WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard. A. Muench
Vice President Technical Services

MAR 11 2002

ET 02-0012

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: Docket No. 50-482: Wolf Creek Generating Station Annual 50.59 Evaluation Report

Gentlemen:

This letter transmits the Annual 50.59 Evaluation Report for Wolf Creek Generating Station (WCGS), which is being submitted pursuant to 10 CFR 50.59(d)(1)(2). Attachment I provides a summary of the evaluation results. Attachment II provides the WCGS Annual 50.59 Evaluation Report.

This report covers the period from January 1, 2001, to December 31, 2001, and contains a summary of 50.59 evaluations performed during this period that were approved by the WCGS onsite review committee.

There are no commitments contained in this correspondence.

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,

Richard A. Muench

RAM/DMH/pb

Attachments

cc: J. N. Donohew (NRC), w/a
D. N. Graves (NRC), w/a
E. W. Merschoff (NRC), w/a
Senior Resident Inspector (NRC), w/a

IE47

WOLF CREEK NUCLEAR OPERATING CORPORATION

Wolf Creek Generating Station

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

ANNUAL 50.59 EVALUATION REPORT

Report No.: 17

Reporting Period: January 1, 2001 through December 31, 2001

SUMMARY

This report provides a brief description of changes, tests, and experiments performed at Wolf Creek Generating Station and evaluated pursuant to 10 CFR 50.59(c)(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, 2001 and ending December 31, 2001. This report is submitted in accordance with the requirements of 10 CFR 50.59(d)(1)(2).

On March 13, 2001, Wolf Creek Nuclear Operating Corporation implemented the revised 10 CFR 50.59 regulation. One evaluation has been generated and included in this report that was performed in accordance with the revised rule.

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 either not to involve an unreviewed safety question or require a license amendment.

Evaluation Number: 59 1999-0057 Revision: 0
Title: Updated Safety Analysis Report (USAR) Revision to Clarify Inservice Inspection Examinations

Activity Description

This activity clarifies the USAR to reflect the performance of in-service inspection examinations with the unit operating. Performance of ISI examinations is non-destructive to equipment. Approved plant processes control the plant configuration. Therefore, there are no expected effects to the plant as a result of this activity.

50.59 Evaluation

Performance of ISI examinations while the plant is on-line has no impact on the probability of occurrence of an accident described in the USAR. Performance of ISI examinations is not a credible initiator for any accident, whether the exam is performed with the plant on-line or shut down. ISI examinations do not manipulate plant systems, structures or facilities. ISI examinations do not affect the initial conditions or accident dynamics as described in the USAR. Therefore, there are no design bases accidents discussed or referenced in the that are impacted by this activity.

Whether an ISI exam is performed with the unit shut down or the unit on-line is inconsequential to radiological consequences of accidents described in the USAR. ISI exams are non-destructive and are not credible initiators of accidents nor do they impact the accident dynamics as described in the USAR. Therefore, ISI exams have no impact on accident consequences.

There are no new credible accidents that could be created by the performance of ISI exams on-line. ISI exams are non-destructive, and do not manipulate plant equipment or facilities in any fashion. If equipment must be removed from service to perform the ISI exam, the plant configuration is controlled in accordance with plant procedures. ISI exams are non-destructive, and do not direct equipment manipulation or facility alteration. Therefore, performance of ISI exams will not create an accident of a different type, regardless of whether the exams are performed with the unit shut down or on-line.

ISI exams are non-destructive and do not direct equipment manipulation therefore; the possibility of equipment malfunction as a result of ISI exams is not credible. Performance of ISI exams is intended to reduce the probability of equipment malfunctions by detecting service-induced defects prior to failure. Performance of the exams with the plant on-line or shut down does not impact that probability.

Whether an ISI exam is performed with the unit shut down or the unit on-line is inconsequential to radiological consequences of malfunctions of equipment important to safety as described in the USAR. ISI exams are non-destructive and are not credible initiators of equipment malfunction, nor do ISI exams impact malfunction dynamics as described in the USAR. Therefore, ISI exams have no impact on radiological consequences due to equipment malfunction.

There are no new credible malfunctions of equipment important to safety as a result of ISI exams. ISI exams are non-destructive, and do not require system or equipment manipulation.

The ASME Section XI acceptance criteria for the specific ISI exams are not being altered by this change. Technical Specification 4.0.5 (and Bases 4.0.5) requires that the ISI program be administered in accordance with ASME Section XI. The acceptance criteria for indications found through ISI examination are located within Section XI and are not altered by this USAR change. Since no acceptance criteria are being altered or changed, the margin of safety is not impacted.

Evaluation Number: 59 1999-0157 Revision: 0
Title: System Changes To Boron Recycle System, Liquid Radwaste System, And Secondary Liquid Waste System

Activity Description

The activity reflects changes to the Boron Recycle System, Liquid Radwaste System, and Secondary Liquid Waste System.

- The Recycle (HE) Evaporator will be considered permanently removed from service. The Secondary Waste (HF) Evaporator will also be considered permanently removed from service. The Recycle Evaporator, Secondary Waste Evaporator, and related equipment including the boron recycle radioactivity monitor, HERE0016, are permanently out of service. Improved technology installed over the past several years by various change packages and temporary modifications has made it more effective to process reactor coolant effluent as liquid radwaste rather than to recycle it. HERE0016 monitors the flow path from the recycle evaporator condensate pump to the reactor makeup water storage tank. A high radiation signal from HERE0016 redirects flow back to the recycle evaporator feed demineralizers. The flow path from the recycle evaporator to the reactor makeup water storage tank reactor is and will continue to be isolated by locked closed valves. Because the recycle and secondary liquid waste evaporators are not operated, they place no heat load on the component cooling water and auxiliary steam systems. Since evaporation is not being used to process liquid radioactive wastes, the Primary Bottoms Tank is no longer the limiting source for iodine. Instead it appears that maximum potential radioactive iodine inventory would be in the drum dryer installed by Temporary Modification 98-018-HB. Rather than perform a specific analysis for this temporary modification, a bounding analysis was performed and documented in calculation change notice AN-99-014-00-CN002. This bounding analysis assumes that all of the iodine entering the liquid radwaste system is contained in a hypothetical hold up tank with no iodine removal other than radioactive decay. Chapter 15.7.2 is being revised to replace the analysis of a Primary Bottoms Tank failure with a bounding hypothetical tank failure.
- The USAR stated function of the Boron Recycle System is being changed to indicate that RCS effluent (letdown and equipment drains) is processed as liquid radwaste rather than recycled into makeup water and boric acid. (The USAR was previously updated to reflect this change, but some sections were missed.) The stated function of the Boron Recycle System is being changed from "*receive and recycle RCS effluents by separation into boric acid and makeup water*" to "*receive reactor coolant effluent for the purpose of storage until it can either be reused or processed by the Liquid Radwaste System*".
- The current method for processing clean tritiated liquid radwaste is by filtration, reverse osmosis, and demineralization rather than by processing through the Liquid Radwaste (HB) Evaporator. The Liquid Radwaste Evaporator remains available for use if needed. (Again, the USAR has been previously updated to reflect some of these changes, but additional obsolete descriptions are being removed.) Statements which indicate tritiated waste streams are recycled into reactor grade makeup water and supplied to the reactor makeup water storage tank (RMWST) are being changed to indicate the RMWST is supplied primarily from the demineralized water system. The Liquid Radwaste (HB) Evaporator, while available, is generally not used. The Liquid Radwaste System using the Liquid Radwaste Processing Skid normally processes the bulk of the radioactive

liquid discharged from the reactor coolant system for discharge. The affected descriptions and data tables are being revised accordingly. The only radionuclide discharged to the environment in significantly increased quantities as a result of these changes is tritium. USAR Section 15.7.3 refers to section 2.4.13.3 which describes an analysis of a Liquid Radwaste Tank failure. Tritium levels in ground water above the 10 CFR 20 release limit at the point it enters the cooling lake are justified using the argument: "*the tritium concentration would be highly diluted by the uncontaminated waters of the cooling lake*". Because tritium (half life 12.3 years) is present in the lake due to operation of the plant, it is not accurate, strictly speaking, for the USAR to call the lake uncontaminated in this context. Calculation AN-99-029 was performed to document an analysis considering both the higher normal tritium release rates due to not recycling letdown and release of the Refueling Water Storage Tank to ground water. The conclusions of AN-99-029 indicate that the lake tritium concentration still remain well below the 10 CFR 20, Appendix B, Table II, Column 2 limit of 1×10^{-3} μCi/ml even if the Refueling Water Storage Tank is released to ground water after the lake has attained equilibrium tritium levels due to continuous release of all the tritium released to the reactor coolant system during normal operation. Note that the USAR incorrectly states that the 10 CFR 20, Appendix B, Table II, Column 2 limit for tritium is 3×10^{-3} μCi/ml. The USAR is being revised accordingly. The term "Liquid Radwaste Demineralizer Skid" is being replaced with "Liquid Radwaste Processing Skid" and the description is being revised to indicate that it may consist of a series of components employing various processes which may include filtration, reverse osmosis, and/or demineralization.

- Heat tracing is made permanently out of service if it is associated with piping that is permanently out of service.
- Actual release data is contained in Annual Radioactive Effluent Release Reports filed with the NRC in accordance with Offsite Dose Calculation Manual (ODCM) requirements. USAR statements indicating that source terms are calculated using the GALE code are being removed. The GALE code is no longer used for updating source terms. Chapter 11 tables supporting chapter 15 and Chapter 2 accident analyses will continue to be maintained and are not considered historical.

