utilized to backfeed PA02.

Once this configuration is obtained, the above listed loads can be started and the steam dumps can be used to remove secondary side heat. The evaluation includes starting the pumps involved with condenser operation in the worst case sequence for voltage drop analysis.

50.59 Evaluation:

By performing this procedure, the electrical isolation device used during normal operation will be removed and reinstalled. The USAR describes that "electrical interlocks prevent interconnection of the onsite auxiliary power system with the Burlington normal source or the switchyard station power normal source."

With neither XMA02 nor XMR01 supplying power the physical plant would be operating under the condition known as "Loss of Offsite Power". The Design Basis Accidents in chapter's 2, 3, 5, 6, 9, and 15 have been reviewed for potential impact by this new procedure. None of the accidents reviewed are impacted by this new procedure. The SL-2 bus will be de-energized and electrical interlocks will be removed.

There are no credible accidents that the implementation of this procedure could create. The safety related buses are not affected by changing the power source of the SL-2 bus from PA02 to SL-8. The administrative controls placed on the normal supply breakers to PA02 will prevent crosstie of out-of-phase electrical sources.

There are no credible malfunctions of equipment important to safety that are directly or indirectly affected by this procedure. Equipment important to safety is powered off of buses NB01 and NB02 which are not affected by this procedure.

There are no Acceptance Limits in the bases of the Technical Specifications that could be affected by the implementation of this procedure. Therefore the margin of safety is not reduced.

Safety Evaluation: 59 2000-0074 **Revision:** 0

Deletion of Alarm on Charcoal Adsorber

Activity Description:

USAR Sections 9.4.1.2.3 and 9.4.2.2.3 will be revised to eliminate the reference to a 400°F charcoal bed high temperature alarm. The corresponding statement in M-10GG will also be revised to be consistent with USAR Section 9.4.2.2.3. The alarm does not exist, although the instrument range includes 400°F. The 400°F alarm described in USAR Section 9.4.2.2.3 serves no purpose since all corrective action is performed at 200°F and 300°F. Therefore, it has been determined that the description of a 400°F alarm is an error that will be corrected.

50.59 Evaluation:

Procedures ALR 00-061E and ALR 00-062F/063F/062E/063E for filter adsorbers FGG02A/B and FGK01A/01B/02A/02B respectively, indicate that if the adsorber unit temperature exceeds 200°F, the affected adsorber unit is isolated. The affected adsorber unit temperature is then checked and if the unit temperature is not stable or decreasing, or if the unit temperature is 300°F or higher, cooling water flow is initiated. Notes in all procedures indicate that the charcoal filter begins to release any built up radioactive material at temperatures greater than 300°F. The corrective actions are intended to maintain the charcoal filter temperature at or below 300°F to prevent the release of radioactive material.

The proposed changes will correct an error in the descriptions of filter adsorber high temperature alarms contained in USAR Sections 9.4.1.2.3 and 9.4.2.2.3 and System Description M-10GG. The change is consistent with other USAR sections and actual plant operation. The 400°F alarm has no function because all corrective actions are taken at more conservative alarm temperatures. The changes do not impact any procedures, activities, administrative controls, or sequences of plant operations. No plant structures, systems, components or equipment are 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. No credible accidents that could be created were identified. No credible Malfunctions of Equipment Important to Safety are identified. A review of the Acceptance Limits contained in the bases for the technical specifications indicates that none were impacted. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0076 **Revision:** 0

USAR Revision to Fire Test Reports

Activity Description:

The USAR, Section 9.5.1.2.2.3.b provides a list of fire tests that confirm the integrity of the fire penetration seals at WCGS. Four new test reports are being added to the WCGS document control system. The proposed USAR revision is to remove the references to the fire tests and substitute a general statement that reads, "The results of the tests, which confirm the integrity of the fire penetration seals, are documented in various fire test reports."

Removal of specific reference to fire test reports is allowed under Regulatory Guide 1.181, which endorses NEI 98-03, Rev. 1. NEI 98-03, Rev. 1 Section A4.1, Removing Excessive Detail, states in part, "Detailed text and drawings may be removed from the UFSAR to the extent that the information provided exceeds that necessary to present the plant design bases, safety analyses and appropriate UFSAR description."

The four new fire tests that are used to confirm the integrity of the fire penetration seals at WCGS are being approved. The USAR lists existing fire tests used to qualify fire barrier penetration seals. The approval of these tests will add four additional fire tests to the WCGS document control system that are not listed in the USAR. Therefore, implementation of the proposed activity will add tests not listed in the USAR.

50.59 Evaluation:

A review of the SER, through Supplement 5, revealed no reference to the specific fire test reports. The SER does reference specific test criteria which penetration seals are required to be designed to meet. However, the specific test reports are not necessary to present the plant design bases and may be deleted under the provisions of NEI 98-03, Rev. 1.

The design basis accident to be reviewed is Fire. The USAR, Section 9.5.1.1.1 states, in part, "The basic fire protection for safety-related items is achieved by fire inception avoidance and through remote separation of systems serving the same safety function or by fire barriers between such installations."

The proposed activity provides reasonable assurance that fire barrier penetration seals installed in barriers used to separate redundant fire safe shutdown components are qualified by established fire tests. This provides reasonable assurance that a fire on one side of the barrier will not spread to the other side and render the redundant components inoperable.

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 are created.

Since the proposed activity would not affect the system's failure mode, the system's 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 fire test reports, therefore, no acceptance limits could be affected and the margin of safety is unchanged.

