**FENOC**
FirstEnergy Nuclear Operating Company

Perry Nuclear Power Plant
10 Center Road
Perry, Ohio 44081

John K. Wood
Vice President, Nuclear

440-280-5224
Fax: 440-280-8029

September 18, 2001
PY-CEI/NRR-2593L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Perry Nuclear Power Plant
Docket No. 50-440
Annual Report of 10 CFR 50.59
Safety Evaluations for 1999 - 2001

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). Those changes, tests, and experiments reported are for the period May 4, 1999 through March 23, 2001 (the end of the PNPP eighth refueling outage), and in selected cases, more recent evaluations.

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. Gregory A. Dunn, Manager - Regulatory Affairs at (440) 280-5305.

Very truly yours,

John K. Wood

Attachments

cc: Region III Administrator
NRC Resident Inspector
NRC Project Manager

IE47

## Format Description

Each 50.59 Safety Evaluation summary is presented in the following format:

SE No.: A sequentially assigned number from one (0001) to end of the period, preceded by the year; e.g., 98-0025.

Source Document: There are several sources of evaluations. The most frequent are abbreviated as shown.

CR - Condition Report
DCN - Drawing Change Notice
DCP - Design Change Package
LL&JED - Lifted Lead and Jumper and Electrical Device
MFI - Mechanical Foreign Item
ONI - Off-Normal Instruction
PAP - Plant Administrative Procedure
PEI - Plant Emergency Instruction
PIF - Potential Issue Form
PSTG - Perry Specific Technical Guidelines
PTI - Periodic Test Instruction
SCR - Setpoint Change Request
SOI - System Operating Instruction
SVI - Surveillance Test Instruction
TM - Temporary Modification
TXI -Temporary Test Instruction

Description of Change:

A short narrative describing the type of plant change.

Summary:

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?

# PERRY NUCLEAR POWER PLANT

# SAFETY EVALUATION SUMMARY

# PURSUANT TO

# 10 CFR 50.59(d)(2)

# 1999 - 2001

Safety Evaluation: 97-0007
Source Document: Drawing Change Notice (DCN) 5451, Revision 0

Description of Change:

Drawing Change Notice (DCN) 5451 is the result of the continued system review of the other systems required by Potential Issue Form Remedial Action, (PIFRA) 94-2049-002. The result of this review indicated that the Condensate Filtration System (N23) and Condensate Demineralizer System (N24) had similar double acting (air pressure required for valve to change state in either direction) air-operated valves which also had erroneous fail positions indicated on their respective P&IDs. The proposed design change removes the fail open (FO) and fail closed (FC) designations from the system P&IDs for these double acting air-operated valves that actually fail-as-is on loss of instrument air.

Summary:

I. No. The proposed drawing change does not alter or modify the function of the plant, or the N23, or N24 Systems. The DCN corrects an error on design drawings by deleting the FO or FC designations from appropriate system valves. These valves do not perform any safety related function. The system design, operation, and performance remain unchanged with this modification. No physical work is proposed on any structure, system, or component (SSC). This DCN does not constitute a change to any plant structures, systems, or components, and therefore, cannot be an initiator or contributor to an accident and cannot adversely influence any initiators or contributors that currently exist for analyzed accidents. This DCN neither alters nor modifies any radiological conditions to the public or to plant personnel. The proposed changes are a correction in documentation, which cannot prevent, degrade, or change actions described or assumed in USAR evaluated accidents. Since no credit is taken for these non-safety related systems in any design basis accident, there is no increase in the consequences for any accident already analyzed in the USAR. This DCN plays no role in mitigating the radiological consequences of accident, does not alter any assumptions previously made in evaluating the radiological consequences, or affect any fission product barriers. This DCN involves equipment that is not important to safety and does not affect physical piping configurations, mechanical or electrical equipment, or electrical control logic. The drawing changes neither affect the system or component operation or maintenance nor constitute a change to plant structures, systems, or components. No changes to hardware or previous analyses that would affect or degrade system or component reliability are made by this DCN.

II. No. The proposed drawing changes do not change how the N23 or N24 Systems, or any other plant systems, function. The change merely corrects the fail positions for non-safety related valves upon loss of instrument air as they appear on the system P&IDs. No physical work is being performed on any plant structures, systems, or components, and no new structures, systems, or components are being added. There is no relationship between the proposed changes to a new accident initiator or failure mechanism that has not already been considered or analyzed in the USAR. Therefore, the proposed change cannot lead to an accident of a different type. This design change will not alter or degrade any equipment and cannot be an initiator or contributor to any malfunction of equipment installed in the plant. The existing design of the plant, the N23, and N24 Systems remain unchanged with administration that is consistent with the current system design bases. The design change does not reflect any physical changes to any systems and does not affect any system piping configuration, mechanical or electrical equipment or electrical control logic.

III. No. This DCN does not affect the design basis of the N23 or N24 Systems and does not affect the ability of these systems to perform as designed. These systems serve no safety function and system analysis has shown that their failures will not compromise any safety related systems or

prevent safe shutdown. The N23 and N24 systems are not addressed in the Technical Specifications. Therefore, this DCN cannot affect the margin of safety as defined in any Technical Specification, Safety Evaluation Report, or Operational Requirements Manual, or their respective Bases.

Safety Evaluation: 98-0011
Source Document: USAR Change Request (CR) 98-016

Description of Change:

USAR Change Request 98-016 deletes the requirement for performing Inservice Inspections on the Turbine Bypass System/Valves. This issue was initially evaluated in the investigation for Potential Issue Form (PIF) 96-3388. USAR Change Request 98-016 modifies section 10.4.4.4, Turbine Bypass System – Tests and Inspections, by deleting the following paragraph:

"Inservice inspection of the bypass system will be performed during refuel outages. The frequency of the inspections will be determined by considering current industry practice and the history of performance of similar systems."

The USAR section 10.4.4.4 states a refuel outage frequency inservice inspection (ISI) of the bypass system will be performed. This inspection is not performed because it is not required. As investigated in PIF 96-3388, during the licensing process Perry Nuclear Power Plant (PNPP) wrongly committed to performing the ISI inspections. Prior to plant start-up (1986) the ISI Technical Engineer understood the inspections were not required and closed out the inspection commitment. The FSAR was not updated at that point, and the inspection statement in the USAR is still present. For a Boiling Water Reactor (BWR) 6 the turbine bypass system is defined as Quality Group D as referenced in the Standard Review Plan 3.2.2 – System Quality Group Classifications. Quality Group D systems are not within the scope of the ISI program and do not require the inspections referenced in USAR Section 10.4.4.4.

Summary:

I. No. The probability a steam bypass pressure regulator failure as described in the USAR Section 15.1 and 15.2 accident analyses would not be increased by this change because functional testing of the bypass valves will still be performed in accordance with the surveillance requirements (SVIs) of Technical Specification 3.7.6. Turbine bypass capability is ensured by the functional and response testing of the turbine bypass valves per the applicable system SVIs. Further inspection via the Inservice Inspection Program is not required for ensuring the intended function of the turbine bypass system or components. Since system reliability is not affected by deletion of this inservice inspection requirement, there would be no increase in the radiological consequences resulting from the malfunction of a turbine bypass system component.

II. No. USAR accident and transient analyses are currently bounded by events involving active and passive component failures that are beyond those involving any inservice inspection scope associated with the turbine bypass system. Its deletion as a testing requirement does not adversely impact any bypass valves important to mitigating the effects of these events because they are already tested to ensure proper function. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. Performance of the turbine bypass valves is currently monitored by SVIs that address function and response. Inservice Inspection is not applicable to the turbine bypass system based on its Quality Group Classification (Group D) for a BWR 6 plant. Deletion of this testing/ inspection requirement has no impact on the steam bypass systems ability to perform its intended design/important to safety function. The important safety function and non-safety function of the turbine bypass system involving steam bypass in is maintained. Thus, there is no reduction in the margin of safety.

Safety Evaluation: 98-0015
Source Document: Drawing Change Notice (DCN) 5801, Revision 0

Description of Change:

DCN 5801 corrects drawing errors to show the addition of existing Penetration Pressurization and Personnel Airlock Leakage Control System valves 1P53F579A, 1P53F579B, 1P53F580A, 1P53F580B, 1P53F581, 1P53F582, 1P53F593A, 1P53F593B, 1P53F594A, 1P53F594B, 1P53F595 and 1P53F596. Errors were identified under the corrective action program.

Summary:

I. No. The Personnel Airlock Leakage Control and Containment Penetration Pressurization Systems are accident mitigation systems and do not act as an initiator for any accidents or transients evaluated in the USAR. Based on the consideration of system interactions, no systems are potentially affected such that an accident or transient in the USAR is more likely to occur. No adverse system interactions have been identified. Both total containment allowable leakage and bypass allowable leakage as identified in Technical Specifications will be maintained. Acceptance of the containment isolation system was based upon conformance to General Design Criterion 56 and to American Society of Mechanical Engineers (ASME) Section III Class 2 or 3, and Seismic Category I requirements. The proposed change maintains conformance with these criteria. The ability of the personnel airlocks to contain radiological release is not degraded. System design basis is unaffected and any previously analyzed radiological consequences remain unchanged. Therefore, the proposed changes do not increase the likelihood or consequences of a containment failure as previously evaluated in the USAR.

II. No. The proposed changes are limited to components which are required for normal operation of the personnel airlocks and do not affect the safety function of any other system, structures or components including the Annulus Exhaust Gas Treatment System, air supply or mechanical or electrical systems of the Containment personnel airlocks. Application of the referenced valves has been evaluated and determined to be acceptable from a design, material, and construction perspective. The valves are in conformance with the applicable ASME standards for the airlocks and will not result in a degradation of the containment airlock system to perform a containment isolation function. Containment isolation is further ensured through conformance to GDC 56, regarding primary containment isolation. The change does not create any new initiators or contributors and does not increase the probability of a malfunction previously thought to be incredible. The proposed change does not alter the operation of the containment personnel airlocks and their associated support systems or of the Annulus Exhaust Gas Treatment System. Common-mode failures are not introduced by the added valves. Therefore, the proposed changes do not create a different type of malfunction of equipment than any previously evaluated in the USAR.

III. No. Reliable operation of the containment personnel airlocks and their associated support systems is required to assure the integrity of the primary containment and to limit the potential for secondary containment bypass leakage as discussed in Technical Specification Bases Sections B3.6.1.1, B3.6.1.2 and B3.6.1.3. No margin of safety in these areas is affected by this change. The margin of safety established in the previous design is based upon conformance to ASME Section III Class 1 or 2, Seismic Category I qualification, and conformance to GDC 56. The added valves comply with these requirements. Existing containment design was based upon criteria which are independent of the quantity of isolation valves, thus the added valves do not reduce the margin of safety as defined in the Technical Specifications Bases.

Safety Evaluation: 98-0027
Source Document: Simple Modification Request Form (SMRF) 97-4083, Revision 0

Description of Change:

SMRF 97-4083 replaces existing obsolete flow/indication and differential pressure instruments in the Condensate Demineralizer System (N24). The existing Fischer & Porter flow totalizers/indicators and the existing differential pressure switches are obsolete and replacement parts are unavailable. This SMRF replaces these instruments with current state of the art Moore Products Controllers and instrumentation that will perform the same function as the existing instrumentation.

Summary:No. A failure in the N24 System does not increase the probability of occurrence of any of the accidents previously evaluated in the USAR. The N24 System is a water treatment system that is not required for safe operation of the plant. This system cannot initiate an accident, nor is it used to mitigate the consequences of any previously defined accident. The replacement components meet or exceed the same design requirements, codes, and standards as the original components. The tubing and fittings used in the installation comply with specification SP-2200 and the wiring complies with SP-2250. The supporting of the components and the tubing is done in the same manner as the other items located within local panel 1H51P013. Since there are no analyzed USAR accidents associated with the N24 System, there is no increase in the consequences of the accidents that have previously been evaluated in the USAR.

II. No. The modification to the flow and alarm instrumentation has been performed in accordance with the originally installed codes and standards. Since the modification has not changed the function of the N24 System, and there have not been any failure modes added that would impact systems required for safe shut down or safe plant operation, the modification does not create the possibility of an accident of a different type than those previously evaluated in the USAR. The effect of introducing the new digital equipment, relative to Electro-Magnetic Interference (EMI)/Radio Frequency Interference (RFI) susceptibility or emissions, resulted in no concerns with regard to equipment or adjacent component performance. However, the consequences of equipment failure (old or new design) can be tolerated since the net effect would have no impact on a system, structure, or component performing a safety function. Adherence to the codes and standards ensures that the new design meets the requirements of the existing system, and will therefore not create a different type of malfunction of equipment than any previously evaluated in the USAR.

III. No. There are no Technical Specifications, USAR, Operational Requirements Manual, NRC Safety Evaluation Report and Standard Review Plan sections applicable to the N24 System. Technical Specification Section 6.3.1 (Reactor Coolant System Chemistry) and associated Table 6.3.1-1 were reviewed for applicability to this SMRF. These sections pertain to Reactor Coolant Chemistry Limits which are not impacted by this modification. Performance of the N24 System has not been affected by the modification, since it will continue to monitor the flow parameters in the same manner as the replaced equipment. Consequently, there is no effect or change on the performance of the N24 system. Margins of safety for the old mechanical totalizer will remain for the new replacement component such that the failure consequence of this equipment will produce identical results as currently exist. Therefore, this modification does not reduce the margin of safety as defined in the basis for any Technical Specifications.

Safety Evaluation: 98-0065
Source Document: Modification Request Form (MRF) 98-0013, Revision 0

Description of Change:

Modification Request Form (MRF) 98-0013 resolves a pressurization problem by adding a leak-off path from the 20-inch Shut Down Cooling (SDC) header to the suppression pool. This flow path will be open during normal plant operation to relieve system pressure. The leak-off path is achieved by installing a new ¾ inch pipe that connects the SDC header to the suppression pool via the E12 Residual Heat Removal (RHR) Heat Exchanger (HX) vent line. This Safety Evaluation has been submitted to the NRC for review and approval by a letter dated March 17, 1999 (PY-CEI/NRR-2362L).

Summary:

I. Yes. The current penetration configuration was evaluated in the Supplement to Safety Evaluation Report (SSER) 2. Penetration P-431 was originally evaluated as having two closed remote manual valves (1E12-F0073B and 1E12-F0074B) on the outside of the containment and one normally closed automatic valve (check valve 1E12-F0558) inside containment. This modification will now have one normally opened, automatic, containment isolation valve outside of containment and a check valve inside of containment. Changing the containment penetration configuration from two normally closed valves outside of containment to a penetration with one normally opened valve outside of containment may increase the probability of malfunction of equipment important to safety as evaluated in SSER 2. Even though the modified penetration meets the requirements of the General Design Criteria (GDC), the fact that the outboard containment isolation valve will now have to change state to perform its safety function (containment isolation), could increase the probability of occurrence of a malfunction of equipment important to safety that was previously evaluated in the USAR.

II. No. The change will not create any new systems, add any new equipment or compromise the function of any systems, structures or components to the extent that the possibility of an accident of a different type than any previously evaluated in the USAR would be created. This change will create no new initiators or contributors for an event that could be considered a new accident. This change also will not affect any known accident initiators or contributors; therefore, it will not increase the probability of an accident previously thought to be incredible. This activity does not make a previously non-credible event credible, nor does it unbound a previously bounded event.

The change will have no adverse affect 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. The proposed activities will not create any new systems, or add any new equipment that can compromise the function of any systems, structures, or components. This proposed activity will not result in any new equipment failures, therefore, this proposed activity will create no new initiators or contributors for an event that could be considered a new accident. This proposed activity also will not affect any known accident initiators or contributors.

III. No. This modification now directs the leakage past valve 1E12-F0008 to the heat exchanger vent line. By administratively controlling the flow rate through the existing vent line to less than or equal to 0.30 gpm, the water entering the weir pool will be at approximately atmospheric pressure and less than 212° F. In other words, the medium entering the

suppression pool will be water and not steam. To summarize, the initial conditions for a "water cannon" type event, with the modified flow rate, will not be met. The probability of a water hammer type event occurring with the new configuration will be less than the probability of this same event happening in the current configuration. Therefore, there are no new affects on the piping system due to this modification.

Safety Evaluation: 98-0076
Source Document: Modification Request Form (MRF) 98-0024, Revision 0

Description of Change:

Modification Request Form (MRF) 98-0024 will eliminate the physical components of the Steam Condensing Mode (SCM) of the Residual Heat Removal System (RHR). References to the Steam Condensing Mode of RHR were previously eliminated from the USAR via USAR Change Request 98-0064 and Safety Evaluation 98-0072.

Summary:

I. No. There are no accidents or transients noted in the USAR with which this portion of piping is associated. The modified piping configuration will slightly reduce the effects of jet impingement loads from this portion of the piping system. This modification does not affect the RHR system ability to perform its required function as described in General Design Criteria (GDC) 34 Residual Heat Removal. Since this portion of the RHR piping system is not used to mitigate an accident, the change cannot increase on-site radiation doses that would impede actions necessary to mitigate the consequences of an accident, nor can this change directly or indirectly affect mitigation of radiological consequences of an accident. The change will not increase on-site radiation doses that would impede actions necessary to mitigate the consequences of an accident. The change will not alter, degrade or prevent actions described or assumed in an accident discussed in the USAR. This change will not directly or indirectly affect mitigation of radiological consequences of an accident. The change will not adversely affect any fission product barriers because the actual piping (radiological barrier) is not changed. With the installation of the blind pipefitting, the probability of malfunction of the RHR system due to voiding of the piping system or a water hammer event is reduced. No other equipment relies upon the SCM relays or process instrumentation, so the removal, de-energization, or deactivation of this equipment cannot create a malfunction of any equipment important to safety. Active SCM electrical components are not relied upon to prevent an accident or mitigate the consequences of an accident.

II. No. The change will not create any new systems, add any new equipment or compromise the function of any systems, structures or components to the extent that the possibility of an accident of a different type than any previously evaluated in the USAR would be created. This change will create no new initiators or contributors for an event that could be considered a new accident. This change also will not affect any known accident initiators or contributors; therefore, it will not increase the probability of an accident previously thought to be incredible. This activity does not make a previously non-credible event credible, nor does it unbound a previously bounded event.

The proposed change will not affect specific component or overall system performance in a manner which could introduce a new initiator or failure not previously considered in the USAR or increase the probability of a malfunction of equipment important to safety previously thought to be incredible to the point where it becomes as likely to occur as the malfunctions currently discussed in the USAR. The proposed changes will not create a situation that could result in a common mode failure of equipment important to safety.

III. No. No margins of safety will be affected by either the removal of hand switches of deactivated Motor Operated Valves, or the removal or de-energization of process instrumentation of the unused SCM of the RHR system. The additions of the pipe blind fittings and pipe caps do not have any impact on the flow characteristics or performance of the RHR system. Structurally, the modified piping system was modeled and reanalyzed to ensure that the stress limits for design, normal, upset, emergency and faulted conditions are in accordance

with the piping design requirements. The piping analysis methodology did not credit any component performance, such as allowable stress, above the currently defined acceptance level of the American Society of Mechanical Engineers (ASME) Code. No other equipment relies upon the SCM relays or process instrumentation, so the removal, de-energization, or deactivation of this equipment will not reduce any margin of safety as defined in the bases of the Technical Specifications.

Safety Evaluation: 99-0001
Source Document: Perry Security Plan, Revision 26

Description of Change:

These changes to the Security Plan pertain to personnel access control measures for the protected and vital areas. These changes are considered to be safeguard information and as a result are managed in accordance with 10CFR73.21.

Summary:

I. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 99-0004
Source Document: Design Change Package (DCP) 98-0052, Revision 0

Description of Change:

The objective of the Design Change Package (DCP) 98-052 is to install several modifications to improve the overall reliability and performance of the Feedwater penetrations. Specifically, the Feed Water Level Control System (FWLCS) is being rerouted to the bonnet of the Motor Operator Valves (MOV) to increase the time available for the operator to take action. This increase in time, results from a significantly shorter system fill time for the proposed design. In addition, the plant is being modified to add the capability to close the MOVs in the event that the normal and emergency alternating current (AC) electrical power sources to the MOVs are not operable. This Safety Evaluation has been submitted to the NRC for review and approval by letters dated September 9, 1998 (PY-CEI/NRR-2322L), January 1, 1999 (PY-CEI/NRR-2352L), March 4, 1999 (PY-CEI/NRR-2370L), and March 18, 1999 (PY-CEI/NRR-2376L).

Summary:

I. Yes. The FWLCS is an accident mitigation system. The FWLCS is a design feature that helps to mitigate the consequences of a Loss Of Coolant Accident (LOCA). The system is not associated with the initiation of a LOCA. The current design affects the consequences of the Feedwater line break outside containment and the loss of coolant accident (LOCA). For the Feedwater line break outside containment, the current design provides isolation of the Feedwater lines by closure of the check valves in the Feedwater lines. The proposed design change will not alter this. The current FWLCS consists of two independent subsystems that provide redundant water seals for each Feedwater line, thereby isolating containment following a LOCA. The proposed design provides equal capability while improving the reliability of the system. Adding the capability for supplementary electrical power for the MOVs increases the reliability for high integrity leakage protection. However, the proposed design change reduces the redundancy and independence of the FWLCS and the independence of the electrical distribution system. This reduction could potentially increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

II. Yes. With the proposed design change, in the event that the Feedwater MOVs failed to close as a result of Division 1 AC electrical power sources not being operable, Division 3 electrical power would be available to close the valves. The system requirements would be met by the operable portions of the FWLCS subsystems. As a result, the proposed design change decreases the consequence of this event. However, in the event that a Feedwater MOV failed to close for a different reason, the FWLCS would no longer be capable of mitigating the radiological consequences of the event. Leakage through the failed MOV would no longer be limited to no greater than 1 gpm because the check valves in the Feedwater lines and the FWLCS would no longer be providing a tested seal to limit the leakage. This is an increased consequence relative to the current system design.

III. No. The margin of safety is not reduced since the containment penetration and FWLCS remain capable of performing their design functions. The proposed design change for the FWLCS will be subject to the requirements of Technical Specification 3.6.1.8, which are identical to the requirements for the current FWLCS design. The leakages through the Feedwater line containment penetrations, and the Residual Heat Removal and Reactor Water Clean Up branch lines that are connected to the Feedwater lines outside of containment will be maintained in

accordance with the current regulatory limits, and will remain consistent with the assumptions of the radiological analyses. No NRC radiological acceptance limits will be exceeded as a result of this design change. Therefore, the margin of safety as defined in the Technical Specifications Bases is not reduced.

Safety Evaluation: 99-0005
Source Document: Plant Administrative Procedure (PAP) 0126, Revision 1

Description of Change:

This revision incorporates the guidance on shift staffing and shift complement previously contained in PAP-0110. PAP-0110 is being cancelled. There are no significant changes made to this guidance. Attachment 10, Control Room Areas, was revised to remove the desk in the Shift/Unit Supervisor's work area to be consistent with USAR Fig. 13.5-1. Attachment 10 was also changed to delete the reference to the Shift Technical Advisor's (STA) office and to conservatively reduce the size of the Horseshoe Area when compared with USAR Fig. 13.5-1.

Summary:

I. No. The proposed change to the Control Room Area attachment conservatively reduces the distance the supervising operator at the controls will be away from critical control room panels. This will help to ensure the complete monitoring of plant parameters and will ensure adequate operator response time to manipulate plant equipment controls when required. Deleting the designation of the STA office in the Control Room is purely administrative and will have no impact on the ability of the STA to perform required functions.

The proposed change will ensure the proper monitoring and control of plant systems and 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. Reducing the normal operating area of the operator at the controls in the Control Room will enhance operator access and visibility. This procedural change has no physical impact on any plant system, structure, or component (SSC); does not change the manner in which any SSC is maintained, operated, or tested; and does not create any new SSC interactions. Therefore, the possibility of an accident or the malfunction of equipment important to safety different than any previously evaluated in the USAR is not created as a result of this PAP revision.

III. No. The proposed change to the Control Room Area attachment conservatively reduces the distance the supervising operator at the controls will be away from critical control room panels. This will help to ensure the complete monitoring of plant parameters and will ensure adequate operator response time to manipulate plant equipment controls when required. Deleting the designation of the STA office in the Control Room is purely administrative and will have no impact on the ability of the STA to perform required functions.

Reducing the normal operating area of the operator at the controls in the Control Room will enhance operator access and visibility. The margin of safety as defined in the Technical Specifications and Bases, Operational Requirements Manual, Operating License, Safety Evaluation Report, and USAR is not reduced.

Safety Evaluation: 99-0049
Source Document: Design Change Package (DCP) 96-0107, Revision 1

Description of Change:

This DCP implements physical changes to upgrade the Meteorological Monitoring System (MMS). The majority of the existing instruments at the 10-meter and 60-meter heights on the meteorological tower are being retained, some are removed, and no new instruments are added. The instrumentation at the tower is being changed from a multiple channel configuration (primary, primary validation, backup, backup validation) to a two channel configuration. System "A" consists of wind speed, wind direction, ambient temperature, dewpoint, precipitation and barometric pressure instruments. System "B" consists of wind speed, wind direction, and ambient temperature instruments. The two existing equipment shelters (primary and backup) at the base of the tower are being replaced by one new shelter. The signal conditioning and data acquisition hardware are being installed in the new shelter and mounted in racks dedicated to each system. Sensor channels will be disconnected from the existing system and transferred to the new shelter in such a way as to ensure that none of these channels will be inoperable in excess of seven days.

Summary:

I. No. The MMS does not interact with the plant beyond providing data to the plant computer, and therefore, is not an accident initiator, so it cannot cause events or transients resulting in USAR evaluated accidents. The upgrade to the MMS does not interact with any component or plant system in a manner that could adversely affect the performance of systems, such as the pressure retaining capabilities of either the Reactor Coolant Pressure Boundary, the Primary Containment, or Emergency Core Cooling Systems, etc. Redundancy within the MMS and independence from other plant systems will not be reduced. The MMS is remotely located from the plant. Offsite power is provided to the MMS, plant power is not involved. This change does not affect any initial conditions or assumptions of any accident. Therefore, there is no impact on accident mitigating systems or the Primary Containment.

The only interface between the MMS and the plant is a new fiber optic communication link that connects the new MMS sensor data acquisition system to the plant computer. This fiber optic link cannot generate Electro-Magnetic Interference/Radio Frequency Interference (EMI/RFI). No cables connected to, associated with, or routed with safety related cables are present in the MMS shelter to provide a physical medium for coupling of any EMI/RFI signals to safety related systems or components. The MMS does not interface with any accident mitigation or safety related control function. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. This system does not interface with any system other than the plant computer and the scope of this change is limited to modifications of the meteorological monitoring system. The change will not create any new systems, nor compromise the functions of any systems, structures, or components affected by this change. This change will not result in any new failures of equipment important to safety, nor create new initiators or contributors for an event which could be considered a new accident. Therefore, this activity does not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than any previously evaluated in the USAR.

III. No. The function of the MMS is unchanged. The design continues to conform with at least the minimum required number and types of meteorological instruments as defined in the Operational Requirements Manual (Section 6.2.8). The new system design is equal to or superior to the existing system, and the new design does not reduce the degree of conformance

with the Regulatory Guides (RG) or NUREGs which form a part of the plant's licensing basis as stated in the USAR. The Technical Specifications do not address operability of the MMS, nor does the MMS affect, impede, or interface with any systems or equipment addressed by the Technical Specifications or their bases. Therefore, the margin of safety is not reduced.

Safety Evaluation: 99-0050
Source Document: Drawing Change Notice (DCN) 5712, Revision 0;
USAR Change Request (CR) 97-068

Description of Change:

The items being evaluated consist of drawing and USAR updates that correct deficiencies identified by Potential Issue Forms (PIF) 97-0346, 97-0469, and 97-0943 for the Emergency Closed Cooling Water (ECCW) System. The proposed changes to the affected USAR figures, tables, drawings and pages will add component operating flow rate information and component design flow rate information. Clarification of minor items such as, table information, titles, and component information will also be performed.

Summary:

I. No. The proposed drawing changes and USAR changes do not alter or modify the function of the plant or the ECCW System, including interfacing systems. The system's design, operation, and performance remain unchanged. Design flow rates to essential equipment have not been changed and are being added to the P&ID/USAR for convenience only. Notes on the P&ID related to the Nuclear Closed Cooling (NCC)/ECCW isolation under accident conditions are consistent with the system's established design basis. The proposed changes do not alter or modify any radiological conditions to the public or onsite personnel or any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. The proposed changes do not affect any fission product barriers. The proposed changes do not degrade system or component reliability. The proposed changes cannot prevent actions described or assumed in an accident discussed in the USAR. Therefore, the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. No physical work is being performed on any plant structures, systems, or components; and no new structures, systems, or components are being added. The proposed changes are not associated with an accident initiator or failure mechanism not already considered in the USAR. The proposed changes cannot lead to an accident of a different type. The proposed drawing changes and USAR changes will not alter or degrade any equipment; and can not be an initiator or contributor to any malfunction of equipment installed in the plant. The proposed changes, which clarify and enhance existing information and correct minor discrepancies, are based on currently established design basis information, data, and system configuration. Therefore, the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. Since the proposed changes will have no impact on the function or operation of the systems, the margin of safety and availability of the ECCW System and interfacing systems will not be reduced. The proposed changes will not degrade the capability of the ECCW System and systems it interfaces with to mitigate the effects of postulated transients and accidents. The proposed drawing changes and USAR changes do not impact the Technical Specification (TS), Safety Evaluation Report (SER), or Operational Requirements Manual (ORM), and their respective Bases, or other related documents and Bases. Therefore, the review of the Technical Specifications, bases for the technical specifications, the ORM and the USAR has shown that there is no clear trend toward a reduction in the margin of safety as a result of these changes.

Safety Evaluation: 99-0052
Source Document: Perry Security Plan, Revision 27

Description of Change:

These changes to the Security Plan pertain to personnel access control measures for the protected and vital areas. These changes are considered to be safeguard information and as a result are managed in accordance with 10CFR73.21.

Summary:

I. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 99-0053
Source Document: Simple Modification Form (SMRF) 98-5049, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 98-5049 will improve the reliability of the Reactor Core Isolation Cooling (RCIC, E51) system turbine steam exhaust drain pot drain line. The drain line has experienced repeated problems affecting its ability to drain resulting from orifice (1E51-D004) clogging and check valve (1E51-F047) sticking due to iron oxide build-up in the piping. The drain pot must be able to drain to prevent potential damage to the RCIC turbine. SMRF 98-5049 will remove check valve 1E51-F047 and orifice 1E51-D004 from the drain line and replace them with a piping spool piece.

Summary:

I. No. The modified drain line does not interface with any plant systems in such a manner as to increase the probability of occurrence of a previously evaluated accident, nor does it create a condition that would increase the probability of occurrence of any previously evaluated accidents. The modifications to the drain line will not create a condition that would increase the radiological consequences of any other previously evaluated accidents, nor does the modification increase on-site radiation doses such that actions to mitigate the radiological consequences of an accident would be impeded, and the modification does not directly or indirectly affect the ability of any other plant system to mitigate the radiological consequences of an accident.

The modification does not compromise the accident mitigating capability of the RCIC system, nor does it affect or interface with any equipment important to safety, and it is not associated with any malfunctions of equipment important to safety previously evaluated in the USAR. Therefore, this modification does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. This activity cannot increase on-site radiation doses such that actions to mitigate the radiological consequences of a malfunction of any other equipment important to safety would be impeded, nor does it directly or indirectly affect the ability of any other system to mitigate the radiological consequences of a malfunction of equipment important to safety.

II. No. The modified drain line does not interface with any plant systems in such a manner as to create the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR. This modification will not create any new systems, add any new equipment, or compromise the function of any systems, structures, or components. This modification does not increase the probability of malfunction of the RCIC system. Therefore, the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. Modification of the RCIC turbine exhaust drain pot drain line does not affect operation of the RCIC system during injection to the vessel, and therefore does not reduce the margin of safety for the RCIC system as defined in the bases for the Technical Specifications. The drain line does not interface with any plant systems, structures, or components in such a manner as to reduce the margin of safety as defined in the basis for any other Technical Specification.

Safety Evaluation: 99-0054
Source Document: USAR Change Request (CR) 99-068

Description of Change:

USAR Change Request 99-068 modifies the USAR to replace the existing Perry Nuclear Power Plant (PNPP) Operational Quality Assurance Program description, established in Chapter 17.2, with the First Energy Nuclear Operating Company Quality Assurance Program Manual (FENOC QAPM). Additionally, USAR Table 1.8-2, "Compliance With QA Related NRC Regulatory Guides," is being replaced with Table 1 of the FENOC QAPM. Other editorial changes to the USAR are made to support the FENOC QAPM.

Summary:

I. No. The Quality Assurance Program for the operations phase of Perry has no input to the accident analysis described in USAR Chapter 15. The Quality Assurance (QA) Program changes made in USAR Change Request 99-068 do not impact the design, function, or operation of the plant. The changes are administrative in nature and do not involve any hardware or operational changes to the plant. Since the accident analysis is not affected, the probability of occurrence of an accident previously evaluated in the USAR is not increased. Similarly, the radiological consequences for any accident previously evaluated in the USAR are not increased since the accident analysis is not being affected.

There are no changes to plant equipment, systems or operations procedures made by this USAR change request. The operational QA Program establishes quality assurance requirements and controls for performing safety related and augmented quality processes and does not direct the design or operation of plant equipment. Therefore, the QA Program changes being made do not increase the probability of occurrence of a malfunction of equipment previously evaluated in the USAR, nor do they increase the radiological consequences of a malfunction of equipment important to safety previously evaluated.

II. No. This USAR Change Request is applicable to the operational QA Program description. It does not involve or impact any plant systems, equipment, or operations procedures. There are no hardware changes to structures, systems, or components being made. The QA Program does not factor into the accident analysis contained in USAR Chapter 15. As a result, the USAR Change Request and associated QA Program changes do not create the possibility of an accident of a different type than previously evaluated in the USAR. The changes do not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated.

III. No. There are no hardware changes or system operation changes being made with this USAR Change Request. The quality assurance program description provides quality assurance controls for safety related and augmented quality systems and processes. It has no direct impact on the operation or design function of any structures, systems or components or their applicable technical specification limits. There is no specific margin of safety associated with USAR Chapter 17.2 or any of the other proposed changes under this USAR Change Request. Therefore, this USAR Change Request does not reduce the margin of safety for any technical specification item.

Safety Evaluation: 99-0055
Source Document: Equivalency Change Package (ECP) 99-8048, Revision 0

Description of Change:

Equivalency Change Package 99-8048 makes changes to USAR Figure 9.4-23 and the Building Heating System (P55) P&ID, as follows:

- Replaces 1P55-F552A through E, 2-1/2" manual globe valves with manual ball valves.
- Changes outlet connector on Heater Bay heating coils (1M41-B0001A through E) from 1-1/2" to 2-1/2".

Summary:

I. No. This change alters the heater coil outlet valve type and the heater coil outlet connection size on USAR Figure 9.4-23. These changes maintain the current heating coil capacity and so do not reduce the heating capability of the Heater Bay Ventilation (M41) System. The M41 and P55 Systems in the Heater Bay, their components, and related systems are non-seismic, non-safety and are not required for the safe operation or safe shutdown of the plant. Failure of the M41 and/or P55 System in the Heater Bay is not considered an initiating event for the accidents evaluated in the USAR. Changing these details does not, and will not, result in failure to meet the design, material, and construction standards applicable to the M41 or P55 Systems. These changes will not affect overall system performance in any manner. They do not alter any assumptions made in evaluating the consequences of an accident or malfunction of equipment important to safety as described in the USAR. These changes do not play any role in mitigating the consequences of an accident and do not affect any fission product barrier. Therefore, this ECP does 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 changes to the M41 and P55 System components on USAR Figure 9.4-23 do not affect the operation of any equipment required to support the safe operation of the plant. These changes do not result in any increase in the probability of the failure of any equipment that is considered an accident initiator. These changes do not affect the design, operation, availability or response of any important to safety equipment to any transients/accidents as described in the USAR. None of the M41 and/or P55 equipment is addressed in any USAR FMEA. These changes will not result in any other failure mode or failure effect not already addressed in the USAR. Therefore, this ECP will not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The changes to the M41 and P55 System components on USAR Figure 9.4-23 do not affect the operation of any equipment required to support the safe operation of the plant. These changes do not modify, degrade or prevent any actions for any accident discussed in the USAR. They do not alter any assumptions made in evaluating the consequences of an accident as described in the USAR. These changes do not play any role in mitigating the consequences of an accident and do not affect any fission product barrier. These changes do not degrade reliability of any SSC important to safety. These changes do not affect the design, operation, availability or response of any important to safety equipment to any transients/accidents as described in the USAR. Therefore, these changes will not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 99-0056
Source Document: Partial Closure of Modification Request Form (MRF) 98-0013, Revision 0

Description of Change:

The partial implementation of this MRF changes penetration P-431 to a penetration that is in full compliance with General Design Criteria GDC-56. The MRF 98-0013 has been fully installed and tested, however, valve 1E12-F0073B cannot be left in the normally opened position until receipt of NRC approval of a License Amendment Request associated with this MRF.

Summary:

I. No. Partial implementation of this MRF changes penetration P-431 to a penetration that is in full compliance with General Design Criteria GDC-56. However, valve 1E12-F0073B will not be changed to normally open until PNPP receives approval of the LAR. The failure of the piping integrity was not considered as an initiator to the loss of Residual Heat Removal (RHR) during shutdown cooling since the piping was designed in accordance with the requirements of ASME Section III. The piping stresses are within the ASME Code allowables and therefore the piping is not assumed to fail. The change can not cause events or transients resulting in USAR evaluated accidents. The change can not be an accident initiator. The change can not increase onsite radiation doses that would impede actions necessary to mitigate the consequences of an accident. The change will not increase onsite radiation doses that would impede actions necessary to mitigate the consequences of an accident. The change will not alter, degrade or prevent actions described or assumed in an accident discussed in the USAR. This change will not directly or indirectly affect mitigation of radiological consequences of an accident. The change will not adversely affect any fission product barriers because the containment isolation capability and the total leakage outside of containment are not changed. There is no increase in the probability of malfunction of the RHR system or its supported systems. The change will not increase onsite radiation doses that would impede actions necessary to mitigate the consequences of a malfunction of equipment important to safety. This change will not directly or indirectly affect mitigation of radiological consequences of a malfunction of equipment important to safety. This change will also not increase the radiation dose to the public. This change will not adversely affect any systems that could impact fission product barriers.

II. No. The change will not create any new systems, add any new equipment or compromise the functioning of any systems, structures or component to the extent that the possibility of an accident of a different type than any previously evaluated in the USAR would be created. This change will create no new initiators or contributors for an event that could be considered a new accident. This change also will not affect any known accident initiators or contributors; therefore, it will not increase the probability of an accident previously thought to be incredible. This activity does not make a previously noncredible event credible, nor does it unbound a previously bounded event.

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. The proposed activities will not create any new systems, or add any new equipment that can compromise the functioning of any systems, structures, or components. This proposed activity will not result in any new equipment failures, therefore, this proposed activity will create no new initiators or contributors for an event that could be considered a new accident. This proposed activity also will not affect any known accident initiators or contributors.

III. No. Partial closure of this MRF will allow the pressure buildup in the RHR Shut Down Cooling (SDC) header to be vented into the suppression pool via the new line. The venting

operation will not establish full flow through this line and will only be momentary. The water entering the weir pool will be at approximately atmospheric pressure and therefore 212°F or less. In other words the medium entering the suppression pool will be water and not steam. To summarize, the initial conditions for a "water cannon" type event, during venting operations, will not be met. The probability of a water hammer type event occurring with the new configuration will be less than the probability of this same event happening in the current configuration. Therefore, there are no new effects to the piping system due to this modification.

Safety Evaluation: 99-0058
Source Document: Simple Modification Request Form (SMRF) 99-5007, Revision 0

Description of Change:

SMRF 99-5007 will remove Residual Heat Removal (RHR) relief valve 1E12-F036 and install a blind flange at the connection to the process pipe, and remove a portion of the associated relief valve discharge piping and install a pipe cap at the open end of the discharge pipe. Relief valve 1E12-F036 is located on the RHR condensate return line to the Reactor Core Isolation Cooling (RCIC) system which is no longer used since the RHR Steam Condensing Mode was eliminated via DCP 98-0024. Removal of this valve will provide a permanent solution to the leakage outside containment that occurs through the seat of the relief valve.

Summary:

I. No. The modified RHR to RCIC condensate return line and the modified 1E12-F036 relief valve discharge line do not affect the operation of the RHR and RCIC systems, and therefore the accident mitigating capability of the RHR and RCIC systems is not compromised. The modified RHR to RCIC condensate return line and the modified 1E12-F036 relief valve discharge piping do not adversely impact any modes of operation of the RHR or RCIC systems. The accident mitigating capability of the RHR and RCIC systems is not compromised by this change and consequently the radiological consequences of any malfunction of equipment important to safety that relies on the RHR or RCIC system for mitigation will not be increased. The RHR to RCIC condensate return line and relief valve discharge piping do not affect or interface with any other equipment important to safety, and are not associated with any malfunctions of equipment important to safety previously evaluated in the USAR. Therefore, this change does not increase the probability of occurrence of a malfunction of equipment important to safety, nor does it increase the radiological consequences of any malfunction of equipment previously evaluated in the USAR.

II. No. The modified RHR to RCIC condensate return line and the modified 1E12-F036 relief valve discharge piping do not interface with any plant systems in such a manner as to create the possibility of an accident of a different type than previously evaluated in the USAR. The modified piping cannot create an accident of a different type than previously evaluated in the USAR since it will be installed in accordance with the requirements of American Society of Mechanical Engineers (ASME) Section III, American National Standards Institute (ANSI)/ASME B31.1, and International Standards Society (ISS)-2000; and since pipe ruptures are not postulated in the modified piping. The modified piping cannot create a malfunction of the RHR or RCIC system, nor does it interface with any other plant system in such a manner as to create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The piping modifications to the condensate return line and relief valve discharge piping will be installed in accordance with the requirements of ASME Section III, ANSI/ASME B31.1, and ISS-2000. Installation in accordance with these requirements will ensure that the margin of safety inherent to the piping codes is maintained. Consequently, the pressure boundary provided by the RHR to RCIC condensate return line and the pressure boundary of the Auxiliary Building Equipment drain header tree will be retained subsequent to this activity. Therefore, the modified 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 basis for any Technical Specification.

Safety Evaluation: 99-0060
Source Document: Simple Modification Request Form (SMRF) 98-5043, Revision 0

Description of Change:

The purpose of this modification is to replace and relocate the temperature transmitter sensing element that controls the hot water valve for the Heater Bay Ventilation (1M41) System heating coils to a location that will provide a better mixed air sample. The existing transmitter 1M41-N040 takes an average temperature across all 5 of the heating coils. The sensing bulb is mounted directly behind the heating coils. This transmitter is being replaced by 2 individual transmitters with sensing bulbs (1M41-N040A/B) that will be mounted on each M41 Air Supply fan housing. The relocation will produce a better sample of the average supply air temperature and will more effectively control the supply air temperature and assist in the freeze prevention of the heating coils. In addition, a three-way solenoid valve will be installed in the air signal lines from the temperature transmitters to temperature controller 1M41-R0042. The temperature controller controls the water supplied to the heating coils.

Summary:

I. No. The materials of the design, and the transmitter itself are of the same manufacturer and model series. The installation conforms to the same design codes and standards as the originally installed instrumentation. The accuracy of the instrumentation remains the same. The system design is improved since the air now being monitored includes any air that may be drawn into the air supplied by the supply air fans. The M41 instrumentation being changed by this modification is not a direct accident initiator. The change does not result in any changes to system interfaces. This design modification does not increase the probability of occurrence of an accident previously described in the USAR, and does not increase the radiological consequences of an accident or malfunction of equipment important to safety as previously described in the USAR. Consequently, there is no increase in the probability of an occurrence of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. This design change increases the reliability of the M41 System. The relocation and the addition of a transmitter increases system reliability and the supply air temperature regulation accuracy. The modification to the M41 System has been performed using the same codes and standards as the original design. This modification will not change the function of the M41 System, and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. Therefore, this design modification does not create, or increase the possibility of, a different accident or malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. There are no applicable Technical Specifications, USAR, Operational Requirements Manual, NRC Safety Evaluation Report or Standard Review Plan sections for the M41 System. Performance of the M41 System has not been affected by this modification since the system will continue to provide conditioned air to all the affected spaces currently being supplied. Consequently, there is no negative affect on, or change to, the performance or reliability of the M41 system. Therefore, the proposed change does not reduce the margin of safety as defined in the basis for the Technical Specifications.

Safety Evaluation: 99-0061
Source Document: Simple Modification Request Form (SMRF) 97-5060, Revision 0; Setpoint Change Requests (SCRs) 1-99-1047 and 1-99-1048, both Revision 0

Description of Change:

This design change performs several changes to the Hot Water Heating System (P55). The relief valve (1P55-F0516) on the system expansion tank (1P55-A0002) is replaced to increase the setpoint from 75 psig to 110 psig and the tank level operating band is reduced from 24 inches to 15 inches by the replacement of level switch 1P55N0032. A new reverse flow check valve (1P55-F0806) is being installed in the expansion tank makeup supply piping. A pressure switch (1P55-N0091) that annunciates system low flow is being replaced with a similar unit having an adjustable reset. In addition, a new design flag on the Piping and Instrument Diagram (P&ID) is being added to correctly identify the design pressure of the expansion tank 1P55-A0002. SCRs 1-99-1047 and 1-99-1048 are included to update the setpoint for 1P55-N0032 and required reset value for 1P55-N0091.

Summary:

I. No. The P55 System is physically separated from safety related equipment and is not considered a direct initiator or mitigator of any USAR evaluated accident or transient, and does not fulfill any function as a fission product barrier. There is no increased potential for flooding or spraying. Building heating capability will be maintained, thus the changes will not impede necessary actions to mitigate the consequences of malfunctions of equipment important to safety. The changes do not alter, degrade or prevent any actions related to the USAR accident analyses. The modification does not add any interactions with Systems, Structures, or Components (SSC) important to safety or equipment considered as initiators of any event. System design/safety functions are maintained. The changes comply with the original American Society of Mechanical Engineers (ASME)/American National Standards Institute (ANSI) B31.1, and ASME Section VIII code requirements.

II. No. The P55 system continues to be a moderate energy system such that there is no increased potential for line break type accidents. The changes do not alter or create any new systems, system interactions or operating functions and no new equipment types are introduced. The changes will not alter the redundancy or independence of any systems. Implementation of the proposed changes will not cause any previously bounded event to become unbounded and will not cause any previously evaluated event that was considered incredible to become credible. The new/replacement components are of a similar design to the existing equipment and continue to satisfy the applicable ASME/ANSI B31.1 and ASME VIII codes. The changes do not alter any redundancy or separation of any important to safety equipment, thus susceptibility to common mode or common cause failures is not possible. The modifications do not introduce any new failure modes or effects.

III. No. The P55 system is not addressed in the Technical Specifications (TS), TS Bases or the Operational Requirements Manual. In addition, the NRC's Safety Evaluation Report and Supplements do not discuss the P55 system. The system provides non-safety related heating. Thus, the system can only indirectly affect structures, systems and components important to safety and their associated margins of safety. The changes do not alter the system's capability to provide heat to maintain the plant buildings at a minimum of 60°F. All the changes are within ASME/ANSI code requirements and the existing equipment design. The USAR Section 9.4.10 discussion of maintaining a minimum system pressure above the extraction steam heating source pressure may be considered an implied margin to prevent the spread of contamination, and the modifications will maintain this margin.

Safety Evaluation: 99-0063
Source Document: Simple Modification Request Form (SMRF) 97-5100, Revision 0

Description of Change:

Lighting panel R71-P045 is identified on drawing 220-746 as an essential lighting panel. A plant design feature is that essential lighting panels have the capability of being provided diesel backed power in the event of a Loss Of Offsite Power (LOOP). This design change is written to change the power feed to panel R71-P045. The power feed will be changed from 2R25-S0009 (F2C08), that is not diesel backed, to panel 1R25-S0009 (F1C08) which is diesel backed.

Summary:

I. No. The R71-P045 essential lighting panel does not contribute in any manner to the occurrence of an accident evaluated in USAR. The R71-P045 lighting panel does not directly interface with a System, Structure, or Component (SSC) important to safe plant operation. The addition of the R71-P045 panel load to the Division 2 Emergency Diesel Generator (EDG) forced shutdown load will not impact the EDG system or response capability. The R71-P045 essential lighting panel is separate from other equipment evaluated in the USAR and is not capable of increasing the doses realized by the public or onsite personnel. A failure of the R71-P045 panel would not alter existing accident radiological consequences or create radiological consequences for accidents not evaluated as having radiological consequences. Therefore, the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. This design change to lighting panel R71-P045 will not result in a change or potential effect on an SSC that performs a safety related function. There is no change or potential effect on equipment used to mitigate the consequences of accidents. The design change does not result in a credible malfunction or potential affect on an SSC used to mitigate the consequences of an accident. No modifications will be made to safety related plant equipment or safety related system functions. There is no significant effect on the electrical power system, the associated breakers, or breaker coordination. The addition of the purposed load is within the capabilities of the presently installed breakers and the present breaker setpoints. Therefore, the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. There are no references to panel R71-P045 or Essential lighting in the Technical Specifications or the Technical Specifications Bases. Additionally, there are no acceptance limits or limiting conditions associated with the R71-P045 lighting panel. There are no operating margins associated with the R71-P045 lighting panel. There are no margins of safety in the USAR that are associated with this panel. Therefore, there is no clear trend toward a reduction in the margin of safety as a result of these changes.

Safety Evaluation: 99-0064
Source Document: USAR Change Request (CR) 99-060

Description of Change:

This USAR Change Request (CR) incorporates revisions to USAR Table 11.2-3 and Figures 11.2-1, Sheets 1, 2 & 3. These changes illustrate the permanent isolation of the Liquid Radwaste Sumps System (G61) from the abandoned portions of Unit 2 equipment, floor and chemical drain sumps. In addition, the Unit 2 sump pumps were eliminated from Table 11.2-3.

Summary:

I. No. This CR illustrates the separation of the G61 System from abandoned portions of the Unit 2 Buildings and has no impact on systems required to support Unit 1 operation. The installed welded caps provides permanent isolation between the abandoned Unit 2 portions of the G61 system from the operating plant and serves to provide and maintain positive control of the system boundary. These changes to the system were done to the same design specification, codes and standards as the existing design. Valves 0G61F0651 and 0G61F0652 are "administratively" controlled by Radwaste Lineup Instruction (RLI)-G61, in a locked closed position to isolate the 6" lines to abandoned Unit 2 Drain Sumps. The addition of a locked closed device to these valves does not affect the Unit 1 system performance. 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, or change directly or indirectly mitigation of radiological consequences of any malfunction of equipment important to safety. In addition, no fission product barriers are affected. Radionuclides, release rates, release mechanisms, or impact to radiological barriers are not affected by this change. No equipment has been removed which would affect, compromise or impact the performance, function, and interaction with Unit 1. This system is not relied upon in any accidents evaluated in USAR nor is it an accident initiator.

II. No. This CR will not create any new systems, nor alter or impact the function of any operating system. No equipment has been removed or altered which would affect the function and interaction with Unit 1. The installations of permanent welded caps provide permanent isolation between the abandoned Unit 2 portions of the G61 system and the operating plant. These changes to the system were done to the same design specification, codes and standards as the existing design. The addition of a locked closed device to the valves does not affect the system. This device serves to mitigate any radiological or environmental consequences by providing and maintaining positive control of the system boundary. This change does not increase the effects of any event that was previously bounded by other accidents to 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 currently evaluated in the USAR.

III. No. This change affects the abandoned Unit 2 portion of the LRW system in which no Technical Specifications exist. The Technical Specifications (TS) and its Bases, Operating License (OL), and Operational Requirements Manual (ORM) are not affected by this change to this system. Therefore, there is no change to any margin of safety provided in the design of the system as discussed in the USAR, TS and its Bases, the OL, the ORM, or the NRC Safety Evaluation Report.

Safety Evaluation: 99-0065
Source Document: Specification Change Notice (SCN) 676, USAR Change Request (CR) 99-089

Description of Change:

Specification Change Notice (SCN) 676 incorporates revisions to Drawing D-320-733 and D-320-737 to illustrate permanent isolation of the Liquid Radioactive Waste System (G50) from the abandoned portions of Unit 2 Spent Resin System of the Solid Radwaste System (G51) and Reactor Water Cleanup Filter Demineralizers System (G36). Changes to Non-Design Specification Drawing D-302-731 and D-302-736 have also been included to illustrate isolation from other abandoned Unit 2 systems associated with the G50 system. This is accomplished by including permanent asset numbers to the drawing where blind flange and butt weld caps were previously installed. In addition, the USAR CR deletes Unit 2 asset numbers for abandoned equipment listed in Table 11.2-5.

Summary:

I. No. This change illustrates the separation of the Liquid Radioactive Waste System (LRW) from abandoned portions of the Unit 2 facility. The installed blind flange and welded caps provide permanent isolation between the abandoned Unit 2 portions of the G51 and G36 systems from the operating plant. This change to the system was performed under the same design specification, codes and standards as the existing design, and serves to provide and maintain positive control of the system boundaries. Radionuclides, release rates, and release mechanisms are not affected by this change. Radiological barriers are not impacted by this change. No equipment has been removed which would affect, compromise or impact the performance or function of the originally evaluated design. No new failure modes or effects are created by this change. This system is not relied upon in any accidents evaluated in USAR nor is it an accident initiator. This drawing change to show the isolation of the LRW System from the abandoned Unit 2 Buildings has no impact on systems required to support Unit 1 operation. The proposed change does not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR, or change directly or indirectly the mitigation of radiological consequences of any malfunction of equipment important to safety.

II. No. This change will not create any new systems, nor alter or impact the function of any system. This change makes no structure, system or component changes to the plant that would impact any design function. No equipment has been removed or altered which would affect the function and interaction with Unit 1. The locked closed boundary valve has no impact on Unit 1 operation. The installation of a permanent blind flange and welded caps provides permanent isolation between the abandoned Unit 2 portions of the G50, G51 and G36 Systems and the operating plant. This change to the system was performed under the same design specification, codes and standards per the existing design, and serves to provide and maintain positive control of the system boundaries. This change does not increase the effects of any event that was previously bounded by other accidents. This change does not increase the probability of any significant event previously thought to be incredible in the USAR, or increase the probability of any malfunction of equipment important to safety.

III. No. This change affects the abandoned Unit 2 portion of the LRW system in which no Technical Specifications exist. The Technical Specifications (TS), Operating License (OL), and Operational Requirements Manual (ORM) are not affected by this change. Therefore, there is no change to the margin of safety provided in the design of the system as discussed in the USAR, TS and Bases, the OL, the ORM, or the NRC Safety Evaluation Report.

Safety Evaluation: 99-0067
Source Document: Drawing Change Notice (DCN) 5607, Revision 0

Description of Change:

DCN 5607 adds the following design details to the Emergency Service Water (ESW) system P&ID: identification of the Service Water/Emergency Service Water intake head and intake tunnel riser nominal diameters, Service Water/Emergency Service Water discharge tunnel riser and discharge nozzle nominal diameters, and location and elevation of the weir wall in the ESW pump house. These items do not represent physical changes to the plant nor do they represent changes to the operation of the plant; they merely provide an additional level of detail on the P&ID. These details are presently shown on USAR Figures 3.8-64, 3.8-65, and 3.8-66.

Summary:

I. No. The changes implemented via DCN 5607 do not represent physical changes to the plant or changes to the plant operating procedures. Since the activity does not constitute a physical change to any plant system, structure, or component, nor does it represent a change to the operation of any plant system, the activity cannot create any conditions that would increase the probability of occurrence of any previously evaluated accidents. The changes do not affect or alter the capability of the ESW system to perform its safety function. Further, since the activity does not constitute a physical change to any plant system, structure, or component, nor does it constitute a change to the operation of any plant system, the activity cannot create any conditions that would increase the radiological consequences of any previously evaluated accidents.

II. No. The changes to the ESW System P&ID do not represent physical changes to the existing plant configuration nor do they represent a change to any operating procedures. The activity cannot conceivably hinder or prevent the ESW system from performing its safety function. This activity will not create any new systems or compromise the functioning of any existing systems, structures, or components and therefore this activity will not create the possibility of an accident of a different type than any previously evaluated in the USAR. The activity cannot hinder or prevent the ESW system from performing its safety function and the activity cannot create a malfunction of the ESW system. The activity does not affect any other plant systems in such a manner as to create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR. The activity cannot conceivably introduce any new failure modes or result in any new or adverse failure effects. Therefore, this activity does not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The changes to the ESW System P&ID do not represent physical changes to the existing plant configuration nor do they represent a change to any operating procedures. The proposed changes do not and cannot affect the design basis of any system, structure, or component and therefore do not prevent the ESW system from performing its safety function. Since the proposed changes will not adversely affect the function or operation of any system, structure, or component; the margin of safety and availability of any plant systems, structures, and components will not be reduced. Therefore, this activity does not reduce the margin of safety defined or implied in the basis for any Technical Specification.

Safety Evaluation: 99-0071
Source Document: Design Change Package (DCP) 97-6090, Revision 0

Description of Change:

The design modification will replace two obsolete (analog) recorders with a single (digital) recorder. The 21 input signals from the two existing recorders will be connected to the new single recorder. The existing Low Pressure Core Spray (LPCS) recorders (1E21-R612 and R616) are utilized for monitoring the "Valve Stem Leak-Off Temperature" in the Reactor Coolant System (RCS) Leak Detection System and only provide alarm and indication. Both existing recorders have no controlling function and are non-safety related, as will be the new recorder. They are located in the Control Room on Panel 1H13-P865.

Summary:

I. No. The "Valve Stem Leak-Off Temperature" recorders do not provide any control function to any system that could increase the probability of occurrence of any accident. Failure of the recorder itself will not initiate any of the accidents described in the USAR because the purpose of the recorder is to record and alarm, with no controlling functions. Although a new failure mode has been identified, no new failure effects have been identified related to replacing the existing recorders with the new recorder that would be different than those previously identified for the original recorders. There are not failure modes resulting from this activity, which would create an accident. This modification does not increase the probability of occurrence or consequences of an accident, or malfunction of equipment important to safety, as previously evaluated in the USAR.

II. No. The recorder is a monitoring/indication device that does not perform any actuation function that could initiate any accident. In the unlikely event that a common mode failure renders the replacement recorder inoperable, diversity exists in other plant instrumentation which can be utilized by the operator to assess the situation and take appropriate action. This activity does not cause any event evaluated in the USAR as being incredible to become credible, and does not cause any event that was previously bounded to become bounding. There are no failure modes resulting from this activity which would create an accident. However, should a gross malfunction occur of the replacement recorder, operations personnel would still have indication of the valve stem leakage from the 21 valves by means of the Drywell Equipment Drain Sump level (LR-R619). This modification will not create the possibility of an accident of a different type or create a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. This modification does not impact the required number of channels, or their operability, or the minimum number of channels required to be operable, or the "Action Statements", or the frequency of channel checks/calibrations. The recording and display ranges of the replacement recorder remain the same as the existing. Furthermore, replacing the two existing recorders with one recorder is considered an enhancement in the reliability of the RCS Leak Detection. Therefore, this activity does not reduce the margin of safety as defined in the bases for any technical specification.

Safety Evaluation: 99-0072
Source Document: USAR Change Request (CR) 99-095

Description of Change:

This CR incorporates the permanent isolation of the Emergency Service Water (ESW) System (P45) from the abandoned portions of Unit 2. USAR text and figures were revised to indicate one unit operation and to eliminate the Unit 2 heat loads.

Summary:

I. No. This CR illustrates the separation of the P45 System from abandoned portions of the Unit 2 facility. This was accomplished by electrically disabling and locking valves 2P45-F0040A, 2P45-F0040B, and 2P45-F0160 as well as including existing spectacle flanges 2P45-D0013 and 2P45-D0014, and associated piping as isolation boundaries. The addition of a locking device to these valves does not affect the system. Existing spectacle flanges 2P45-D0013 and 2P45-D0014 will continue to isolate Unit 2 de-ice lines from the operating unit and will be monitored to ensure flange gasket integrity. No equipment has been removed which would affect, compromise or impact the performance, function, and interaction with Unit 1. Unit 1 systems, structures, and components are unaffected by these changes. Hence, accident analysis has not been affected. The radiological consequences of an accident described in the USAR are unaffected by the change made to any of these systems. In addition, no fission product barriers are affected.

II. No. This CR will not create any new systems, nor alter or impact the functioning of any operating system. The locked closed boundary valves and the elimination of the Unit 2 heat loads have no impact on Unit 1 operation. The existing spectacle blind flanges provide permanent isolation between the abandoned Unit 2 portions of the systems and the operating plant. These changes to the system were done to the same design specification, codes and standards per the existing design. The addition of a locking device to the valves does not affect the system. These devices serve to mitigate any radiological or environmental consequences by providing and maintaining positive control of the system boundaries. This CR made no structure, system or component changes to the plant that would impact its design function. No equipment had been removed or altered which would affect the function and interaction with Unit 1.

III. No. The Technical Specifications and its Bases, Operating License, and Operational 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 SE Report. The existing spectacle blind flanges provide permanent isolation between the abandoned Unit 2 portions of the systems from the operating plant. These changes to the system were done to the same design specification, codes and standards per the existing design. The addition of a locking device to the valves does not affect the system and serves to provide and maintain positive control of the system boundary. The boundary isolation valves have been evaluated for long term isolation and found to be acceptable.

Safety Evaluation: 99-0073
Source Document: USAR Change Request (CR) 99-096

Description of Change:

Incorporates Offgas (N64) component shielding source term changes in USAR Table 12.2-5.

Summary:

I. No. The change to the shielding source terms used for Offgas shielding determination does not result in any physical changes to the plant or to any Offgas system component. These shielding source term changes do not result in any changes to plant shielding and are not the result of any physical plant modification. They do not affect the radiological environment for any important to safety equipment or any equipment qualification zone. It does not change in any way the manner in which the system operates or responds. Failure of Offgas components is addressed in USAR section 15.7.1.1 with the source terms from Table 15.7-3A. This change does not affect those source terms or any part of the associated analyses. The as designed shielding continues to provide the desired dose rate reductions such that dose rates are within applicable radiation zoning criteria (see USAR Figures 12.3-1 through 5). These changes do not modify, degrade or prevent any actions needed to mitigate the consequences of any accident or any malfunction of equipment important to safety discussed in the USAR. They do not alter any assumptions made in evaluating the consequences of an accident or malfunction of equipment important to safety as described in the USAR. Therefore, this change does 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 change to the shielding source terms for the applicable Offgas components in USAR Table 12.2-5 does not affect the operation of any equipment required to support the operation of the plant. The shielding source terms being changed are used to determine shielding requirements to ensure conformance to the radiation zoning criteria as set forth in USAR Figures 12.3-1 through 5. None of the affected components are in or adjacent to any area/zone containing important to safety components. Accordingly, there is no affect on the radiological environment for any important to safety equipment or equipment qualification zone. These changes do not affect the design, operation, availability or response of any important to safety equipment to any transients/accidents as described in the USAR. These changes will not result in any other failure mode or failure effect not already addressed in the USAR. Therefore, this change will not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The change in the shielding source terms for the applicable N64 system components in USAR Table 12.2-5 does not affect the operation of any equipment required to support the operation of the plant. These changes do not modify, degrade or prevent any actions for any accident discussed in the USAR. They do not alter any assumptions made in evaluating the consequences of an accident as described in the USAR. These changes do not play any role in mitigating the consequences of an accident and do not affect any fission product barrier. These changes do not degrade reliability of any Structure, System, or Component (SSC) important to safety. These changes do not affect the design, operation, availability or response of any important to safety equipment to any transients/accidents as described in the USAR. These changes do not cause dose to personnel to exceed 10CFR20 limits nor to exceed USAR Chapter 12.3 radiation zoning criteria. Therefore, these changes will not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 99-0074
Source Document: Drawing Change Notice (DCN) 5870, Revision 0

Description of Change:

DCN 5870 was initiated against Drawing 021-001-001 to revise the reference location and the required Net Positive Suction Head (NPSH) for Residual Heat Removal (RHR) Mode A-2 (Low Pressure Coolant Injection (LPCI), Reactor Pressure 0 psig).

Summary:

I. No. The RHR system is required to mitigate several accident scenarios. The proposed USAR change to revise the NPSH reference location and add the additional NPSH data for extended pump flowrates does not impact the probability that an accident will occur or the radiological consequences of an accident. The NPSH available is in excess of the NPSH required for all RHR flow conditions. Therefore, the proposed USAR change will not have an impact on the capability of any Engineered Safety Feature (ESF) to mitigate the radiological consequences of an accident. The proposed USAR change does not impair the availability of the RHR or any other ESF system, structure or component, and therefore the initial conditions assumed in an accident are not affected. The proposed USAR change does not affect the function or operation of the RHR system and no additional system interactions are being created. Therefore, the proposed change will not increase the probability of occurrence of an accident or a malfunction of equipment important to safety or increase the radiological consequences as a result of an accident or a malfunction of equipment important to safety.

II. No. The proposed change does not create a new failure mode or mechanism for any system, structure or component in the plant. The proposed change does not create any additional accident initiators. The change in the NPSH reference point, the increased NPSH requirement at runout flow, and the extension of the flowrate shown on the pump performance curves will not cause a change in the RHR pump performance. Due to the large amount of NPSH Available, this activity will not result in degradation of the RHR pumps. No ESF function or performance is affected by the proposed USAR change. Therefore, the proposed USAR change will not create a different type of accident or malfunction of equipment important to safety.

III. No. There are no Technical Specifications that control the NPSH requirements for the RHR pump. The RHR system has design limits associated with safety class systems, structures, and components; however, this change does not affect the ability of the system to perform its intended safety related functions. The margin of safety, as related to this change, would be best described as having at least the NPSH Required for all design flowrates. The NPSH Available is in excess of the NPSH Required for all design flowrates. Therefore, since the change in the NPSH reference point and the increased NPSH Required for runout flow do not affect the RHR pump operation or performance, no reduction in the margin of safety will be experienced.

Safety Evaluation: 99-0076
Source Document: USAR Change Request (CR) 99-102

Description of Change:

This change to the USAR identifies the potential for small batch type releases of low activity from areas that would not be discharged via the normal effluent points, and would not follow the flow paths for ventilation as described in the USAR.

Summary:

I. No. The proposed change will not result in a significant release of activity from the site, nor will it increase the doses to the public that would challenge the 10CFR20, 10CFR50 and 40CFR190 limits. If low-level radioactivity is detected, the radionuclide mix will be incorporated into the effluent release calculations. Batch releases would only be allowed from low-activity areas that do not have the potential for a significant release. These areas are approved and controlled with chemistry instructions. This change will not change the source term for the gaseous effluent releases that were evaluated. This is based on the termination of the batch release through administrative controls in the event that sampling activities or plant conditions indicate that limits would be challenged. Therefore, this USAR change will not directly or indirectly affect the probability of occurrence, or the radiological consequences of, an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. This activity does not modify existing equipment. Temporary equipment would be used for these small batch releases. A batch-type release is anticipated to occur as a result of maintenance activities, surveillances, or a potentially hazardous chemical atmosphere may exist in an area or building where discharge through the normal effluent points are not possible or operationally desirable. These releases are short in duration with administrative controls to terminate the activity if plant conditions change. This equipment will be capable of properly assessing the release using the guidelines of Reg. Guide 1.21 to ensure compliance with 10CFR20 and 10CFR50. Therefore, this USAR change will not directly or indirectly affect the probability of occurrence, or the radiological consequences of, an accident or malfunction of equipment important to safety not previously evaluated in the USAR.

III. No. The batch releases being performed with this change would only be allowed from low activity areas that do not have the potential for a significant release. These releases will be monitored for activity using the methodology established in the Offsite Dose Calculation Manual for inoperable effluent monitoring equipment. The margin of safety is considered to be the nuclide activity released from the site and the doses associated with those releases. If activity is detected, a quick dose rate will be performed, and if results are > 1% of the 10CFR50 dose rate limits the release will be terminated. This conservative limitation falls in line with the current source term that represents a very small fraction of the allowable limits specified in 10CFR20, 40, 190 and Appendix B, or 10CFR50 Appendix I. As such, no margin of safety will be reduced as a result of the proposed change.

Safety Evaluation: 99-0077
Source Document: Equivalency Change Packages (ECP) 99-8055, 8056, and 8057, all Revision 0; Setpoint Change Requests (SCR) 1-99-1092 through 1098, all Revision 0

Description of Change:

The purpose of this modification is to replace the obsolete Ultra-Sonics Level detectors located on Division 1, 2, and 3 Fuel Oil Storage Tanks (FOST) 1R45-A0002A(B) and 1R45-A0004. The existing transmitters 1R45-N0188A/B and 1R45-N0008 monitor the fuel oil inventories and alarm at the 7 day and 24 hour inventory levels. The transmitter also provides a 4-20 milli-amp signal to the control room for indication and alarming purposes.

Summary:

I. No. The level measurement and instrumentation are of the same design and type as the originally approved capacitance level measurement system licensed for plant startup. The installation conforms to the same design codes and standards as the originally installed instrumentation. The accuracy of the instrumentation remains equal or better. The Standby Diesel Generator Fuel Oil (R45) instrumentation being changed by this modification is not an accident initiator. Neither the loss of alternating current (AC) Power or the probability of Station Blackout increases due to the change in this level instrumentation. The changes do not reflect any changes in the R45 System that will cause it to operate outside of applicable design or testing limits. The change does not result in any changes to system interfaces. The failure analysis indicates that this design modification will not increase the probability of a diesel system failure or transient. This design modification does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously described in the USAR.

II. No. The modification to the R45 System is performed using the same codes and standards as the original design. This modification will not change the function of the R45 System, and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This change also does not cause any event evaluated in the USAR to be incredible to become credible or any event that was previously bounded to become bounding. This design change does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. This change has no effect on the Technical Specifications (3.8.3.1), the Operational Requirements Manual, NRC Safety Evauation Report or Standard Review Plan. Performance of the R45 System has not been affected by this modification since the system will continue to provide FOST level monitoring using the existing methods of indication and alarming. Consequently, there is no negative affect on the performance or reliability of the R45 System. Operator action to dipstick the FOST is part of the original license basis and not a new requirement. This proposed change does not eliminate or alter these alarms or indication, or compromise the ability to independently verify FOST level using the dipstick. Therefore, these changes do not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 99-0078
Source Document: Drawing Change Notice (DCN) 5527, Revision 0

Description of Change:

DCN 5527 incorporated descriptions of skid mounted equipment for Reactor/Turbine Generator Trip (N32), Steam Bypass and Pressure Regulation (C85), and Reactor Recirculation (B33) system drawings.

Summary:

I. No. The proposed drawing changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The proposed changes do not impose increased testing requirements on important to safety systems or equipment. The proposed drawing changes do not create any new failure modes or failure effects for equipment important to safety; do not degrade the reliability of any plant system, or introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of all systems are not affected by the proposed changes. The proposed drawing changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed drawing changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, will not increase the probability of an accident previously thought to be incredible, do not make a previously non-credible event credible, and do not create any new failure modes or failure effects for equipment important to safety. The proposed drawing changes do not affect the design and operation of any plant Systems, Structures, or Components (SSC), and does not affect how SSCs react to normal and abnormal transients; does not degrade any equipment; and will not create an initiator or contributor to any malfunction of equipment installed in the plant not previously evaluated in the USAR.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplements to the SER are not adversely affected by the proposed drawing changes. The proposed drawing changes are not related to Technical Specification Bases. The proposed drawing changes will not adversely affect the design basis of any SSC, or adversely affect the ability of any SSC to perform as designed. Since the proposed drawing changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. The proposed drawing changes will not degrade the capability of SSCs to mitigate the effects of postulated transients and accidents. Therefore, the proposed drawing changes will not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 99- 0079
Source Document: USAR Change Request (CR) 99-109

Description of Change:

The proposed USAR changes are clarifying; that the "normal" position of the Suppression Pool Cleanup (SPCU) system return line isolation and SPCU suction valves are closed which allows the High Pressure Core Spray (HPCS, E22) suction to be aligned to its normal position, i.e., the Condensate Storage Tank (CST).

Summary:

I. No. This activity does not increase the probability of occurrence, or the radiological consequences, of an accident or malfunction of equipment important to safety as previously evaluated in the USAR. The insertion of the "normally closed" designator maintains the original design and licensing basis. The HPCS System, as a part of the Emergency Core Cooling (ECC) System, is provided to respond to a reduction in the inventory of reactor coolant. The position of SPCU suction and return valves does not interact with any accident initiators or contributors. Additionally, the initial position of the SPCU system valves does not affect any single failures or operator errors associated with each accident. With the valves in the closed position at the time of an accident, the probability of successful containment isolation is intuitively higher than if the valves are in the open position. Therefore, the safer mode of operation is to keep the valves in the post-accident position when they are not in use. Based on this evaluation, this does not affect any of the initiators or contributors to the accidents previously evaluated in the USAR because the initiators and contributors to the accidents do not involve and are not affected by the proposed changes.

II. No. This activity does not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than any previously evaluated in the USAR. The designation of the valves as "normally closed" as opposed to N.O. or N.C. will not add or remove Systems, Structures or Components (SSC) to or from the plant. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not instigate or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible.

III. No. This activity does not reduce the margin of safety as defined in the basis for any Technical Specification. The Technical Specifications, ORM, and the SER/Supplement to SER are not adversely affected by the proposed changes. The proposed changes will not adversely affect the design basis of any SSC. The changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of any SSCs the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 99-0080
Source Document: Work Management Instruction (WMI) 004, Revision 2;
USAR Change Request (CR) 99-110

Description of Change:

The proposed changes provide clarification of the Nuclear Safety Operations Analysis (NSOA) described within the USAR and portions of the NSOA that have been superseded by other license basis documents in order to reflect individual plant operating practices.

Summary:

I. No. The proposed USAR and procedure changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The proposed changes do not impose increased testing requirements on systems or equipment important to safety. The proposed changes do not create any new failure modes or failure effects for equipment important to safety; do not degrade the reliability of any plant system, and do not introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of protective systems are not affected by the proposed changes.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible, do not make a previously non-credible event credible, and do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not affect the design and operation of any plant Structures, Systems or Components (SSCs), and do not affect how SSCs react to normal and abnormal transients, do not degrade any equipment; and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR.

III. No. The Technical Specification, Operational Requirements Manual and the Safety Evaluation Reports (SER/SSER) are not adversely affected by the proposed changes. The proposed changes are specifically allowed by the TS bases. The proposed changes will not adversely affect the design basis of any SSC, or adversely affect the ability of any SSC to perform as designed. The proposed changes will not degrade the capability of SSCs to mitigate the effect of the postulated transients and accidents. Therefore, the proposed changes will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0081
Source Document: Simple Modification Request Form (SMRF) 99-5023, Revision 0

Description of Change:

This Simple Modification Request Form (SMRF) replaces concrete in a Heater Bay El. 647'-6" knockout wall that was left open by Field Change Request (FCR) 018834 during Re-Fueling Outage (RFO) 4 with four removable ¼" thick aluminum panels that will be bolted to the 5'x7' embedded steel framing. Closing the knockout wall opening with the aluminum panels is expected to facilitate wall removal during equipment outages. Radiation measurements with the wall removed have shown that radiation levels on both sides of the knockout wall are well below the Zone III design guidance levels.

Summary:

I. No. Replacing the concrete in the knockout wall at elevation 647'-6" with aluminum panels or leaving the knockout wall open cannot effect Heater Bay flooding. This change does not affect available water volumes or free volume storage in the Heater Bay. USAR Figure 12.6-4 shows the heater bay does not add post accident radiological consequences. This change will not adversely affect or impede the ability of plant operators to perform post accident functions or to access any required areas. This change does not alter, degrade or prevent any actions related to the USAR accident analyses. Radiological assessments show that radiation levels on both sides of the knockout wall are well below the level III design guide levels and that the concrete is not necessary. This change does not create any interactions with other systems. The concrete wall in question is not a building boundary wall and is not credited as a fire rated barrier. The knockout wall is classified as non-seismic and has no environmental qualification requirements. Leaving the knockout wall open or bolting aluminum panels to the existing steel framing does not impose additional loads or impose testing requirements on important to safety (ITS) equipment, structures, or systems. Therefore, the proposed change does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The aluminum knockout wall is a passive, non-safety related component and is not considered an initiator of any USAR transient or accident. Adding the aluminum panels does not add any new component interconnections. Replacing the 1N27-C003A knockout wall concrete with aluminum panels or leaving the knockout wall open will not cause any previously bounded event to become bounding and will not cause any previously evaluated event that was considered incredible to become credible. There are no potential failures associated with the knockout wall changes and no new malfunctions or failure effects are identified. The knockout wall and proposed changes are non-seismic, non-code, non-pressure retaining, non-load carrying and the wall is passively designed for radiation shielding only. The 1N27-C003A knockout wall change has no direct or indirect adverse affect on any equipment including ITS equipment, and therefore can not influence any malfunction of equipment ITS.

III. No. The knockout wall and the proposed changes for the "A" Feedpump Turbine Room are located 5'-9" east of column line HB-1 and are not addressed in the Technical Specifications or the Operational Requirements Manual. This change does not result in the creation of, or change to, any high radiation boundaries as defined by Technical Specification 5.7. The proposed change does not change, create, or delete any radioactive effluent release pathways as defined by Tech Spec 5.5.4 and the Offsite Dose Calculation Manual. Therefore, no margin of safety has been reduced.

Safety Evaluation: 99-0082
Source Document: USAR Change Request (CR) 99-107

Description of Change:

This change will eliminate the Plant Chemist position and title and replace it with the Supervisor of Chemistry Unit positions and titles, as maintaining the ANSI N18.1-1971 requirement for supervisors not requiring NRC licenses for radiochemistry.

Summary:

I. No. The replacement of the Plant Chemist position and title represents a change that is limited to the description of the organizational structure and title that complies with the ANSI N18.1-1971 requirements. No functions or activities have been eliminated, only re-assigned. There is no impact upon the regulatory guidelines of Reg. Guide 1.8 or ANSI N18.1-1971. The requirements in ANSI N18.1-1971 for supervisors not requiring NRC licenses for radiochemistry will be met with the Chemistry Unit Supervisors. The design or operation of the plant will not be affected by this organizational change. Hence, the probability of occurrence of an accident or malfunction of equipment evaluated in the USAR will not be impacted.

II. No. This change is solely organizational in nature and all applicable licensing commitments will remain satisfied. Therefore, this change will not affect the design or operation of the plant in any manner which could be construed to create a new mode of, or increase the probability of, a new accident or malfunction of equipment important to safety. Hence, accident analyses as described in the USAR will not be impacted. Therefore, a malfunction of equipment important to safety or an accident of a different type than previously evaluated in the USAR is not created.

III. No. No plant functions or activities have been altered or eliminated. The site remains in compliance with Reg. Guide 1.8, ANSI N18.1-1971, the level of commitment to Reg. Guide 1.8 and ANSI N18.1-1971 as detailed in USAR Table 1.8-2, the Technical Specifications, the Operating License, the Operational Requirements Manual, the Quality Assurance Program, and the bases for these documents. The design or operation of the plant will not be affected. Hence, no margin of safety as described in the bases for any Technical Specification has been reduced.

Safety Evaluation: 99-0084
Source Document: USAR Change Request (CR) 99-111

Description of Change:

This USAR Change Request revises USAR Figure 10.1-4 Sheet 2 of 2 to document a location change for a low point drain appendage. USAR Figure 10.1-4 contains the piping diagram of the Condensate System (N21). The drain appendage is on the normal Condensate Storage Tank (CST) make-up water line that supplies water from the Mixed Bed Demineralizer and Distribution System to the CST via the High Pressure Condenser.

Summary:

I. No. There are no accidents or transients evaluated in the USAR that are initiated or impacted by the failure of the drain appendage on the CST make-up water line. The location of the drain appendage upstream of the CST make-up water control valve 1N21-F0395 instead of downstream will not affect the operation of any Structure, System or Component (SSC), or their anticipated response to USAR analyzed accidents. No equipment important to safety is affected by the proposed change. The proposed change does not directly or indirectly affect the design, material, or construction standards of any SSC. No changes are being made to any assumptions or inputs previously made to assess radiological consequences and no fission product barriers are affected. Therefore, the proposed change will not increase the probability of occurrence of an accident or malfunction of equipment important to safety. In addition, no changes to the radiological consequences as a result of an accident or malfunction of equipment important to safety will occur.

II. No. The proposed USAR change represents a location change for the CST make-up water line low point drain. The proposed change does not create a new failure mode or mechanism for any SSC in the plant. The proposed change does not create any additional accident initiators. The drain appendage location will not degrade or prevent acceptable SSC performance. No Engineered Safety Feature (ESF) function or performance is affected by the proposed USAR change. Therefore, the proposed USAR change will not create a different type of an accident or malfunction of equipment important to safety.

III. No. There is no specific margin of safety defined for the drain appendage or the CST make-up water line to which the drain appendage is attached. The drain appendage location upstream of the CST make-up water control valve 1N21-F0395 (versus downstream) has no impact on any SSC and has no operational plant impact. The proposed USAR change does not adversely impact the design or licensing basis for any SSC. SSC reliability, redundancy, operation and availability are unchanged. Therefore, the proposed USAR change can have no impact on the margin of safety implied or specifically stated by any licensing document, including the USAR, Safety Evaluation Report, Operational Requirements Manual, Core Operating Limits Report (COLR), Offsite Dose Calculation Manual (ODCM), Plant Operating Procedures (POP), Plant Process Control Program (PCP), and Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0085
Source Document: USAR Change Request (CR) 99-112

Description of Change:

USAR CR 99-112 removes USAR Figure 9.4-11 for the Unit 2 Emergency Service Water Pump House Ventilation System (M32) and modifies the USAR description of M32 regarding Units 1 and 2. Unit 2 has been abandoned and the M32 System for Unit 2 was never installed.

Summary:

I. No. Unit 2 has been officially abandoned. The Unit 2 portion of the M32 System is not required for Unit 1 operation, nor is credit taken for the Unit 2 M32 System in mitigating any accident in Unit 1. Therefore, its removal does not increase the probability or consequences of an accident for Unit 1. The removal of equipment that is not necessary to support Unit 1 in any way, and was never installed. Therefore, it does not increase the probability of an accident or malfunction of equipment important to safety, or the radiological consequences of any accident or equipment malfunction.

II. No. The abandonment of Unit 2 and cessation of construction activities reduces the potential for Unit 1 equipment being affected by a failure of Unit 2 equipment. Removal of a Unit 2 figure and text from the USAR that depicts and describes a system that was never physically installed results in no physical impact on any Unit 1 System, Structure, or Component (SSC), and makes no change to the manner in which any Unit 1 SSC is operated or tested. Therefore, the possibility of this change creating a new accident or malfunction of equipment is precluded.

III. No. The Unit 2 equipment being deleted is not described in the Technical Specifications, their Bases, the Operational Requirements Manual, the Offsite Dose Calculation Manual, the Process Control Program or the Operating License for Unit 1. Therefore, because credit is not taken for this Unit 2 equipment in the Unit 1 analysis, its removal does not reduce the margin of safety for Unit 1.

Safety Evaluation: 99-0086
Source Document: USAR Change Request (CR) 99-113

Description of Change:

This CR removes Unit 2 equipment from USAR Figure 6.9-1, Sheet 2 of 2, for the Unit 2 Feed Water Leakage Control system (N27). Unit 2 has been abandoned and this system for Unit 2 was never installed.

Summary:

I. No. Unit 2 has been officially abandoned. The Unit 2 portion of the N27 System is not required for Unit 1 operation, nor is credit taken for Unit 2 N27 System in mitigating any accident in Unit 1. Therefore, its removal does not increase the probability or consequence of an accident for Unit 1. The removal of equipment from a USAR figure that is not necessary to support Unit 1 in any way, and was never installed. Therefore, it does not increase the probability of a malfunction important to safety or the radiological consequences of any malfunction.

II. No. The equipment being deleted from USAR Figure 6.9-1 (Sheet 2 of 2) has been evaluated in USAR Section 1.10 and determined to be unnecessary to Support Unit 1. The abandonment of Unit 2 and cessation of construction activities eliminates the possibility of accidents or equipment failures of Unit 2 systems from interfering with Unit 1 equipment. Therefore, the proposed activity does not create the possibility of an accident or malfunction of equipment important to safety different than previously evaluated in the USAR.

III. No. The Unit 2 equipment being deleted from the USAR figure is not described in the Technical Specifications, their Bases, the Operational Requirements Manual (ORM), the Offsite Dose Calculation Manual (ODCM), the Process Control Program (PCP) or the Operating License for Unit 1. Credit is not taken for this Unit 2 equipment in the Unit 1 analysis, therefore, its removal does not reduce the margin of safety for Unit 1.

Safety Evaluation: 99-0087
Source Document: Simple Modification Request (SMRF) 99-5011, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 99-5011 will: (1) install blind pipe fittings in place of Residual Heat Removal (RHR) check valves 1E12-F0054A/B; (2) remove the valve internals (stem and plug), yoke, and actuator from control valves 1E12-F0065A/B, and install a blind bonnet on the valve body; and (3) disconnect the power and control wiring and instrument air to valves 1E12-F0065A/B including removal of their associated solenoid valves (1E12-F0465A/B) and Control Room Panel hand switches . With the conversion of 1E12-F0065A/B to a permanently-open status, 1E12-F0011A/B will become the RHR system Shutdown Cooling/Suppression Pool Cooling boundary in the Appendix R shutdown model.

Summary:

I. No. The modifications performed under SMRF 99-5011 do not create any conditions that would increase the probability of occurrence of any previously evaluated accidents since (1) the RHR and Reactor Core Isolation Cooling (RCIC) system operation will be the same as its current operation, and (2) the piping installations will be performed per the requirements of the original design code for the system, and therefore the original integrity and quality of the system will be maintained. The plant retains its current post fire safe shutdown capability subsequent to the implementation of SMRF 99-5011 and since no new system interactions are introduced, the probability of occurrence of any previously evaluated accidents cannot be increased. This SMRF will not create any conditions that would increase the radiological consequences of any previously evaluated accidents. The modified RHR to RCIC condensate return line does not affect the operation of the RHR and RCIC systems, and therefore the accident mitigating capability of the RHR and RCIC systems is not compromised. The change to the Appendix R safe shutdown model does not increase the radiological consequences of any previously evaluated accidents since the Perry Nuclear Power Plant (PNPP) Fire Protection Program retains its capability to achieve and maintain both hot and cold shutdown, and thus the accident mitigating capability of the RHR system is not compromised.

II. No. The modifications performed under SMRF 99-5011 do not interface with any plant systems (1) in such a manner as to create the possibility of an accident of a different type than previously evaluated in the USAR, or (2) in such a manner as to create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR. The modifications do not change the present operation of the RHR or RCIC system nor do they prevent the RHR or RCIC system from performing their safety functions. Therefore, given that plant operation remains unchanged from its current state, these modifications cannot create the possibility of a different type of accident or malfunction of equipment important to safety. The post fire safe shutdown capability of the plant is retained, and therefore different accidents or malfunctions are not created.

III. No. Installation of the piping modifications in accordance with the appropriate codes will ensure that the margin of safety associated with the piping system pressure boundary is maintained. The modified piping and electrical and pneumatic systems do 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 basis for any Technical Specification. The ability to achieve and maintain post fire safe shutdown will be retained, and thus the change to the Fire Protection Program does not reduce the margin of safety defined in the basis for any technical specification.

Safety Evaluation: 99-0088
Source Document: Condition Report (CR) 99-0243, Revision 0

Description of Change:

Condition Report (CR) 99-0243 evaluated the use of an alternate leak sealant material to stop an existing leak in the 1N11-F0430A valve. This valve isolates the Aux. Steam supply to the Moisture Separator Reheaters (MSR). Due to the use of an alternate leak sealant material, it cannot be assured that the 1N11-F0430A valve will continue to be able to be opened via the motor operator. Accordingly, the 1N11-F0430A switch, which is located in the Control Room, will be tagged as part of the temporary modification prohibiting operation of the valve using the motor.

Summary:

I. No. This temporary modification / leak sealant injection affects the ability of the 1N11-F0430A valve to provide blanketing steam to the Second Stage Reheater of MSR #4 (1N25-B0001A) as depicted in USAR Figure 10.1-1. Steam blanketing is not required or needed for any mode of normal operation and is not required for the safe shutdown of the plant. Its only use is to provide a steam layup of the MSRs during shutdown using Auxiliary Steam (P61) or to provide advanced warm-up of the MSRs as desired during startup prior to the availability of normal steam supplies.

This temporary modification does not and will not result in failure to meet the design, material, and construction standards applicable to the Main and Reheat Steam System (N11), nor does it affect overall system performance in any manner that could increase the probability of an accident. It does not result in any increase in the probability of failure of equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This temporary modification will not change N11 or P61 systems characteristics such as system vibration, fatigue, corrosion, thermal cycling or degradation of the environment of equipment important to safety that would exceed the design limits. Therefore, this change does 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 temporary modification to 1N11-F0430A does not affect the operation of any equipment required to support the operation or safe shutdown of the plant. It does not result in any increase in the probability of failure of equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This temporary modification does not increase the probability of occurrence of any accident/transient evaluated in the USAR nor do these changes cause previously bounded accidents to become bounding. This temporary modification has no affect on any environment or conditions for any important to safety equipment or equipment qualification zone. It will not result in any other failure mode or failure effect not already addressed in the USAR. This temporary modification does not affect the design, operation, availability or response of any important to safety equipment to any transients/accidents as described in the USAR. Therefore, this change will not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The steam blanketing function supported by the P61 system for the MSRs is not addressed nor does it affect any Technical Specification. Therefore, this temporary modification will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0089
Source Document: Drawing Change Notice (DCN) 5883, Revision 0;
USAR Change Request (CR) 99-119

Description of Change:

The reactor building refueling floor drawing E-015-044 (USAR Figure 9.1-27) and the fuel handling building fuel handling floor drawing E-015-045 (USAR Figure 9.1-26) are changed to delete the identification of specific storage locations for equipment components and tools. Notes are added to these drawings to require the storage and restraining of items on these floors to be as required by plant administrative procedures.

USAR section 9.1.4.2.7.1 which stated "A safety railing is provided to keep unauthorized personnel from entering the platform track area." is changed to state "A safety railing adjacent to the pools is provided to keep personnel from entering the pool area."

Summary:

I. No. Accidents previously evaluated in the USAR that may be affected by this change include the fuel handling accidents, and the control rod drop accident. Equipment important to safety that could be impacted by this change includes the refueling tools used during refueling outages, the fuel pool storage racks, stored new, and stored spent fuel. Other safety significant equipment in the vicinity of stored equipment; tools and components include containment vacuum breakers, containment spray equipment, hydrogen igniters and the combustible gas control system. The storage area requirements of Plant Administrative Procedure (PAP) 0204 will assure that stored items at the reactor building refueling floor and the fuel handling building fuel handling floor satisfy the requirements of Regulatory Guide 1.29. Regarding the changes to the handrail at the reactor building refueling floor, since there will still be handrail on all sides of the pools, plant personnel and equipment will be protected from falling into the pools. As demonstrated by the analysis, this change meets the requirements for protection of equipment important to safety from damage due to interaction with stored equipment components and tools during a seismic event. Therefore the probability of the occurrence or the consequences of an accident or the probability of the malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. Equipment, tools and components at the reactor building refueling floor and the fuel handling building fuel handling floor are required to be stored per PAP-0204. The storage area requirements of administrative procedure PAP-0204 will assure that stored items at the reactor building refueling floor and the fuel handling building fuel handling floor are seismically restrained and have adequate seismic clearance from the pools and safety related and augmented quality equipment. The potential for improperly stored items to fall into the fuel pools during a seismic event is prevented. The potential for improperly stored items to impact plant safe shutdown components during a seismic event is prevented. These changes do not cause any event previously considered to be incredible to become credible, nor do these changes cause any previously bound event to become bounding. Therefore, the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. Identifying PAP-0204 seismic restraint requirements for the storage of items at the reactor building refueling floor and the intermediate building fuel handling facilities will assure that loose items are secured. Thus, equipment required for safe shutdown will not be impacted by loose items during a seismic event. There is no clear trend toward a reduction in the margin of safety as a result of this proposed change. Therefore, these changes will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0090
Source Document: USAR Change Request (CR) 99-117

Description of Change:

The ability to transfer water from the upper containment pool to the suppression pool is a normal feature of the plant Suppression Pool Makeup (G43) System that helps ensure an adequate water supply for long term post-accident recirculation. USAR Section 6.2.7.3.3 states that following an inadvertent upper containment pool dump, the upper containment pool volume can be transferred from the suppression pool back to the upper containment pool through the Residual Heat Removal (RHR) pumps with a 20 minute pumping time. Potential Issue Form (PIF) 97-1488 documents that it would take longer than 20 minutes to transfer water back to the upper containment pool when the RHR spectacle flanges are in the "blank installed" position (normal position). Therefore, USAR Section 6.2.7.3.3 will be revised to clarify the suppression pool water level restoration following an inadvertent upper containment pool dump.

Summary:

I. No. The proposed USAR change provides clarification regarding the restoration of an inadvertent upper containment pool dump. The Technical Specifications, through Limiting Conditions for Operation (LCOs), govern suppression pool and upper containment pool water levels. An increase in the suppression pool level does not increase the probability that an accident will occur or adversely affect any Engineered Safety Feature (ESF) equipment. The function and performance of all ESF components are unaffected. The proposed USAR change does not create any additional system interactions. No accidents or equipment important to safety are affected by the proposed change and the radiological consequences of an accident or a malfunction of equipment important to safety are unchanged.

II. No. The proposed USAR change does not impact the operation of the Suppression Pool Makeup System or any other plant Structure, System or Component (SSC). The ability of an SSC to perform its functions and operate as designed has not been changed. Since no new system interactions are created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed USAR change does not impact the operation of any other plant SSC and does not create any new failure modes. Therefore, the proposed USAR change will not create a different type of accident or malfunction of equipment important to safety.

III. No. The proposed USAR change does not affect the ability of the Suppression Pool Makeup System or any ESF system or structure including any Emergency Core Cooling System to mitigate an accident or transient as defined in the USAR. The Technical Specifications provide requirements for restoration of suppression pool and upper containment pool water levels. Therefore, this situation has already been analyzed and incorporated into the plant's licensing basis via the Technical Specifications. The proposed USAR change supports the Technical Specifications and does not represent a change to the operating license. Therefore, the proposed USAR change does not change any margin of safety.

Safety Evaluation: 99-0091
Source Document: USAR Change Request (CR) 99-116

Description of Change:

Potential Issue Form (PIF) 98-2091 was written to document that the building names shown on Drawing D-302-382 did not match the Valve Lineup Instruction (VLI) for the Potable Water (P71) system. Therefore, this USAR CR was written to document the acceptability of changing the building names in the USAR figures.

Summary:

I. No. The change in the building names does not affect the operation of any equipment. The building names are not specifically addressed in any accident analyses and the names serve no safety function. The proposed USAR change does not affect or degrade the design, material, performance or construction standards of any system or component. System operation, availability, and response to transients remain the same as already described in the USAR. No changes are being made to any assumption or inputs previously made to assess dose consequences and no fission product barriers are being affected. The proposed USAR change does not change, degrade, or prevent actions described or assumed for any accident discussed in the USAR. Therefore, the proposed change will not increase the probability of occurrence of an accident or a malfunction of equipment important to safety. The proposed change will not increase the radiological consequences of an accident or a malfunction of equipment important to safety.

II. No. Changing the buildings names will not change the ability of any component that is important to safety to perform its intended design function and therefore, does not create the possibility of a different type of accident or malfunction. No Engineered Safety Feature systems, structures, or components are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed USAR change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create a different type of accident or a malfunction of equipment important to safety.

III. No. The change in the building names does not impact the design, function, or operation of any plant system. The Safety Evaluation Report, Technical Specifications, and Operational Requirements Manual do not specify building names and therefore, are not impacted by the proposed change. The change in the building names does not impact the margin of safety as defined by any licensing documents.

Safety Evaluation: 99-0092
Source Document: Simple Modification Request Form (SMRF) 99-5002, Revision 0

Description of Change:

SMRF 99-5002, Revision 0 removes the ability to regenerate N24 (Condensate Demineralizer) resin by permanently isolating the acid and caustic storage tanks, the hot water tank, and the delivery system from the N24 regeneration vessels. This is accomplished by adding welded pipe caps, blind flanges, spacer, and tube plugs to the interconnecting piping. This change will eliminate the major input to the Chemical Waste subsystem of the Liquid Radwaste System.

Summary:

I. No. The installation of blind flanges, plugs, spacer, and caps has no impact on the remaining portion of the N24 system to support Unit 1 operation. No equipment has been removed or process variables changed, which would affect, compromise, or impact the remaining portion of the N24 system to support Unit 1 operation. The change reduces the threat to Control Room habitability with the removal of hazardous materials from the condensate demineralizer area. The addition of the blind flanges, plugs, spacer, and caps were installed to the same design specification, codes and standards per the existing design, and serves to provide and maintain positive control of the system boundary. The use of procured regenerated resin reduces the inventory of the radioactive fluid streams. The resin will perform its ion exchange function regardless of where it is regenerated. The radiological consequences of any accident described in the USAR are unaffected by the change made to the N24 System. This change makes no structure, system or component changes to the plant that would impact its safety related function. No adverse system interactions are created by the implementation of this change. No new failure modes or effects are created 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 malfunction of equipment important to safety evaluated previously.

II. No. The resin will perform its ion exchange function regardless of where it is regenerated. The blind flanges, plugs, spacer, and caps were designed and installed to the same design specification, codes and standards per the existing design. This change does not increase the effects of any event that was previously bounded by other accidents to 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 does not affect the function or performance of the Condensate Demineralizer System as evaluated previously.

III. No. The Technical Specifications (TS) and its Bases, Operating License, and Operational Requirements Manual are not affected by this change to this system. There is no adverse 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. The blind flanges, plugs, spacer, and caps were designed and installed to the same design specification, codes and standards per the existing design and serve to provide and maintain positive control of the system boundary. These 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.

Safety Evaluation: 99-0093
Source Document: Plant Administrative Procedure (PAP) 1914, Revision 5, PIC 11

Description of Change:

This Procedure/Instruction Change (PIC) to Plant Administrative Procedure (PAP) 1914 establishes the maximum limit for the amount of carbon dioxide ($CO_2$) to be contained in Fire Protection tank 1P54-A009 at 1000 lbs. It also changes the minimum amount of $CO_2$ from 1000 lbs. to 375 lbs.

Summary:

I. No. $CO_2$ tank 1P54-A009 serves fire areas 1CC-4i (Computer Room Sub-Floor) and 1CC-5a (Control Room Sub-Floor). Fire area 1CC-5a is subdivided into three separate areas. Four independent $CO_2$ systems and a common tank (1P54-A009) provide fire suppression for these areas. The fire analysis for area 1CC-4i provides for manual total $CO_2$ flooding of the sub-floor. Analysis has determined that a volume of 300 lbs. of $CO_2$ will accomplish effective total flooding. Fire area 1CC-4i does not contain any equipment required for safe shutdown. The fire analysis for fire area 1CC-4i indicates a 1 hour fire resistance rated wall would contain a fire in the area. The probability of occurrence of a malfunction of equipment important to safety as a result of limiting the amount of $CO_2$ contained in tank 1P54-A009 is not increased or decreased. Carbon dioxide is not used in the operation of any equipment within the plant. Any radiological consequence as a result of a malfunction of equipment important to safety is unchanged. This change does not affect the operation of any equipment important to safety.

II. No. Fire area 1CC-4i does not contain any equipment required for safe shutdown. Fire area 1CC-5a contains the control equipment required for operation of Unit 1. Both divisions of safe shutdown components are located within fire area 1CC-5a. The $CO_2$ systems supplied by tank 1P54-A009 do not protect any equipment in the plant necessary for safe shutdown. Fire area 1CC-4i does not contain any equipment necessary for safe shutdown. Although both divisions of safe shutdown components and circuits are located in fire area 1CC-5a, the redundant means of control for safe shutdown of the reactor is located outside the control room. The ability to detect, control and suppress a fire in fire area 1CC-4i and 1CC-5a remains unchanged. This change does not create the possibility of a different type of activity or malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. Technical Specifications (TS) contain only the reporting requirements for fire protection as described in TS 5.6.6 "Special Reports." USAR Appendix 9A describes the fire protection program and the requirements for protection of the safe-shutdown capability. The fire protection program provides a "defense in depth" approach that involves prevention, detection, and suppression combined with post-fire safe shutdown capabilities. The margin of safety is based on maintaining one train of equipment and circuits necessary to achieve and maintain safe shutdown free of damage in the event of a fire. Therefore, the change in the maximum quantity of $CO_2$ will not impact the margin of safety.

Safety Evaluation: 99-0094
Source Document: Drawing Change Notice (DCN) 5614, Revision 0; and DCN 5791, Revision 0

Description of Change:

DCN 5614 and DCN 5791 both update the Potable Water System P&ID to reflect the addition of several valves, the removal of several toilet trailers, and correction of various building names.

Summary:

I. No. The USAR analysis of flooding shows that the site topography precludes the occurrence of significant flooding. The added valves and deleted toilet trailer supply lines are off of the distribution mains and outside of the power block and isolated from all equipment important to safety. They are not adjacent to any safety related equipment or equipment important to safety, and they do not interface with any systems or components involved with radiological consequences. DCNs 5614 and 5791 do not alter the analyzed functions of the P54 (Fire Protection) or P71 (Potable Water Supply) Systems. Analysis of the effects of incorrect valve positioning or isolated main breaks involving these valves have been bounded by existing analyses. There is no equipment important to safety affected by these systems, no equipment important to safety located in the area where the newly discovered valves and deleted toilet trailer supply line are/were located, and there is no interface with systems required to mitigate radiological consequences. The equipment associated with these changes does not directly affect any fission product barrier. The equipment served by the new valves is not "important to safety" or, in the case of the toilet trailers, no longer exists. None of this equipment has any effect upon any fission product barrier.

II. No. The only foreseeable problems with these valves are inadvertent positioning or a water line break. However, because each valve is connected to an isolated dead end leg from the P54 or P71 distribution system, there would be no adverse impact to the power block portions of the plant. The valves associated with this modification and the pipe lines in which they are mounted are all smaller than other associated piping in the same system, and they are located in the same general areas as that larger piping. The consequences of failure are therefore less severe. There are no important-to-safety systems or equipment in the vicinity of piping associated with these DCNs. Therefore, a problem with either the P54 or P71 system in the area affected by these DCNs cannot create any new initiators for accident scenarios. Further, the proposed changes introduce only manually operated valves and do not revise system functions or interactions. Thus, this activity does not create the possibility of a different type of accident or malfunction of equipment import to safety other than previously evaluated in the USAR.

III. No. Neither the P54 nor P71 system connections to the Procedures & Records (P&R) Building or to the toilet trailer interfere with the overall designed operation and function of these systems. The additional valves provide a second location for line isolation. These systems are not addressed in the Technical Specifications or in the Operational Requirements Manual. The applicable design and installation criteria for non-safety related components establish a margin of safety which is not diminished by the addition of these valves to the P&ID drawing.

Safety Evaluation: 99-0095
Source Document: Drawing Change Notice (DCN) 5864, Revision 0

Description of Change:

DCN 5864 modifies the drawing representation of various solenoid valves and process valve actuators in the Condensate Filtration and Demineralizer System. The solenoid valve representations are changed to show 4-way solenoid valves. In all cases, the failure positions of the process valves associated with these solenoid valves remain unchanged by DCN 5864. This DCN also adds a shut-off valve to a vendor drawing. The valve is already installed in the plant and it is shown on the P&ID drawing.

Summary:

I. No. The proposed changes will not alter process valve or system operations, therefore, there is no increase in the probability of occurrence of any USAR evaluated event. Since both the expected and the actual operation of the subject valves is unaffected by the proposed changes, there is no change in operability of components or systems, and no affect upon the mitigation of any radiological consequences. The proposed changes do not affect any process pressure boundaries and do not alter any radiological boundaries or fission product barriers. DCN 5864 has no effect upon the actual or anticipated operation of any component included in it. None of the changes included in DCN 5864 affect any equipment other than the valves listed in this Safety Evaluation. For the valves in question, it has been established that the failure positions remain unchanged despite the solenoid valve changes described in DCN 5864. The valves associated with the changes described in DCN 5864 retain the same behavior upon loss of air and power as presently indicated on the P&IDs. This DCN has no impact upon the failure modes or effects as they relate to the existing component and system design.

II. No. Since the process valves and system functions are not altered, the changes cannot cause any event that was considered bounded to become bounding or cause any event that is considered to be incredible to become credible. The proposed changes also have no effect upon any other plant equipment. The failure effects of the process valves are not changed. The proposed changes also have no effect upon system operation or upon system interactions. This DCN does not introduce any new system functions. Thus, susceptibility to common mode and common cause failures is not possible. These changes do not alter the redundancy or independence of any components or systems.

III. No. The proposed changes have no effect upon any aspect of component or system operation. Since the design function and operation of the equipment addressed in the proposed changes is not affected by those changes, its ability to continue to support the requirements of Technical Specification 3.7.5 / B3.7.5 remains unchanged. Therefore, the margin of safety as defined in the bases for any Technical Specification is not reduced.

Safety Evaluation: 99-0096
Source Document: Design Change Notice (DCN) 5267, Revision 0; DCN 5876, Revision 0; DCN 5884, Revision 0

Description of Change:

These DCNs incorporate the following changes: clarification the American Society of Mechanical Engineers (ASME) Code designations for piping of the Standby Diesel Generator Jacket Water and Lube Oil Systems, revised the code break implemented as part of the installation of Scram Discharge Volume level switches, revised the code boundaries applicable to the portion of the Inclined Fuel Transport System that serves the containment isolation function, and revised the physical location of a discharge relief valve relative to other components on the header.

Summary:

I. No. The proposed changes will not degrade System, Structure, and Component (SSC) reliability. No additional loads are being imposed as a result of the proposed changes. No equipment protection features are being deleted or modified by the proposed changes. The support system performance necessary for reliable operation of the important to safety equipment has not been downgraded as a result of the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not increase the frequency of operation of important to safety systems or equipment. The proposed changes do not impose increased testing requirements on important to safety systems or equipment. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. None of the changes addressed in this Safety Evaluation have any real or perceived effect upon any of the systems or equipment with which they are associated. The proposed changes will not add or remove SSCs to or from the plant. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event.

III. No. The "margin of safety" for each of the components and systems affected by the changes addressed in this Safety Evaluation is established in the design and installation requirements for each component including compliance with ASME and American National Standards Institute (ANSI) code requirements. None of the changes have an adverse impact upon the compliance of each item within each applicable requirement, and the operation of each item is not affected by any of the proposed changes. Since all requirements continue to be met, the safety margins inherent in those requirements cannot be diminished. Therefore, there is no reduction in the margin of safety as defined in the bases for any Technical specification.

Safety Evaluation: 99-0097
Source Document: USAR Change Request (CR) 99-125

Description of Change:

The USAR Change Request (CR) removes obsolete information from USAR Section 4.3, Nuclear Design, and also proposes editorial and clarifying changes. Information associated with the initial fuel cycle, Unit 2, and information affected by changes that have been made to General Electric Standard Application for Reactor Fuel (GESTAR II) are being removed because they are obsolete. The editorial changes affect reference subsection numbers, the sheet number for a figure, and delete two of the section references. The clarifying changes remove two circular references.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade System, Structure, or Component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report/Supplements to Safety Evaluation Report are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 99-0098
Source Document: USAR Change Request (CR) 99-124

Description of Change:

The USAR Change Request (CR) removes non-essential detail and obsolete information from the USAR and also proposes an editorial change. The USAR provides a description of the computer codes used for the analysis of the reactor internal components. The descriptions of the program version, history of use, and extent of application are being removed from the USAR because it is excessive detail. The CR also removes the description of the Unit 2 neutron fluence calculation because it is obsolete information. The USAR states, "A description of the supplementary burnable poison is provided in Section 4.2." This statement is being removed from the USAR because the supplementary burnable poison is not described in the USAR.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident. The proposed changes will not cause a change to any system interface in a way that would increase the likelihood of an accident or transient. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. The proposed changes will not increase offsite doses that would result from plant accidents and transients. The proposed changes will not degrade System, Structure, or Component (SSC) reliability. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not degrade the reliability of any plant system, nor do they introduce any new failure mechanisms for any plant system. The current operation, function, performance, and expected response of all systems are not affected by the proposed changes. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes do not make a previously non-credible event credible. The proposed changes do not unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not create any new potential failure modes, interactions, or operational sequences that could result in degradation or failure of systems or equipment important to safety.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report/Supplements to Safety Evaluation Report are not adversely affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. The proposed changes will not degrade the capability of SSCs to mitigate the effects of postulated transients and accidents.

Safety Evaluation: 99-0099
Source Document: Condition Report (CR) 99-2157

Description of Change:

CR 99-2157 reported cracks in the vertical section of the concrete lining of the Service Water (SW) and Emergency Service Water (ESW) tunnels. The CR investigation concluded that the reported cracks are tension cracks in the vertical sections (between 2 and 4 o'clock, and 8 and 10 o'clock) of the tunnel concrete lining and are caused by the settlement of the lower portion of the tunnel cross section. The settlement is attributed to the original construction methods of the tunnel. CR investigation concluded that the rock formation is in good condition and that the cracks do not affect the structural integrity of the tunnel lining and the concrete lining is capable of withstanding all applicable loadings. The cracked concrete lining has no impact on flow, volume, pressure of water in the cooling tunnels, the reliability or capacity of the tunnel to supply water to ESW and SW pump houses, and hence has no impact to safety systems supported by SW or ESW, including the loss of condenser vacuum event. The cracks act as joints as required by the American Concrete Institute (ACI) 322-72 code. Cracks of this type in large plain concrete structures are anticipated by the ACI 322 code. The potential for cracks was anticipated in the design of the tunnel lining and was reviewed and accepted by the NRC. The review concluded that the cracking of the tunnel lining and unlikely localized collapse of the tunnel lining would not impair the safety function of the tunnel structure and will not endanger the safe operation of the ESW pumps under any reactor emergency shutdown or lake level.

Summary:

I. No. As stated above, the cracks were anticipated in original design and were reviewed by the NRC. Therefore, existing licensing bases bound the reported cracks. The cracks do not alter any fission product barrier nor affect any process, test, system, component, equipment, or their operations. The cracks will not influence assumptions previously made in evaluating the radiological consequences nor prevent actions required to mitigate radiological consequences of an accident described in the USAR. No new system or component interactions are introduced by the cracks and the cracks have no impact on the operation of any system, equipment, or component including ESW pumps. Since the ESW function is maintained and not compromised by the worst case effect of the cracks (partial tunnel lining failure), the supported functions of the ESW system are maintained. These include maintaining fission product barriers (adequate core cooling, Reactor Pressure Vessel cooling/integrity, and Containment cooling/integrity). Therefore, neither the probability of occurrence of an accident, the consequences of a previously analyzed accident, the probability of malfunction, nor the radiological consequence of malfunction of equipment important to safety previously evaluated in the USAR will be increased.

II. No. This activity does not alter, modify or affect any equipment or function of any equipment important to safety. The failure of the tunnel is a postulated event in the SSER No. 1 that bounds the effects of this change. The cracked section will not compromise the safety function of the concrete lining or the tunnel, nor their ability to withstand all applicable loading. The cracked concrete lining has no impact on supply of water to SW or ESW pumps nor operation of any system, equipment, or component. Hence, it will not create the possibility of a different type of malfunction of equipment important to safety nor an accident of a different type than any previously evaluated in the USAR.

III. No. As stated above, the reported cracks are tension cracks in the non-reinforced (plain) sections of the tunnel concrete lining due to minor settlement. Subsequent compression loads on the lining will cause the tensile crack gaps to close and will develop compression stresses in the lining (as intended by the code). These tensile cracks do not affect the compressive load

resisting capability of the tunnel lining and once the lining is subjected to compression loads and stresses, it will behave as designed. Review of the design reports for the PNPP tunnel shows that the tunnels and lining were designed to satisfy the ACI 322-72 code allowables and that compression failure is the primary failure mode of the tunnel lining that can lead to collapse of the tunnel. The NRC final acceptance of the tunnel lining structure is based on evaluation of a postulated crack and localized collapse of a large section of the tunnel lining. The staff acknowledged that the tunnel lining could experience local failures without compromising the water supply through the tunnel to the ESW pumps. Since this change does not affect the staff's basis for acceptance, the margin of safety (ESW function) is not changed or reduced.

Safety Evaluation: 99-0100
Source Document: Drawing Change Notice (DCN) 5894, Revision 0

Description of Change:

The purpose of this DCN is to incorporate onto the Two Bed Demineralizer (P21) system P&ID drawing D-302-711, the existing level switch 0P21-N0702, thus showing its function as an input to the local annunciator as fed from level transmitter 0P21-N0465.

Summary:

I. No. The low level instrument is of the same design and type as the other level measurement instruments already described in the monitoring of the P21-A001 storage tank as originally licensed for plant startup. The installation conforms to the same design codes and standards as the originally installed instrumentation. The P21 instrument that is being added by this DCN is not an accident initiator. Neither the loss of Two Bed Storage Tank level monitoring or probability of P21 system inoperability increases due to the addition of this low level switch. This addition to the level monitoring capability does not reflect any change to the P21 System that will cause it to operate outside of applicable design or testing limits. The addition of this level switch to applicable plant documentation does not result in any changes to system interfaces. The failure analysis as described above indicates that this drawing change will not increase the probability of a P21 system failure or transient. This drawing change does not increase the probability of occurrence of an accident previously described in the USAR.

II. No. The installation of the 0P21-N0702 level monitoring switch to the P21 system was performed using the same codes and standards as the original design. This drawing change will not change the function of the P21 Two-Bed Storage Tank level monitoring system and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This change also does not cause any event evaluated in the USAR as incredible to become credible, or any event that was previously bounded to become bounding. This drawing change does not create the possibility of an accident of a different type than any previously evaluated in the USAR.

III. No. This change has no effect on the Technical Specifications, the Operational Requirements Manual, NRC Safety Evaluation Report or Standard Review Plan. Performance of the P21 System is not affected by this additional P21 Two-Bed Storage Tank level switch since the system will continue to provide P21 level monitoring using the existing methods of indication and alarm as is currently supplied. Consequently, there is no negative affect on, or change to, the performance or reliability of the P21 system. There is no margin of safety described in the Technical Specifications that relates to this level switch or its function. Therefore, these changes do not reduce the margin of safety as defined in the bases for the applicable Technical Specifications.

Safety Evaluation: 99-0101
Source Document: Drawing Change Notice (DCN) 5829, Revision 0

Description of Change:

This DCN has been written to implement the documentation changes necessary to change lighting panel R71-P0127 from an "Essential" lighting panel to a "Normal" lighting panel. This panel provides stairway lighting on all elevations of the Service Building. The change of description is the result of the engineering review associated with Potential Issue Form Remedial Action (PIFRA) 97-1328-001.

Summary:

I. No. The R71-P0127 lighting panel will continue to provide the required lighting for the Service Building. No special tests will be necessary that would challenge safe plant operation. The R71-P0127 Service Building lighting panel does not contribute in any manner to the occurrence of an accident evaluated in USAR. The R71-P0127 panel is non-safety and will continue to provide lighting for normal plant operations. The emergency lights in the stairways will provide illumination for emergency access or egress in the event of a loss of normal power. The R71-P0127 panel does not interface with safety systems important to safe plant operation. The R71-P0127 panel does not provide lighting required for access to equipment important to safety or required for response to USAR Chapter 15 accident analysis. There is no equipment important to safety in the Service Building. Additionally, the Service Building does not provide the most direct route for plant personnel to access plant equipment.

II. No. This description change has not changed the function of the R71-P0127 lighting panel and there is no negative effect or change to systems required for safe shut down or safe plant operation. The lighting panel is intended for the purpose of Service Building (SB) stairway lighting and can not create the possibility of a new accident condition. This change does not cause the effect of any event that was previously bounded to become bounding. No changes to plant equipment will be required to implement this documentation change. Therefore, this description change does not create the possibility of an accident of a different type than those previously evaluated in the USAR.

This description change does not introduce any new failure modes. There are no modifications that inhibit or change the function of any system important to safety. The lighting panel will continue to be capable of effectively providing illumination for the SB stairway areas. Neither the panel nor any of the panel loads are utilized to mitigate the consequences of an accident condition. This description change does not constitute a new failure mode or compromise the integrity of any equipment important to safety. This documentation change to panel R71-P0127 does not result in a credible malfunction of equipment important to safety. No modifications will be required to plant equipment.

III. No. The performance of the R71-P0127 lighting panel is not affected by this description change. The panel will continue to be capable of providing the required stairway illumination. This change neither impacts the ability of the R71-P0127 panel to provide normal lighting, nor does it reduce the margin of safety of equipment important to safety. There is no impact on personnel activities that would be associated with responses to a Station Blackout condition. Consequently, there is no effect or change of the performance of the required plant lighting systems. Therefore, this description change does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0102
Source Document: Simple Modification Request Form (SMRF) 97-5098, Revision 0

Description of Change:

This modification will "Abandon-in-Place" the electrical and mechanical portions of the Peripheral Deicing System. The electrical equipment to be labeled as "Abandon-in-Place " consists of four (4) motor operators on Circulating Water (CW) valves 1N71-F375, -F380, -F385, and -F390, their indicating lights and manual control switches. The mechanical equipment change consists of tagging the valves as "Locked Closed" and labeling the piping downstream of the valves as "Abandoned-in-Place." The Peripheral Deicing System will no longer be utilized in conjunction with the Central Deicing System to help prevent the build-up of ice on the Cooling Tower's diagonal supports.

Summary:

I. No. This modification implements physical changes in that it will "Abandoned-in-Place" the Peripheral Deicing System, which is a subsystem of the Circulating Water (CW) System (N71). This deicing system is only manually operated from a local panel, has no automatic controls and does not provide any control functions or signals to any system. Abandoning this system in-place will not degrade the performance of the CW System or the Central Deicing Systems and will not impact other systems that could initiate the accident described in the USAR. No portion of the N71 System has a safety related function, nor is it required to support any safety function, and it does not support safe shutdown of the reactor. No new failure modes or system failures have been identified related to the abandonment of this system. This deicing system (before or after abandonment) does not control or mitigate radiological activity. No other Systems, Structures, or Components (SSC) important to safety are affected by this change. This activity does not increase the probability of occurrence or consequences of an accident, nor malfunction of equipment, important to safety as previously evaluated in the USAR.

II. No. The N71 System is unaffected by the abandonment of the Peripheral Deicing System. The consequences of a common mode failure to the N71 system due to the abandonment this deicing system has not changed. The Peripheral Deicing System, once abandoned, has no means in which to render the N71 system inoperable or cause it to fail in a different manner. The Central Deicing System is the principal system in the prevention of ice built-up at the Cooling Tower's diagonals supports and it will remain operational and unaffected by this modification. In addition, it has been determined that the Peripheral Deicing System is neither effective nor necessary in the prevention of ice build-up at the Cooling Tower per an industry study. There are no failure modes resulting from this activity, which would create an accident. This activity will not create the possibility of an accident of a different type, or a different type of malfunction of equipment important to safety, than previously evaluated in the USAR.

III. No. The current design of the N71 System, including the portion affected by this modification, supports no safety function nor does it support safe shutdown of the reactor. This deicing system has no instrumentation channels or inter-locking controls with any system. No margins of safety were found in the Technical Specifications, USAR, Safety Evaluation Report, Supplements to Safety Evaluation Report design standard or its specifications that relates to this deicing system. This activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 99-0103
Source Document: USAR Change Request (CR) 99-131

Description of Change:

A USAR Change Request (CR) is being initiated to revise USAR Sections 3.6.2.3.5.2, 6.2.4.2.2.2.b, 6.3.2.6 and 9.3.3.2.1 to clarify the existing design and licensing basis for flooding in the Emergency Core Cooling System (ECCS) pump rooms.

Summary:

I. No. Flooding of the ECCS pump rooms is already discussed in the USAR and is part of the design and licensing basis of the plant. However, the existing text is located in several USAR sections and would benefit from clarification. Since the proposed change is only providing clarification of the USAR and does not affect the design or licensing basis currently identified within the USAR, the proposed change will not increase the probability of or radiological consequences associated with a previously identified accident or malfunction of equipment important to safety. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change will clarify the USAR discussion of flooding in the ECCS pump rooms. The proposed change does not affect the design or licensing basis contained in the USAR. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed USAR change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed USAR changes provide clarification of an issue that has a well defined licensing and design basis. During accident conditions, application of single failure criteria to the ECCS eliminates consideration of piping failures other than those that initiate the accident. Passive mechanical failures of pump seals and valve packing are isolable through closure of the suction isolation valve. Analyses have shown that unisolable failures during normal operation affect only the pump room that sustained the failure. Since the proposed USAR changes are being made to clarify the meaning of the USAR statements related to ECCS pump room flooding and do not create any alteration or revision to the meaning, the changes have no impact on any margin of safety. The margins of safety defined for Technical Specification Sections 3.5.1, 3.5.2 and 3.6.2.2 are unaffected by the proposed USAR changes.

Safety Evaluation: 99-0104
Source Document: Simple Modification Request Form (SMRF) 99-5003, Revision 0, Setpoint Change Request (SCR) 1-99-1107

Description of Change:

This modification is being installed to prevent a Reactor Core Isolation Cooling (RCIC) logic failure from causing a plant shutdown. A 4 ½ minute time delay relay is being added to the turbine trip contact (Reactor Feed Pump Turbines (RFPTs), Main Turbine, and associated trip annunicator) located in Control Room panel 1H13-P0629. This provides time for plant operators to assess plant conditions and secure the RCIC system if not required for safe plant operation. This modification will eliminate unnecessary plant transients due to spurious RCIC system initiations. This Safety Evaluation has been submitted to the NRC for review and approval by a letter dated June 5, 2000 (PY-CEI/NRR-2465L).

Summary:

I. Yes. Updated Safety Analysis Report (USAR) Chapters 3, 10, and 15 describe turbine events and licensing basis accidents. The discussion of each issue and the accidents described in the USAR have been reviewed with respect to SMRF 99-5003. The proposed modification does add a new failure mode and a new failure effect as evaluated in the Failure Modes and Effects Section of this Safety Evaluation. The addition of a time delay relay does add an insignificant but distinct increase to the probability of occurrence of a malfunction of the turbine (failure of the time delay relay) and main steam line piping and isolation valves. The increase in probability is not considered measurable. Specifically, incorrect operation of the new control switch could result in exceeding the design requirements for current analyzed design basis events in conjunction with a spurious RCIC initiation (equipment failure) while below 70% power. However, it is concluded that this modification does increase the probability of occurrence of an accident previously evaluated in the USAR.

II. Yes. Updated Safety Analysis Report (USAR) Chapters 3, 10, and 15 describe turbine events and licensing basis accidents. The discussion of each issue and the accidents described in the USAR have been reviewed with respect to SMRF 99-5003. The proposed modification does add a new failure mode and a new failure effect as evaluated in the Failure Modes and Effects Section of this Safety Evaluation. The addition of a time delay relay does add an insignificant but distinct increase to the probability of occurrence of a malfunction of the turbine (failure of the time delay relay) and main steam line piping and isolation valves. The increase in probability is not considered measurable. Specifically, incorrect operation of the new control switch could result in exceeding the design requirements for current analyzed design basis events in conjunction with a spurious RCIC initiation (equipment failure) while below 70% power. However, it is concluded that this modification does increase the probability of occurrence of a malfunction of equipment important to safety other than any previously evaluated in the USAR.

III. No. Although there is no margin of safety associated with the turbine, the regulatory requirement for acceptance of the turbine for use at Perry Nuclear Power Plant (PNPP) is based upon a calculated value of probability of external turbine missile interaction with safety related equipment. The barriers (Turbine casing and surrounding structures) and barrier interaction analyzed in the report will not be changed by this modification. The location of safety related equipment as it relates to the turbine missiles will not be changed. The strike probability will not be increased by the 4 1/2-minute time delay relay added by SMRF 99-5003. Thus, there is no reduction in the margin of safety by this modification.

Safety Evaluation: 99-0105
Source Document: USAR Change Request (CR) 99-132

Description of Change:

This USAR CR incorporates the manual initiation mode for the transformer deluge systems for the main and start-up transformers.

Summary:

I. No. There is no safe shutdown equipment in the Turbine Building. Safety related circuits in the Turbine Building consist of the Turbine Stop Valve inlet instrumentation located approximately forty feet north of the southeast corner of the Turbine Building at grade (620' elevation). Circuits for those instruments are routed inside steel conduit approximately fifty feet west, then south, approximately fifteen feet into the Turbine Power Complex. The closest point of the circuits is approximately two hundred feet from the wall common to the transformers. Turbine Building elevation 620' is also provided with an automatic sprinkler system. There is no change in fire detection capabilities. Full functional tests will continue at the prescribed interval for each transformer deluge system. The probability that a transformer fire will involve the safety related circuits and instrumentation in the Unit I turbine building will not increase. This change will not degrade or prevent any operator actions described or assumed for any accident described in the USAR and therefore will not increase the radiological consequences of any accident previously evaluated in the USAR. The two hour barriers, distance separation, and Turbine Building automatic sprinkler system provide adequate protection for safety related circuits and equipment. Changing the fire suppression system from automatic to manual will not cause a malfunction of any equipment important to safety previously evaluated in the USAR. Therefore, the proposed change will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. This change represents an incremental reduction in the fire suppression capability as described in the Fire Protection Evaluation Report (USAR Appendix 9A) by reliance on manual initiation only for the stated transformer deluge systems. However, all other aspects of the fire protection program and system capabilities remain unchanged. Evaluation of the fire hazard as well as the applicable elements of the Fire Protection Evaluation Report (USAR Appendix 9A) indicate that the proposed change will not create a different type of malfunction of equipment important to safety than previously evaluated in the USAR. The threat of fire induced damages will not be increased for safety related structures, systems or components as a result of this change

III. No. Technical Specifications (TS) contain only the reporting requirements for fire protection as described in TS Section 5.6.6 "Special Reports." The separation and protection provided for the redundant trains and credited for supporting the post fire shutdown capability is unchallenged by this change. Therefore, this change will not compromise the "defense-in-depth " measures established by the Fire Protection Program.

Safety Evaluation: 99-0106
Source Document: USAR Change Request (CR) 00-006

Description of Change:

This USAR CR incorporates several minor changes/additions to notes onto USAR Figure 10.4-1, "Steam Bypass and Pressure Regulation System." Notes 2 and 3 added the Perry vendor drawing number to the General Electric drawing reference. Note 4 was redefined to exclude the tubing shown on the Hydraulic Control Unit skid (HCU).

Summary:

I. No. The Note 4 revision did not change the tubing margin of safety. All the system tubing was designed to ANSI/ASME B31.1. The Note 4 change did not change the acceptance limit in ANSI/ASME B31.1. Therefore, the margin between the acceptance limit and the design failure point (or any other system limitation) is unchanged. The drawing note clarifications do not cause a physical change in the plant or a change in the operation of the plant. Additionally, the changes do not affect the design or function of the Steam Bypass and Pressure Regulation System nor any systems connected to the Steam Bypass and Pressure Regulation System. Therefore, no new accident causes can be introduced by the drawing clarifications. Similarly the causes and results of the accidents analyzed in the USAR are unchanged, and thus there can be no change (i.e., no increase) in the probability of occurrence of an accident previously evaluated in the USAR.

There are no changes in causes, results, or the means of mitigating accidents analyzed in the USAR. There are no new accident causes introduced by these drawing clarifications. The drawing clarifications do not affect any fission product barriers. Therefore, there can be no increase the radiological consequences of an accident previously evaluated in the USAR.

II. No. There are no physical changes to the plant. There is no change to the design, function, operation or system interaction of any SSC in the plant as a result of the changes. Hence, there can be no new accident initiators, failure initiators, or failure mechanisms introduced. Since there are no new accident initiators, failure initiators, or new failure mechanisms present, there is no possibility of an accident of a different type than was previously evaluated in the USAR being present.

Similarly, the drawing clarifications can not introduce any new system interactions, new equipment failure initiators or failure mechanisms. Therefore, the drawing clarifications can not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The turbine bypass (Steam Bypass and Pressure Regulation System) is addressed by Technical Specification 3.7.6 that requires the system to be operable when the reactor thermal power is greater than 25 percent. There are no physical changes to any SSC in the plant as a result of the drawing changes. The design, function, system interaction, or operation of the system is unchanged. No margins in the Technical Specifications (TS), TS Bases, or Operational Requirements Manual have been changed as a result of these drawing clarifications. Hence, the margins of safety, as defined in the bases for any Technical Specification, are unchanged (i.e. not reduced).

Safety Evaluation: 00-0001
Source Document: Drawing Change Notice (DCN) 5886, Revision 0;
Condition Report (CR) 99-0419

Description of Change:

CR 99-0419 was issued to document that the Relative Humidity (RH) in several plant areas could drop below the 20% minimum, as stated on the B-022 series drawings, during winter months. Environmental qualification of equipment is not affected by conditions of minimum relative humidity. Low relative humidity in the affected plant areas, during plant operation, is not a bounding technical requirement. The short-term reduction in relative humidity during cold, dry weather will not affect the operation or function of any equipment. Therefore, a note is being added to the affected environmental drawings to state that the relative humidity is allowed to drop below the 20% minimum RH shown on the environmental drawings for certain plant areas.

Summary:

I. No. The proposed change provides clarification of several USAR figures regarding the minimum allowable relative humidity. The proposed change does not affect the operation of any System, Structure, or Component (SSC). Minimum relative humidity during plant operation is not a bounding technical requirements. The short term reduction in relative humidity during very cold weather will not affect the operation or function of any equipment. Relative humidity is not an initiator for any accident defined in the USAR. The availability of plant systems, structures, and components is not affected. The function and performance of all Engineered Safety Function (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not impact the operation of any system, structure, or component. Since low relative humidity is not a bounding technical requirement, it will not invalidate current design analyses. There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF systems, structure or components are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed USAR change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident of a different type or malfunction of equipment important to safety.

III. No. The proposed change does not affect the ability of any ESF system or structure including any Emergency Core Cooling System (ECCS) to mitigate the radiological consequences of an accident or transient as defined in the USAR. The margin of safety with regards to equipment environmental qualifications is that defined by the equipment's or component's capability to perform its design safety functions when exposed to normal, abnormal, accident, and post-accident environments. The proposed change will not degrade equipment or component capability. Previously analyzed equipment or component capability to function as anticipated and fulfill its design requirements will not be degraded.

Ambient temperature limits specified by Technical Specification Table 3.3.6.1-1 remain unaffected. The proposed change will not result in an increase in any accident or non-accident operating temperature in any plant area. Technical Specification 3.6.1.12 provides limitations for containment relative humidity, but these environmental zones are not being impacted by the proposed change. Therefore, no change to the operating license is required and the margin of safety is not reduced.

Safety Evaluation: 00-0002
Source Document: Drawing Change Notice (DCN) 5885, Revision 0

Description of Change:

DCN 5885 corrects drawing errors on Piping and Instrument Diagrams (P&ID) and related design drawings associated with the Emergency Diesel Generators (R46), Residual Heat Removal (E12), Radwaste (G50), Equipment Drains (G61), Combustible Gas Control (M51), Reactor Plant Sampling (P35) and Post Accident Sampling (P87) Systems. Errors were identified under corrective action documents and a general engineering review of the P&IDs.

Summary:

I. No. None of the affected systems are considered to be a direct initiator of any USAR evaluated accident except for the G50, Liquid Radwaste System. The change to the G50 system is a drawing coordinate reference correction. Thus, Radwaste failures discussed in USAR Section 15.7.2 and 15.7.3 cannot be affected. The changes do not add or revise any interactions with other Structures, Systems or Components (SSCs) important to safety or SSCs considered as initiators of any event. The original American Society of Mechanical Engineers (ASME)/American National Standards Institute (ANSI) B31.1, ASME Section III codes are maintained. The changes do not alter system design/ safety functions or create any new interactions that could adversely affect the mitigating capability of the systems. Further, the changes proposed do not require any new or altered post-accident operator actions. Thus, the changes do not alter, degrade or prevent any actions related to the USAR accident analyses. The redundancy and independence of the systems and equipment important to safety are not affected by the modifications. These drawing changes do not create any new or revised failure modes or effects. Fission product barrier performance is not adversely affected. The analysis of the various items indicates that the safety and design bases of the affected systems are maintained with the proposed drawing changes. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The proposed changes do not alter or create any new systems, system interactions or operating functions. The changes will not alter the redundancy or independence of any systems. Based on the analysis section, design functions and capabilities of the affected systems are maintained. Changes to pressure boundary components continue to satisfy the applicable ASME/ANSI B31.1, ASME Section III codes. The potential failures associated with the modifications have been evaluated and no new failure effects were identified. The changes do not alter any redundancy or separation of any important to safety equipment. Thus, susceptibility to common mode or common cause failures is not created. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The E12, M51, P87 and R46 systems are SSCs important to safety and are addressed in Technical Specifications and/or the Operational Requirements Manual (ORM). The design functions of the affected systems are maintained, and none of the drawing changes alter or compromise the statements or underlying assumptions associated with any of the reviewed documentation. The changes in safety related pressure boundary components continue to comply with the ASME Section III code. Therefore, the margin of safety has not been reduced.

Safety Evaluation: 00-0003
Source Document: Simple Modification Request Form (SMRF) 99-5018, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 99-5018 removes Unit 2 480 volt alternating current (AC) buses 2R23-S0011 and 2R23-S0012 (Buses EF-2-C and EF-2-D), Motor Control Centers (MCC) 2R23-S0018 (EF2A07), 2R23-S0023 (EF2C07), and relocates Motor Control Center 0R24-S0037 (EF2A09). All loads connected to this equipment are being transferred to the relocated EF2A09 bus except the battery charger 2R42-S0008 (EFD-2-B), and the ground alarm for the Unit 2 Division 2 direct current (DC) system. The battery charger is connected to Unit 2 breaker F2D10 and the ground detector alarm is eliminated. Additionally, drawing changes are included to eliminate Unit 2 electrical equipment that is not installed and abandoned.

Summary:

I. No. The Unit 2 equipment listed above along with their loads is not credited for the safe operation and/or safe shutdown of Unit 1. The loads that are currently energized from these buses and MCCs will be powered from other permanent sources that are equally reliable. While the new supply to Battery Charger EFD-2B will be non-safety rather than safety related, this charger is not used when the battery is used to support Unit 1 operation. In addition, the Unit 1 ground detector will be utilized when the Unit 2 battery is in service supplying Unit 1. Thus, the equipment deletions and reconnection of loads do not increase the probability of an accident or the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. There is no interface between the equipment listed above and the Unit 1 reactor coolant pressure boundary or any system or component which could result in an accident or transient in Unit 1. The operation of the Unit 2 Division 2 DC system will be identical when used to support Unit 1. The design changes and the drawing changes that delete Unit 2 electrical equipment that is not installed do not add any new equipment types and do not alter or add any system interactions. There is no increased probability of an accident of a different type or a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The equipment listed above is not required for Unit 1 operation or safe shutdown and is not addressed in the Technical Specifications. It also has no effect on the offsite or onsite power sources for Unit 1. Therefore, this modification does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0004
Source Document: Simple Modification Request Form (SMRF) 99-5018, Revision 0

Description of Change:

The non-safety related Unit 2, 4.16kV switchgear 2R22-S006 (EH22/XH22) and 2R22-S009 (EH23) will be de-energized and abandoned/removed. Switchgear XH22 currently supplies power to Service Water Pump 0P41-C001D. The Service Water pump load will be moved to the non-safety related Unit 2 switchgear XH21. The existing protective relay settings will be maintained to properly protect each load and Perry Nuclear Power Plant (PNPP) standard cable sizing will be utilized. The Service Water pump is non-safety related and is not required to mitigate any accident scenarios. There are no loads currently on switchgear 2R22-S009.

Summary:

I. No. The existing Unit 2 switchgear 2R22-S006 is not diesel backed or classified as a 1E power source. Only one pump is required to be operation during a Loss Of Offsite Power (LOOP) event. This is currently supplied by service water pump 0P41-C001B which can be powered from the Unit 1, Division 2 diesel generator. Since the proposed change is removing a non-safety related load from a Unit 2 switchgear and re-powering the load from a separate Unit 2 switchgear, the proposed change will not increase the probability of, or radiological consequences associated with, a previously identified accident or malfunction of equipment important to safety. The Unit 2 transformer, LH-2-A, provides the alternate Class 1E offsite power source to support Unit 1. The movement of the Service Water pump load from switchgear EH22 to switchgear XH21 will have no net effect on the Unit 2 transformer and will therefore, not affect Unit 1. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change to remove a non-safety related load from the Unit 2 switchgear, EH22, and re-power that load from another Unit 2 switchgear, EH21, will not result in the creation of new types of accidents or malfunctions of equipment important to safety. The protective relay settings will ensure that the loads are adequately protected. The re-powering of the Service Water pump load will be performed in accordance with PNPP cable sizing standards. Therefore, no new failure modes are created. The proposed change does not adversely affect the design or licensing basis contained in the USAR. The proposed change does not adversely impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed modification does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed change does not affect any margin of safety. The elimination of a non-safety related Unit 2 switchgear load and the placement of that load on another non-safety Unit

2 switchgear will not impact any Unit 1 safety related SSCs. The proposed changes do not adversely affect the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0005
Source Document: USAR Change Request (CR) 00-005

Description of Change:

This USAR Change Request (CR) removes non-essential detail and obsolete information from USAR Section 4.4, "Thermal and Hydraulic Design", and also proposes editorial and clarifying changes. Information associated with the description of two computer codes is excessively detailed and is being removed. Information associated with the initial fuel cycle, information affected by changes that have been made to General Electric Standard Application for Reactor Fuel (GESTAR II), and information that should have been updated with the implementation of an earlier design change are being removed because they are obsolete. The editorial changes affect reference subsection numbers, delete a circular reference, and delete text references and a section reference associated with obsolete material that is being removed. The clarifying changes identify the organization responsible for making specific calculations for each reload cycle, and clarify the description of two process instruments.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade System, Structure or Component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplement to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. Therefore, the proposed change will not create the possibility of an accident of a different type or a different malfunction of equipment important to safety.

Safety Evaluation: 00-0006
Source Document: Drawing Change Notice (DCN) 3771, Revision 0

Description of Change:

DCN 3771 corrects drawing errors on Piping and Instrument Diagrams (P&ID) and related design drawings associated with the Emergency Diesel Generators and support systems. Errors were identified under corrective action documents and a general engineering review of the P&IDs.

Summary:

I. No. None of the affected systems are considered to be a direct initiator of any USAR evaluated accident. The changes do not add or revise any interactions with other Structures, Systems or Components (SSCs) important to safety or SSCs considered as initiators of any event. The original American Society of Mechanical Engineers (ASME)/American National Standards Institute (ANSI) B31.1, ASME Section III codes and manufacturer's standards are maintained. The changes do not alter system design/safety functions or create any new interactions that could adversely affect the mitigating capability of the systems. Further, the changes proposed do not require any new or altered post-accident operator actions. Thus, the changes do not alter, degrade or prevent any actions related to the USAR accident analyses. The redundancy and independence of the systems and equipment important to safety are not affected by the modifications. These drawing changes do not create any new or revised failure modes or effects. The analysis of the various items indicates that the safety and design bases of the affected systems are maintained. The changes will not lead to a possible malfunction of equipment important to safety. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The proposed changes do not alter or create any new systems, system interactions or operating functions. The changes will not alter the redundancy or independence of any systems. Based on the analysis section, design functions and capabilities of the affected systems are maintained. Changes to pressure boundary components continue to satisfy the applicable ASME/ANSI B31.1, ASME Section III codes and manufacturer's standards as applicable. The potential failures associated with the modifications have been evaluated and no new failure effects were identified. The changes do not alter any redundancy or separation of any important to safety equipment. Thus, susceptibility to common mode or common cause failures is not created. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The affected systems are SSCs addressed in Technical Specifications 3.0.7, 3.3.8.1, 3.8.1, 3.8.2, 3.8.3, 5.5.9 and the Operational Requirements Manual Sections 6.4.6 and 7.5.5. The margin of safety for the diesel generators is their ability to supply the required electrical loads as defined in the referenced Technical Specifications and associated bases. The design functions of all affected systems are maintained and none of the drawing changes alter or compromise the statements or underlying assumptions associated with any of the reviewed documentation. The changes associated with pressure boundary components continue to comply with the ASME/ANSI B31.1 or ASME Section III code, as applicable. SSC reliability, redundancy, operation and availability are unchanged. Therefore, the proposed change will not create the possibility of an accident of a different type or a different malfunction of equipment important to safety.

Safety Evaluation: 00-0007
Source Document: Design Change Package (DCP) 96-0044, Revision 0

Description of Change:

This design change eliminates the Main Steam Isolation Valve (MSIV) Leakage Control System (LCS) based upon the granting of a license amendment to adopt the Revised Accident Source Term Methodology. The MSIV LCS and the outboard MSIV drain lines are being eliminated with physical separation (cut and cap) from any operating system and are abandoned in place.

Summary:

I. No. The current operation, function, performance and expected response of the Main Steam system, the Main Steam Line Drain system, the Nuclear Steam Supply Shutoff system, the Plant Standby Electrical Power system, and the systems, structures and components interfacing with, or in the vicinity of, these systems will not be adversely affected by this modification. The design change eliminates a large amount of high energy piping and reactor coolant pressure boundary piping. The replacement of manual and automatic containment isolation valves with pipe caps and test valve appendages will reduce the amount of post-accident leakage. Interface systems are not adversely impacted. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. The change does not create any new systems or add any equipment that can affect the functioning of any systems, structures or components. No new equipment failures, event initiators or event contributors are created. Interfacing systems are not adversely impacted. The reactor pressure boundary and primary containment are not affected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. The current operation, function, performance and expected response of the interface systems, and the systems, structures and components in the vicinity of these systems will not be adversely affected by this modification. Design margins that existed have not been compromised. There is no adverse impact on the ability of any system to mitigate the effects of postulated transients and accidents. The reactor pressure boundary and primary containment are not affected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 00-0008
Source Document: USAR Change Request (CR) 00-027

Description of Change:

This USAR Change Request (CR) removes duplicate information from the USAR and also proposes an editorial change. USAR Sections 4.5.1.2 and 4.5.2.4 provide descriptions of the plant's conformance with Regulatory Guide (RG) 1.31, "Control of Stainless Steel Welding." A portion of this material describes the General Electric (GE) test program that demonstrated that controlling weld filler metal ferrite at 5 percent minimum produces production welds that meet the regulatory positions of RG 1.31. The description of the GE test program is being removed from the USAR because it duplicates information on USAR Table 1.81. The portions of these sections that directly address the plant's conformance to RG 1.31 are being retained in the USAR.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade System, Structure or Component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplements to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0009
Source Document: Perry Security Plan, Revision 28

Description of Change:

These changes to the Security Plan pertain to personnel access control measures for the protected and vital areas. These changes are considered to be safeguard information and as a result are managed in accordance with 10CFR73.21.

Summary:

I. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 00-0010
Source Document: Simple Modification Request Form (SMRF) 99-5038, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 99-5038 has been prepared to install a nitrogen supply from the Building Heating System (P55) nitrogen generator (1P55-D0031) to the Auxiliary Steam Boilers (0P61-B0001A/B) by connecting to existing equipment. The modification will use an existing connection on the nitrogen ($N_2$) system piping that is used as the source of $N_2$ gas for the Building Heating System surge tank. Nitrogen is used to provide a purge blanket to prevent corrosion of internal Auxiliary Boiler components during periods when the boiler is not operating. In addition, the existing $N_2$ bottles that currently supply nitrogen to the Auxiliary Boilers, will be removed from service.

Summary:

I. No. The P55, Building Heating System, and the P61, Auxiliary Boiler System, are non-safety related systems and do not act as initiators for any accident scenarios. The proposed change does not increase the probability of instrument air system failure. The availability of plant systems, structures, or components (SSC) is not affected. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. The proposed modification will not impact the pressure differential that is maintained across the Building Heating system heat exchangers and no new failure modes have been introduced. Therefore, there is no increase in the probability to spread radioactive contamination. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change will not impact the ability of the $N_2$ generator or associated components to maintain the pressurized nitrogen blanket on the Building Heating System expansion tank. The proposed change employs appropriate pressure controls to ensure that the $N_2$ supply to the P55 system expansion tank is unaffected. Additionally, a control room alarm will alert plant operators, and corrective actions can be taken, should P55 expansion tank pressure become too low. The proposed change does not impact the operation of any system, structure, or component. There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed modification does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident of a different type or malfunction of equipment important to safety.

III. No. There is no specific margin of safety defined for the Auxiliary Boiler, the Building Heating System or the $N_2$ generator. The installation of a nitrogen supply line to the Auxiliary Boiler has no impact on any SSC, and has no operational plant impact. The proposed change does not adversely impact the design or licensing basis for any SSC. SSC reliability, redundancy, operation and availability are unchanged. Therefore, the proposed change can have no impact on the margin of safety implied or specifically stated by any licensing documents.

Safety Evaluation: 00-0011
Source Document: Drawing Change Notice (DCN) 5900, Revision 0;
USAR Change Request (CR) 00-033

Description of Change:

This Change Request is being initiated to revise the USAR to include additional on-site chemicals which have been evaluated for potential impact on Control Room habitability and Emergency Diesel Generator (EDG) operation. In addition, DCN 5900 has been prepared to show the nitrogen tube trailer as a permanent plant structure on drawing D-302-950. The chemicals were analyzed using the guidance of NUREG-0570 and Regulatory Guide 1.78, and have no adverse impact on Control Room habitability or EDG operation.

Summary:

I. No. The proposed changes do not affect control room habitability or emergency diesel generator operability. The proposed changes are limited to documentation and no plant modifications will be performed by the proposed change. As documented by calculation, the nitrogen tube trailer does not have to be postulated as a tornado missile. In addition, based on Department of Transportation requirements, the tube trailer will not become an external missile as a result of a rupture of one or more tubes. The function and performance of all ESF components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed changes document the addition of chemicals stored on-site which have been analyzed for Control Room habitability and impact on EDG operation. This additional analysis is consistent in methodology and assumptions with the existing analysis. The analysis of additional chemicals for onsite storage makes no change in System, Structure or Component (SSC) design, operation and availability, and response to transients remains the same as already described in the USAR. The nitrogen tube trailer does not pose a threat to any safety related systems, structures, or components as a result of a tornado or by becoming a missile The proposed change does not impact the operation of any system, structure, or component. There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No Engineered Safety Feature (ESF) SSC are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident of a different type or malfunction of equipment important to safety.

III. No. The potential toxic and asphyxiant effects of the additional substances added to USAR Table 2.2-10 on Control Room habitability were evaluated utilizing the methodology provided by NUREG-0570 and Regulatory Guide 1.78. The probability of incapacitation of the Control Room staff due to a chemical release does not increase as a result of this change. There is no adverse impact from the proposed changes on EDG operation. Oxygen concentrations at the EDG intakes remain within acceptable limits for all analyzed gaseous releases.

The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0013
Source Document: USAR Change Request (CR) 00-029

Description of Change:

USAR Change Request 00-029 was initiated to remove the dissolved oxygen requirement from USAR Table 5.2-6. In addition, the typical values for iron and copper in the Condensate, Condensate Treatment Effluent, Feedwater, Reactor Water and Control Rod Drive Cooling water are also being removed from the table.

Section 5.2 of the USAR discusses the use of water chemistry to maintain the integrity of the reactor coolant pressure boundary. Over several years, the dissolved oxygen and other mineral requirements have been revised. A review of Regulatory Guide 1.56 provides no guidance for the control of oxygen, iron, or copper and therefore, maintaining these values in the USAR is not necessary.

Summary:

I. No. The proposed change does not authorize any changes to the plant. The Regulatory Guide 1.56 requirements to control general corrosion and stress-corrosion cracking are not being changed by the revision to the USAR. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not eliminate or revise any existing water chemistry criteria. The Regulatory Guide 1.56 requirements will be maintained through the appropriate plant procedures. Therefore, the proposed USAR change will have no impact on the physical plant. The proposed change does not impact the operation of any system, structure, or component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The margin of safety is defined as the ability to control corrosion and stress-corrosion cracking to prevent component failure. As previously discussed, Regulatory Guide 1.56 requirements will continue to be met. In addition, all General Electric (GE) Fuel Warranty requirements will continue to be met. The removal of the oxygen, iron, and copper parameters does not have an affect on the plant water chemistry. Any changes to plant water chemistry (via chemistry procedures) will be addressed by separate 50.59 evaluations.

The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0014
Source Document: Simple Modification Request Form (SMRF) 99-5010, Revision 0

Description of Change:

The existing Unit 1 and 2, Division 3 High Pressure Core Spray (HPCS) diesel batteries (1E22-S0005 and 2E22-S0005) are C&D (vendor) type 3DCU9 with each battery rated for 100 Ampere-hours (Ah). SMRF 99-5010 will replace these batteries with slightly larger batteries, C&D type KCR7 that are rated for 250 Ah. The SMRF will also replace the existing battery mounting racks with new mounting racks.

Summary:

I. No. The anticipated operational transients and design basis accidents in the USAR Sections 2, 3, 6, 9, and 15 have been reviewed with respect to this design change and it is concluded that these activities do not increase the probability of occurrence of an accident previously evaluated. The HPCS diesel (E22) system and its supporting systems operations to mitigate the effects of a design basis accident will not be altered. Performance of fission product barriers will not be affected by this activity. There will be no change in radiation dose to the public or onsite doses by this activity. As such, this design change cannot increase the consequences of an accident. No new failure modes or resultant equipment/system failure effects will be introduced as a result of this activity. In addition, the change does not alter equipment qualification or physical separation requirements for Division 3 equipment. No safe shutdown circuits are affected by this change. This activity does not alter any role in mitigating the radiological consequences of an accident and does not affect any fission product barrier. The proposed activity will not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. This proposed activity does not impact the Division 3 battery capability to provide 125 VDC power to required Direct Current (DC) loads. No accident of a different type may be expected due to this activity. The possibility of an accident of a different type than any previously evaluated in the USAR will not be created. This change does not introduce any new equipment failure modes nor results in any changes in system operation or function. This activity does not affect the design, operation, availability or response of any equipment important to safety. No new failure modes or resultant equipment/system failure effects will be introduced as a result of this activity. Therefore, there is no potential for a different type of a malfunction of equipment important to safety than any previously evaluated by this activity.

III. No. Based on the analysis conducted, this activity does not impact the Division 3 Class 1E DC system function to provide a reliable DC power source to mitigate accident consequences and station blackout conditions. Technical Specification Sections 3.8.4, 3.8.5, 3.8.6, 3.8.7, 3.8.8, 3.5.1, 3.5.2, Table 3.8.6-1, Table 3.3.5.1-1 and their associated bases are not affected by this activity. The replacement batteries are larger than the existing batteries and have greater margin to provide 125 VDC power for operation of HPCS and its supporting systems. The margin of safety for the Division 3 batteries as defined in Technical Specification bases B3.8.4 and USAR section 8.3.2.1.3.2 is to have adequate storage capacity to carry the required load continuously for at least 2 hours. Battery margin is defined as the difference between the battery capability (corrected for aging, design margin, temperature correction factor) and the design basis required load. The replacement batteries have a higher rating (250 Ah) compared to the existing batteries (100 Ah). Thus, the proposed modification will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0015
Source Document: USAR Change Request (CR) 00-039

Description of Change:

This USAR change request deletes USAR Figure 13.5-1 that describes the Horseshoe and the Surveillance Areas in the Control Room. This information is contained in Plant Administrative Procedure PAP-0126 and does not need to be repeated in the USAR.

Regulatory Guide 1.114, Guidance to Operators At The Controls and to Senior Operators in the Control Room of a Nuclear Power Unit, provides information related to complying with the NRC's requirements for the presence of an operator at the controls of a facility. This Regulatory Guide recommends that administrative procedures be established that define and outline the specific area within the control room designated as the "surveillance area" where the operator at the controls should remain. The "surveillance area" is designated in PAP-0126 as the Horseshoe Area and is the section of the Control Room in which the Supervising Operator at the controls performs his normal shift functions including visual surveillance of safety related annunciators and instrumentation.

An additional area, the Operations Area, has also been defined in PAP-0126 and is the section of the Control Room in which the Supervising Operator at the controls may enter in the event of an emergency affecting the safety of operations, in order to verify receipt of an annunciator alarm or to initiate corrective actions.

These definitions are consistent with the guidance listed in Regulatory Guide 1.114 and are not being changed by this USAR change request. USAR Figure 13-5-1 is redundant with the information contained in PAP-0126 and is not required to be maintained in the USAR. No reduction in commitments is being made by this change request.

Summary:

I. No. The proposed change to the USAR results in the deletion of a redundant figure. Defined Control Room areas will remain consistent with Regulatory Guide 1.114. Operator control board monitoring and response times will remain the same. No USAR evaluated accidents are initiated or affected by this figure in the USAR. The USAR change does not represent any physical or process change that could impact System, Structure, or Component (SSC) operating parameters or could cause a change to any SSC that would increase the likelihood or frequency of an accident. Therefore, the proposed change will not increase the probability of occurrence of an accident previously evaluated in the USAR. The proposed change will ensure the proper monitoring and control of plant systems and 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. This USAR change does not involve any credible failure modes or mechanisms for any SSC, therefore the change could not initiate any sequence of events resulting in any type of accident. The USAR change does not represent any physical or process change that could impact SSC operating parameters. The possibility of an accident of a different type or the malfunction of equipment important to safety than any previously evaluated in the USAR is not created as a result of this revision.

III. No. The removal of Figure 13.5-1 from the USAR does not affect any specific margin of safety. This USAR change does not adversely impact the design or licensing basis for any SSC. SSC reliability, redundancy, operation and availability are unchanged. There is no impact on the margin of safety as defined in the Technical Specifications and Bases, Operational Requirements Manual, Operating License, Safety Evaluation Report, and USAR.

Safety Evaluation: 00-0016
Source Document: Surveillance Instruction (SVI) B33-T1168, Revision 4, PIC 3

Description of Change:

This instruction verifies reactor coolant temperature differentials and recirculation flow rates are within Technical Specification limits prior to startup of an idle recirculation loop. This is accomplished by using process instrumentation. This Procedure/Instruction Change (PIC) adds administrative controls for installing/removing a jumper to bypass the reactor recirculation pump start/upshift thermal shock interlocks in the event a component deficiency renders the interlocks non-functional. All plant parameters confirmed by the thermal shock interlocks are monitored by this surveillance

Summary:

I. No. The addition of a jumper to manually control the thermal shock interlocks meets the same design intent as the original automatic interlocks. All other control functions will not be affected by manually controlling the thermal shock interlocks; therefore the original design intent has not been changed. The addition of a jumper is not a design basis accident initiator or contributor. Manual control in this case does not reflect any changes in the Reactor Recirculation Control (B33) System that will cause it to operate outside of applicable design or testing limits. The change does not result in any changes to system interfaces. The Plant Operators must verify the design basis delta temperatures limitations are acceptable to meet the Technical Specification requirements anytime a startup/upshift of the recirculation pump is done no matter if using manual or automatic control of the thermal shock interlocks. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The manual control of the thermal shock interlock will not change the function of the B33 reactor recirculation pumps or system and there will not be any impact to systems required for safe shutdown or safe plant operation. This change does not result in any increase in the probability of the failure of any equipment that is an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. The procedure changes comply with the existing design basis and intent. The change does not affect the design, operation, availability or response to any transients/accidents of any equipment important to safety as described in the USAR. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The procedure change has no effect on the Technical Specifications (SR 3.4.11.3, SR 3.4.11.4), the Operational Requirements Manual (ORM), NRC Safety Evaluation Report and the Standard Review Plan. Performance of the B33 recirculation pumps has not been affected if manual control of the thermal shock interlocks is required since the design intent and design logic will not change. Consequently, there is no negative impact or change to the performance or reliability of the B33 system. The verification of the differences between bottom head coolant temperature and the Reactor Pressure Vessel (RPV) coolant temperature plus the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant must still be done to comply with the Technical Specifications. The startup/upshift circuitry is separate from the shutdown circuitry; therefore, shutdown of the plant in any plant condition will not be affected. Operator actions will not change. Therefore, this procedural change does not reduce the margin of safety as defined in the Technical Specification.

Safety Evaluation: 00-0017
Source Document: USAR Change Request (CR) 00-043

Description of Change:

Post Accident Sampling System (P87) conduit 1P87C71B associated with Design Change Package (DCP) 87-0524B was added to drawing D-215-667 sheet 501, and USAR Figure 8.3-19 (this design change package was previously evaluated prior to implementation, however a formal change request was never submitted to up-date USAR Figure 8.3-19). Conduit 1P87C71B is routed from containment electrical penetration 1R72-S009 (inside Reactor Building) to solenoid valve 1P87-F049.

Summary:

I. No. The conduit in question was evaluated for structural loading on the containment liner/concrete walls and installed in accordance with approved plant procedures and standards including separation criteria. The cable in conduit 1P87C71B was made 'spare' per approved plant procedures thus rendering the circuit de-energized/inoperable. For electrical separation, Perry Nuclear Power Plant (PNPP) is committed to Regulatory Guide 1.75 revision 2, Physical Independence of Electrical Systems, and Institute of Electrical and Electronics Engineers (IEEE) 384-1974, Criteria for Separation of Safety Related Class IE Equipment and Circuits. Conduit 1P87C71B conforms to the aforementioned Reg. Guide, IEEE Standard and all Perry separation requirements. Installation of this conduit to penetration 1R72-S009 will not compromise the leak tight barrier that is required for integrity to the primary containment boundary. This conduit installation has not altered the design requirements associated with the containment penetration leakage rate testing as described in Sections 3.8.2.1.6 and 6.2.6.2 of the USAR. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The potential for common mode or common cause failures is not increased and no new failure modes or effects are created. Compliance with PNPP and industry standards during installation provided added assurance against possible malfunctions. Installation of this conduit to electrical penetration 1R72-S009 will not compromise the integrity of this isolation device. Therefore, the addition of conduit will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. Installation of conduit 1P87C71B to electrical penetration 1R72-S009 will not compromise the integrity of this isolation device. Conduit 1P87C71B is a Division 2 safety related conduit routed to a Division 2 electrical penetration, thus meeting the requirements of IEEE-384-1974 standard. Therefore, the addition of this conduit on USAR Figure 8.3-19 does not reduce the margin of safety as defined in the bases for any Technical Specification or other license documents.

Safety Evaluation: 00-0018
Source Document: USAR Change Request (CR) 00-046

Description of Change:

USAR Change Request 00-046 was initiated to revise USAR Figure 3.7-17 to re-orient the seismic vibration monitor D51-R160 to help demonstrate that it is accessible for removal of the recording plates. The physical orientation of this device was performed during the construction/startup phase of the plant via Engineering Change Notice (ECN) 25964-86-1129, Rev. -. The orientation of the seismic vibration recorder has no impact on the operability of the recorder. The only purpose of the recorder is to signal the control room when an Operating Basis Earthquake (OBE) has been recorded. No Engineered Safety Features (ESF) or safety related systems are activated by the seismic vibration recorder. The seismic vibration recorder is discussed in the Operational Requirements Manual (ORM), however, the actual orientation of the recorder is beyond the level of detail presented in the discussion.

Summary:

I. No. The function and performance of all ESF components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change will have no impact on the physical plant. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs, and will not reduce the margin of safety implied or specifically stated by any licensing documents, including the USAR, Safety Evaluation Report (SER), ORM, Core Operating Limits Report (COLR), Offsite Dose Calculation Manual (ODCM), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0020
Source Document: Simple Modification Request Form (SMRF) 98-5014, Revision 0

Description of Change:

This design change will delete the Halon fire suppression system from the Central Alarm Station (CAS), located in the 620' elevation of the Service Building. The change revises the wiring in the control Panel 0H51-P0893 to allow for the fire detectors in the CAS to remain in service as area detection only. The deletion of the subject Halon suppression system will require changes to the USAR Sections 9.5.1.2.5 and 9A.4.18.1, and supporting fire protection drawings. Changes to plant operating procedures (Pre-Fire Plan, system operating procedures and others) will also be required as the result of this change. The basis for this evaluation is to determine/establish that the manual fire fighting capability via the Fire Brigade and manual suppression means (extinguishers, fire hoses) is a viable and practical means of maintaining the required level of fire safety.

Summary:

I. No. Adequate fire barriers separate the Diesel Generator Building and Control Complex from any postulated fire occurring in the Service Building, there is no potential for the effects of a fire in the CAS adversely affecting equipment important to safety. In addition, the existing fire hazards and the associated worst-case fire scenario will be adequately mitigated by the manual fire suppression capability, providing an adequate level of protection The elimination of the CAS Halon system will not affect the performance of these fire barriers. The Security Plan does not require a fire suppression system for the equipment or alarm station. T herefore, the elimination of this fire suppression system will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The elimination of the Halon suppression system will not increase the threat or the consequence of fire-induced damage to safety related structures, systems, nor is a new or unique type of equipment failure potential introduced by this proposed change. Protection of safe shutdown equipment and equipment important to safety in adjacent buildings is provided by fire rated barriers between the buildings and the associated worst-case fire scenario will be adequately mitigated by the manual fire suppression capability. From the perspective of nuclear safety impact, these changes will not cause potential single failures to become common mode failures, nor cause events previously considered incredible to become credible. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR

III. No. The change to the Fire Protection Program implemented through this SMRF is permissible under the Perry Nuclear Power Plant (PNPP) Fire Protection License Condition to the extent that the change does not adversely impact the credited post-fire safe-shutdown capability. The separation and protection of redundant trains credited to support the post-fire shutdown capability is unaffected by the elimination of the Halon fire suppression system. Therefore, equipment and circuits required to achieve and maintain safe shutdown will remain free of fire damage and the margin of safety established by the Fire Protection Program as reviewed and approved by the NRC is maintained

Safety Evaluation: 00-0021
Source Document: Simple Modification Request Form (SMRF) 99-5056, Revision 0

Description of Change:

A failure of the recirculation suction temperature RTDs (Upscale High Temp) has previously resulted in the cavitation interlock setpoint being exceeded (Condition Reports 93-0024 and 93-0133) which resulted in a plant manual Scram. The cavitation interlock protects the Reactor Recirculation System (RRS) pumps and jet pumps from cavitation damage due to insufficient subcooling. The interlock caused the RRS pumps in both loops to transfer to low speed, which resulted in entering the exclusion region of the power-flow map. This required a manual scram per station procedures. This modification will allow Operations to bypass the cavitation interlock at high thermal power levels. Bypassing this interlock will prevent the tripping of the recirculation pumps inadvertently.

Summary:

I. No. The removal of the wires from selector switches S125A/B "DOME/PUMP Δt INTLK BYPASS STEAM" will not change, degrade, or prevent actions described or assumed in an accident discussed in the USAR. This design change is to eliminate an annunciator wiring for a switch position and will not alter any assumptions with respect to radiological consequences, nor does it play a role in mitigating the radiological consequences of an accident described in the USAR. The proposed design modification does not affect any fission product barriers. Therefore, this design modification does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR.

II. No. The ΔT Interlock Bypass Switch and the associated instrumentation are not required for plant safe shutdown or to mitigate the consequences of an accident. In case there is a malfunction in the recirculation pumps, the flow control valves, or the jet pumps, the consequences of this malfunction can not be affected by the status/operation of the ΔT Interlock Bypass Switch or the associated instrumentation or the elimination of the annunciator for the bypass switch position. This design change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the USAR.

III. No. This proposed design modification has no effect on the Technical Specifications, the Operational Requirements Manual (ORM), NRC Safety Evaluation Report, and Standard Review Plan. The margin of safety associated with this change is to maintain the reactor coolant pressure boundary. Based on the controls placed upon plant operations via the plant operating instructions, this margin will not be reduced. This equipment/components are not included in the basis for any Technical Specifications margin of safety or in the Technical Specifications Bases. This design modification will not have any effect on the margin of safety of any structures, systems, and components which are addressed in the basis for the Technical Specifications. Therefore, these changes do not reduce the margin of safety as defined in the bases for the Technical Specifications.

Safety Evaluation: 00-0022
Source Document: Simple Modification Request Form (SMRF) 99-5049, Revision 0

Description of Change:

This SMRF replaces Reactor Water Cleanup System (RWCS, G33) globe valves 1G33-F0505A/B and 1G33-F0506A/B with double disc gate valves. These valves are installed in the 2" drain lines for the reactor water cleanup system. The routing of the drain lines is changed to eliminate the snubbers and spring can pipe supports. One new tie-back support is added for each drain line. The valves and piping up to the second valve on each drain line are part of the reactor coolant pressure boundary. The pipe schedule of the 6" length of pipe and end cap at the end of the drain lines is increased to be the same pipe schedule as the rest of the drain lines.

Summary:

I. No. Accidents evaluated in the USAR that may be affected by this modification include loss-of-coolant accidents inside containment as described in USAR section 15.6.5. The portion of the RWCS piping being modified is part of the Reactor Coolant Pressure Boundary (RCPB) located inside the drywell. The safety design basis of the RCPB portion of the RWCS is that the requirements of Regulatory Guides 1.26 and 1.29 be met (prevent excessive loss of reactor coolant, prevent the release of radioactive material from the reactor, and isolate the major portion of the RWCS from the RCPB). To comply with the above design safety basis, the modified RCPB portion of the RWCS was analyzed and found to comply with seismic category 1, ASME Class 1 piping requirements. The requirements of USAR section 3.6.1.2 to protect systems and components required for safe shutdown from postulated pipe rupture by physical arrangement has been complied with regarding this modification. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. The change in pipe routing for these 2" drain lines will be not significantly changed. The location of the two 2" drain lines has been rotated approximately 30 degrees in the horizontal plane, but they remain located under the recirculation piping and recirculation pump bowls and continue to be protected from pipe rupture from other sources. The physical location of any pipe breaks is not changed, therefore there cannot be any additional pipe rupture effects to other equipment required for safe shutdown. During the modifications, the isolation requirements of System Operating Instruction SOI-B33 section 7.14 for draining reactor recirculation loops A and B will be complied with. The possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. Upon completion of the modifications, a pressure test with VT-2 will be performed per ISI-1B21-T1300-1. This test is in compliance with ASME code case N-416-1 for welded repairs or installation of replacement items by welding, class 1, 2 and 3 Section XI, Division 1. These valves are not identified in USAR Table 3.9-30 as active valves for the RWCS. The new double disc gate valves will be procured as seismic category 1 ASME Class 1 valves. The new valves will be designed to the seismic requirements of USAR section 3.7 and design transients per USAR section 3.9.1.1 and as shown in USAR Figure 3.9-30 for the RWCS. The modification to be made complies with all of the design requirements and the piping and supports remain within the allowable stresses for this ASME Class 1 piping system. A review of the Technical Specifications, Bases for the Technical Specifications, the Operational Requirements Manual, and the USAR has shown that there is no clear trend toward a reduction in the margin of safety of the RCPB as a result of this change.

Safety Evaluation: 00-0023
Source Document: Simple Modification Request Form (SMRF) 98-5054, Revision 0

Description of Change:

Reactor Recirculation (B33) Hydraulic Power Unit (HPU) Isolate/Operate valves (1B33-F603A/B and 1B33-F604A/B) must be cycled/exercised periodically to prevent binding or sticking. To accomplish this, Operations performs a HPU subloop transfer weekly. This is Operations manhour intensive and is also a reactivity concern. This modification is a change to the Modicon Programmable Logic Controller (PLC) logic to periodically cycle the Isolate/Operate valves for a short duration (1/4 second). This will accomplish the valve exercise to prevent the binding and sticking of these valves. Implementation of this design modification will free up Operations resources and will not require manual subloop transfers every week. The implementation of this design change will be accomplished by the reprogramming of the B33 Modicon PLC. In addition, to address equipment obsolescence and reliability concerns, the existing PLC design will be upgraded/replaced with a Modicon 484 Form Factor Processor Unit which is an equivalent replacement.

Summary:

I. No. The modification is being installed to provide reliability to the solenoid valves and the HPU so they perform their required function upon demand. Based on analysis conducted, it is believed that this modification will in fact reduce the probability of failure of the solenoid valves with respect to initiation of either of these two events. This design modification does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR.

II. No. If a failure were to occur as a result of instituting the automatic ¼ second cycle function, the most severe result would be a lockup of the flow control valve. As stated previously, this is an analyzed condition and is a condition that can occur in the current design. The ¼ second pulse that exercises the solenoid valves does not result in operating the flow control system in a new or different manner. The bounding event remains a Recirculation Flow Control Failure with Increasing Flow. This design change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the USAR.

III. No. This proposed design modification has no effect on the Technical Specifications, the Operational Requirements Manual, NRC Safety Evaluation Report, and Standard Review Plan. The margin of safety associated with this change is to maintain the reactor coolant pressure boundary (Technical Specification Bases B3.4.1) or the Minimum Critical Power Ratio Limits (Technical Specification Bases B3.4.2). The equipment/components affected by this modification are not included in the basis for any Technical Specifications margin of safety or in the Technical Specifications Bases. This design modification will not have any effect on the margin of safety of any structures, systems, and components which are addressed in the basis for the Technical Specifications. Therefore, these changes do not reduce the margin of safety as defined in the bases for the Technical Specifications.

Safety Evaluation: 00-0024
Source Document: Simple Modification Request Form (SMRF 00-5013), Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 00-5013, Revision 0 will (1) install a new inflatable seal and seal retainer on the Emergency Service Water (ESW) forebay side of each of the two existing sluice gates, (2) install a temporary sluice gate during the time the permanent sluice gate is removed and being modified, (3) install a new air supply hose and spring loaded hose take-up reel for each inflatable seal, (4) install components to provide for local, manual seal inflation/deflation, (5) install a new selector switch to enable or defeat the manual raise/lower control circuit and the automatic opening signal of the sluice gates, and (6) install a new remote mounted thermometer in the ESW forebay with local indication in the ESW pump house. SMRF 00-5013, Revision 0 does not connect the seals to an air supply source.

Summary:

I. No. The modifications described above (1) do not create any conditions or interface with any plant systems, structures, or components in such a manner that would increase the probability of occurrence of any previously evaluated accidents, and (2) do not create any conditions that would increase the radiological consequences of any previously evaluated accidents. The ESW System, including the sluice gates, is an accident mitigating system and is not an accident initiator. Therefore, the modifications to the ESW sluice gates cannot increase the probability of occurrence of any previously evaluated accidents. The activities implemented via SMRF 00-5013, Revision 0 cannot increase on-site radiation doses such that actions to mitigate the radiological consequences of an accident would be impeded, nor does it directly or indirectly affect the ability of any other plant system to mitigate the radiological consequences of an accident. The modifications to the sluice gates do not adversely affect the operation of the ESW System, and therefore the accident mitigating capability of the ESW System is not compromised.

The activities implemented via SMRF 00-5013, Revision 0 do not adversely impact any modes of operation of the ESW System, and therefore this change cannot create a malfunction of the safety related ESW System. Further, this activity does not introduce any new failure modes nor result in any new failure effects related to the other components of the ESW System or any other plant system, and therefore cannot increase the probability of occurrence of malfunction of equipment important to safety. The accident mitigating capability of the ESW System is not compromised by this change and consequently the radiological consequences of any malfunction of equipment important to safety that relies on the ESW System for mitigation will not be increased. This activity does not reduce the functional performance of the Emergency Core Cooling System (ECCS) nor any other accident mitigating systems requiring ESW such that the radiological consequences due to equipment malfunction would increase. The modifications to the sluice gates do not affect or interface with any other equipment important to safety and are not associated with any malfunctions of equipment important to safety previously evaluated in the USAR.

II. No. The modifications to the sluice gates do not interface with any plant systems (1) in such a manner as to create the possibility of an accident of a different type than previously evaluated in the USAR, or (2) in such a manner as to create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR. The ESW System, including the sluice gates, is an accident mitigating system and is not an accident initiator. The change implemented via SMRF 00-5013, Revision 0 will not create any new systems or add any new equipment that can compromise the functioning of any existing

Systems, Structures, or Components (SSC). This change will not result in any new failure modes of equipment important to safety nor result in any new or adverse failure effects, and therefore this change will not create any new initiators or contributors for an event that could be considered a new accident. Operation of the ESW System, sluice gates, and normal and alternate intake paths remain unchanged from their current state after installation of the new components, and therefore a different type of malfunction of equipment important to safety cannot occur.

III. No. Subsequent to this activity, the sluice gates will still be capable of automatic opening upon receipt of a low forebay water level signal and therefore the margin of safety in regard to the availability of cooling water for the ESW pumps is not reduced. The changes do not interface with or affect any plant SSCs in such a manner as to reduce any margin of safety. The operational and functional configuration of the ESW System will not be changed, and therefore any margins of safety associated with the ESW System will not be reduced.

Safety Evaluation: 00-0025
Source Document: Modification Request Form (MRF) 99-0024, Revision 0

Description of Change:

The purpose of Design Modification 99-0024, Revision 0 is to install non-intrusive ultrasonic feedwater flow instrumentation supplied by Caldon Incorporated. This proposed design change modification will install the Caldon Leading Edge Flow Meter 2000 (LEFM2000) to monitor feedwater flow through the two feedwater lines currently monitored by the feedwater flow venturi (1N27-N001A/B).

Summary:

I. No. The design of the flow measurement and the instrumentation are of the same technology as previously installed in November 1995 to conduct Feedwater Flow Measurement Testing at both low and high power levels. This testing was completed under TXI 242 and the associated 10CFR50.59 evaluation. The installation conforms to the same design codes and standards as the previously installed instrumentation. The accuracy of the instrumentation remains equal or better. There are no accidents previously evaluated in the USAR since this modification interfaces with Process Computer System (C91) plant computers and the Distribution Panels System (R25) 120v/240v/480v distribution panels and they are not associated with any accident initiators.

The Feedwater Control System (C34) instrumentation being installed by this modification is not an accident initiator. The changes do not reflect any changes in the Feedwater (N27) or C34 Systems that will cause it to operate outside of applicable design or testing limits. The installation does not result in any changes to system interfaces other than the C91 and R25 systems, neither of which are systems that can initiate any accident. The signal obtained at the C91 computer cannot be used until further evaluations are completed on the Instrument Uncertainty Analysis. The failure analysis as described above indicates that this design modification will not increase the probability of a Feedwater System failure or transient. This design modification does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR.

II. No. The addition to the N27/C34 System instrumentation will be performed using the Perry Nuclear Power Plant (PNPP) installation codes and standards. The instrument is of the non-intrusive type with no control or usable indication function. This design modification is for the installation of the instrument and not the use of the output signal. This design package will not change the function of the N27/C34 feedwater flow control or monitoring and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This change also does not cause any event evaluated in the USAR to be incredible to be come credible or any event that was previously bounded to become bounding. This design change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the USAR.

III. No. This change has no effect on the Technical Specifications, the Operational Requirements Manual, NRC Safety Evaluation Report, and Standard Review Plan. Performance of the N27/C34 Systems have not been affected by this design package since the current feedwater flow instrumentation will continue to provide feedwater flow monitoring using the existing methods. The instrument is of the non-intrusive type with no control or usable indication functions. This design modification is for the installation of the instrument and not the use of the output signal. Consequently, there is no negative effect or change to the performance or

reliability of the N27/C34 system. There is no margin of safety described in the Technical Specifications. There are no new operator actions required or affected as described in any part of the original license basis. This proposed design package does not eliminate or alter any feedwater flow controls, monitoring or calculations, nor compromise the ability to independently monitor feedwater flow. Therefore, these changes do not reduce the margin of safety as defined in the bases for the Technical Specifications.

Safety Evaluation: 00-0026
Source Document: Design Change Package (DCP) 99-5048, Revision 0

Description of Change:

DCP 99-5048 addresses Heating Ventilating & Air Conditioning (HVAC) modifications required to support the conversion of Control Complex (CC) 620' Unit 2 Div. 1 and Div. 2 areas into the new Radiologically Restricted Area (RRA) Access Area. This DCP adds three Air Handling Units, with associated ventilation components and controls, to serve the new RRA Access office areas. Additionally, an exhaust fan, with associated fire damper, will be installed in the Service Building (SB) RRA Access Hallway. This fan is being added to ensure airflow is from the Control Building, a radiological clean area, and into the Intermediate Building (IB) or Hot Machine Shop, which are potentially contaminated areas.

The addition of this exhaust fan, in conjunction with the removal of door IB-330, per DCP 99-5046, requires that the airflow distribution for the Intermediate Building Ventilation System (M33) and the Hot Shop HVAC System (M54) be modified.

A count room, with enclosed changing room, is being included as part of the new 620' RRA Access Control Point. Since the potential for low level airborne contamination may arise within the count room and changing room, an exhaust to the Controlled Access and Miscellaneous Equipment Areas HVAC System (M21) will be provided per this DCP. To more accurately reflect equipment temperature requirements in the different areas of environmental zone CB-2, three separate zones were created. For each of these environmental zones the maximum and minimum temperatures, for normal plant operation, are being modified to reflect the limiting Equipment Qualifications.

Summary:

I. No. The equipment that is changed by this modification is not an accident initiator. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The systems altered per this modification are HVAC Systems that provide environmental control and ventilation requirements for the plant. Since the environmental control and ventilation requirements are not impaired, these HVAC changes cannot be direct precursors of an accident. This modification will continue to ensure compliance with all Environmental Qualification (EQ) parameters, therefore, the possibility of a failure of equipment in an area served by these HVAC systems is not created. The proposed activity does not create the possibility of an accident of a different type than any previously evaluated in the USAR nor does the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The systems modified per this DCP are ventilation systems outside of the nuclear island that have no direct impact upon the margin of safety as defined in the basis for any Technical Specification. These modifications do not result in any changes in operation of the Control Room boundary, the Control Room HVAC System, the Annulus Exhaust Gas Treatment System, nor the Fuel Handling Building Ventilation System.

Safety Evaluation: 00-0027
Source Document: Design Change Package (DCP) 99-5046, Revision 0

Description of Change:

The purpose of this modification is to renovate the current Unit 2 Motor Control Center (MCC) and Switchgear Rooms on Elevation 620'-6" of the Control Complex for use as a Radiological Restricted Area (RRA) Health Physics (HP) Control Point and HP support facilities. The majority of the renovation activities are concentrated within the existing Control Complex area. Minor changes will be required within the existing Service and Intermediate Building to provide an access corridor to and from the new HP Control Point.

Summary:

I. No. Passive architectural components such as the partition walls, doors, and ceilings of Control Complex floor elevation 620'-6" are not precursors to any accident described in USAR Chapter 15 or other analyzed plant events such as high winds, design basis tornado, flood, fire, and seismic events. The seismic design of these architectural components will prevent any seismic interaction with existing safety related components. Changes to the structure and architectural features as described within this modification do not increase the probability of occurrence of an accident or other plant event previously evaluated in the USAR. The proposed design modification to the architectural feature of Control Complex floor elevation 620'-6" is being implemented in accordance with criteria set forth in NRC General Design Criteria (GDC) 1, GDC 2, GDC 3, GDC 4, GDC 5, and the structural design criteria set forth in the USAR. The addition of a second RRA access point will facilitate more effective RRA access. Changes to internal Control Complex doors and walls will not change any barriers to radiological releases or release paths. Changes to the abandoned equipment foundation pads and architectural features as described within this modification do not increase the radiological consequences of an accident previously evaluated in the USAR.