The most significant radiological effects are:

- Since evaporation is not being used to process liquid radioactive wastes, the Primary Bottoms Tank is no longer the limiting source of iodine in the radwaste tank rupture analysis. A bounding analysis was performed and documented. This analysis shows an increase in the calculated Exclusion Area Boundary thyroid dose from 0.124 rem to 1.49. This increase is much less than 10% of the available margin and therefore, is not considered an increase in consequences.
- Since RCS effluent is processed as liquid radwaste, rather than recycled, more tritium is released to the cooling lake. A calculation was prepared to evaluate and justify this change. This calculation shows that cooling lake tritium concentration will remain below 3×10^{-5} μCi/ml which is well below the applicable 10 CFR 20 limit of 1×10^{-3} μCi/ml.
- Minimal hardware changes are being made. These changes consist of lifting leads, pulling fuses, and removal of the Recycle Evaporator Concentrates Pump.

50.59 Evaluation

USAR described accidents considered in this activity are:

1. Airborne release due to a waste gas decay tank failure (USAR section 15.7.1).
2. Airborne release due to a liquid radwaste tank failure (USAR section 15.7.2). Two liquid tanks were analyzed and discussed in section 15.7.2.
   a) Boron recycle hold-up tank (RHUT) (highest total inventory of radioactivity)
   b) Primary evaporator bottoms tank (highest inventory of halogens)
3. Liquid release due to liquid tank failures (USAR section 15.7.3 which refers to USAR section 2.4.13.3). Three liquid tanks were analyzed and discussed in section 2.4.13.3.
   a) Primary spent resin storage tank
   b) Boron recycle hold-up tank (RHUT)
   c) Refueling water storage tank (RWST)

The relevant accidents identified are tank failures. None of the changes being evaluated increase the pressure or temperature, or otherwise adversely challenge the integrity of any tank identified above. Therefore, there is no increase in the probability of an accident previously evaluated in the USAR.

The source of all of the radioactivity in all of the tanks identified in the analysis above is the reactor coolant system (RCS). The changes described above under headings *1-a* and *1-b* actually tend to decrease RCS radioactivity levels, since make-up water and boric acid do not have the low levels of residual activity that would exist in the recycled products. The affects of the process changes on each relevant event will be discussed individually.

- <u>Waste Gas Decay Tank Failure</u> (USAR section 15.7.1).

  Waste gas decay tanks are used to permit the decay of fission product gases as a means of minimizing the release of radioactive materials to the atmosphere. The principal radioactive components of the waste gas decay tanks are the noble gases krypton and xenon, the particulate daughters of some of the krypton and xenon isotopes, and trace quantities of halogens. The main source of these is the RCS via volume control tank (VCT) purging. RCS radioactivity levels are not being increased, therefore the main source of the radioactivity in the tanks is not increased. Smaller sources of gasses include vents from the recycle evaporator gas stripper, the reactor coolant drain tank, the pressurizer relief tank, and the recycle holdup tanks. The only one of these sources affected is the vent from the recycle evaporator gas stripper, which is no longer used. Therefore the total amount of radioactive material assumed to be in a failed waste gas decay tank is still bounding and radiological consequences are not increased.

- <u>Airborne Release Due To A Liquid Radwaste Tank Failure</u> (USAR section 15.7.2).

  1. <u>Boron Recycle Hold-Up Tank</u>

     The USAR states this tank was selected for evaluation because of all of the tanks listed in USAR Table 11.1-6 it contains the maximum total inventory of radioactivity. All of the noble gases and 10% of the iodine in the tank are assumed to escape and become airborne. The amount of radioactivity assumed for the accident analysis is based on an assumed 1% fuel defects and the process flow depicted in USAR Figures 11.1A-2 sheets 1 and 2, which show an RHUT receiving RCS effluent at the rate of 2140 gallons per day (1840 gpd letdown shim bleed and 300 gpd from the reactor coolant drain tank). Although

USAR Figure 11.1A-2 sheet 2 is affected and is being revised, all changes are downstream of the RHUT. Therefore the source terms used for evaluating an RHUT rupture are still bounding and radiological consequences are not increased.

2. Primary Evaporator Bottoms Tank

The USAR states this tank was selected for evaluation because of all of the tanks listed in USAR Table 11.1-6 it contains the maximum total inventory of radioactive iodine. (Some demineralizers actually contain more, but it is considered fixed in the resin and not a credible airborne source.) Since evaporation is not being used to process liquid radioactive wastes, the Primary Bottoms Tank is no longer the limiting source. Instead it appears that maximum potential radioactive iodine inventory would be in the drum dryer installed by Temporary Modification 98-018-HB. Rather than perform a specific analysis for this temporary modification, a bounding analysis was performed and documented in calculation change notice AN-99-014-000-CN002. This bounding analysis assumes that all of the iodine entering the liquid radwaste system is contained in a hypothetical hold up tank with no iodine removal other than radioactive decay. The estimated Exclusion Area Boundary 0-2 hour thyroid dose due to releasing the equilibrium iodine curie content of this hypothetical tank is less than 1.5 rem. IE Circular 80-018 "10 CFR 50.59 Safety Evaluations For Changes To Radioactive Waste Treatment Systems" states that "radiological consequences of unexpected and uncontrolled gaseous releases of radioactivity stored in a waste system must be a small fraction of the 10 CFR 100 guidelines, i.e., less than 1.5 rem thyroid."

In addition, the USAR is revised to describe this scenario as having dose consequences that are a small fraction of the 10 CFR 100 guidelines (10 percent or 2.5 rem whole body and 30 rem thyroid). The 10 CFR 100 thyroid limit is 300 rem. An increase in dose of less than 10% of the available margin is generally not considered an increase in consequences.

- Liquid Release Due To A Liquid Tank Failures (USAR section 15.7.3 / 2.4.13.3).

Radioactive liquids from the plant are postulated to enter the ground water as a result of the accidental rupture of tanks. Concentrations in ground water at the point it would enter the cooling lake (or the tributary to Wolf Creek in the event the cooling lake is drained) are calculated and compared to 10 CFR 20 limits.

The rupture of each of three tanks is postulated as a separate isolated event. The tanks were selected based on radioisotopes of relatively long half-lives of concern to human health, Sr-90, Cs-137, Co-60, and H-3.

Highest curie contents for Sr-90, Cs-137, and Co-60 are in the Primary Spent Resin Storage Tank. The highest concentration of H-3 is in the Boron Recycle Holdup Tank, while the greatest curie content of H-3 is in the Refueling Water Storage Tank.

USAR Table 2.4-35 gives each tanks curie content for the important radionuclides used in the analysis.

1. Primary Spent Resin Storage Tank

(Sr-90, Cs-137, and Co-60)

Bechtel mechanical calculation HB-21 determined the curie content for this tank as given in USAR Table 11.1-6 sheet 17 and USAR Table 2.4-35. These values were based on the primary coolant activities determined by Bechtel mechanical

calculation HB-19 given in USAR Table11.1-1 and the process flow paths, flow rates, and decontamination factors shown on USAR Figure 11.1A-1 sheets 1 & 2. A careful review of these shows that virtually all of the calculated activity is from the CVCS Mixed Bed Demineralizers, the CVCS Cation Bed Demineralizer, and the Recycle Evaporator Demineralizer. Since the changes being evaluated are downstream of these demineralizers, the source terms used for evaluating a Primary Spent Resin Storage Tank failure are unaffected and radiological consequences are not increased.

2. Boron Recycle Hold-Up Tank; and
3. Refueling Water Storage Tank

(H-3)

Since the Reactor Water Make Up Tank is being supplied exclusively from uncontaminated sources, and not from the Recycle Evaporator, the tritium levels in the RCS, and therefore the tritium levels in these tanks, would be decreased due to the changes described above under heading *1-a*. This results in a reduction of the radiological consequences. However, the USAR analysis justifies tritium levels in ground water above the 10 CFR 20 release limit at the point it enters the cooling lake using the argument: "*the tritium concentration would be highly diluted by the uncontaminated waters of the cooling lake*". Because tritium (half life 12.3 years) builds up in the lake over the life of the plant, it is not accurate, strictly speaking, for the USAR to call the lake uncontaminated in this context. Calculation AN-99-029 was performed to document an analysis considering both the higher normal tritium release rates due to not recycling letdown and release of the refueling water storage tank to ground water as described in USAR section 2.4.13.3. IE Circular 80-018 "10 CFR 50.59 Safety Evaluations For Changes To Radioactive Waste Treatment Systems" states that radiological consequences of unexpected and uncontrolled liquid releases of radioactivity stored in a waste system must be less than the radionuclide concentrations of 10 CFR 20, Appendix B, Table II, Column 2 for liquid releases at the nearest water supplies. The conclusions of AN-99-029 indicate that the lake tritium concentration still remain well below the 10 CFR 20, Appendix B, Table II, Column 2 limit of 1×10^{-3} μCi/ml even if the refueling water storage tank is released to ground water after the lake has attained equilibrium tritium levels due to continuous release of all the tritium released to the reactor coolant system during normal operation.

No new types of credible accidents have been identified.

The equipment and systems being evaluated are not considered important to safety. USAR Table 11.2-2 identifies tank uncontrolled release protection provisions. The Recycle Evaporator is listed on this table. Since it will now be permanently isolated the potential for an uncontrolled release from it will be reduced. No new credible malfunctions have been identified.

USAR Table 11.2-2, "Tank Uncontrolled Release Protection Provisions", does list and discuss the Recycle Evaporator. Since the Recycle Evaporator will be isolated, the probability of an evaporator malfunction resulting in an uncontrolled release will be reduced.

USAR Table, 11.2-2, "Tank Uncontrolled Release Protection Provisions", does list and discuss the Recycle Evaporator. Since the Recycle Evaporator will no longer be used it will contain less radioactivity. Therefore, radiological consequences of an equipment malfunction will not be increased.

No new or unique procedures, processes, etc. are involved. No new types of credible accidents have been identified.

The changes all involve equipment and systems in the Radwaste building, which are physically distant and separated from equipment important to safety. Therefore, the possibility of a different type of a malfunction of equipment important to safety is not created.

No acceptance limits have been identified. Since no acceptance limits were identified, no related margins of safety have been defined.

Evaluation Number: 59 1999-0161 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the modification/replacement of Main Turbine lines with a low alloy steel (2 1/4 Cr - 1 Moly) This modification will mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC) and improve piping resistance to FAC.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed activity will restore degraded sections of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

50.59 Evaluation

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being associated with this change. Therefore, the probability of occurrence of an accident is not increased and radiological consequences are not increased.

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents could be created.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. There are no malfunctions of equipment important to safety identified. Therefore, the probability of occurrence of an equipment malfunction is not increased and the radiological consequences are not increased.

Since, the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are identified. Therefore, the margin of safety has not been decreased.

Evaluation Number: 59 1999-0162 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the modification/replacement of Main Turbine lines with a low alloy steel (2 1/4 Cr - 1 Moly). This modification will mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC) and improve piping resistance to FAC.

The proposed pipe replacement did not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses will remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed activity will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

50.59 Evaluation

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents are identified as being associated with this change. Therefore, the probability of occurrence of an accident is not increased and radiological consequences are not increased.

Since the proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings), no accidents could be created.

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that a new credible failure mode is not introduced. There are no malfunctions of equipment important to safety identified. Therefore, the probability of occurrence of an equipment malfunction is not increased and the radiological consequences are not increased.

Since, the proposed change will restore a degraded section of the affected piping system, to perform its original design intent, no acceptance limits which could affect the basis for any Technical specification are identified. Therefore, the margin of safety has not been decreased.

Evaluation Number: 59 2000-0011 Revision: 0
Title: Updated Safety Analysis Report Change to Address Plant Safety Review Committee Responsibilities

Activity Description

This activity changes the wording of two sections in the Updated Safety Analysis Report (USAR) that discuss Plant Safety Review Committee (PSRC) responsibility. These sections did not contain specific wording do describe the types of information that require PSRC review. Therefore, it was left to interpretation what these sections of the USAR referred to. Due to the generic in nature it could be understood that the review requirements included a greater set of information than was in the PSRC procedure. The sections are being revised to describe the actual intent of the review requirements in terms of the documentation associated to the specific regulations and or programs.

50.59 Evaluation

There are no design bases accidents discussed or referenced in the USAR that take credit for PSRC activities. Therefore the probability of an accident or the consequences of an accident have not been increased

This is an administrative change to PSRC review responsibilities. A change to these administrative sections will not create any credible accidents.

This change is administrative in nature. It will not affect equipment. Therefore, there will be no increase in equipment malfunctions or consequences of malfunctions. No new equipment malfunctions will be created.

There are no acceptance limits documented that this change would affect. Therefore, the margin of safety has not been reduced.

Evaluation Number: 59 2000-0049 Revision: 0
Title: Changes to Fuel Elevator and Fuel Transfer System

Activity Description

The proposed activity will correct the following USAR inaccuracies:

- USAR Figure 9.1-11, New Fuel Elevator, will be revised to accurately reflect the physical arrangement of the new fuel elevator drive system. This figure incorrectly shows the hoist mounted to the north of the elevator, with cables and sheave assemblies supporting this arrangement. The actual configuration is with the hoist to the east of the elevator. The actual orientation of the drive system is per design (reference M-716-00167-W07, Instruction Manual for New Fuel Elevator.
- USAR Section 9.1.4.2.2, Fuel Transfer Tube and Associated Components, Item b, Lifting arm (transfer car position), will be revised to accurately reflect the physical location of the mechanical latch device for the mechanical interlock which allows fuel transfer system lifting arm operation only when the transfer car is at the end of its travel. The USAR incorrectly states that the mechanical latch device is located on the lifting arm. The actual location in on the transfer car per design (reference M-716-00158-W16, Instruction Manual for Fuel Transfer System).

For each of these items, this 50.59 evaluation credits the design process for ensuring that the new fuel elevator drive system and the fuel transfer system mechanical interlock meets all appropriate design requirements.

50.59 Evaluation

Since the affected equipment handles irradiated fuel assemblies, the fuel handling accident is reviewed. No other design basis accident is applicable. The actual condition of the new fuel elevator and the fuel transfer system are per design requirements. Therefore, no credible accidents are identified as affected.

Neither the actual orientation of the new fuel elevator drive system nor the location of the fuel transfer system mechanical interlock increases the probability of dropping design loads. Therefore, the proposed change does not increase the probability of dropping a fuel assembly.

The fuel handling accident in the fuel building assumes that all fuel rods in the dropped assembly are ruptured. Since the new fuel elevator and the fuel transfer system can handle a maximum of one fuel assembly, there is no potential for increasing the radiological consequences of a fuel handling accident.

For the same reasons the activity does not create accidents of a different type than any previously evaluated.

The only equipment important to safety that interfaces with the new fuel elevator and the fuel transfer system is nuclear fuel. The actual orientation of the new fuel elevator drive system and the location of the fuel transfer system mechanical interlock is per design requirements. No credible malfunctions of equipment important to safety are identified.

The only safety related equipment that the new fuel elevator and the fuel transfer system potentially affects is nuclear fuel. The actual orientation of the new fuel elevator drive system

and the location of the fuel transfer system mechanical interlock has no impact on fuel assemblies, thus does not increase the probability of occurrence of a malfunction of equipment important to safety.

Either the new fuel elevator or the fuel transfer system may contain an irradiated fuel assembly. The actual orientation of the new fuel elevator drive system and the location of the fuel transfer system mechanical interlock do not affect the assembly in any manner, therefore the activity does not increase the radiological consequences of a malfunction of equipment important to safety.

For the same reasons, the activity does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated.

There are no acceptance limits in licensing basis documents are identified. Therefore, the change does not reduce the margin of safety.

Evaluation Number: 59 2000-0058 Revision: 0
Title: Replacement of Obsolete Instrumentation and Controls

Activity Description

This activity replaces obsolete equipment in the Condensate Demineralizer (AK) system that is no longer supported by the manufacturers and resolves several technical problems in the system. The following changes are included in the scope of the activity:

- Instruments and controls in panel AK182 will be replaced with a programmable logic controller (PLC) and a panel mounted industrial computer.
- Conductivity and pH analyzers in panel AK182 will be replaced with new Rosemount analyzers.
- The annunciators in panels HF133 and AK182 will be replaced with new Panalarm Series 90 annunciators.
- Flow transmitters will be replaced with Rosemount differential pressure transmitters with LCD displays (0-100% flow indication).
- The three valve manifolds used to isolate the flow transmitters will be replaced with five-valve manifolds.
- A pulsation dampener and isolation valves will be added to the Acid Regeneration Pump discharge header to reduce pressure spikes which cause the Acid Concentration Analyzer, AKCIS0312, to false alarm and the pumps to trip.
- A temperature sensor and transmitter will be installed in the Sluice Water supply pipe upstream of valve AKPCV0311.
- A digital display will be installed in panel AK182 to display the Sluice Water temperature.
- The circuitry for relays in panel AK182 will be modified so that the relays can be reset without removing the interior panel cover to depress a manual reset button.
- A latching relay will be installed in each circuit between hand-switches and the corresponding solenoid valves to prevent the solenoid valves from vibrating out of position.

The changes to be implemented within this activity will reduce system maintenance and improve operability. Overall system reliability is also expected to increase. However, the new PLC to be installed in panel AK182 may exhibit greater vulnerability to the nuclear power plant EMI/RFI environment and electrical transients than the existing Tenor drum controller. Installation and wiring of the PLC in accordance with the modification instructions and the manufacturer's literature is expected to minimize the PLC vulnerability.

50.59 Evaluation

The changes proposed by this activity do not affect the design basis accidents discussed or referenced in USAR. Therefore, the probability of occurrence or radiological consequences of an accident are not increased. The changes proposed by this activity do not create any new credible accidents.

The Condensate Demineralizer System (AK) serves no safety function. The equipment affected by this activity is not relied upon during or following design basis events to assure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shut down condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to 10 CFR 100.11 guidelines. The changes proposed to be implemented under this activity do not affect equipment important to safety. Therefore, the probability of malfunctions of equipment

important to safety is not increased. Consequences of equipment malfunctions are not increased tan the possibility of a different type of malfunction is not created.

The changes to be implemented by this activity do not affect the Technical Specifications or plant licensing basis documents other than the USAR figure changes previously identified. Therefore, no acceptance limits associated to those documents are affected, and the margin of safety has not been reduced.

Evaluation Number: 59 2000-0079 Revision: 0
Title: Modification of Spent Fuel Pool Bridge Crane

Activity Description

The objective of this activity is to enable modification of the Spent Fuel Pool Bridge Crane (HKE04). The seismically designed modification, temporarily adds an extension on the north side of the existing crane. The modification is necessary to enable fuel-handling access to the north two rows of the new spent fuel storage racks that were installed as part of another modification. The extension will be installed, using structural steel bolting, for a short duration to enable access to the otherwise inaccessible storage cells of the new racks for post-installation drag testing as well as storage of spent fuel assemblies into those cells. The existing 2-ton motorized monorail trolley will be relocated onto the extension structure (monorail). Once the post-installation drag testing and the storage of spent fuel assemblies in these presently inaccessible storage cells have been completed, the 2-ton motorized monorail trolley will be replaced back to the south monorail beam, and the crane extension will be removed from the Spent Fuel Pool Bridge Crane. The only permanent change will be the additional bolt holes used for attaching the crane extension to the SFP bridge crane. The local change to the cross section of the existing members have been evaluated and found acceptable. The temporary configuration has been analyzed to insure the crane will maintain its structural integrity during a Safe Shutdown Earthquake (SSE).

The functional requirements of the Spent Fuel Pool (SFP) Bridge Crane are identified in the USAR. The existing 2-ton motorized monorail trolley is primarily used for moving fuel assemblies in the fuel storage pool area. The existing 2-ton motorized monorail trolley will be temporarily relocated onto the temporarily added monorail beam on the north side of the SFP Bridge Crane. As such, the SFP Bridge Crane will be operated within the same parameters as presently defined in the USAR. The USAR describes interlocks between the bridge crane and the new fuel elevator, fuel transfer canal gate and the cask loading gate, a limited maximum lift height (to maintain fuel shielding), a weight-operated hoist upper limit switch in addition to the geared-type upper and lower limit switches, and that the SFP Bridge Crane to be controlled from a pendant station supported from the trolley. Since the existing 2-ton motorized monorail trolley will be re-located to the north monorail, all of these functions will be retained with the hoist on the north monorail.

The Spent Fuel Pool Bridge Crane (HKE04) will be modified by temporarily adding a monorail beam on the north side of the bridge crane. The existing 2-ton electric hoist will be also temporarily relocated, onto the temporary monorail beam. This temporary change in the configuration of the Spent Fuel Pool Bridge Crane will be a configuration different than presently depicted on USAR Figures.

50.59 Evaluation

The fuel handling accident was reviewed. It consists of dropping of a spent fuel assembly onto the fuel storage area floor, resulting in the rupture of the cladding of all the fuel rods in the assembly despite many administrative controls and physical limitations imposed on fuel handling operations.

There are no new or different types of credible accidents that the proposed modification to the Spent Fuel Pool Bridge Crane could introduce since the same administrative controls and physical limitations remain in effect as prior to the modification. Fuel assemblies will be

handled with the same motorized monorail trolley hoist utilizing the same controls and procedures as presently defined in the USAR. Therefore there is no increase in the probability of occurrence or no increase of radiological consequences of an accident previously evaluated in the USAR.

The proposed change has been analyzed to ensure structural integrity during an SSE. Therefore, there are no credible malfunctions of equipment important to safety that will be changed or consequences that will be altered by the proposed modification to the Spent Fuel Pool Bridge Crane.

Fuel assemblies will be handled with the same motorized monorail trolley hoist utilizing the same controls and procedures as presently defined in the USAR. Therefore, the proposed modification to the Spent Fuel Pool Bridge crane will not create the possibility of an accident or equipment malfunction of a different type than any previously evaluated in the USAR.

There are no acceptance limits in the USAR that could be affected by the proposed modification to the Spent Fuel Pool Bridge Crane. Therefore, the margin of safety has not been reduced.

Evaluation Number: 59 2001-0001 Revision: 0
Title: Addition of Spool Piece to Replace Automatic Temperature Control

Activity Description

This activity deletes Service Water temperature control valve EATV0007 and associated instruments and adds a spool piece in its place. This activity will eliminate the possibility of EATV0007 binding closed. The spool piece will allow for full cooling water flow. However, the automatic temperature control performed by EATV0007 for maintaining generator hydrogen temperature will be lost. Therefore, Operations will throttle Service Water return isolation valve EAV0037 manually for the temperature control. Lake temperature does not change rapidly and the Generator load is fairly constant. Due to these conditions, Operations can maintain the Generator Hydrogen temperature. Replacing EATV0007 with approximately 18 inches of spool pipe and throttling EAV0037 does not change flow significantly. This modification should not cause any erosion on the hydrogen cooler tubes or challenge the hydrogen temperature limits. The control room also has an alarm and generator hydrogen temperature indication. Operating procedures provide actions in response to abnormal conditions.

50.59 Evaluation

The transient describing a decrease in heat removal by the secondary system on a turbine trip and the initiation signal of generator trip was reviewed for potential impact. Once the generator trips, the results of the turbine trip remain the same. Generator hydrogen temperature control is not relied on for this event. Therefore, this activity will not increase the probability or consequences of an accident described in the USAR.

Current design postulated that EATV0007 will fail open and EAV0037 is throttled to maintain generator gas temperature. Exchanging EATV0007 with a spool piece on the generator hydrogen cooling line mimics this condition. Therefore, this activity does not create any credible accidents.

This is a non-safety related system. Changing EATV0007 to a spool piece in the generator hydrogen cooler line reduces valve malfunctions and does not increase cooler degradation. Therefore, it does not cause or create any credible malfunctions of equipment important to safety. Nor will it increase the consequences of a malfunction of equipment.

There were no acceptance limits associated to this activity. Therefore the margin of safety has not been reduced.

Evaluation Number: 59 2001-0002 Revision: 0
Title: Compensatory Measures for Loss of Sprinkler and Spray Systems

Activity Description

This activity revises the compensatory measures for out of service containment fire suppression systems. The revision allows the verification of operable fire detection in the area and requires no additional compensatory measures if detection is operable.

This activity also adds compensatory measures for the loss of supervisory circuits for the valve for the underground fire main to be sealed or locked in its required position when the electrical supervisory circuit is demonstrated inoperable.

50.59 Evaluation

Fire protection features provided inside containment consist of manually charged sprinkler systems in the North and South cable penetration areas, a manually charged standpipe and hose system, portable fire extinguishers, automatic linear heat detection above each reactor coolant pump, automatic linear heat detection above concentrations of cable trays, and automatic ionization smoke detection inside containment cooler ducts. The automatic fire detection system alarms in the control room.

Separation of fire safe shutdown components within containment meets the intent of Appendix R, Section III.G.2. The Fire Hazards Analysis (USAR Appendix 9.5B) summarizes the fire safe shutdown methodology inside containment and concludes that no postulated fire from fixed or transient combustibles in the reactor building will prevent safe shutdown of the plant.

Compensatory action options of an hourly fire watch, 8 hour tours, or hourly temperature monitoring for inoperable fire suppression and hose standpipe systems inside containment are impractical and considered inferior to operable detection. An hourly fire watch inside containment while at power is not recommended due to ALARA considerations. Also, an hourly fire watch and 8 hour tours of containment are not as reliable as operable detection, since detection is present at all times. Monitoring containment air temperature is also not as reliable an indicator of fire as fire detection since the fire detection system will operate long before a noticeable increase in containment air temperature. Operable fire detection inside containment provides a more reliable and faster indication of fire than an hourly fire watch, 8 hour tours, or hourly temperature monitoring.

During normal operation, total reliance is placed on the fire detection system to initiate action. With an operable water supply and fire suppression systems, the only way to initiate fire suppression activities is by the automatic fire detection system. The currently analyzed condition specified that fire detection is required before a fire in containment is recognized. Therefore, relying on fire detection when the suppression systems or standpipe hose system is inoperable is not different than the normal operation of the plant.

In accordance with Wolf Creek Generating Station procedures, when a fire is detected by the automatic detection system, an operator is sent to investigate the source of the alarm. If there is a true fire emergency, then the Fire Brigade is called out. Operators only open the valve that supplies water to the hose stations and sprinkler systems in containment if directed by the Fire Brigade Leader. Not all situations require charging the fire protection piping inside containment.

It has been established that fire extinguishers are a suitable backup to hose stations inside containment during postulated fires in Modes 1 through 4. During Mode 5, the air lock can be breached. This allows a backup water supply to be brought into containment. The Fire Hazards Analysis (USAR Appendix 9.5B) describes the fire safe shutdown methodology for a fire in containment. An RCP lube oil fire is not postulated during power operation due to the oil collection system. Adequate separation exists between redundant components within the containment such that a fire in any one area will not impact the redundant train. Aside from RCP lube oil, the only significant fire hazard in containment during operation is cable insulation. The cable insulation is IEEE 383 rated. Tests have shown that IEEE 383 cable insulation does not propagate fire and self-extinguishes when flame is removed. Since the quantity of transient combustibles is low, and an RCP lube oil fire is not postulated, the probability of a significant fire of such magnitude that the cable insulation will continue to burn is very low. Under these postulated scenarios, it is reasonable to conclude that a sufficient alternate to hose stations for a fire inside containment is fire extinguishers. Therefore, a postulated fire inside containment can be extinguished with fire extinguishers in the absence of suppression and hose stations.

In addition to the above, one of the corrective actions for PIR 2000-2976 was to add compensatory measures for fire protection valve supervision out of service. This change adds the appropriate compensatory measures for loss of fire protection valve supervision. System 7, Fire Protection Valve Supervision will be added to USAR Table 9.5.1-3.

There are no design basis accidents affected by this change.

Since there are no physical changes and the design basis function of the system is not affected by this change, no new types of accidents not previously analyzed could be created. In addition, the probability of occurrence of an accident, or consequences of and accident has not been increased.

Since the proposed change would not affect the system's failure mode, the systems design function, the level of qualification, or equipment important to safety, no credible malfunctions of equipment important to safety are identified. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits associated with these administrative changes. Therefore, no acceptance limits could be affected and the margin of safety has not been reduced.

Evaluation Number: 59 2001-0003 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of a six and eight inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0004 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of a twelve and twenty inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0005 Revision: 0
Title: Revision to Quality Auditor Qualifications

Activity Description

This activity takes an exception to ANSI N45.2.23 Section 2.3.4 which states "The prospective Lead Auditor shall have participated in a minimum of five (5) quality assurance audits within a period of time not to exceed three (3) years prior to the date of qualification, one audit of which shall be a nuclear quality assurance audit within the year prior to his certification." The exception will require prospective Lead Auditors to demonstrate their ability to effectively implement the audit process and effectively lead an audit team. The activity will also allow a prospective Lead Auditor to have participated in at least one nuclear quality assurance audit within the year preceding the individual's date of qualification. This activity will not affect the quality of the qualification requirements for a lead auditor. Changing the quantity of audits performed will not change the focus on the qualification process to demonstrate the ability of perspective lead auditors to effectively implement the audit process and effectively lead audits. Requirements of 10 CFR 50 Appendix B criterion for Audits will continue to be met. The nuclear industry has greatly expanded in the number and type of processes where the skills necessary for being an effective lead auditor can be obtained. None of these processes met the literal definition of an audit. However, candidates have opportunities to train in audit like activities which include internal self assessments, trending for the identification of recurring or emerging problems, INPO evaluations, Management oversight processes, and performance monitoring through indicators. Thus changing the quantity of involvement from 5 audits to 1 audit will not change the quality of the process for certifying a lead auditor. The requirements continue to require perspective lead auditors to demonstrate their ability to effectively implement the audit process and effectively lead an audit team.
This activity adds a new exception for Regulatory Guide 1.146. The exception states, "Prospective Lead Auditors shall demonstrate their ability to effectively implement the audit process and effectively lead an audit team. This process is described in written procedures which provide for evaluation and documentation of the results of this demonstration. A prospective Lead Auditor shall have participated in at least one nuclear quality assurance audit within the year preceding the individual's date of qualification. Upon successful demonstration of the ability to effectively implement the audit process and effectively lead audits, and having met the other provision of Section 2.3 of ANSI/ASME N45.2.23-1978, the individual may be certified as being a Lead Auditor."

50.59 Evaluation

This proposed activity does not impact the design basis accidents as discussed in the USAR since it affects the certification of Lead Auditors for internal and supplier audits. Audits are performed by observation and documentation of review and do directly not affect WCNOC structures, systems or components. There are no design basis accidents whose initiation or mitigation rely on auditor qualifications. There are no credible accidents that could be created as a result of this change since it affects the certification of Lead Auditors for internal and supplier audits. The abilities of an auditor have not been compromised by this change. Audit quality has not been reduced.
There are no credible malfunctions of equipment important to safety that may be directly or indirectly affected by the proposed activity. Since this change only affects the administrative requirements for internal audits the probability of an equipment malfunction and consequences have not been increased.

This activity does not affect the acceptance limits for the technical specifications that could be affected by this proposed change USAR since it affects the administrative requirements for internal audits. Therefore, the margin of safety has not been affected.

Evaluation Number: 59 2001-0006 Revision: 0
Title: Clarification to Technical Specification Bases Associated to Diesel Generator Jacket Water Keep Warm System

Activity Description

This activity revises the Technical Specification Bases to reflect that the OPERABILITY limit associated with the diesel generator jacket water keep warm system is 105°F instead of the low temperature alarm condition. It has been determined that the diesel generator is OPERABLE at the alarm set point of 135°F and that the diesel generator can still perform its specified safety function.

When the diesel generator is on standby, the jacket water keepwarm pumps circulate water through the electric heater and the cylinder jacket. This keeps the engine warm to facilitate starting. A temperature switch controls the heater. A failure of the keepwarm system will lower the jacket cooling water temperature. The jacket cooling water temperature is alarmed locally and annunciated in the control room as a common trouble alarm.

The jacket water keepwarm system runs continuously when the diesel generator is shut down, and contains a heater that cycles as required to maintain the temperature between 145°F and 150°F. The jacket water low temperature alarm is set at 135°F. Discussions with the diesel generator vendor identified that the diesel generator will perform its specified function below the alarm setpoint. This value was chosen based on operating experience as the value that supports quick starting of the diesel generator. The alarm setpoint is fixed at a lower temperature than normal operating temperature to avoid nuisance alarms while still providing warning should water temperature fall. The vendor manual states that each day the jacket water temperature should be verified to be a minimum of 105 °F. Therefore, the diesel generator remains OPERABLE as long as jacket water temperature is maintained above 105°F. At 105°F the diesel generator will still perform its specified function.

50.59 Evaluation

The activity will not alter the operation of any plant equipment, or otherwise increase their failure probability. Therefore the proposed changes to the Bases do not affect any design basis accidents. The proposed change would not impact the initial conditions of a design basis accident 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 design basis accident or transient; or Is not installed instrumentation used to detect and indicate a significant abnormal degradation of the reactor coolant pressure boundary. As such, the proposed changes would not affect design basis accidents or create a credible accident. The changes provide clarification of the existing requirements and consistency with the design and licensing basis. Therefore, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated. The changes also will not increase the consequences of an accident previously evaluated in the Updated Safety Analysis Report. The consequences of a previously analyzed event are dependent on the initial conditions assumed for the analysis, and the availability and successful function of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The changes will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients by the plant safety analysis or licensing basis. The changes do not impact the operability of components assumed to actuate when necessary for the prevention or mitigation of accidents or transients

or impact the plant variables necessary to satisfy the assumption for initial conditions in the safety analyses.

The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. These changes will not increase the probability of occurrence of a malfunction of equipment important to safety because they will not involve any physical changes to plant systems, structures, or components (SSCs). The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. The manner in which these SSCs are operated, maintained, or inspected continues to ensure equipment important to safety is maintained operable. The changes do not involve a physical alteration of the plant. No new equipment is being introduced, and no installed equipment is being operated in a new or different manner. There are no alterations to the parameters within which the plant is normally operated. No changes are being proposed to the procedures governing normal plant operation or those procedures relied upon to mitigate a design basis event. The changes do not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. As such, no new failure modes are being introduced. The changes will not increase the probability of occurrence of a malfunction of equipment important ot safety previously evaluated in the Updated Safety Analysis Report. The changes will not alter the operation of any plant equipment, or otherwise increase their failure probability. The probability that equipment failures resulting in an analyzed event will occur is unrelated to the changes. The changes also will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the Updated Safety Analysis Report. The consequences of a previously evaluated malfunction of equipment important to safety are dependent on the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated.

The proposed change to the Technical Specification Bases 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 these changes. No setpoints are adversely affected, and no changes are being proposed in the plant operational limits as a result of these changes. Since there are no changes to any acceptance limits, the margin of safety as defined in the bases of any Technical Specification is not reduced.

Evaluation Number: 59 2001-0007 Revision: 0
Title: GE Magne-Blast Circuit Breaker Replacement

Activity Description

This activity replaces the General Electric (GE) Magne-Blast circuit breakers in the 4.16 KV Class 1E switchgear (NB001) with Siemens vacuum circuit breakers. Due to differences in circuit breaker spring motor charging times (4 seconds for GE vs. 9 seconds for Siemens) the setpoint for the102/DG timing relay in the Diesel Generator breaker control circuit must be increased from 3 seconds to 4.5 seconds. Since the new set point is beyond the 0.2-4.0 second timing range of the existing relay, the 102/DG will be replaced with a new relay that has a range of 1.5-30.0 seconds.

50.59 Evaluation

The Loss of Coolant Accident (LOCA), loss of off-site power and LOCA combined with loss of off-site power accidents were reviewed. The circuit breakers and time-delay relay perform the same functions as the original devices. The proposed changes cannot create any credible accidents. The proposed modifications have no affect on the probability of occurrence of a LOCA, loss of off-site power or LOCA combined with a loss of off-site power. The proposed modifications do not affect the existing Wolf Creek accident analysis. Therefore, the radiological consequences of an accident previously evaluated in the USAR are unaffected. The proposed breaker and relay changes cannot create any credible accidents. Therefore, the proposed changes cannot create the possibility of an accident of a different type than previously evaluated in the USAR.

The circuit breakers are interrupting devices used for switching loads and the isolation of faults and overloads. Separate protective relays provide inputs to the circuit breakers for the fault and overload protection functions. When the diesel generator is in the automatic mode, the 102/DG timing relay is a permissive in the control circuit for the diesel generator output breaker. The relay creates a time delay between isolation of the NB bus and connection of the diesel generator to the bus. The purpose of the time delay is to allow back Electro Motive Force (EMF) from loads connected to the bus to decay prior to connection of the diesel generator. The only change in malfunctions of equipment important to safety is associated with the vacuum interrupters on the replacement Siemens breakers. If the connected load is a grounded circuit, the loss of vacuum in the interrupter will result in excessive arcing across the breaker contacts when they part to interrupt current flow. The arc would continue to exist until backup protection operates. The result would be destruction of the interrupter. The Siemens vacuum circuit breakers are expected to require less maintenance and to have greater reliability than the GE Magne-Blast breakers which they replace. The 102/DG timing relay which is being replaced is identical to the original except for timing range and setpoint. Therefore, the probability of occurrence of a malfunction of equipment important to safety will not increase. Replacement of the NB breakers, 102/DG timing relay and its setpoint change have no affect on the radiological consequences of a malfunction of equipment important to safety. Therefore, the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR will not increase. Although the loss of vacuum for the Siemens breakers is a different failure mode than for the GE breakers, the affect on breaker function is the same as for a GE breaker. If an arc chute and/or other portions of the arc quenching mechanism on a GE breaker fails to interrupt the arc, catastrophic failure of the breaker pole will result. For both the GE and Siemens breakers, failure to extinguish the load electrical arc will result in loss of the breaker. Therefore, the proposed changes will not create

a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

Technical Specification 3.8.1 requires that, upon loss of off-site power, the Standby Diesel Generator auto-starts from standby condition and energizes permanently connected loads in £ 12 seconds. The same time requirement is stated for loss of off-site power in conjunction with a Safety Injection signal (SIS). Increasing the 102/DG timing relay setpoint from 3 to 4.5 seconds will not cause the 12 second time limits to be exceeded. In the case of loss of off-site power, the plant will see no difference in diesel generator operation. The coil for the 102/DG relay is connected in parallel with the remainder of the control circuit for the diesel generator breaker. Upon loss of off-site power, shedding of the NB bus loads and isolation of the bus, the diesel generator is started and the 102/DG relay is energized. The relay will time out and its contacts close at 4.5 seconds. Since the diesel generator will take 6-10 seconds to reach the required voltage and speed, the change in relay operating time has no affect on diesel generator operation. In the case of loss of off-site power concurrent with a LOCA safety injection signal, the Load Shedding/Emergency Load Sequencing (LSELS) loads could be affected. The diesel generator will start upon occurrence of the SIS signal. If the loss of off-site power occurs after the diesel generator has already reached the required speed and voltage, the time delay between shedding NB bus loads and connecting the sequencer step 1 loads back on the bus will increase from 8 seconds to 9.5 seconds. This is still below the 12-second maximum time requirement.

Evaluation Number: 59 2001-0008 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of a six and twelve inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0009 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of a portion of a six inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0010 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of degraded sections of a ten inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0011 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of degraded sections of six and eight inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0012 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of a degraded portion of a sixteen inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0013 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of degraded portions of a six and eight inch line in the Feedwater Heater Extraction, Drains and Vents system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0014 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of degraded sections of a four inch line in Auxiliary Steam system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0015 Revision: 0
Title: Piping Replacement Due to Flow Accelerated Corrosion

Activity Description

This activity provides guidelines for the replacement of degraded portions of two fourteen inch lines in the Feedwater system, to mitigate abnormal pipe-wall thinning due to Flow Accelerated Corrosion (FAC), with a low alloy steel (2 1/4 Cr - 1 Moly) which improves piping resistance to FAC.

The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the geometric configuration. The mechanical properties such as tensile strength and code allowable stresses remain unchanged. The lower yield strength and higher Young's Modulus is judged to have insignificant impact on the original analysis. Therefore, the change does not adversely affect the existing safety margins or structural integrity of the affected piping system. The piping stresses will remain acceptable within code allowables.

The proposed replacement does not adversely affect any system, component or procedures required to mitigate the consequences of an accident previously evaluated in the USAR. The proposed change will restore a degraded section of the affected piping system, to its original design configuration (piping geometry, cross section, support location, fittings).

Ductile fracture, corrosion, erosion/corrosion, loss of mechanical properties, excess strain, mechanical creep etc., are credible failure modes for which the proposed piping replacement has been evaluated, through a critical characteristics comparison to the existent piping system design. Based on the evaluation, it was concluded that no new credible failure mode is introduced.

The proposed change will restore a degraded section of the affected piping system, to perform its original design intent. The proposed replacement does not involve or affect any safety related system or component. All system functions will continue to be performed as designed.

50.59 Evaluation

No accidents are identified as being associated with or impacted by this change. The proposed change will restore a degraded section of the affected piping system to its original design configuration (piping geometry, cross section, support location, fittings).
Therefore, the probability of occurrence of an accident, or consequences of and accident has not been increased. No credible accidents could be created.

No credible malfunction of equipment important to safety are identified which could be associated with or affected by this change. The proposed pipe replacement does not change the cross sectional properties (section modulus, moment of inertia), or the mechanical properties (tensile strength, and/or code allowable stresses), or the geometric configuration of the piping. Therefore, the probability and consequences of a malfunction of equipment important to safety has not been changed. No new malfunctions have been created.

There are no acceptance limits contained in the bases of the technical specification or licensing basis documents that could be affected by this change. Therefore, the Margin of safety has not been reduced.

Evaluation Number: 59 2001-0016 Revision: 0
Title: 7300 Modification for Feedwater Control Valve Loops

Activity Description

This activity is applicable to the control loops for the feedwater control valves. Control loop modifications will be made in the 7300 process control cabinets. Computer inputs currently spared will be used in cabinet RJ049. Driver cards in each control loop of feedwater control will be removed and two tracking driver cards will be installed. The new cards will be installed in parallel. In the event of a failure the control will default to manual with the control signal going to the feedwater control valves. They will remain at the same level as when last in automatic. The feedwater control valve will fail as is and the operator can then control it in manual from the manual station in the control room.

An analog signal is provided across a 250 ohm resistor (1-5 Volts) to a 7300 NCI1 input card. This signal is routed to RJ049 by installed spare wires to spare computer inputs.

In addition, there are 4 lead/lag cards in the loop that are set with a gain of 1 and delay to 0 seconds that provide no useful purpose. These cards will be removed and the pin to pin wiring modified.

This modification does not change normal operation in automatic or manual of the feedwater control valves because the tracking driver cards being installed provide the same operational functions. However, post modification-testing process of the control loops will be revised to demonstrate that the specific card failures have been addressed. This does not affect any testing discussions in the USAR for the 7300 process equipment.

50.59 Evaluation

In the Loss of Normal Feedwater accident it is assumed that feedwater control fails due to a pipe break. The analysis does not assume that feedwater control valves are available. Though the feedwater control valves close on a Feedwater isolation signal they are not safety related or credited in the design bases accident discussion. The feedwater control valves provide a non-safety backup function to the safety related isolation function of the feedwater isolation valves. The analysis for feedwater system malfunctions that result in an increase in feedwater flow is attributed to the loss of two feedwater control valves due to control failures (the valves fail open). However, the control failures that would result from the activity addressed by this evaluation would be to fail the feedwater control valves closed. This change does not affect these analyses, the feedwater isolation valves or the backup function of the feedwater control valves.
This change will not create additional accidents. It will fail feedwater control position as-is for certain driver or control card failures (such as blown fuse). Therefore, there are no new accidents created, and the probability of an accident or its consequences have not been increased.

This change is in the non-safety control portion of the 7300. The functions for the feedwater control valves are not safety related. Loss of feedwater and feedwater addition accidents do not take credit for feedwater control valve operation. Equipment important to safety is not affected. Therefore, there are no new equipment malfunctions created, and the probability of an equipment malfunction or its consequences have not been increased.

Discussion of feedwater control valves is in Bases B 3.7.3 as part of feedwater system. However, there are no acceptance limits associated to these valves that are affected by this activity. Therefore, the margin of safety has not been affected.

Evaluation Number: 59 2001-0017 Revision: 0
Title: Addition of Isolation Valve to Nitrogen System

Activity Description

The proposed activity includes addition of a 3/4" NUPRO isolation valve that will function as a second isolation valve between the nitrogen system pressure and the pressure sensing bellows of the nitrogen pressure controller (BGPC8155). BG8155 is a 1" Fisher Gas Process control valve that functions to regulate the pressure of nitrogen supplied to the Volume Control Tank during refueling outages. The pressure downstream of BG8155 is sensed by the BGPC8155 pressure controller via 3/8" tubing. The BGPC8155 modulates the BG8155 valve as needed to maintain ~25-30 psig N2 nom. The BGPC8155 Pressure Controller has a history of failure due to over-ranging of its Bellows Sensing Element. When the BG8155 control valve is not in service, nitrogen pressure of approximately 125 psig is applied to the sensing element. Per the vendor technical manual, a brass construction bellows has a maximum allowable static pressure limit of 40.0 psig. The addition of the new isolation valve is essentially a secondary isolation device beyond the non-safety related BGV0076, whose purpose is to ensure equipment protection. Therefore, the design function of the parent valve (BG8155) is not altered by the additional valve. The required procedure revisions (CKL BG-120 to ensure proper lineup and SYS HA-208 for manipulations) will be completed prior to implementation of the activity. It is not anticipated that the proposed change 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 indicates that no design basis accidents credit the operation of this valve or the parent valve (BG8155). Therefore, there is no potential impact due to the addition of the new isolation valve.

50.59 Evaluation

There is no new design function, unapproved method of operation change, or credible accident scenarios created by the addition of the new isolation valve. Its positioning and operation will be governed by the above referenced procedures, and no credible accidents will be created. Since the proposed 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. Since no malfunctions of equipment important to safety were identified, the probability of occurrence of a malfunction of equipment important to safety is not affected by this the addition of a new isolation valve. Since no malfunctions of equipment important to safety previously evaluated in the USAR were identified, the radiological consequences of a malfunction of equipment important to safety are not affected by this change. Since no malfunctions of equipment important to safety were identified, no different type of malfunctions of equipment could be created. No equipment malfunction type different than previously evaluated in the USAR could be identified to occur as a result of the proposed changes.

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. Since an examination of design basis accidents discussed or referenced in the USAR indicated that no design basis accidents credit the operation of this valve, the addition of a new isolation valve will not increase the radiological consequences of an accident previously evaluated in the USAR.

Since an examination of design basis accidents discussed or referenced in the USAR indicated that no design basis accidents credit the operation of this valve, the addition of a new isolation valve 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 no credible accidents that could be created are identified, no accidents of a different type than any previously evaluated in the USAR could be created. The addition of a new isolation valve will redundantly assure isolation and thus protection of the non-safety related BGPC8155 pressure sensing element. The proposed changes will not effect plant equipment such that a new initiating event could occur. The additional isolation valve only provides redundant protection for non-safety related equipment.

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.

Evaluation Number: 59 2001-0018 Revision: 0
Title: Pressure Instrument for Heat Exchanger Abandoned In Place

Activity Description

This activity proposes to abandon the instrument EGFI-0117 in place. This is a locally mounted instrument indicating the CCW out flow for Excess Letdown Heat Exchanger. This indicator is located inside the containment and is infrequently used. This instrument is abandoned in place to improve maintenance efficiency and avoid future modifications. Two instrument root valves will be closed isolating the instrument from the process line. This instrument is only used to balance CCW out flow for the Excess Letdown Heat Exchanger. Once the flow is balanced the valve BGV-204 is locked open so that the flow does not change. There is no routine use for the indicator. If required the CCW flow in the line can be measured by using temporary flow instrumentation. The proposed activity will simply isolate the instrument by closing the root valves and will not affect any other component directly or indirectly.

50.59 Evaluation

EGFI-0117 has an indication function only and it does not initiate, sequence, or actuate any equipment or affect any operator action or mode of control. This proposed activity would not impact any design basis accidents. The flow indicator has indication function only and it does not initiate sequence or actuate any equipment or affects any operator actions or mode of control. The only safety related function of the indicator is a pressure boundary. The pressure boundary is now further secured above and beyond the original design by closing the instrument root valves. The proposed activity cannot initiate any accidents nor does the instrument have any mitigating functions. The proposed activity requires simply closing the root valves and isolating the indicator from the process line. Therefore, the proposed activity will not increase the probability or consequences of an accident. Since no new failure modes are introduced and plant equipment operation will not change, no accidents are created.

The design function of the Excess Letdown Heat Exchanger is not affected by the proposed change. The flow in the line is balanced and the valve BGV-204 is locked open in an intermediate position. This instrument is located inside containment without any routine function. The proposed activity will not impact, directly nor indirectly, any SSC's ability to perform its intended design function including both Safety Related and non Safety Related SSCs. The proposed activity will not affect, directly nor indirectly, the failure modes of any Safety Related nor non Safety Related equipment. Operation of plant equipment will not change. Therefore, the activity does no increase the probability or consequences of an equipment malfunction. Nor does the activity introduce any new fairlur modes.

The proposed activities will not alter any design basis limits, and will not impact, directly nor indirectly, any design functions or operations of the plant. No acceptance limits are identified. Since the activity has no impact on acceptance limits, the activity will not impact the margin of safety.

Evaluation Number: 59 2001-0019 Revision: 0
Title: Addition of Temporary Mixed Bed Resin Skid

Activity Description

The temporary modification (TMO) provides a temporary mixed bed resin skid connected in series with the existing makeup water demineralizer trains. The temporary skid will remove ion leakage from the existing demineralizer train to enhance the water quality supplied to the Demineralized Water (AN) System for plant use. The temporary mixed bed resin skid is expected to reduce the existing sodium leakage from 0.2 to 0.3 ppb to less than 0.1 ppb as discussed in the Updated Safety Analysis Report (USAR)

The function of the Makeup Demineralizer (WM) System is to provide reactor quality water for normal operations. The Makeup Demineralizer (WM) System is comprised of two parts, the pretreatment and demineralizer equipment. A raw water source is supplied to the pretreatment equipment form John Redmond Reservoir or service water. The raw water is chlorinated, passed through two parallel lime softeners and three parallel sand and carbon filters to provide organic free water to the makeup demineralizer equipment. The demineralizer equipment consists of two demineralizer trains. Each train consists of one strong cation unit, one weak base anion unit, one strong base anion unit, and one mixed bed unit that supplies demineralized water to the Chemistry requirements for the Power Block.

50.59 Evaluation

There are no design basis accidents identified or evaluated for the non-safety related WM system in the USAR. Since no accidents have been identified, the activity does not increase the probability or consequences of an accident.

Since the systems' functions are not changed, no credible accidents that could be created are identified. Therefore, the activity will not crate a different type of accident.

The makeup water system is located in the Shop Building outside of the power block with the only interface being the supply connection to the non-safety related Demineralizer Water Storage (AN) System. The temporary mixed bed resin skid has the same failure modes as the existing demineralizer equipment and is therefore bounded by the original analysis. Any failures of the temporary mixed bed resin skid will be handled by the existing design features of the Shop Building and would not directly or indirectly affect any safety related equipment important to safety. Since no credible malfunctions of equipment are identified the activity will not increase the probability or consequences of a malfunction. Nor will the actively create a different type of malfunction.

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

Evaluation Number: 59 2001-0020 Revision: 0
Title: Addition of Vent to Residual Heat Removal Piping

Activity Description

A calculation was prepared that indicated that eight inch Emergency Core cooling system (ECCS) piping downstream of the check valve EJ8969A could be susceptible to gas accumulation. It has been determined that all ECCS pumps are fully capable of performing their intended design function, in the current configuration. However, it is good engineering design to provide venting capabilities for susceptible portions of piping.

The evaluation determined the amount of gas that could be carried to the Centrifugal Charging Pump (CCP) and Safety Injection Pump (SIP) during ECCS recirculation if gas is accumulated downstream of valves EJHV8804A and EJ8969A. The analysis determined that the peak void fraction of the water entering CCP-A and SIP-A pumps is less than 2% and 5% respectively. The void duration in the water flow entering CCP-A and SIP-A is less than 20 seconds. The Engineering Evaluation determined that with the low void fraction and short duration the gas intrusion will not cause any degradation or damage to SI or CC pumps during ECCS recirculation phase. Therefore, the integrity and performance of the CC and SI pumps will not be affected.

It is recognized that it is good engineering design to provide high points in piping systems to allow for venting. Therefore, a vent is being proposed in the piping system. This activity allows for installation of the ¾" vent line in the 8" run pipe. The added weight of the vent line (pipe, valve and fittings) is approximately 20 lb. Stress intensity factor at the ¾" vent connection for the 8" diameter run pipe is 1.00. Therefore, the addition of the vent line will not change any stresses in the piping system. The vent line will be about 18" long. This is a standard design vent line. The standard design vent line does not need additional support. There is an in-line anchor, EJ01-A001/132 about 18" away from the proposed location which can take additional load due to the added weight of 20 lb. The system is rigid enough so that there is no significant impact on the piping stresses and support loads. This vent line will extend the pressure boundary up to valve, EJV0203. The piping downstream of the vent valve will be non-safety related.

The appropriate procedures will be revised to vent from EJV0203 during system fill and vent, and on a monthly basis as a minimum.

50.59 Evaluation

The analyzed events are assumed to be initiated by the failure of plant structures, systems or components. The proposed change will not have a detrimental impact on the integrity of any plant structure, system, or component. The proposed changes will not alter the operation of any plant equipment, or otherwise increase the probability of an accident initiation. Therefore, the proposed change does not affect any design basis accidents and the probability and consequences of accidents are not increased..
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 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. No new accidents are identified.

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. This change will not increase the probability of occurrence or consequences of a malfunction of equipment important to safety. The added vent line will be an extension of the pressure boundary, it will have passive function which will be consistent with previous design and operation of the equipment.

The proposed change will 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 does not affect any acceptable limits contained in the bases of the Technical specifications. There are no design changes or equipment performance parameter changes associated with these changes. No set points are adversely affected, and no changes are being proposed in the plant operational limits. Therefore, the margin of safety has not been affected.

Evaluation Number: 59 2001-0021 Revision: 0
Title: Temporary Procedure for Tuning Letdown Temperature and Pressure

Activity Description

A temporary procedure has been written to provide instructions for tuning Chemical and Volume Control System (CVCS) (system designator – BG) Letdown Temperature and Pressure Loops BG LPT-130 and BG LPP-131. This activity will ensure that newly installed globe valve BGTV0130 is controlling letdown temperature with optimized tuning parameters. This procedure will cycle valves BGTV0130 and BGPCV0131 under conditions of 75-gpm letdown in both automatic and manual control while monitoring and recording several process system variables. A temporary flow instrument will be used to measure Component Cooling Water (CCW) flow. This procedure will connect a vendor supplied "Loop Scanner" to the 7300 Process Control System at six process system test points in order to observe control system response. A vendor supplied "Portable Stem Travel Transducer" will be used only to store flow data from the flow instrument and will not be connected to the valve.

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 (Ref. USAR section 3A, pg 3A-47 R/0) 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 temporary procedure are considered a part of control systems not required for safety as defined in USAR Section 7.7; therefore, they exceed the requirements described in the USAR.

The Fisher Signal Conditioning Module (SCM) will be connected to six design test points in the 7300 Process Control System. These six parameters are letdown heat exchanger outlet temperature, letdown temperature controller output, letdown flow, letdown pressure, letdown pressure controller output, and regenerative heat exchanger letdown outlet temperature. Prior to use in the plant, personnel will ensure that all 16 channels of the Fisher SCM have at least a 1 megohm input impedance. This criterion meets the minimum load resistance for 7300 Process Control Cards. This is acceptable isolation for test equipment to ensure no appreciable card loading.

2. Potential Affects to Seismic Analysis:
The portion of the flow instrument that attaches to the piping (i.e., transducer assembly) is insignificant in mass compared to the mass of the piping system it is attached to. The remainder of the flow instrument (i.e., flow meter) will be situated on the floor with flexible wiring routed from the flow meter to the transducers, data logger and to a power source. The data logger will rest on the floor near the valve or as otherwise approved by System Engineering to ensure no seismic interaction. There will be an insignificant effect on the seismic spectra because of this temporary installation.

3. Pressure / Temperature Effects
The high-pressure alarm, via BGPB0131A, will sound if Letdown reaches 500 psig per Alarm procedure ALR 00-039A, "LTDN HX DISCH PRESS HI". The associated Letdown piping is protected from over-pressurization by relief valve BG8117 with a setpoint of 600 psig.
The performance of this temporary procedure will cause letdown pressure (BG PI-131) and temperature (BG TI-130) to be cycled around existing control points. The procedure specified

upper and lower control limits for letdown pressure of 400 psig and 300 psig; respectively, are acceptable for normal operation. The lower limit of 300 psig is the lowest normal control point for letdown pressure. This lower limit provides sufficient margin to prevent flashing of the letdown coolant before it enters the letdown heat exchanger. This is true even if letdown leaving Regenerative Heat Exchanger EBG07 is at the alarm point of 380° F. The upper limit of 400 psig, although outside the normal range of 300-350 psig, for operation with a steam bubble in the pressurizer, produces no negative effects on plant operation and is within the design limits. Changes of ±50 psig around the control point of 350 psig will produce small changes in letdown flow since the change in total differential pressure across the letdown orifices is relatively small. The high-pressure alarm value where Operations would enter ALR 00-039D is 500 psig. The pressurizer level control system will adjust charging flow as required to maintain the program level. Therefore, based on the lower pressure limitation remaining within the normal operating range and the upper pressure limitation being well within the alarm setpoint, this temporary procedure does not exceed the design pressure limits of the system.

The temporary procedure specifies an upper limit for letdown temperature of <120°F. Alarm procedure ALR 00-039B, "LTDN HX DISCH TEMP HI", would be entered if temperature were to reach 120F. Operation between the initial starting temperature of 110° and up to 120°F produces no negative effects on operation. Some increase in the Reactor Coolant Pump (RCP) seal leakoff should be expected as Volume Control Tanks (VCT) temperature rises, however approach to the high seal injection temperature of 135°F is not expected even at VCT temperatures of 120°F. This is primarily due to Seal Water heat exchanger EBG03 cooling of Normal Charging Pump (NCP) or Centrifugal Charging Pump (CCP) recirculation and RCP seal return that would bring pump suction temperatures below VCT temperature. A lower boundary of 90°F is also specified in the "Precautions and Limitations" section of the procedure to bound the test range. This is the lowest temperature observed during full power operations while the Boron Thermal Regeneration System (BTRS) is in service. Although the BTRS will not be inservice, this provides lower bounds for normal plant operations.

4. Reactivity Effects
The CVCS demineralizers will be bypassed in the procedure prerequisites 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 is emphasized in the Precautions and Limitations section and in notes throughout the procedure. 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 design basis accidents, effects of natural phenomena, and other hazards were reviewed for potential impact by the temporary procedure. In particular, the following accidents were reviewed:

- CVCS Malfunction That Results in a Decrease in the Boron Concentration in the RCS (15.4.6)
- Inadvertent Operation of the ECCS During Power Operation (15.5.1)
- CVCS Malfunction That Increases RCS Inventory (15.5.2)

· Break in Instrument Line or Other Lines From RCS Pressure Boundary That Penetrate the Containment (15.6.2)
· LOCAs Resulting From a Spectrum of Postulated Piping Breaks Within the RCS Pressure Boundary (15.6.5)

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.

Reactor letdown is automatically isolated on low pressurizer level in the event of a LOCA and is manually isolated in the event of a leak where pressurizer level cannot be maintained by one operating charging pump. Normal letdown is not required to bring the plant to a safe shutdown condition.

The following USAR failure mode analyses were reviewed for potential impact by the temporary procedure;
· Table 5.4A-3, RHR and Safety Related Cold Shutdown Operations-Failure Modes and Effects Analysis
· Table 7.7-3, Loss of Any Single Instrument
· Table 7.7-4, Loss of Power to a Protection Separation Group
· Table 7.7-5, Loss of Power to a Protection Separation Group
· Table 9.2-13, CCW System Single Active Failure Analysis
Table 9.3-10, 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 procedure or off normal procedure.

The seismic response of the CVCS and CCW systems is not affected by this procedure due to the controls on 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. Therefore, the margin of safety has not been affected.

Evaluation Number: 59 2001-0023 Revision: 0
Title: Evaluation of Penetration Room Cooler Declared Out of Service

Activity Description

The activity provides an evaluation of the operability of supported equipment for the condition of having an Electrical Penetration Room Cooler out of service with room temperature >101°F. It specifies required compensatory actions for an out of service cooler. The evaluation also clarifies under what circumstances these room coolers are a required support system. The room coolers SGL15A and SGL15B are located in rooms 1409 and 1410 respectively on elevation 2026' of the Auxiliary building. This activity provides a pre-determined operability determination giving the bounding conditions for the electrical penetration room cooler operation which will insure that the specified safety function of the equipment located within the room will be unaffected if the room coolers (SGL15A or SGL15B) become non-functional during normal plant operation, or during a DBA. For example during planned or emergent maintenance activities if one of these room coolers would be both inoperable and unavailable. This is not intended to be a change to the design or licensing basis but rather an operability determination for this condition. UFSAR Table 3.11(B)-1 is being revised to add a footnote indicating the potential increase in room temperature due to an inoperable electrical penetration room cooler in rooms 1409 & 1410. The supported systems safety functions will be unaffected during normal operation, or Safe Shutdown Earthquake (SSE), fire, or Design Bases Accident (DBA). This activity is a condition allowed by the guidance provided in NRC Generic Letter 91-18 "Degraded and Non-Conforming Conditions and on Operability", section 6.12 of the attachment, which provides guidance allowing licensees to make determinations as to which support equipment is required under various conditions.

50.59 Evaluation

Although the disposition allows a condition that is contrary to that which is described in the USAR, it is being addressed as an abnormal and temporary condition and not a change to the design/licensing basis. The penetration room temperature is not an initiator of any accident previously analyzed. The room coolers are not included in the Probabilistic Risk Assessment model and they are not considered risk significant. The frequency of accidents is thus unaffected.

Based upon the review of environmental qualification reports it is concluded that there would be no more than a minimal increase in the likelihood that some component in the affected rooms might malfunction due to a higher ambient temperature. Thus, there is not more than a minimal increase in the likelihood of a malfunction of equipment.

The loss of coolant accident was reviewed. The USAR section 9.4.3.2 discusses operators taking manual actions in the south electrical penetration room following a loss of coolant accident. The ability to access the room by operators is unaffected by this activity. The ambient temperature will not be so adverse as to prevent access to the rooms. The rooms are classified as a harsh environment due to radiation but not due to temperature. Safe shutdown capability following a fire was also reviewed. The electrical cable separation design will be unaffected by this activity. The fire protection function, safety function and the normal operation & DBA function of the room coolers is to maintain room ambient temperature. The room ambient temperature will be maintained within operable bounds and the supported SSC functions unaffected.

The radiological consequences of a malfunction of an system, structure or component due to the room cooler being out of service is unaffected. Although the room temperature is predicted to rise beyond the engineering design values stated in USAR Table 3.11(B)-1 and 3.11(B)-2, the room ambient temperature was shown to be within a range that would allow the supported SSC function to be unaffected.

Although the inoperability of the room cooler will allow the room temperature to rise beyond the design maximum values stated in Table 3.11(B)-1 and 3.11(B)-2, the room ambient temperature was shown in the evaluation to be in a range during an accident that would allow the supported SSC function to still be achievable. Thus, no new type of event is created.

Based upon the activity evaluation the required cooling function of the room cooler is not required as a support equipment during normal plant operation when the temperature is less than 101°F and during a subsequent DBA the safety related equipment located in the affected room will still perform their specified safety function. If the room temperature exceeds 101°F during normal plant operation the room cooler is required support equipment. The room temperature will be administratively maintained no more than 101°F per the Area Temperature Monitoring Specification of the Technical Requirements Manual (TRM). The condition of having the room cooler inoperable with room temperature < or = 101°F is evaluated to be acceptable. Any increase in room temperature due to a subsequent DBA will not prevent equipment in the room from performing their safety function. No new credible malfunctions would be created that would have a different result, by implementing this operating condition.

There are no design basis limits for fission product barriers specifically affected by an out of service penetration room cooler. The TRM section 3.7.22 addresses area temperature monitoring. The Bases of the TRM Specification as it might affect fission product barriers are unaffected by this activity.

The method of evaluating environmentally qualified equipment is unaffected. The affected rooms are still classified as a harsh environment due to radiation and not temperature.