Safety Evaluation: 59 2000-0077 **Revision:** 0

Radiological Emergency Response Plan Revision 2

Activity Description:

The Radiological Emergency Response Plan (RERP) is being changed to delete commitments that have been achieved. Additional changes are incorporated into the RERP to reflect duty title changes that are listed in the Technical Specifications. Other changes to the RERP include a change to the description of "Coffey County Ambulance" to "Coffey County Emergency Medical (EMS)" and a change in description of the Federal Telecommunication System (FTS 2000) to Emergency Telecommunications System (ETS).

50.59 Evaluation:

The changes described above do not impact Emergency Plan procedures or processes, nor do they decrease the effectiveness of the RERP. Therefore there are no USAR design basis accidents that could be impacted by these changes.

Since no Emergency 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. 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, therefore, there is no impact on acceptance limits and the margin of safety has not been reduced.

Safety Evaluation: 59 2000-0078 **Revision:** 0

Emergency Action Level Revision

Activity Description:

The Emergency Action Levels (EALs) are being revised to provide identified enhancements and delete commitments. A change is made to add words in the bases for several different EALs to allow for the manual closure of Containment (CTMT) Phase A isolation devices before declaring an event. Operators are trained to try every method possible while performing emergency procedures (EMG) including placing the equipment in the status directed by the EMGs. If the equipment cannot be placed in the desired position, then the classification is made. The readings for equipment is being added to the bases to avoid the need to reference other document to determine the classification criteria has been met due to instrument readings.

Commitment references are being removed that have been determined to no longer be applicable.

50.59 Evaluation:

Safety Analysis Engineering has examined and verified the upper range indications of the continuous containment atmosphere radioactivity monitors GT-RE-31 and GT-RE-32. The upper range indication values for both GT-RE-31 and GT-RE-32 are E-2 micro-Ci/cc for gaseous radioactivity and E-6 micro-Ci/cc for Iodine radioactivity. These upper range monitor indication values are consistent with the values presented in the WCNOC Total Plant Setpoint Document (pages 605 to 607), USAR Table 11.5-3, and also the calibration documentation.

Assuming the instantaneous release and dispersal of the reactor coolant noble gas and Iodine inventory associated with normal operating concentrations in the containment atmosphere, an additional examination of the upper range are acceptable based on the industry guidance documents. In conclusion, an additional examination of the GT-RE-31 and GT-RE-32 upper range indication values determined they are acceptable.

The changes described above do not impact Emergency Plan procedures or processes, nor do they decrease the effectiveness of the RERP. Therefore there are no USAR design basis accidents that could be impacted by these changes. Since no Emergency 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. 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. Therefore, the margin of safety has not been reduced.

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. Karl A. (Tony) Harris, Manager Regulatory Affairs, at Wolf Creek Generating Station, (620) 364-4038.

COMMITMENT	Due Date/Event
None	N/A

LIST OF ACRONYMS USED

AC	Alternating Current
AMSAC	(ATWS) Mitigation System Activation Circuitry
APCSB	Auxiliary and Power Conversion Systems Branch
ASME	American Society of Mechanical Engineers
BIT	Boron Injection Tank
BTP	Branch Technical Position
CACS	Containment Air Coolers
CCN	Calculation Change Notice
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CeCWS	Central Chilled Water System
CIS	Containment Isolation Signal
CDF	Core Damage Frequency
CPIS	Containment Purge Isolation Signal
CSAS	Containment Spray Activation Signal
CST	Condensate Storage Tank
CTMT	Containment
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DWMS	Demineralizer Water Make-up System
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EER	Engineering Evaluation Request
EMG	Emergency Procedures
EQ	Equipment Qualification
ESFAS	Engineered Safety Feature Activation System
ESW	Essential Service Water
ETS	Emergency Telecommunications System
FBIS	Fuel Building Isolation Signal
FHA	Fuel Handling Accident
HIS	Hand Indicating Switch
HVAC	Heating, Ventilation, Air Conditioning
IFM	Intermediate Flow Mixing
ILRT	Integrated Leak Rate Testing
IPS	Integrated Plant Scheduling
ISEG	Independent Safety Engineering Group
IST	In-Service Testing
ITIP	Industry Technical Information Program
L&CA	Licensing & Corrective Action
LOCA	Loss of Coolant Accident
LOSP	Loss of Off Site Power
MIC	Micro-biologically Induced Corrosion

MSLB	Main Steam Line Break
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSE	Nuclear Safety Engineering
NSSS	Nuclear Steam Supply System
P/T	Pressure/Temperature
PM	Preventative Maintenance
PSA	Probabilistic Safety Assessment
PSRST	Primary Spent Resin Storage Tank
PSTT	Portable Stem Travel Transducer
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RERP	Radiological Emergency Response Plan
RFA	Robust Fuel Assembly
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SER	Safety Evaluation Report
SFPE	Society of Fire Protection Engineers
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIS	Safety Injection Signal
SSC	System, Structure, Component
SSE	Safe Shutdown Earthquake
SSED	Security System Explosive Detectors
SUT	Start-Up Transformer
TMO	Temporary Modification
TMP	Temporary Procedure
TRM	Technical Requirements Manual
TS	Technical Specifications
UAT	Unit Auxiliary Transformer
USAR	Updated Safety Analysis Report
USQD	Unreviewed Safety Question Determination
VCT	Volume Control Tank
VOC	Volatile Organic Content
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation