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Perry Nuclear Power Plant
Docket No. 50-440
Report of 10 CFR 50.59 Evaluations for 2001 - 2003

Ladies and Gentlemen:

Pursuant to 10 CFR 50.59(d)(2), enclosed is the report of facility changes, tests, and experiments for the Perry Nuclear Power Plant (PNPP). The changes, tests, and experiments reported are for the period from the last submittal dated September 18, 2001 to the present.

Attachment 1 defines the acronyms and format description. Attachment 2 provides the summaries of the safety evaluations listed above.

If you have questions or require additional information, please contact Mr. Vernon K. Higaki, Manager - Regulatory Affairs at (440) 280-5294.

Very truly yours,

Attachments

cc: Region III Administrator
NRC Resident Inspector
NRC Project Manager

JE47

Format Description

The revised 10 CFR 50.59 regulation was implemented at the Perry Nuclear Power Plant (PNPP) on June 28, 2001. For the 50.59 Evaluations prepared prior to that date, the summary is presented in the following format:

<u>SE No.</u>: A sequentially assigned number from one (0001) to end of the period, preceded by the year; e.g., 99-0025.

<u>Source Document</u>: Common sources of evaluations, with the associated acronym, are listed below.

CR - Condition Report
COLR - Core Operating Limits Report
DCN - Drawing Change Notice
DCP - Design Change Package
ECP - Engineering Change Package
ECR - Engineering Change Request
TM - Temporary Modification
ODCM - Offsite Dose Calculation Manual
PAP - Plant Administrative Procedure
PEI - Plant Emergency Instruction
PSTG - Perry Specific Technical Guidelines
PTI - Periodic Test Instruction
SCR - Setpoint Change Request
SMRF - Simple Modification Request Form
SOI - System Operating Instruction
SVI - Surveillance Test Instruction
TM - Temporary Modification
TXI -Temporary Test Instruction
USAR - Updated Final Safety Analysis Report

<u>Description of Change</u>:

A short narrative describing the type of plant change.

<u>Summary</u>:

I. Response - Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report increased?

II. Response - Is the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report created?

III. Response - Is the margin of safety as defined in the basis any Technical Specification reduced?

For 50.59 Evaluations prepared after June 28, 2001, the summary is presented in the following format:

SE No.: The PNPP 50.59 program requires the performance of a 50.59 pre-screening document for activities being evaluated pursuant to 50.59. The pre-screening document is sequentially assigned a number starting with one (0001) to the end of the year, preceded by the year, e.g., 01-0001. If the pre-screening document and follow-on review documents indicate that the performance of a 50.59 Evaluation is required, then one is prepared. The number assigned to the 50.59 is the same number that is assigned to the pre-screening document. Since not all pre-screening documents require the preparation of a 50.59 Evaluation, gaps in the numbering of the 50.59 Evaluations are expected.

Source Document: Common sources of the evaluations, with the associated acronym, are listed above.

1.1 Activity Description:

A short narrative describing the type of plant change.

1.2 Summary of Evaluation:

A summary of the 50.59 Evaluation responses and conclusions associated with the eight (8) criteria contained in 10 CFR 50.59(c)(2).

1.3 Is a license amendment required prior to implementation of the change?

A simple response indicating if the 50.59 Evaluation required the performance of a license amendment pursuant to 10 CFR 50.90.

PERRY NUCLEAR POWER PLANT

SAFETY EVALUATION SUMMARY

PURSUANT TO

10 CFR 50.59(d)(2)

2001 - 2003

SE No.: 01-0009
Source Document: DCN 5909, Rev. 0

Description of Change:
This DCN adds a note to the Plant Underdrain (P72) system P&ID D-302-0861 (USAR Fig 2.4-71) to indicate that cleanout ports may be provided on the P72 sump pump discharge piping in various locations to allow for the cleaning of the discharge piping as needed.

Summary:

I. No. The Plant Underdrain system provides protection of the plant in the event of a seismic event to preclude instability/overturning of the nuclear island as a result of high groundwater elevations. The portion of the system affected by this change to the P72 P&ID is the non-safety sump pump discharge piping. The function of the discharge piping is not affected by the note nor by the potential addition of cleanout ports. These ports are fittings similar to others provided for the system/piping that would be provided and installed consistent with line specification L1-4 and applicable station procedures. The sump pump discharge piping is not credited with accident mitigation and is not an accident initiator. The operational requirements for the system are not modified. No changes are made to any assumptions or inputs previously made to assess dose consequences and no fission product barriers are affected. Therefore, neither the probability of occurrence nor the consequences of a previously analyzed accident nor malfunction of equipment important to safety will be increased.

II. No. The change does not modify or affect the function of the P72 system. The addition of this note and/or the potential future addition of cleanout ports to the non-safety, sump pump discharge piping, are not accident initiators. This note and/or the potential future addition of cleanout ports does not result in any interface with any plant systems, structures, or components in such a manner as to create the possibility of an accident of a different type than any previously evaluated in the USAR. This change does not result in making any previous non-credible events credible, nor does this activity result in making a previously bounded event no longer bounded. The change will not create any new systems, or add any new equipment that can compromise the functioning of any systems, structures, or components. This change also will not affect any known accident initiators or contributors and therefore it will not increase the probability of an accident previously thought to be incredible. Therefore, the proposed changes will not create the possibility of an accident or malfunction of a different type than previously evaluated.

III. No. This change does not affect the operation of the P72 system. The system will still be capable of performing its design function with or without cleanout ports. These ports only serve to facilitate maintenance and cleaning of the sump pump discharge piping. Postulated radioactive releases due to liquid-containing tank failures evaluated in Section 15.7.3 of the USAR remains unaffected regardless of the presence of cleanout ports in the sump pump discharge piping. This change cannot increase the dose to the public nor on-site radiation doses such that actions to mitigate the radiological consequences of an accident would be impeded, nor does this change directly or indirectly affect the ability of any other plant system to mitigate the radiological consequences of an accident. The sump pump discharge piping portion of the P72 system is non-safety related and is not referenced in any of the technical specifications or technical specification bases. In addition, the change does not affect any other systems that are referenced in any of the Technical Specifications or Technical Specification Bases. The USAR revision does not reduce the margin of safety implied or specifically stated by any licensing documents, including the USAR, Safety Evaluation Report (SER), Operational Requirements Manual (ORM), Core Operating Limits Report (COLR), Offsite Dose Calculation Manual (ODCM), or Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the basis for any Technical Specification.

SE No.: 01-0025
Source Document: TM 1-01-002, Rev. 0

Description of Change:
This Temporary Modification disengages the Emergency Diesel Generator (EDG) room cooling inlet air damper 1M43F0220A from its actuator, and maintains it fully open with temporary mechanical clamps. This Temporary Modification removes the 1M43F0220A actuator from its installed configuration to a secure location proximal to the damper.

Summary:

I. No. The only event that the Division 1 EDG is considered to affect the probability of occurrence is the station blackout event. For all other accidents, the Division 1 EDG serves to mitigate the accidents rather than affect their probability. These changes show that the function and the long-term reliability of the Division 1 EDG are maintained. Thus, the accidents/events would be mitigated as previously evaluated and the dose consequences would remain unchanged. These changes do not place additional demands on the control room staff outside of the control room following an accident, such that the operator dose is increased. The change will not add, alter, degrade or prevent actions described or assumed in an accident discussed in the USAR. The Temporary Modification (TM) does not impact the level of redundancy inherent to the Div 1 EDG room cooling system. The TM does not compromise seismic or separation criteria. The TM does not change any postualted accident release rates, duration or mechanisms. It has no impact upon any radiation release barriers. It does not physically change the plant such that there will be an increase in personnel exposure due to actions taken during an accident.

II. No. The implementation of the TM does not create any new failure modes. There are no new effects resulting from previously evaluated failure modes. Single failure considerations are unchanged. To implement this TM, reliable mechanical devices will be placed on damper 1M43F0220A such that it will remain in the open position. The resulting damper position supports the operability of the Division 1 EDG. No common mode or common cause failures are probable as a result of this action.

III. No. The function of the M43 system is a support system for the EDG. It is required to be operable such that the diesel generator and other supporting components remain within the Equipment Qualification analyzed temperatures for operability. The TM and associated administrative controls maintain operability of the M43 system to supply the required support to the EDG. Therefore, there is no reduction of margin of safety as defined in the basis for any Technical Specification.

SE No.: 01-0028
Source Document: SMRF 99-5030, Rev. 0

Description of Change:
The Sulfuric Acid Tank Containment Area and Building (P83) is being abandoned and removed from USAR Figures 9.2-16 (DWG D-302-382), 1.2-2 (DWG E-036-022), as well as system drawing D-302-311. In support of this, SMRF 99-5030 installed a welded cap to isolate the Potable Water system (P71) to the building as shown on USAR Figure 9.2-16. The associated USAR Change Request (CR) removes the associated Sequencer (1H51P063) and instrumentation interface to the Circulating Water system (N71) from USAR Figure 10.1-7 (DWG D-302-201) and drawing D-302-311. In addition, the sample piping to the abandoned Sequencer is isolated by locking valve 1N71F0531 closed as shown on USAR Figure 10.1-7.

Summary:

I. No. This change isolates the abandoned Unit 1 Sulfuric Acid Tank Containment Area and Building of the P83 System and has no impact on the remaining portion of the P71 and N71 systems to support Unit 1 operation. Unit 1 systems, structures, and components are unaffected by these changes. Therefore, this change does not increase the dose consequences of any accident described in the USAR. The locked closed boundary valve 1N71F0531, as well as the welded cap have no impact on Unit 1 operation by providing and maintaining positive control on the system pressure boundary. The addition of a locked closed device to the N71 system valve does not affect the system function or performance of the originally evaluated Unit 1 design. These changes to the system were done to the same design specification, codes and standards per the existing design. No equipment has been removed which would affect, compromise or impact the performance, function, and interaction with Unit 1. No interfacing equipment has been modified or removed which would affect, compromise or impact the function or performance of their originally evaluated Unit 1 design. The SMRF does not alter any assumptions previously made in the USAR accident analysis. In addition, no fission product barriers are affected. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. Operator actions needed to mitigate accidents are not impeded by this change. The P83, P71, and N71 systems do not have any direct connection with any plant systems or equipment important to safety. The removal of the Sulfuric Acid Storage Tank does not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR, nor change directly or indirectly mitigation of radiological consequences of malfunctions of equipment important to safety. Radionuclides, release rates, release mechanisms, or impact to radiological boundaries are not affected by this change.

II. No. This CR will not create any new systems nor negatively affect the functioning of any operating system. The abandonment of equipment as described in this CR cannot affect the possibility of any Unit 1 accident. This change does not increase the effects of any event that was previously bounded by other accidents such that these new effects become bounding. This change does not increase the probability of any significant event previously thought to be incredible to be as likely to occur as any accident in the USAR. This CR made no structure, system or component changes to the plant that would impact design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1.

III. No. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. The Technical Specifications and its Bases, Operating License, and Operations Requirements Manual are

not affected by this change to this system. There is no change to any accident analysis or margin of safety provided for in the design of the system as discussed in the USAR, Technical Specifications and its Bases, or applicable NRC SE Report. The addition of a locked closed device to the N71 system valve and the installation of a cap serve to provide and maintain positive control of the system boundary. The boundary isolation valve and cap have been evaluated for permanent isolation and found acceptable. Therefore, by compliance with the design specifications, codes, and standards, the margin of safety is unchanged. These boundary isolation devices have been evaluated for permanent isolation, which will not introduce any new radiological or environmental concerns.

SE No.: 01-0030
Source Document: DCP 99-5001, Rev. 0

Description of Change:
DCP 99-5001, Rev. 0 added hardware to the plant to facilitate the use of the Inclined Fuel Transfer System (IFTS) in plant operating Modes 1, 2, and 3. License Amendment 100, that allowed operation of the IFTS during plant operation, had already been submitted and approved by the Nuclear Regulatory Commission. Independent review of the License Amendment identified that the hardware changes were not adequately described in the submittal. Condition Report 99-3035 was written to identify this condition and Condition Report Remedial Action-002 required the safety evaluation be written. Thus, this safety evaluation was written to address the hardware changes made to the plant under DCP 99-5001 with the IFTS blind flange installed. The hardware changes included the addition of a drain valve (1F42F007) in the IFTS tube drain line to facilitate leak testing of the IFTS tube drain line, the seismic qualification of the drain line that resulted in a support modification (1F42H010), and the addition of a drain valve (1F42F006) in place of a drain plug in a pipe spool between the blind flange and the non-safety manual isolation valve to the upper pool (1F42F002). License Amendment 123 was later approved to allow use of IFTS (rotation of the blind flange) in plant operating Modes 1, 2 and 3.

Summary:

I. No. The proposed activity was found not to increase the probability of an accident previously evaluated since the analysis demonstrated that the seismic and pressure boundary integrity function/capability of the IFTS was maintained. The radiological consequences of an accident previously evaluated were not increased since the structural capability of the new components met or exceeded those already existing. Therefore, no decrease in barrier performance, and prevention, or degradation of operator actions, result from these changes. The modification does not increase the probability of a malfunction of existing equipment important to safety. This is because the additions met or exceeded existing design Codes, new components had equal or higher pressure rating, and the drain line was seismically qualified. Thus, no new failure mechanisms were introduced. There will be no increase in the radiological consequences of a malfunction of equipment previously evaluated. This is because the new components exceed or are of equal structural capability of existing components. No decrease in barrier performance or prevention/degradation of operator actions is anticipated as a result of the implementation of the design changes when the IFTS blind is installed. Amendment 100 and supplemental Amendment 123 addressed the use of IFTS with the IFTS blind flange removed in operating Modes 1, 2, and 3.

II. No. The proposed activity does not create the possibility of an existing accident of a different type previously evaluated in the USAR because existing design requirements are maintained and the drain line is now seismically qualified. Additionally, no new failure mechanisms are introduced as a result of this change for the same reasons. Therefore, there is no possibility of a different type of malfunction of equipment important to safety than previously evaluated.

III. No. The proposed activity does not reduce the margin of safety defined in the basis for any Technical Specification with or without the blind flange removed. The IFTS drain line remains non-safety as a result of this change. A seismic analysis was performed for the drain line and a piping support was added at the analytical boundary to this analysis. The USAR Table 9.1-4 identifies the drain line as non-safety class with a quality group of ANSI B31.1. The piping analysis stress results were within the ANSI B31.1 and ASME Section III stress limits. The design change is also consistent with the changes evaluated under Amendment 100. It is concluded that the margin of safety established in Technical Specifications is maintained.

SE No.: 01-0031
Source Document: SMRF 96-5075, Rev. 0

Description of Change:
The Low Pressure Coolant Injection (LPCI) "C" injection line needs a vent added so that air can be bled from the system. The recommended solution to this concern is to add a ¾" diameter pipe to the top of the 12" diameter LPCI line and tie it to the 8" diameter flush supply line. This ¾" line will be slightly sloped up to the 8" diameter flush supply line where it could then be vented at valve 1E12F0583. The new vent line does not bypass any valves or system components. This configuration will allow for a passive venting system where no additional vent valves will be required. SMRF 95-5075 was initiated to add a "passive" venting system as described above.

Summary:

I. No. The passive vent line is less than 1" in diameter and is designed and installed per the ASME code. No moderate energy or high energy pipe breaks are postulated in 1" or less piping. The new vent is not in the LPCI flow path and therefore does not affect LPCI flow rates. The RHR-C system will still function as designed such that the radiological consequences of previously evaluated accidents remain unchanged. This change will not increase the radiation dose to the public nor will it increase on-site radiation doses that would impede actions necessary to mitigate the consequences of a malfunction of equipment important to safety. This change will not adversely affect any systems that could impact fission product barriers. Therefore, neither the probability of occurrence nor the consequences of a previously analyzed accident nor malfunction of equipment important to safety will be increased.

II. No. The change will not create any new systems, add any new equipment or compromise the functioning of any systems, structures, or components. This change will not result in any new equipment failures, therefore, this change will create no new initiators or contributors for an event which could be considered a new accident. The change will have no adverse effect on any system important to safety, and does not affect the way any system will react to normal and abnormal transients. Plant systems and their operation will not be adversely impacted by the change. This proposed activity will not result in any new equipment failures, therefore, this proposed activity will create no new initators or contributors for an event that could be considered a new malfunction of equipment. This proposed activity also will not affect any known accident initiators or contributors.

III. No. The addition of the vent does not have any impact on the flow characteristics or performance of the RHR system. Structurally, the modified piping system was modeled and re-analyzed to ensure that the stress limits for design, normal, upset, emergency, faulted are in accordance will the piping design requirements. The piping analysis methodology did not credit any component performance, such as allowable stress, above the currently defined acceptance level. The piping stress analysis (Ref: 1E12G022 (A)) documents that the pipe stress in the new vent line is below the allowable ASME Code allowables as defined in the ASME Design Specification. The margin of safety when evaluating piping analysis is considered the difference between the ASME code allowables and the failure point of the material. Therefore, the DCP will not reduce the margin of safety as defined in the basis for any Technical Specification.

SE No.: 01-0032
Source Document: USAR Change Request 01-049

Description of Change:
1. Incorporated 2000 Census figures where applicable.
2. Incorporated current school population figures.
3. Deleted 2 camps which no longer exist.
4. A new prison was added and hospital and prison capacities were updated.

Summary:

I. No. This change updates the USAR with current population figures and facilities available off-site. It has no effect on the operation or accident analysis of the plant, or on any equipment or components in the plant. Therefore there is no safety significance.

II. No. This change updates off-site population figures and names of industries and facilities. It has no effect on any equipment, components or structures on or near the plant. Therefore, no safety concern exists.

III. No. All information in this change is in reference to off-site facilities and population figures. This change has no effect on any Technical Specification, the Bases, the Operational Requirements Manual, or the Safety Evaluation Report.

SE No.: 01-0033
Source Document: USAR Change Request 01-051

Description of Change:
1. Updated the names of railroads.
2. Updated the names of local industries.
3. Updated the information on the use of local airports.

Summary:

I. No. The changes to the USAR are administrative changes to names and functions of off-site facilities, which do not effect the operation of the plant or any equipment. In addition, there is no effect on off-site dose assessment capabilities nor the response of the Emergency Response Organization (ERO). There is no effect on the accident analysis of the plant.

II. No. The changes to the USAR are administrative changes to names of off-site facilities that do not alter existing plant operating or abnormal procedures. Nor do they effect plant equipment or the probability of any accidents.

III. No. The changes do not effect the operational requirements of Technical Specifications, the Bases, the Operational Requirements Manual, or the Safety Evaluation Report.

SE No.: 01-0034
Source Document: SMRF 99-5034, Rev. 0

Description of Change:

SMRF 99-5034 abandons the Chemical Storage Equipment and the Decontamination Station of the original cement based G51 Solid Radwaste system (SRW). This is accomplished by electrically disconnecting all electrical equipment such that no abandoned field device will be energized and locking closed three valves that provide air and water to the abandoned equipment. Therefore, this change incorporates a revision to USAR Section 1.2.2.9.3 to delete the first sentence that referred to the Solid Radwaste system being common to Unit 1 and Unit 2. USAR Chapter 11.4 is revised to remove any reference to cement or solidified waste. While no USAR figures relative to the G51 system are affected by this change, non-USAR drawing D-302-745, Radwaste Solidification system, is being revised to eliminate and isolate equipment and components associated with the cement portion of the G51 system. As a result of the above change, service air (P51 system) is no longer required for the Cement Air Slide or other supporting services. Therefore, USAR Figure 9.3-29 (DWG D-302-242) Service Air Distribution is revised to show valve 0P51F0520 as locked closed, with a reference to indicate that all piping downstream is abandoned.

Summary:

I. No. SMRF 99-5034 abandons the Chemical Storage Equipment and the Decontamination Station of the original cement based G51 Solid Radwaste system (SRW). This change illustrates the separation of the Unit 1 Solid Radwaste system (G51) from abandoned portions of the cement system portion of the system. All valves are administratively controlled per the associated Valve Lineup Instructions (VLIs). The addition of a locked closed device to these valves does not affect the system function or performance of the originally evaluated Unit 1 design. The addition of a locked closed device to valve 0G51F0511 and electrically disabling of the normally closed valve 0G51F0370 does not affect the system function or performance of the originally evaluated Unit 1 design. The USAR Sections and Tables were revised to indicate single unit operation and to eliminate any reference to the term "solidification/dewatering" or any similar term. No interfacing equipment has been modified or removed which would affect, compromise or impact the function or performance of their originally evaluated Unit 1 design. This change does not alter any previous accident assumptions. The radiological consequences of any accident described in the USAR are unaffected by the changes made to the G51 and P51 systems. In addition, no fission product barriers are affected. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. Operator actions needed to mitigate accidents are not impeded by this change. Therefore, the changes do not increase the radiological consequences of an accident previously evaluated in the USAR. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR, nor change directly or indirectly mitigation of radiological consequences of equipment important to safety. This change does not affect the function of any fission product barrier. The handling and processing of liquid radioactive waste is in compliance with the Branch Technical Position ETSB 11-3, 10CFR71 and applicable ICC and DOT regulations. Radionuclides, release rates, release mechanisms, or impact to radiological barriers are not affected by this change.

II. No. This change will not create any new systems, nor negatively affect the functioning of any operating system. The locked closed boundary valves have no impact on Unit 1 operation. The addition of a locked closed device and/or electrically disabling the valves serves to mitigate any radiological or environmental consequences by providing and maintaining positive control of the system pressure boundary. The abandonment of equipment as described in this change cannot affect the possibility of any Unit 1 accident.

This change does not increase the effects of any event that was previously bounded by other accidents such that these new effects become bounding. This change does not increase the probability of any significant event previously thought to be incredible to be as likely to occur as any accident evaluated in the USAR. This change made no structure, system or component changes to the plant that would impact design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1. No new failure modes or failure effects are created by this change.

III. No. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage that were previously analyzed in USAR Chapter 15. The Technical Specifications and Bases, Operating License, and Operations Requirements Manual are not affected by this change to this system. There is no change to any accident analysis or margin of safety provided in the design of the system as discussed in the USAR, Technical Specifications and Bases, or applicable NRC Safety Evaluation Report. The addition of a locking device to valve 0G51F0511 and electrically disabling valve 0G51F0370 serves to provide and maintain positive control of the system pressure boundary. These boundary isolation devices have been evaluated for permanent isolation, which will not introduce any new radiological or environmental concerns. Therefore, by compliance with the design specifications, codes, and standards, the margin of safety is unchanged. They will not introduce any new radiological or environmental concerns.

SE No.: 01-0035
Source Document: SMRF 99-5035, Rev. 0

Description of Change:
The Offgas Charcoal Vault Refrigeration System (N64A) step controller (1N64R0055) is being abandoned and removed. SMRF 99-5035 disconnects and removed the step controller from the solenoid valves and pressure switches of the N64 Brine Cooling Packages as shown on USAR Figures 9.4-24, sheets 2, 3, 4 and 5 of 5 (DWGs D-913-009, -010, -011 and -012, respectively). The associated USAR Change Request removes these items from the USAR Figures as well as USAR Section 9.4.11.

Summary:

I. No. This change disconnects, abandons, and removes the step controller (1N64R0055). This change to the system was done to the same design specification, codes and standards per the existing design. No equipment has been removed which would affect, compromise or impact the performance, function, and interaction with Unit 1. No interfacing equipment has been modified or removed which would affect, compromise or impact the function or performance of their originally evaluated Unit 1 design. Therefore, this change does not increase the dose consequences of any accident described in the USAR. The installation of the new tubing runs were done to the same design specification, codes and standards per the existing design and serve to provide and maintain positive control of the system boundaries. This change does not alter any assumptions previously made in the USAR accident analysis. In addition, no fission product barriers are affected. Radionuclides, release rates, release mechanisms, or impact to radiological barriers are not affected by this change. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. Operator actions needed to mitigate accidents are not impeded by this change. The proposed change does not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR, nor change directly or indirectly mitigation of radiological consequences of equipment important to safety.

II. No. This change will not create any new systems nor negatively affect the functioning of any operating system. The abandonment and removal of equipment as described in this change cannot affect the possibility of any Unit 1 accident. This change does not increase the effects of any event that was previously bounded by other accidents such that these new effects become bounding. This change does not increase the probability of any significant event previously thought to be incredible to be as likely to occur as any accident in the USAR. This change made no structure, system or component changes to the plant that would impact design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1.

III. No. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. The Technical Specifications (TS) and its Bases, Operating License, and Operations Requirements Manual are not affected by this change to this system. There is no change to any accident analysis or margin of safety provided in the design of the system as discussed in the USAR, TS and its Bases, or applicable NRC Safety Evaluation Report. Therefore, by compliance with the design specifications, codes, and standards, the margin of safety is unchanged and will not introduce any new radiological or environmental concerns.

SE No.: 01-0036
Source Document: SMRF 99-5020

Description of Change:
This SMRF and USAR Change Request (CR) incorporates revisions to USAR Figures: 11.2-1, Sheet 4, 11.2-2, 9.3-29 (DWG D-302-242); 9.3-31 (DWG D-302-243); 9.2-4, Sheet 1 (DWG D-302-611); 9.7-4 USAR drawings, described in greater detail below, illustrate the permanent isolation and abandonment of portions of the Liquid Radioactive Waste system (G50). These revisions also illustrate the isolation and abandonment of this system (G50) from the supporting Service Air system (P51), Instrument Air system (P52), Radwaste Building Ventilation systems (M31), Nuclear Sampling system (P34), and portions that are associated with the Detergent Drain system (G50) and Solid Radioactive Waste system (G51). This is accomplished by locking closed valves and installing various caps, plugs and blind flanges. The Nuclear Closed Cooling system (P43) piping and function are unchanged by this effort, however, the USAR figures and tables have been revised to reflect that while G50 system components are to be abandoned, the P43 flow paths will remain available to maintain system flow balance.

Summary:

I. No. This CR/SMRF illustrates the separation of the Liquid Radioactive Waste System (LRW) from abandoned portions of the Unit 1 facility. The installation of caps, plugs, and blind flanges provide permanent isolation between the abandoned portions of the G50, P51, P52, M31, P34, and G51 systems from the operating plant. The locked closed boundary valves, as well as the caps, plugs, and blind flanges, have no impact on Unit 1 operation and serve to mitigate any radiological consequences by providing and maintaining positive control on the system pressure boundary. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. Operator actions needed to mitigate accidents are not impeded by this change. The addition of a locked closed device to the various system valves does not affect the system function or performance of the originally evaluated Unit 1 design. These changes to the system were done to the same design specification, codes and standards per the existing design. The USAR Sections and Tables were revised to eliminate any reference or discussion relative to the G50 system and associated abandoned equipment or supporting systems. No interfacing equipment has been modified or removed which would affect, compromise or impact the function or performance of their originally evaluated Unit 1 design. Although various LRW components are abandoned, the dose limits provided in USAR Tables 12.4-8, 15.7-12, 15.7-14, and 15.7-16 were conservative. The SMRF does not alter any assumptions previously made in the USAR accident analysis. In addition, no fission product barriers are affected. This SMRF made no structure, system or component changes to the plant that would impact its safety-related function. No equipment has been removed or altered which would impact the originally evaluated function or performance. No new failure modes or effects are created by this change.

II. No. This CR will not create any new systems nor negatively affect the functioning of any operating system. The abandonment of equipment as described in this CR cannot affect the possibility of any Unit 1 accident. This change does not increase the effects of any event that was previously bounded by other accidents such that these new effects become bounding. This change does not increase the probability of any significant event previously thought to be incredible. This CR made no structure, system or component changes to the plant that would impact design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1.

III. No. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage as analyzed in USAR Chapter 15. Although various LRW components are abandoned, the dose limits provided in USAR Tables 12.4-8, 15.7-12, 15.7-14, and 15.7-16 are conservative. The addition of a locked closed device to system valves and the installation of caps, plugs, and blind flanges serve to provide and maintain positive control of the system boundary. The boundary isolation valves and other isolation devices have been evaluated for permanent isolation and found to be acceptable. Therefore, by compliance with the design specifications, codes, and standards, the margin of safety is unchanged. These boundary isolation devices have been evaluated for permanent isolation, which will not introduce any new radiological or environmental concerns.

SE No.: 01-0037
Source Document: DCN 5493, Rev. 0

Description of Change:
DCN 5493 adds a new design temperature flag to the Nuclear Closed Cooling (NCC) system P&ID (D-302-613) to identify a higher design temperature for the NCC system piping downstream of the Reactor Water Clean-Up (RWCU) non-regenerative heat exchanger than is currently shown on the P&ID. DCN 5493 increases the design temperature for this portion of piping from 150°F to 200°F. The original piping design temperature of 150°F has reached 175°F during RWCU blowdown operation and therefore this condition necessitates a change to the design temperature of this portion of piping.

Summary:

I. No. This change reflecting the design temperature of the NCC piping downstream of the RWCU non-regenerative heat exchanger bounds the highest temperature (175°F) that has been observed in this portion of piping and therefore provides margin over the expected maximum operating temperature. The piping analyses prepared to support DCN 5493 have shown that the affected NCC piping and pipe supports are fully qualified for a design temperature of 200°F. This piping design temperature change will not affect the ability of the NCC system to provide cooling water to plant equipment, does not affect the NCC system's capability to mitigate a high drywell pressure condition resulting from a LOCA (if this action is desired by the operators), and does not affect the NCC system's ability to support the drywell and containment cooling systems during plant transients.

II. No. This change will neither create any new systems nor negatively affect the functioning of any operating system. The change in temperatures cannot affect the possibility of any Unit 1 accident, so long as temperatures stay within the given range. This change does not increase the probability of any significant event previously thought to be incredible. This change made no structure, system or component changes to the plant that would impact design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1.

III. No. The affected NCC piping and pipe supports are qualified for the higher design temperature and therefore the margin of safety inherent to the piping codes is maintained. The margin of safety associated with containment penetration P311 is retained given that the penetration remains fully qualified to the requirements of ASME Section III. The containment isolation valves at penetration P311 are not affected by this activity and will remain fully operable subsequent to implementation of DCN 5493. Consequently, this DCN will not reduce the margin of safety associated with containment leakage as defined in the basis for Technical Specification Section 3.6.1.1 for Primary Containment. The affected piping does not interface with any other plant systems, structures, or components in such a manner as to reduce the margin of safety as defined in the bases for any other Technical Specifications.

SE No.: 01-0038
Source Document: SMRF 99-5033, Rev. 0; Technical Assignment File 81777

Description of Change:
In support of isolating the condensate return to Auxiliary Steam system (P61) from the nonfunctioning Radwaste Evaporators portion of the Liquid Radwaste (LRW) – Waste Evaporate/Condenser system (G50), this USAR Change Request (CR) incorporates revisions to various USAR sections, tables, and figures as follows:

USAR Sections 2.2.3.1.3.1, 9.2.7, 9.5.10, and Table 9.5-2 have been revised to remove information and/or references relative to the radwaste evaporators and associated abandoned equipment or supporting systems, as well as references to dual unit operation.

USAR Figure 9.2-4 and Table 9.2-24 has been revised to add a note to indicate that even though the Radwates Evaporator Pump Coolers will be taken out of service and provide no heat load, the Nuclear Closed Cooling (P43) system flow path will remain available to maintain system flow balance.

USAR Figures 9.5-18 is revised to remove all but Unit 1 portions of the P61 system as a result of permanently isolating the abandoned Unit 2 portion of the P61 from the Unit 1 operating plant.

USAR Figure 9.5-20 is revised to identify valve 0P41F0563 and 0P41F0564 as locked closed to isolate the Service Water (P41) system to the abandoned Auxiliary Steam (P61) Radwaste Evaporator Condensate Cooler and associated Conductivity Sample Panels.

Summary:

I. No. This change illustrates the abandonment of the Radwaste Evaporator condensate return portions of the P61 system as well as changes to the supporting P41 and P43 systems. This change has no impact on the remaining portion of systems' support of Unit 1 operation. The addition of a locked closed device to these valves does not affect the system function or performance of the originally evaluated Unit 1 design. The addition of a locking device to the valves serves to provide and maintain positive control of the system pressure boundary. The CR does not alter any assumptions previously made in the USAR accident analysis. In additions, no fission product barriers are affected. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hyphothesized during periods of operation at design basis fuel leakage, as analyzed in USAR Chapter 15. Operator actions needed to mitigate accidents are not impeded by this change. The proposed change does not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR, nor change directly or indirectly mitigation of radiological consequences of equipment important to safety. Radionuclides, release rates, and release mechanisms are not affected by this change.

II. No. This CR will not create any new systems nor negatively affect the functioning of any operating system. The locked closed boundary valves have no impact on Unit 1 operation and serve to mitigate any radiological consequences by providing and maintaining positive control on the system pressure boundary. The abandonment of equipment as described in this CR cannot affect the possibility of any Unit 1 accident. This change does not increase the effects of any event that was previously bounded by other accidents such that these new effects become bounding. This change does not increase the probability of any significant event previously thought to be incredible to be as likely to occur as any accident in the USAR. This CR made no structure, system or component changes to the plant that would impact design function.

III. No. This change does not impact any of the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage, as analyzed in USAR Chapter 15. The Technical Specifications (TS) and Bases, Operating License, and Operations Requirements Manual are not affected by this change to this system. There is no change to any accident analysis or margin of safety provided in the design of the system as discussed in the USAR, TS and Bases, or applicable NRC Safety Evaluation Report. The addition of a locked closed device to valves serves to provide and maintain positive control of the system boundary. Therefore, by compliance with the codes and standards, the margin of safety is unchanged. These boundary isolation devices have been evaluated for permanent isolation, which will not introduce any new radiological or environmental concerns.

SE No.: 01-0039
Source Document: SMRF 00-5005, Rev. 0

Description of Change:
USAR Figure 3.11-11, environmental zone AB-2 is being revised to include 154°F as the maximum temperature for the High Pressure Core Spray (HPCS) pump room during accident conditions. It was noted that during the modeling of the HPCS pump room, the previous analysis modeled the room as isolated from other rooms. It was determined that this was not accurate since there is a pipe chase that connects the HPCS pump room with the HPCS penetration room (AB-8). The GOTHIC model determined that the post accident temperature of the HPCS penetration room should also be increased from 126°F to 147°F as documented in calculation ECA-064.

SMRF 00-5005 has been prepared to install insulation on the HPCS large bore piping in an effort to reduce the post-accident HPCS room heat load. The SMRF is also documenting an increase in the post-accident temperatures in the HPCS pump room and the HPCS penetration room (environmental zones AB-2 and AB-8), as previously described. As a result, environmental zone AB-2 will be split into AB-2E [Low Pressure Core Spray (LPCS) pump room] and AB-2W (HPCS pump room).

The post-accident temperature in environmental zone AB-2W (HPCS pump room) in increasing from 143°F to 154°F and the post-accident temperature in environmental zone AB-8 (HPCS penetration room) will be increased from 126°F to 147°F.

Summary:

I. No. The HPCS system is used to mitigate various Design Basis Accident (DBA) scenarios. The proposed modification to install insulation on the piping has been analyzed and found to be acceptable. In addition, the proposed changes do not affect the design, material or construction standards of any system or equipment. No new or existing accident initiators have been affected by the proposed change. System operation, availability, and response to transients remains the same as already described in the USAR. The equipment qualification for all safety-related components is maintained. The HPCS pump and the associated safety-related equipment will still be able to mitigate the consequences of an accident.

   The equipment important to safety for this room has been evaluated at the elevated temperatures, the evaluation concluded that the equipment would continue to perform its safety function at elevated temperatures. This activity does not impair the availability of any plant system; therefore, the assumption of equipment operability in the initial conditions for the accident analysis is not affected. This activity does not degrade the performance below the design basis, by affecting the environmental, seismic or separation criteria, of the plant systems which are assumed to operate during the DBA. This activity does not increase challenges to the safety-related systems that are assumed to function in the accident analysis by imposing increased or more severe testing requirements. The activity does not impair system reliability by imposing transients not analyzed in the design basis for the system, or equipment protective features, degrade support or attendant system performance, or reduce system redundancy or independence. Therefore, the increase in the post accident HPCS pump and penetration room temperatures will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The proposed increase to the HPCS pump and penetration room post-accident temperatures will not create a possibility for an accident or malfunction of a different type than previously evaluated in the USAR. The installation of insulation on the HPCS piping within the HPCS pump room provides greater operating margin and is not a requirement for the operation and function of the HPCS system. Therefore, the insulation can not

create an accident of a different type than previously evaluated in the USAR. The equipment qualification for all safety-related components is maintained. The addition of insulation will reduce the heat load within the HPCS pump and penetration rooms and will not adversely affect any equipment important to safety. Both the direct and indirect effects of the change on the ability of the plant systems to perform their safety function have been considered.

III. No. The margin of safety for the environmental qualification of components is best described as the ability of the equipment to maintain operability of Engineered Safety Feature (ESF) equipment and the fission product barriers. The environmental qualification for ESF component is being maintained and therefore, the margin of safety has not been reduced. The addition of insulation to the HPCS piping is beyond the level of detail discussed in the Technical Specification or the bases.

SE No.: 02-00076
Source Document: TM 1-02-001

1.1 Activity Description

This activity installs temporary pumps inside the Plant Underdrain (P72) system manholes while permanent pumps are out of service. The temporary pump will be placed in the bottom of the manhole and the discharge will be directed to the gravity drain system. This results in bypassing the radiation monitors (0D17K0820A/B) and the flow indicators (0P72R0700A/B) and flow totalizers (0P72N0700A/B) that are in the East and West pump discharge lines.

1.2 Summary of Evaluation

The purpose of the use of temporary pumps is to retain the capability of maintaining the groundwater level below the prescribed design basis requirements. Since the original function included automatic pump operation, it also included automatic tripping to mitigate the consequences of automatic pump operation following a radwaste tank failure in addition to the operator action to manually trip the pumps.

The temporary modification does not include either unmanned or automatic sump pump operation, such that the automatic trip of automatically operating sump pumps following a radwaste tank failure event can be considered to be not applicable. Furthermore, the immediacy of the termination of temporary pump operation following a radwaste tank failure based on communication from the radwaste control room or the main control room may be considered an improvement compared to the USAR evaluated operator action. Since the initiating event for a radwaste tank failure would require a substantial plant event (seismic or other), termination of the sump pump operation upon occurrence of a seismic event or a plant transient represents an improved response compared to the accident analysis.

The manual operation of the temporary pump replaces the automatic system to control groundwater level. However, the pumps are classified as non-safety related, as they are not relied upon to prevent or mitigate accident conditions. Accumulation of groundwater in the designed operating level band occurs slowly, such that the reliability of the temporary change to manually control groundwater level below prescribed requirements is considered to be equivalent to the automatic/ permanent system described and evaluated in the USAR.

Therefore, the use of temporary pumps as described does not require a license amendment prior to implementation of the change.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 02-00111
Source Document: DCP 01-5030, Rev. 0

1.1 Activity Description
This activity changes the P48 system [Service Water (SW)/Emergency Service Water (ESW)] chlorination from an automatic system to a manual system. Throughout the USAR and this evaluation, the generic term biocide is frequently used rather than chlorine or chlorination. The actual biocide used is sodium hypochlorite. The USAR in Chapters 7 and 9 indicates that chlorination of the ESW system is automatic. No mention of the automatic operation vs. manual is discussed in the USAR for the SW system. This evaluation only addresses the change of ESW biocide injection from automatic to manual.

1.2 Summary of Evaluation
The purpose of this change is to replace the Biocide Injection system (P48) that was originally intended to have the capability for automatic as well as manual biocide injection into the SW/ESW systems. The purpose of biocide injection is to minimize algae and plant growth and preclude bio-fouling of the system loops. The present system with solenoid controlled air valves, timers and other interlocks has been difficult to maintain and operate in automatic mode. Even as an automatic system, it required intervention by plant chemistry technicians to set timers and sample the biocide levels in the loops. Typically, biocide injection to the ESW system has been done manually, either by use of the installed system or by an alternate means of drum addition when the automatic equipment was unavailable. Ultimately this has resulted in effective treatment.

The design change substitutes manual operator action (chemistry technician) for automatic. Since programs and procedures have been in place that have been proven effective, this evaluation determines the change can not have more than a minimal effect on the safety related ESW system. No reliance is placed on the P48 system for normal operation of the ESW system or during plant transients or accidents. The ESW system will be able to perform its function as stated in the USAR for all plant conditions including accident analysis. No new malfunctions or accidents, or changes to existing malfunctions or accidents are credible to be postulated due to this change. There will be no increase of radiological effects of accidents due to this change.

Therefore, the change from automatic biocide injection to manual injection to the ESW system does not require a license amendment prior to implementation of the change.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 02-00414
Source Document: USAR Change Request 02-035

1.1 Activity Description

The USAR is being revised to change Table 1.8-1 page 1.8-12 in the USAR for Regulatory Guide 1.23 revision 0, 1972, to take exception to the instrument accuracies stated in Section C.4. Table 1.8-1 is also being modified on page 1.8-41 to add a reference within the Degree of Conformance statement for Regulatory Guide 1.97 to the Regulatory Guide 1.23 Degree of Conformance Statement. The reference to the Emergency Plan is being eliminated since the contents of the Emergency Plan description of the Meteorological monitoring system were added to USAR section 2.3.3.4 as part of DCP 96-107, Rev. 1 and it's licensing basis evaluated by safety evaluation 99-0049 and USAR Change Request 99-123. This change takes exception to the meteorological monitoring system's instrumentation accuracy guidance described in Regulatory Guide 1.23, Rev. 0, 1972. The accuracies will be listed in Table 2.3-31 in the USAR. Table 2.3-32 "Automated Backup Sequence for Onsite Perry Meteorological Data" is being deleted as it is no longer necessary based upon the text added in USAR section 2.3.3.4 as part of DCP 96-107, Rev. 1. USAR section 2.3.3.4 is also being revised to take exception to the meteorological monitoring system's instrumentation accuracy guidance described in Regulatory Guide 1.23, Rev. 0, 1972 and deletes the reference to table 2.3-32. The USAR is also being revised to correct a typographical error in the List of Tables for Table 2.3-31.

Revision to section 6.2.8 of the Plant Data Book (PDB-R0001) Meteorological Monitoring Instrumentation bases to state instruments meet the accuracies listed in table 2.3-31 of the USAR. These changes are being proposed based upon USAR Change Request 02-035, which was initiated, based upon the investigation of Condition Report 01-3773. The Condition Report investigation revealed that newer industry standards were appropriate to use for determining the Meteorological Monitoring Instrumentation accuracies. The remaining "cleanup" issues were discovered during the review of the USAR in preparation of changing the commitment to a newer standard.

This 50.59 evaluation is required because some of the instrument accuracies are being expanded to values greater than what are listed in our current commitment to Regulatory Guide 1.23 Rev. 0, 1972. Therefore, this activity is considered to affect a USAR described design function.

1.2 Summary of Evaluation

This activity will decrease the accuracy requirements (i.e. expand the allowed values) for the Meteorological Monitoring System parameters for wind speed, delta temperature and dew point. This activity is being proposed to provide the use of the standard method for calculating accuracies within the calibration trip data sheets which are used as the source documents for instrument accuracies for the Meteorological Monitoring System surveillance instructions. The proposed method for calculating the instrument accuracies uses the square root of the sum of the squares method. This method is used for other safety-related instrumentation within the plant and is the method described in section 2.3.3.1 of the USAR as to how the Meteorological Monitoring System calculates accuracies. In fact, the calculations for the calibration trip data sheets and subsequently the accuracies listed in the Meteorological Monitoring System surveillance instruction are more restrictive. The investigation of Condition Report 01-3773 documents concerns with the way the accuracies were calculated and what equipment was included within these accuracy calculations. Currently the accuracies listed within the surveillance instructions are in accordance with our USAR commitments, however, based upon historical trending of the surveillance data the accuracies cannot be consistently maintained if we continue to use the "old" standard for accuracy as listed in Regulatory Guide 1.23, Rev. 0, 1972. The proposed values for the meteorological monitoring instrumentation accuracy is in accordance with Regulatory Guide 1.97, Rev. 3 and ANSI/ANS 2.5-1984. Regulatory Guide 1.97, Rev. 3 only addresses the meteorological parameters directly associated with dose assessment [e.g., Wind Speed, Wind Direction and Delta Temperature (used for atmospheric stability)]. Therefore, ANSI/ANS 2.5-1984 is the standard used for the Dew Point

parameter. ANSI/ANS 2.5-1984 is the approved industry standard that was proposed for endorsement by the NRC in proposed Rev. 1 to Regulatory Guide 1.23. This proposed Regulatory Guide is referenced in NUREG 0654 revision 1, which is an NRC approved document for emergency planning. The accuracy values in ANSI/ANS 2.5-1984 for Wind Speed, Wind Direction and Delta Temperature are the same as the accuracies listed in Regulatory Guide 1.97, Rev. 3.

The design basis for the Meteorological Monitoring Instrumentation is to ensure that data is available for estimating potential radiation dose to the public as a result of routine or accidental releases of radioactive materials to the atmosphere. This will provide the basic meteorological information required for reasonable estimates of atmospheric transport and diffusion. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. Decreasing the accuracy of the Meteorological Monitoring Instrumentation in accordance with the NRC guidelines as described in Regulatory Guide 1.97, Rev. 3 and NUREG 0654, Rev. 1, does affect the design basis for the Meteorological Monitoring System but does not impact the ability for estimating potential radiation dose to the public as a result of routine or accidental releases of radioactive materials to the atmosphere. This is evident by the NRC approval of Regulatory Guide 1.97, Rev. 3 and NUREG 0654, Rev. 1.

NUREG/CR 3011 "Dose Projection Considerations for Emergency Condition at Nuclear Power Plants" states that meteorological data is the most accurate data used in dose assessment. This is because of the large uncertainties (not in percent but in factors of two or more) within the dose assessment models for atmosphere dispersion and transport. The meteorological data accuracies will remain within the industry standards. The measurements for the instrumentation for the meteorological monitoring system will be responsive to the criteria of NUREG 0654, Rev. 1 and Regulatory Guide 1.97, Rev. 3.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 02-00994
Source Document: ECR 02-0045, Rev. 0

1.1 Activity Description
The purpose of this Engineering Change Package (ECP) is to ensure that the Emergency Service Water (ESW) system can perform its design safety function of heat removal. This goal is achieved by re-defining the design basis heat removal requirements for each Heat Exchanger (HX) cooled by ESW and establishing minimum design basis ESW flow requirements for each HX. Based on this information, minimum flows for testing and operating flows and temperatures for each mode of system operation are also defined. Supporting documentation (calculations, procedures, vendor manuals) for the ECP is included in this evaluation.

1.2 Summary of Evaluation
As justified below, this activity does not represent a change to Technical Specifications and does not require a license amendment per 10CFR50.59 paragraph (c)(2).

The ESW system is a key mitigating system for accidents and transients evaluated in the USAR since it provides heat removal to the ultimate heat sink and dilution water for radwaste tank rupture accidents. However, the ESW system and its supplied HXs are not considered initiators of any USAR evaluated event. Furthermore, based on the lack of adverse failure modes or effects, there is no potential that any new failure mechanism could cause the initiation of a USAR described event.

Compliance to applicable ASME Codes, Tubular Exchanger Manufacturers Association (TEMA) standards and Equipment Qualification (EQ) requirements is maintained for affected piping and components. Including postulated failures of the non-safety related portions of ESW yard piping that do not meet ANSI/ASME B31.1 stress allowables, there are no adverse failure modes or effects that could increase the possibility of System, Structure, or Component (SSC) malfunctions. The changes in heat exchanger fouling utilized in the design basis calculations are conservative compared to historical Generic Letter [GL] 89-13 test results. In the case of the Fuel Pool Cooling and Cleanup [FPCC] heat exchangers the fouling factors utilized in the calculations have been increased for added conservatism since these exchangers are not tested per GL 89-13. Continued performance monitoring under GL 89-13 provides added assurance that required heat removal capabilities will be maintained. Thus, there is minimal likelihood of any increase in SSC malfunctions.

The proposed changes will maintain heat removal capability for all supplied systems that support post-accident mitigating safety systems including the Residual Heat Removal (RHR) system. Room temperatures and equipment qualifications are not adversely affected and the design functions and operation of the ESW system are maintained. The new operating flows for ESW have been evaluated as acceptable to provide the necessary dilution flow for the postulated Radwaste tank rupture accidents such that the dose consequences continue to meet the 10CFR20 limits. In addition, the affected piping when subjected to the new higher operating temperatures will continue to meet applicable ASME Section III Code requirements. Thus, the supported systems will continue to provide post-accident mitigating functions. Therefore, there will be no increase in the consequences of an accident previously evaluated in the USAR.

Compliance to pertinent design standards, ASME Code and EQ requirements for all safety-related components will assure no increased equipment malfunctions. The ability to remove heat from supported systems is maintained. The ability of ESW to remove decay heat from the RHR system and provide protection for nuclear fuel cladding, reactor coolant pressure boundary and containment following an accident or transient assures that these three fission product barriers are maintained by the proposed changes.

The ESW system is a support system to those systems with credited accident mitigating functions. The proposed changes do not create any new system interconnections or electrical interlocks that could alter or add any new system or component interactions. Thus, there is no potential for creating an accident of a different type than previously evaluated in the USAR.

The ESW system design safety function as heat removal is maintained and thus there is no potential for any new or revised malfunctions of SSCs important to safety.

The ESW system is not directly connected with any of the fission product barriers. However, the barriers that could be affected are fuel cladding, reactor coolant pressure boundary and containment as a result of any loss of ESW heat removal capability. Design calculations document that the revised heat load and heat removal capabilities of each supported system are fully adequate at the design basis minimum required ESW flows. Thus, the proposed changes maintain the capability to cool fuel cladding, reactor coolant pressure boundary and containment and therefore, the changes do not directly or indirectly alter or exceed limits on these barriers.

The subject of this design change is the re-calculated ESW design and operating parameters (i.e., heat loads, flows and temperatures). Although the HX calculations have been performed with Category B software, no specific licensing basis required software or analysis methodology was required or discussed in the USAR, NUREG and SER. In addition, as discussed in Sections 3.9 and 3.10, although the original ESW flow of 7300 gpm and the design fouling factor for the RHR HXs are identified and discussed in the USAR, these parameters were not inputs to the containment long-term accident analyses and were not a basis for NRC review or acceptance. Further, the heat removal rate for the RHR heat exchangers is not revised by this ECP. Thus, there is no departure from a USAR described methodology for establishing the design bases or safety analyses.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 02-01210
Source Document: ECP 02-0237, Rev. 0

1.1 Activity Description
This ECP eliminates the Main Steam Line Isolation Valves (MSIV) stem packing leakoff piping from the outboard MSIV's 1B21F028A, B ,C & D including valves 1B21F027A, B, C & D on the leakoff piping and temperature elements 1B21N0710A, B, C & D. The piping will be capped and partially removed.

Outboard MSIV's 1B21F0028A, B, C, & D have a leak-off line between the first and second stage stem packing. This leak-off line directs first stage packing leakage to the containment annulus region. In the event of a first stage packing leak, it is possible to isolate the packing leak off line by manually closing the associated 1B21F0027A/B/C/D valve. However, the B21F0027A/B/C/D valve is not part of the penetration boundary and is not tested in accordance with the LLRT test program. Over the course of an operating cycle, the Technical Specification (TS) limit might be exceeded following a first stage packing leak and leak off line isolation, since the leakage is not measurable until shutdown. Elimination and capping of the leak off piping resolves this problem.

1.2 Summary of Evaluation
The USAR accidents were reviewed with respect to the effects of the proposed design change in the analysis section of this evaluation. None of the accident frequencies were found to be affected by the implementation of the proposed change.

This evaluation analyzes USAR described design functions that are potentially affected by the proposed design change. Both direct and indirect effects of the proposed change on design functions were evaluated. All potentially affected design functions were found to be satisfactorily performed, such that no malfunctions of equipment were identified.

This change does not result in increasing previously evaluated release rates, changing a release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed change does not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

Based on the scope of the proposed change, and the evaluation of possible failure effects of the proposed change, no new events of a significance to be considered an accident, nor new malfunctions of an SSC important to safety, could be identified.

This evaluation analyzes the effects of the proposed change on the According to the Non-Conformance Report (NCR) NEDS 2014, the increased temperature has no detrimental effects on the NCC system piping or components. No effects were identified.

The methods of evaluation that support the proposed design change are consistent with the methods used in establishing the design bases and the safety analyses documented in the USAR.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 02-01329
Source Document: DCP 01-5011, Rev. 0

1.1 Activity Description
This activity proposes eliminating the Vibration and Loose Parts Monitoring System (V&LPMS). System sensors, charge amplifiers, and the control panel will be spared in place. Wiring will be determinated, coiled and spared in the associated panels. The control room annunciator window will be blanked for use in future plant modifications. Associated procedures will either be deleted or altered to reflect this change.

1.2 Summary of Evaluation
The USAR accidents were reviewed with respect to the effects of the proposed design change in the analysis section of this evaluation. None of the accident frequencies were found to be affected by the implementation of the proposed change.

This evaluation analyzes USAR described design functions that are potentially affected by the proposed design change. Both direct and indirect effects of the proposed change on design functions were evaluated. All potentially affected design functions were found to be satisfactorily performed, such that no malfunctions of equipment were identified.

This change does not result in increasing previously evaluated release rates, changing a release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed change does not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

Based on the scope of the proposed change, and the evaluation of possible failure effects of the proposed change, no new events of a significance to be considered an accident, nor new malfunctions of a System, Structure, or Component (SSC) important to safety, could be identified.

This evaluation analyzes the effects of the proposed change on the PNPP fission product barriers. No effects were identified.

The methods of evaluation establishing the design bases and the safety analyses documented in the USAR are not altered or affected by this activity.

Further, the Boiling Water Reactor Owners Group (BWROG) has submitted and received approval of Topical Report NEDC-32975P-A "Regulatory Relaxation for BWR Loose Parts Monitoring Systems", which justifies removal of this system based on expenses, dose and years of Reactor operation without proven benefit. Approval response from the NRC is documented in an included Safety Evaluation and associated cover letter dated January 25, 2001.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 03-00112
Source Document: ECP 02-0213A, Rev. 0

1.1 Activity Description

ECP 02-0213A will permanently eliminate the interconnection between the "A" loop of the Emergency Closed Cooling Water (ECCW - P42) system and the Nuclear Closed Cooling Water (NCCW - P43) system in the vicinity of the Control Complex chillers. This ECP is the first part of a change that is intended to eliminate and/or prevent the occurrence of two continuing problem areas associated with the ECCW system and thus increase the reliability and availability of the ECCW system. The second part of the change (ECP 02-0213) will be implemented independent from this change and will eliminate the interconnection between the "B" loop of the ECCW system and the NCCW system in the vicinity of the Control Complex chillers.

The first problem area for the ECCW system concerns the post accident maximum allowable leakage from the ECCW system. The ECCW system must operate for seven days post accident without any make-up to the system surge tank. Demonstrating that this leakage criterion is satisfied has been a continuing problem due to the currently existing potential leak paths past the valves that isolate the ECCW system from the NCCW system. The second problem area concerns the MOVs that currently must change position on a LOOP/LOCA signal in order to align the safety related ECCW cooling water to the Control Complex chillers. Several of these MOVs have had problems demonstrating their ability to perform their safety functions.

Although the MOVs (0P42F0150A, 0P42F0300A, and 0P42F0330A) affected by this change have not exhibited seat leakage or problems performing their safety related "open/close" functions, implementation of this ECP is viewed as a preventative measure that will prevent these problems from occurring in the future. Installation of piping blinds that would effectively isolate the "A" loop of the ECCW system from the NCCW system would eliminate the most significant potential leakage paths from the "A" loop of the ECCW system and would thus provide a solution to the major problem area that could contribute to inventory loss from the "A" loop of the ECCW system. This design change will eliminate the safety related "open/close" functions associated with MOVs 0P42F0150A, 0P42F0300A, and 0P42F0330A since the "A" loop of the ECCW system will be permanently aligned in its required configuration for post-accident operation. These safety related MOVs will be removed from the Generic Letter 89-10 program thereby reducing maintenance and testing costs.

The scope of work for ECP 02-0213A includes installation of two piping blinds that will isolate the "A" ECCW piping from the supply and discharge headers of the "C" Control Complex Chiller. The piping blinds will contain appendages with a root valve that will allow all portions of the affected piping to be vented. Portions of the existing piping will be cut and removed and the new blinds will be welded to the existing piping. MOV 0P42F0150A will be electrically determinated and placed in the closed position. MOVs 0P42F0300A and 0P42F0330A will be electrically determinated and placed in the open position. These changes will permanently align the ECCW "A" loop in its required post-accident configuration. Consequently, valve 0P42F0150A will no longer receive a close signal at the onset of a LOOP/LOCA and valves 0P42F0300A and 0P42F0330A will no longer receive open signals at the onset of a LOOP/LOCA. Control Room panel 0H13P0904 will be modified to reflect the changes to these valves and the change to the available flow paths. Subsequent to this change, the NCCW system will be capable of supplying cooling water to only the "B" and "C" Control Complex Chillers. The flow path for NCCW supply to the "A" Control Complex Chiller will be isolated.

There is a possibility that the normally stand-by "A" ECCW and "A" Emergency Service Water (ESW - P45) systems would be required to support normal plant operation. This scenario would occur if both the "B" and "C" Control Complex chillers were simultaneously unavailable. In this case, the "A" Control Complex chiller would be run to support normal plant operation, necessitating

the use of the "A" trains of the ECCW and ESW systems to support operation of the "A" chiller. This condition would also require use of the normally stand-by "A" trains of the Emergency Pump Area Ventilation system (M28) and the ESW Pump House Ventilation system (M32).

1.2 Summary of Evaluation

The ECCW and NCCW systems and the "A" trains of the potentially affected safety systems (P42, P45, M28, and M32) will still be fully capable of performing their design functions subsequent to this activity. It was determined that the additional run time on the safety systems would be a relatively small fraction of the planned usage time of these systems and would not degrade the design features of these systems to the point where they would be unable to perform their safety functions. In other words, the additional run time on the "A" train of the affected safety systems does not make the "A" train of these systems any more likely to fail or malfunction than currently assumed. Further, there are no new failure mechanisms introduced to any of these systems as a result of this activity. Consequently, the likelihood of System, Structure, or Component (SSC) malfunctions is not increased and the possibility of new malfunctions with different results is not created.

The change to the manner in which the "A" loop of ECCW is operated to provide cooling water to the "A" Control Complex chiller is not adverse and is actually beneficial since automatic valve positioning will no longer be required to align the safety related ECCW cooling water to the "A" Control Complex chiller. This change enhances the reliability of the "A" loop of ECCW during performance of its safety function since (1) it eliminates the predominant leak paths to the NCCW system by installation of the piping blinds that provide a physical barrier to prevent inventory loss, and (2) eliminates the possibility of failures associated with system re-alignment since the "A" loop of ECCW will be permanently aligned in its required post-accident configuration. This aspect of the change also does not increase the likelihood of SSC malfunctions and does not create the possibility of new malfunctions with different results.

The systems affected and/or modified by this change do not initiate any currently evaluated accidents. The changes implemented by this activity do not create any situations that could initiate a new accident. Consequently, the frequency of occurrence of the currently evaluated accidents does not increase and the possibility of initiating accidents of a different type is not created. The mitigation effectiveness of the affected safety systems remains unchanged and therefore the radiological consequences from accidents and malfunctions do not increase. The affected safety systems will still be fully capable of supporting the Emergency Core Cooling System and therefore the fission product barriers will not be adversely affected. The modified piping system is qualified in accordance with typical analytical methods and does not conflict with any USAR described methodologies. Therefore, the activity does not result in a departure from a method of evaluation described in the USAR.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10CFR50.59 and therefore the evaluation of the proposed change determined that a license amendment is not required.

1.3 Is a license amendment required prior to the implementation of the change?

No

SE No.: 03-00160
Source Document: TM 1-03-003, Rev. 0

1.1 Activity Description
This Temporary Modification (TM) will be installed on the Thrust Bearing Wear Detector while the detector circuitry is degraded, with the intent of remaining installed to support detector circuit operation until repairs can be made.

Installation of a Temporary Modification to the Turbine Thrust Bearing Wear Detector Circuitry. Power will be bypassed around the "Lower Thrust Bearing" annunciator pressure switch straight to the "Lower Thrust Bearing" trip annunciator. This removes the annunciator functionally from the circuitry and allows the trip pressure switch to directly perform the Turbine trip, which is a change to the trip circuitry. Original design of the trip circuitry requires both the annunciator pressure switch and the trip pressure switch to actuate before a Turbine trip occurs. With the annunciator pressure switch bypassed, a Turbine trip could be initated by the Trip pressure switch directly.

1.2 Summary of Evaluation
The USAR accidents were reviewed with respect to the effects of the proposed design change in the analysis section of this evaluation. Since this activity has only a negligible effect on the frequency of a turbine trip event, the accident category remains unchanged. None of the accident frequencies were found to be affected by the implementation of the proposed change.

This evaluation analyzes USAR described design functions that are potentially affected by the proposed design change. All potentially affected design functions were found to be satisfactorily performed, such that no malfunctions of equipment were identified.

This change does not result in increasing previously evaluated release rates, changing a release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed change does not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of reactor accidents.

Based on the scope of the proposed change, and the evaluation of possible failure effects of the proposed change, no new events of a significance to be considered an accident, nor new malfunctions of an SSC important to safety, could be identified.

This evaluation analyzes the effects of the proposed change on the PNPP fission product barriers. No effects were identified.

The proposed TM does not change or affect any of the methods used in establishing the design bases and the safety analyses documented in the USAR.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 03-00197
Source Document: ECP 02-0213, Rev. 0

1.1 Activity Description
ECP 02-0213 will permanently eliminate the interconnection between the "B" loop of the Emergency Closed Cooling Water (ECCW - P42) system and the Nuclear Closed Cooling Water (NCCW - P43) system in the vicinity of the Control Complex chillers. This ECP is the second part of a change that is intended to eliminate and/or prevent the occurrence of two continuing problem areas associated with the ECCW system and thus increase the reliability and availability of the ECCW system. The first part of the change (ECP 02-0213A) will be implemented independent from, and prior to this change, and will eliminate the interconnection between the "A" loop of the ECCW system and the NCCW system in the vicinity of the Control Complex chillers.

The first problem area for the ECCW system concerns the post accident maximum allowable leakage from the ECCW system. The ECCW system must operate for seven days post accident without any make-up to the system surge tank. Demonstrating that this leakage criterion is satisfied has been a continuing problem due to the currently existing potential leak paths past the valves that isolate the ECCW system from the NCCW system. The second problem area concerns the MOVs that currently must change position on a LOOP/LOCA signal in order to align the safety related ECCW cooling water to the Control Complex chillers. Several of these MOVs have had problems demonstrating their ability to perform their safety functions.

Although some of the MOVs affected by this change have not exhibited seat leakage or problems performing their safety related "open/close" functions, implementation of this ECP is viewed as a preventative measure that will prevent these problems from occurring in the future. Installation of new piping blanks and relocation of an existing blank that would effectively isolate the "B" loop of the ECCW system from the NCCW system would eliminate the most significant potential leakage paths from the "B" loop of the ECCW system and would thus provide a solution to the major problem area that could contribute to inventory loss from the "B" loop of the ECCW system. This design change will eliminate the safety related "open/close" functions associated with MOVs 0P42F0150B, 0P42F0300B, 0P42F0330B, 0P42F0295A/B, 0P42F0325A/B, 0P42F0290, 0P42F0320, and 0P42F0550. These safety related MOVs will be removed from the Generic Letter 89-10 program thereby reducing maintenance and testing costs.

The scope of work for ECP 02-0213 includes rotating the location of blank 0P42D0002 and manual valve 0P42F0554, i.e., the blank will be installed in the location of the valve and the valve will be installed in the location of the blank. Manual valve 0P42F0554 will be placed in the open position. Blank 0P42D0001 will be removed and replaced with a full-bore spacer. MOV 0P42F0150B will be electrically determinated (both power and control logic) and placed in the closed position. MOVs 0P42F0550, 0P42F0295B, and 0P42F0325B will be electrically determinated (both power and control logic), removed, and replaced with piping blanks containing drain appendages such that the capability of draining the affected piping is maintained. MOVs 0P42F0300B, 0P42F0330B, 0P42F0295A, 0P42F0325A, 0P42F0290, and 0P42F0320 will be electrically determinated (both power and control logic) and placed in the open position. These changes will permanently align the ECCW "B" loop in its required post-accident configuration, and will align the NCCW cooling water supply to the "C" Control Complex chiller. Consequently, valves 0P42F0150B, 0P42F0295A, 0P42F0325A, 0P42F0290, and 0P42F0320 will no longer receive close signals at the onset of a LOOP/LOCA and valves 0P42F0300B and 0P42F0330B will no longer receive open signals at the onset of a LOOP/LOCA.

Control Room panel 0H13P0904 will be modified to reflect the changes to these valves and the change to the available flow paths. Subsequent to this change, the flow path for NCCW supply to both the "A" and the "B" Control Complex Chiller will be isolated and thus the NCCW system will be capable of supplying cooling water to only the "C" Control Complex Chiller.

1.2 Summary of Evaluation

The ECCW and NCCW systems, the affected safety systems [P42, Emergency Service Water system (ESW - P45), the Emergency Pump Area Ventilation system (M28), and the ESW Pump House Ventilation system (M32)], and the safety related containment and drywell isolation valves in the P43 system will still be fully capable of performing their design functions subsequent to this activity. It was determined that any additional run time on the safety systems to support operation of the "A" or "B" Control Complex chiller during normal plant operation would not degrade the design features of these systems to the point where they would be unable to perform their safety functions. In other words, the additional run time on the affected safety systems does not make these systems any more likely to fail or malfunction than currently assumed. Further, there are no new failure mechanisms introduced to any of these systems as a result of this activity. Consequently, the likelihood of System, Structure, or Component (SSC) malfunctions is not increased and the possibility of new malfunctions with different results is not created.

The change to the manner in which the "B" loop of ECCW is operated to provide cooling water to the "B" Control Complex chiller is not adverse and is actually beneficial since automatic valve positioning will no longer be required to align the safety related ECCW cooling water to the "B" Control Complex chiller. This change enhances the reliability of the "B" loop of ECCW during performance of its safety function since (1) it eliminates the predominant leak paths to the NCCW system by installation of the piping blanks that provide a physical barrier to prevent inventory loss, and (2) it eliminates the possibility of failures associated with system re-alignment since the "B" loop of ECCW will be permanently aligned in its required post-accident configuration. This aspect of the change also does not increase the likelihood of SSC malfunctions and does not create the possibility of new malfunctions with different results.

The systems affected and/or modified by this change do not initiate any currently evaluated accidents. The changes implemented by this activity do not create any situations that could initiate a new accident. Consequently, the frequency of occurrence of the currently evaluated accidents does not increase and the possibility of initiating accidents of a different type is not created. The mitigation effectiveness of the affected safety systems remains unchanged and therefore the radiological consequences from accidents and malfunctions do not increase. The affected safety systems will still be fully capable of supporting the Emergency Core Cooling System and therefore the fission product barriers will not be adversely affected. The modified piping system is qualified in accordance with typical analytical methods and does not conflict with any USAR described methodologies. Therefore, the activity does not result in a departure from a method of evaluation described in the USAR.

In conclusion, the proposed activity does not meet any of the criteria in paragraph (c)(2) of 10CFR50.59 and therefore the evaluation of the proposed change determined that a license amendment is not required.

1.3 Is a license amendment required prior to the implementation of the change?

No

SE No.: 03-00232
Source Document: PAP-0204, Rev. 0

1.1 Activity Description
These changes to PAP-0204 address the administration of the plant housekeeping program. The change scope also addresses the administration needed to implement temporary alterations in support of maintenance in accordance with 10CFR50.65.a.4.

Significantly, these procedure changes prescribe limits for the implementation of some temporary alterations to the facility that are "pre-evaluated" under 10CFR50.59. These pre-evaluated limits for temporary alterations would allow some temporary alterations to exist in the facility for up to an entire nuclear operating cycle. The temporary alterations include: the use of hand carried items; the use of hoses; the use of stanchions; the use of ladders; the use of electrical cords and conductors; the use of measurement and test equipment (electrical and mechanical); the use of items that could create internal missile hazards; control of temporary items in pipe break jet areas; control of activities that could affect external flooding; control of activities that could affect the results of internal plant flooding; temporary storage of items in the plant; the use of temporary shielding in the facility; the use of flexible impermeable material in the facility; the use of leakage containment devices; the use of glove boxes; the use of tents; the use of oil soak materials; and limitations for activities that might affect plant barriers.

These changes are made to clarify design basis requirements in the plant operations manual regarding common maintenance activities, and to administer the implementation of temporary alterations in support of maintenance.

1.2 Summary of Evaluation
The USAR accidents were reviewed with respect to the effects of the proposed procedure changes in the analysis section of this evaluation. None of the accident frequencies were found to be affected by the implementation of the proposed change.

This evaluation analyzes USAR described design functions that are potentially affected by the proposed procedure changes. Both direct and indirect effects of the proposed changes on design functions were evaluated. Potentially affected design functions were found to be satisfactorily performed. A number of the temporary alterations authorized by this procedure change require an action by the plant worker to remove the temporary alteration under all conditions, including an emergency. Since this action satisfies the definition of a maintenance activity, it is governed by 10CFR50.65.a.4 and not by 10CFR50.59. Consideration of the potential for an operator error and an increased likelihood of a malfunction associated with the credited manual action to remove/restore the temporary alteration is not required, since that consideration is made under another more applicable regulation. Therefore, the likelihood of any malfunctions of equipment important to safety are not increased by this change.

This change does not result in increasing previously evaluated release rates, changing a release duration, establishing a new release mechanism or path, or changing mitigation effectiveness. The proposed changes do not result in increased dose consequences that might impede required actions inside or outside the control room to mitigate the consequences of nuclear accidents.

Based on the scope of the proposed changes, and the evaluation of possible failure effects of the proposed changes, no new events of a significance to be considered an accident, nor new malfunctions of an System, Structure, or Component (SSC) important to safety, could be identified.

This evaluation analyzes the effects of the proposed changes on the fission product barriers. No effects were identified.

The methods of evaluation that support the proposed procedure changes are consistent with the methods used in establishing the design bases and the safety analyses documented in the USAR.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 03-00543
Source Document: USAR Change Request 03-032

1.1 Activity Description
This USAR Change Request incorporates an additional methodology that can be used to perform the turbine missile probability analysis. The Siemens Westinghouse methodology for performance of General Electric turbines, which is NRC-approved, will be added to the USAR. This will permit PNPP to use either the General Electric methodology, which is currently described within the USAR, or the Siemens Westinghouse methodology for General Electric turbines.

1.2 Summary of Evaluation
This USAR Change Request incorporates an additional methodology that can be used to perform the turbine missile probability analysis. The Siemens Westinghouse methodology for performance of General Electric turbines, which is NRC-approved, will be added to the USAR. This will permit PNPP to use either the General Electric methodology, which is currently described within the USAR, or the Siemens Westinghouse methodology approved for General Electric turbines.

The alternative turbine missile probability methodology is simply the application of a new process, which is used to obtain missile probabilities, i.e., the value of P_1. The application of this methodology does not physically alter the design or operation of the turbine or any plant System, Structure, or Component (SSC). The NRC has applied an acceptance limit for the results of the methodology. If the acceptance limit is not exceeded, then potential turbine-generated missiles will not generate unacceptable risk of damage to plant SSCs.

1.3 Is a license amendment required prior to the implementation of the change?
No

SE No.: 03-00748
Source Document: Calculation 2.4.6.14, Rev. 0 and Addendum 1; Calculation 2.4.6.15, Rev. 0; USAR Change Request 03-040; Technical Specifications Bases Change 03-039

1.1 Activity Description
This activity revises the design basis analysis that establishes the analytical limit for the Turbine Building leakage detection system high temperature set point. This analysis is revised in order to establish more accurate modeling of the Turbine Building using a different computer code, GOTHIC. The more accurate model was developed in part to respond to issues associated with the simplicity of the model in the existing analysis (Ref. CR 02-02565). The new analysis (Ref. PNPP Calculation 2.4.6.14, Rev. 0 and Rev. 0, Addendum 1) was primarily performed to help prepare for a proposed change to the plant Technical Specifications.

The major change associated with the revised analysis is the requirement to establish a new main steam leakage limit on which the turbine building leakage detection temperature analytical limit and set point are based. However, the turbine building high temperature system isolation set point and analytical limit are not changed by this activity. The new Turbine Building leakage limit is established at 32.9 Lbm/sec (Ref. - Calculation 2.4.6.15, Rev. 0) versus the existing 25 GPM (2.94 Lbm/sec).

This change results in changes to USAR section 5.2.5.1.3.b (Ref. USAR Change 03-040) and Technical Specification Bases section B 3.3.6.1 (Ref. Technical Specification Bases change number 03-039). Both of these changes reflect the increase in allowable leakage from 25 GPM (2.94 Lbm/sec) to 280 GPM (32.9 Lbm/sec).

1.2 Summary of Evaluation
An analysis was performed to determine the Turbine Building temperature assuming a Main Steam leak of 25 GPM (2.94 Lbm/sec). This analysis was performed using the computer code COMPARE. The COMPARE model of the Turbine Building was a simplistic two-volume model that did not specifically account for the effects of heat sinks, HVAC, or seasonal temperature variations. It was recently determined that this simplistic model should be replaced with a more realistic representation to support a proposed license amendment. This 10CFR50.59 evaluation addresses the change in the design basis resulting from the new analytical techniques used to evaluate the Turbine Building temperature response to Main Steam leakage.

This activity impacts the design and licensing bases for the system that has been reviewed against the licensing requirements of the existing leakage detection system as defined in the PNPP USAR and the Safety Evaluation Report (SER). In addition, the change was reviewed against the original system design requirements as defined in various Perry specific documents and generic industry standards. Analyses have been performed to determine the impact of the change on offsite dose consequences as well as maintaining main steam system integrity. The results of these evaluations indicate that the consequences of the higher leakage limit remain bounded by the current licensing base release from a main steam line break. Additionally, the system has been shown to isolate soon enough to avoid the degradation of the leak to a point where Main Steam line integrity is jeopardized. Based on the results of these evaluations it has been determined that this activity is acceptable and can be performed within the auspices of 10CFR50.59.

Specifically, the revised analytical methods and higher leakage limit do not alter the current function of the leak detection system that isolates the Main Steam system prior to the leakage degrading to a point where the system integrity is challenged such that a large steam line break would occur. Therefore, there is no increase in the frequency of occurrence of an accident resulting from this change.

The results of this activity are shown to be bounded by the existing acceptance criteria in that system integrity and reactor vessel makeup capability are not challenged by the increased leak rate. The current temperature analytical limit and set point are not changed by this activity and thus the Turbine Building environment remains unchanged by this activity. Therefore, this activity does not increase the likelihood of occurrence of a malfunction of an System, Structure, or Component (SSC).

Since the revised analytical results have been shown to be bounded by the existing analytical results, there is, therefore, no increase in the consequences of any previously analyzed accident or malfunction of an SSC because of this change.

It has been demonstrated that the revised analytical methodology maintains the Main Steam line isolation at a point where the leak does not propagate into a large break. In addition, since the current instrument temperature set points are not changed, the resulting Turbine Building temperature and steam environment is not impacted. Therefore, the possibility of new accidents or malfunctions of SSCs is not created by this change.

The revised analytical methodology has been shown to allow the leak detection system to continue to isolate the Main Steam system. The isolation time ensures that the radiological consequences of the leak remain bounded by the current Main Steam Line break analysis. In addition, the Main Steam System pressure retaining capability is maintained. That is, the isolation of the main steam system has been shown to occur prior to the crack reaching its critical length, and thus the leak does not propagate into a full steam line break. Therefore, the design basis limit for the fission barrier remains unchanged by this revised methodology.

The methodology for evaluating the Turbine Building steam leakage is not described in the USAR or in any other regulatory document. However, the new method (GOTHIC) was shown to essentially be equivalent to the old method (COMPARE). Therefore, this change is not a departure from any USAR methodology

1.3 Is a license amendment required prior to the implementation of the change?
No