The partition wall enclosures erected around the equipment important to safety are designed and constructed to meet the same seismic and fire protection requirements as the existing partition walls and will have no effect on equipment important to safety. The equipment important to safety within the area remains qualified. Structural removal of abandoned equipment foundation pads has no impact on the equipment important to safety that will remain within the boundary of the area being modified. No new failure modes or effects are created as a result of the changes to the structure and architectural features as described within this modification. Changes to the architectural features as described within this modification will not increase the probability of a malfunction of equipment important to safety previously evaluated in the USAR. This modification will not add any new radiological release paths to the plant or alter any licensed discharge paths. Changes to the architectural features as described within this modification will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The passive structural and architectural features of Control Complex floor elevation 620'-6" are not precursors to any accident in the USAR and are independent of the power block. Changes to the structure and architectural features as described within this modification are designed and constructed to meet the same requirements as the existing design and will not create the possibility of an accident of a different type than any previously evaluated in the USAR. Changes to the architectural features and removal of abandoned equipment foundation

pads as described within this modification are designed and constructed to meet the same seismic and fire protection requirements as the existing design and will not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than any previously evaluated in the USAR.

III. No. The Control Complex elevation 620'-6" structural and architectural features are outside the nuclear island and have no direct impact upon the margin of safety as defined in the basis of any Technical Specification. Changes to the structure and architectural features as described within this modification do not result in any changes to the Control Room boundary or impact any Control Room related system operation or function. Therefore, the margin of safety as described in the bases for any Technical Specification is not reduced.

Safety Evaluation: 00-0028
Source Document: Simple Modification Request Form (SMRF) 99-5045, Revision 0

Description of Change:

This design change will delete carbon dioxide hose stations from the Unit 2 Switchgear Rooms, located in the 620' elevation of the Control Complex. The safety related equipment in the area consists of Unit 2, Division 3 equipment, which serves as a backup for the Unit 1 equipment, and one Motor Control Center (MCC), located along the north wall of the Unit 2 area. The nonsafety equipment in use to support Unit 1 is being consolidated in the west part of the Unit 2, Division 1 Room and a carbon dioxide hose station will remain in service for the switchgear left in place. The new occupancy will be considered light hazard occupancies, therefore, carbon dioxide hose stations are not required in this area.

Summary:

I. No. The remaining water and carbon dioxide hose stations provide an adequate level of protection for equipment important to safety for any expected fire hazards and the associated worst-case fire scenarios. Protection of safe shutdown equipment and equipment important to safety will be maintained by fire rated barriers enclosing the rooms containing safety related equipment. There is no change in the potential effects of a fire or fire fighting activities in the areas covered by these hose stations on safety related equipment or equipment important to safety and the potential impact of the use of water as a fire-fighting agent remains as previously evaluated. There are no postulated post accident radiological sources in the Control Complex area protected by these carbon dioxide hose stations and the availability or reliability of any equipment whose failure could prevent mitigation of the radiological consequences of any analyzed accident will not be affected. Therefore, the elimination of these hose stations will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The elimination of the carbon dioxide hose stations will not increase the threat or the consequence of fire-induced damage to safety related structures, systems, nor is a new or unique type of equipment failure potential introduced by this proposed change. Protection of safe shutdown equipment and equipment important to safety in adjacent buildings is provided by fire rated barriers and the associated worst-case fire scenario will be adequately mitigated by the remaining manual fire suppression capability. The use of the water type hose stations for manual suppression will not increase the extent or consequence of fire-induced damage to safety related structures, systems, or components. From the perspective of nuclear safety impact, these changes will not cause potential single failures to become common mode failures, nor cause events previously considered incredible to become credible. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The change to the Fire Protection Program implemented through this SMRF is permissible under the Perry Nuclear Power Plant (PNPP) Fire Protection License Condition to the extent that the change does not adversely impact the credited post-fire safe-shutdown capability. The margin of safety, as it applies to this modification, is based on maintaining one train of equipment to achieve and maintain safe shutdown free of fire damage. The separation and protection of redundant trains credited to support the post-fire shutdown capability is unaffected by the elimination of the carbon dioxide hose stations. Therefore, equipment and circuits required to achieve and maintain safe shutdown will remain free of fire damage and the margin of safety established by the Fire Protection Program as reviewed and approved by the NRC is maintained.

Safety Evaluation: 00-0029
Source Document: Perry Security Plan, Revision 29

Description of Change:

These changes to the Security Plan pertain to personnel access control measures for the protected and vital areas. These changes are considered to be safeguard information and as a result are managed in accordance with 10CFR73.21.

Summary:

I. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This change is administrative only. The design and operation of the plant are unchanged. Accident analysis is unaffected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 00-0030
Source Document: Condition Report (CR) 99-3037

Description of Change:

Plant Administrative Procedure PAP-0809, "Radiological Environmental Contamination Response", requires the performance of an evaluation if contaminated material is allowed to remain in place, regardless of whether it is inside or outside of the Protected Area. This CR evaluates the radiological impact of leaving sediment with a very low level of Cobalt Co-60 to remain in place. The evaluation shows that the radiological impact is insignificant.

Summary:

I. No. The sediment cannot interact with any System, Structure or Component (SSC) involved in the operation of the Perry Plant. Therefore the sediment cannot increase the probability of occurrence of an accident or increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. The potential radiological consequences of leaving the sediment in place were conservatively assessed and compared to the worst case accident having the same environmental pathway. Results show the impact from leaving the sediment in place is insignificant and will not increase the radiological consequences of an accident or of a malfunction of equipment important to safety previously evaluated in the USAR. Therefore, no safety concern exists.

II. No. The sediment does not interact with any plant SSC. Leaving it in place will not create the possibility of an accident of a different type or a malfunction of equipment of a different type than any previously evaluated in the USAR. Therefore, no safety concern exists.

III. No. The radiological impact of leaving the sediment in place was conservatively assessed and determined to be insignificant. It does not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 00-0031
Source Document: USAR Change Request (CR) 00-054

Description of Change:

The subject USAR Change Request (CR) proposes to remove excessive detail and duplication from USAR Table 3.2-1.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade SSC reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report/Supplements to Safety Evaluation Report are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0032
Source Document: Setpoint Change Requests (SCR) 1-99-1117, Revision 0; SCR 1-99-1118, Revision 0; SCR 1-99-1119, Revision 0

Description of Change:

The above listed SCRs change the condenser high pressure alarms setpoints due to Power Uprate of the Unit. USAR Section 10.4.1.6.b states that the high pressure alarm is provided at approximately 4 inches Hg absolute. Due to Power Uprate of the Unit, the condenser pressure will be slightly increased, resulting in a necessity to increase the alarm settings to avoid a nuisance alarm.

Summary:

I. No. The relevant accidents that apply to condenser vacuum are Turbine Trip (USAR Section 15.2.3, Main Steam Isolation Valves Closure (USAR Section15.2.4), and Loss of Condenser Vacuum (USAR Section 15.2.5). Changing the setpoint for the condenser vacuum alarm will not initiate any of these accidents. These setpoints provide an alarm on control room panel 1H13-P680 to alert the control room operators of a low condenser vacuum condition. There are no automatic actions that are precipitated by this setpoint. The new setpoint has adequate margin from the setpoints that provide automatic initiation functions, such as a Turbine Load Limit Setback and a Reactor Recirculation Flow Control Valve runback. The low vacuum alarm is not an accident initiator.

The new setpoints do not change any of the instrument accuracies or response characteristics, and therefore do not cause a condition that would make one of the above accidents more likely to occur. The new alarm settings will not cause the system to operate outside of its design limits. The setpoint changes do not cause any effect on any interfacing components or systems that would lead to initiation of these accidents. Since the changes do not lead or contribute to any USAR accident it was concluded that the changes would not increase the radiological consequences of any of these accidents. The condenser high alarm is not equipment important to safety. In addition, there were no new failure modes or effects that would lead to a malfunction of equipment important to safety. Consequently it was concluded that the SCRs would not increase the radiological consequences of malfunction of equipment important to safety, since the changes do not effect or impact equipment important to safety.

II. No. It was concluded that the setpoints would not create the possibility of a different type, since the function of the alarm does not cause the actuation of any plant equipment other than the alarm that only provides indication of a high pressure in the condenser. The alarm does not cause this condition. The SCRs do not cause any new failure modes or effects that would impact equipment important to safety.

III. No. The setpoint changes will not affect the design basis of any System, Structure, or Component (SSC) or affect the ability of any SSC to perform its design function. The margin of safety associated with the condenser vacuum setpoints would be the setpoints that perform an actuation or automatic trip function such as a turbine trip initiated on loss of condenser vacuum or when condenser back pressure exceeds approximately 8" HgA. The condenser vacuum alarm is not associated with these functions. Since these setpoint changes only involve the condenser high pressure alarm, no margin of safety has been reduced. Therefore, the setpoint changes will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0033
Source Document: USAR Change Request (CR) 00-055; Simple Modification Request Form (SMRF) 00-5013, Revision 0

Description of Change:

USAR Change Request 00-055 defines the change to the plant that results from use of the sluice gate sealing system. The automatic opening capability of the sluice gates upon receipt of a low Emergency Service Water (ESW) forebay water level signal will be eliminated during the time that the sluice gate seals are inflated. Elimination of the automatic opening feature renders the alternate intake tunnel unavailable and thus the alternate supply source from the ultimate heat sink will not be available. This USAR Change Request also identifies that the sluice gates and the non-safety seals, when inflated, provide a barrier to prevent recirculation of plant discharge water to the ESW forebay. This Safety Evaluation has been submitted to the NRC for review and approval by letters dated June 1, 2000 (PY-CEI/NRR-2492L) and June 30, 2000 (PY-CEI/NRR-2505L).

Summary:

I. Yes. Inflation of the sluice gate seals and disabling of the automatic opening function does not involve any system, structure, or component (SSC) that could be construed to be an accident initiator, and this operation does not affect any SSC such that the probability of occurrence of an accident could be increased. The sluice gates in the closed position and the inflated sluice gate seals provide a barrier between the plant discharge tunnel and the ESW forebay. Performance of this isolation function does not result in any interfaces with any plant SSCs in such a manner as to increase the probability of occurrence of a previously evaluated accident. Opening of the sluice gates and use of the alternate intake tunnel is not relied upon for mitigation of a Loss Of Coolant Accident (LOCA) or other accidents with radiological consequences analyzed in the USAR. Defeating the automatic gate opening signal prevents the sluice gates from accomplishing their safety function of automatically opening the alternate intake tunnel to ensure a cooling water supply. This operational change to the plant will purposely defeat the sluice gate manual raise/lower circuit and the automatic opening feature during the time that the seals are inflated. Although the probability of malfunction of the sluice gates may have increased due to the deliberate defeating of the automatic opening feature, this condition is of no consequence since the probability of failure of the normal intake is extremely low such that the sluice gates would not be required to function during the time period when the automatic open signal is defeated. Therefore, it is concluded that failure of the sluice gates seals does not increase the radiological consequences of any previously analyzed malfunctions of equipment important to safety since failure of the seals is highly unlikely and thus the accident mitigating capability of the ESW system will not be degraded.

II. No. The operational change to the sluice gates, i.e., inflation of the seals and disabling of the sluice gate automatic opening feature, does not result in any interactions or interfaces with other plant SSCs that could create the possibility of an accident of a different type. Availability of only one intake during the time that the automatic opening function is disabled has been demonstrated to be acceptable because a water supply from the normal intake to the ESW pumps will be available. Cooling water supply from only one intake path cannot possibly initiate an accident of a different type than previously evaluated because the cooling water supply paths cannot create or initiate an accident. This operational change does not introduce any new failure modes of equipment important to safety nor result in any new or adverse failure effects. The operational change under evaluation does not adversely affect the satisfactory operation of the ESW system. Therefore, this activity cannot create a different type of malfunction of equipment since the ESW system's capability to perform its safety function is not changed.

III. Yes. The current acceptance limit associated with the availability of the cooling supply from the ultimate heat sink assumes that two supply paths are available and that automatic initiation institutes supply from the alternate path. The failure or limiting point in regard to supply from the ultimate heat sink would correspond to unavailability of both intake paths. This operational change disables the manual/automatic opening feature of the redundant sluice gates during the time the seals are inflated and thus isolates the alternate supply path. This activity represents a reduction to the margin of safety as previously defined since it results in the availability of only one intake path from the ultimate heat sink. The reduction to the previously defined margin of safety is acceptable since the probability of loss of the normal intake is extremely low, and therefore it can be concluded that cooling water from the ultimate heat sink will be available through the normal intake path. The failure or limiting point has an extremely low probability of ever being reached.

Safety Evaluation: 00-0034
Source Document: Perry Specification Technical Guidelines (PSTG), Revision 5, PIC 3

Description of Change:

PSTG, Revision5, PIC 3 incorporates the following:

- Updated values for Minimum Reactor Pressure Vessel Flooding Pressure (MRFP), Minimum Core Flooding Interval (MCFI), Heat Capacity Limit (HCL), Pressure Suppression Pressure (PSP) and Safety Relief Valve Tail Pipe Level Limit (SRVTPLL), and
- Updated NUMAC® (E31, leak detection) setpoints, and updated the calculation revision numbers.

Summary:

I. No. The revisions to MRFP, MCFI, HCL, PSP and SRVTPLL are based on the Cycle 8 core reload and performed using the formulas and methodologies previously used which are in accordance with Emergency Plan Guidelines (EPG)/Safety Analysis Guidelines (SAG) and those previously used in PSTG, Revisions 5. The changes for NUMAC setpoints were previously evaluated in accordance with Plant Administrative Procedure (PAP) 1403. Therefore, the revisions to MRFP, MCFI, HCL, PSP, SRVTPLL and the NUMAC setpoints will not increase the consequences of an accident previously evaluated. The calculation revision numbers are entirely editorial. Therefore, the probability of and consequences of accidents and of equipment malfunctions previously evaluated in the USAR are not affected.

II. No. The PSTG provides the licensing bases for the Plant Emergency Instructions (PEI). The PEIs provide symptom based actions to take in response to an accident or transient to shutdown the reactor, restore and maintain adequate core cooling, and maintain containment integrity. Operation of the systems and components as directed by the PEIs occurs after the accident or transient has begun and therefore does not affect the possible initiators of any accidents or transients. All systems operated by the PSTG/PEIs are operated within the design bases of the system. Therefore, this change does not create the possibility of a accident or a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The revisions to MRFP, MCFI, HCL, PSP and SRVTPLL are based on the Cycle 8 core reload and performed using the formulas and methodologies previously used, and the changes for the NUMAC setpoints were previously evaluated. This change affects operator actions taken post-accident to restore the plant to a safe condition and to reestablish Technical Specification assumptions. These actions are taken in response to an accident which is beyond the USAR design and Technical Specification, Operational Requirements Manual and Offsite Dose Calculation Manual assumed bases. The margin of safety as defined in the bases for any Technical Specification is not reduced.

Safety Evaluation: 00-0035
Source Document: Setpoint Change Requests (SCR) 1-00-1000 through 1-00-1023, Revision 0; Plant Data Book R001, "Operational Requirement Manual (ORM)", Revision 0, PIC 23

Description of Change:

Setpoint changes to the following: Average Power Range Monitor (APRM) – Flow Biased SCRAM (two loop and single loop operation); Average Power Range Monitor – Flow Biased Rod Block (two loop and single loop operation); Main Steam Line Flow High Isolation.

Changes to the ORM to document the following setpoint changes: APRM Control Rod Block Instrumentation Flow Biased Neutron Flux – Upscale Trip Setpoint and Allowable Value for Single and Two Loop Operation (Table 6.2.1-2, Trip Functions 1.a.1) and 1.b.1)); Reactor Protection System Flow Biased Simulated Thermal Power High Trip Setpoint for Single and Two Loop Operation (Attachment 2, Table 1, Functional Units 2.b.1)a) and 2.b.2)a)); Isolation Actuation Instrumentation Main Steam Line Isolation Trip Setpoint (Attachment 2, Table 2, Functional Unit 2.d.); End-of-Cycle Recirculation Pump Trip System Instrumentation Trip Setpoint Supporting Notes (Attachment 2, Table 5, Trip Function 2); Control Rod Block Instrumentation Rod Pattern Control System Trip Setpoint for Low Power and Rod Withdrawal Limiter High Power (Attachment 2, Table 7, Trip Functions a. and b.)

The safety evaluation is based on the approved Power Uprate analysis contained in Amendment 112 which was issued on June 1, 2000 by the NRC in response to an application request dated September 9, 1999 (PY-CEI/NRR-2420L) as supplemented by submittals dated March 1, 2000 (PY-CEI/NRR-2470L), March 13, 2000 (PY-CEI/NRR-2477L) and May 11, 2000 (PY-CEI/NRR-2499L).

Summary:

I. Yes. The changes have been addressed in Power Uprate (PU) Licensing Amendment Request (LAR), PY-CEI/NRR-2420L, which the NRC approved on 6/1/00 as Amendment 112. However, this Amendment is currently not effective at Perry and therefore not considered part of the current Perry USAR. As explained in the Significant Hazards Considerations (SHC) associated with that LAR, the regulatory criteria established for plant equipment are still being complied with and no new operating mode, safety related equipment lineup, accident scenario, or equipment failure mode was identified. As such, the probability of occurrence of an accident or malfunction of equipment important to safety has not been increased. However, as explained in the SHC associated with the Power Uprate LAR, some of the radiological consequences in the USAR were calculated to increase. As such, the radiological consequences of an accident or a malfunction of equipment important to safety has been increased by this change.

II. No. The changes have been addressed in Amendment 112. However, this Amendment is currently not effective at Perry and therefore not considered part of the current Perry USAR. As explained in the SHC associated with the PU LAR, the full spectrum of accident considerations defined in RG 1.70 has been evaluated and no new or different kind of accident has been identified. The changes are the result of Power Uprate, which uses existing technology and applies it within the capabilities of already existing plant equipment in accordance with existing regulatory criteria. No new power dependent accidents have been identified with Boiling Water Reactors (BWR) designed for higher power. No new operating mode, safety related equipment lineup, accident scenario or equipment failure mode was identified. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created

III. Yes. The changes have been addressed in Amendment 112. However, this Amendment is currently not effective at Perry and therefore not considered part of the current Perry USAR.

As explained in the SHC associated with that LAR, Power Uprate does not involve a significant reduction in margin of safety. Since the changes are the result of Power Uprate, a reduction in the margin of safety as defined in the basis for technical specifications is created.

Safety Evaluation: 00-0036
Source Document: Plant Data Book (PDB)-A0006, Revision 7; PDB-A0011, Revision 2; PDB-A0014, Revision 6

Description of Change:

Changes to PDB-A0006 were identified in Sections 2.3.1 and 2.4 of Attachment 1 to PY-CEI/NRR-2420L.

Changes to PDB-A0011 include modifying the Feedwater Temperature vs. Core Thermal Power graph. The graph is revised to reflect power uprate conditions by extrapolating the existing curves and adjusting these curves as necessary to ensure a smooth transition to the new calculated final temperature. As such, the existing feedwater temperature data points are unchanged or conservatively slightly increased for the same Mega Watts (MW) Thermal Power Level. However, since 100% rated thermal power is being redefined to be 3758 MWth vs. 3579 MWth, the curves will shift slightly to the left to account for the rescaled % rated thermal power. These changes are consistent with information presented in Table 1-2 of Attachment 1 to PY-CEI-NRR-2420L.

Changes to PDB-A0014 include modifying the Percent Reactor Power vs. Indicated Steam Flow graphs. The graphs are revised to reflect power uprate conditions by extrapolating data to the uprated steam flow conditions and re-scaling the percent reactor power. As such, the steam flow data points are unchanged for the same MWth power level. However, since 100% rated thermal power is being redefined to be 3758 MWth vs. 3579 MWth, the curves will shift slightly down to account for the rescaled % rated thermal power. These changes are consistent with information presented in Table 1-2 of Attachment 1 to PY-CEI-NRR-2420L.

Summary:

I. No. The changes have been addressed in Power Uprate (PU) Licensing Amendment Request (LAR) PY-CEI/NRR-2420L which the NRC approved on 6/1/00 as Amendment 112. This Amendment became effective at Perry on 6/12/00, and therefore is considered part of the current Perry Licensing Basis. As explained in the Significant Hazards Considerations (SHC) associated with that LAR, the regulatory criteria established for plant equipment are still being complied with and no new operating mode, safety related equipment lineup, accident scenario, or equipment failure mode was identified. As such, the probability of occurrence of an accident or malfunction of equipment important to safety has not been increased. As explained in the SHC associated with the Power Uprate LAR, some of the radiological consequences in the USAR were calculated to increase. However, this effect on the radiological consequences was reviewed and approved by the NRC as part of License Amendment 112 which is now part of the Perry Licensing Basis. As such, the radiological consequences of an accident or a malfunction of equipment important to safety previously evaluated is not increased by this change.

II. No. The changes have been addressed in PU LAR PY-CEI/NRR-2420L which the NRC approved on 6/1/00 as Amendment 112. This Amendment became effective at Perry on 6/12/00 and therefore is considered part of the current Perry Licensing Basis. As explained in the SHC associated with the PU LAR, the full spectrum of accident considerations defined in RG 1.70 has been evaluated and no new or different kind of accident has been identified. The changes are the result of Power Uprate, which uses existing technology and applies it within the capabilities of already existing plant equipment in accordance with existing regulatory criteria. No new power dependent accidents have been identified with BWRs designed for higher

power. No new operating mode, safety related equipment lineup, accident scenario or equipment failure mode was identified. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety previously evaluated has not been created.

III. No. The changes have been addressed in LAR PY-CEI/NRR-2420L, which the NRC approved on 6/1/00 as Amendment 112. This Amendment became effective at Perry on 6/12/00 and therefore is considered part of the current Perry Licensing Basis. As explained in the SHC associated with that LAR, Power Uprate does not involve a significant reduction in margin of safety. Thus, Power Uprate is considered to represent a slight reduction in the margin of safety. However, since this effect on the margin of safety was reviewed and approved by the NRC as part of License Amendment 112 which is now part of the Perry Licensing Basis. As such, the margin of safety as defined in the basis for technical specifications has not been reduced.

Safety Evaluation: 00-0037
Source Document: Plant Data Book (PDB)-F0001, Revision 7

Description of Change:

PDB-F0001, "Core Operating Limits Report", for Perry Nuclear Power Plant (PNPP) Unit 1 Cycle 8 (Reload 7) was updated to include the thermal limits associated with Power Uprate analysis.

Although the Power Uprate was handled by GE topical for Power Uprate analysis, the analysis of the reactor core for Power Uprate conditions was evaluated as a reload analysis using GESTAR II (which is General Electric's topical report which applies NRC approved methodologies to the reload analysis). GESTAR II is referenced by Perry's USAR as the means for reload analysis.

The Safety Evaluation is based on the analysis results from the GESTAR II process and the approved Power Uprate analysis contained in Amendment 112 which was issued on June 1, 2000 by the NRC in response to an application request dated September 9, 1999 (PY-CEI/NRR-2420L) as supplemented by submittals dated March 1, 2000 (PY-CEI/NRR-2470L), March 13, 2000 (PY-CEI/NRR-2477L) and May 11, 2000 (PY-CEI/NRR-2499L). This Safety Evaluation addresses only thermal limit changes.

Summary:

I. No. The Power Uprate analysis was performed in accordance with the requirements of the GESTAR II. The fuel failure mechanisms described in GESTAR II are also unchanged. The cycle specific Safety and Operating Limits will protect the fuel in accordance with the design basis. As a result, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. The Power Uprate analysis was performed in accordance with GESTAR II. The essential components of the fuel are the same as previously analyzed. The function and operation of the fuel remains unchanged. The initiating sequence of the events have not changed. The GESTAR II analysis has been accepted by the NRC as comprehensive for ensuring that the reload design will perform within acceptable bounds. As a result the possibility for an accident or malfunction of a different type than any evaluated previously in the USAR is not created.

III. No. The limits contained in the Core Operating Limits Report (COLR) do not alter the design or function of any plant system. The Power Uprate analysis was produced using NRC-approved methods described in GESTAR II. The design satisfies the acceptance criteria of the other fuel-related Technical Specifications [Minimum Critical Power Ratio (MCPR)] Operating Limits [Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)]. As a result the margin of safety as defined in the bases for any Technical Specification is not reduced.

Safety Evaluation: 00-0038
Source Document: Temporary Instruction TXI 317, Revision 0

Description of Change:

Temporary Instruction, TXI 317, "100% to 105% Power Uprate Implementation", is similar to some of the original startup tests as described in USAR Section 14.2.12. The key difference being the higher reactor power level conditions under which the test will be performed. This test consists essentially of steady state, baseline data collection between approximately 90% and 100% of the pre-uprate licensed power [i.e., approximately 3221 Mega Watt thermal (MWth) to 3579 MWth]. Following completion of the baseline testing at 100% of the pre-uprate power condition (i.e., 3579 MWth), a series of five 1% incremental tests will be conducted through 105% of the pre-uprate power (i.e., 3758 MWth). These power increases will be made along an established flow control/rod line. During each incremental test, data will be taken and evaluated for acceptance against associated Technical Specification (TS) requirements and performance parameters.

Summary:

I. No. The proposed testing activity has been addressed in Power Uprate (PU) Licensing Amendment Request (LAR), PY-CEI/NRR-2420L, which the NRC approved on 6/1/00 as Amendment 112. This Amendment became effective at Perry on 6/12/00 and therefore is considered part of the current Perry Licensing Basis. As explained in the Significant Hazards Considerations (SHC) associated with that LAR, the regulatory criteria established for plant equipment are still being complied with and no new operating mode, safety related equipment lineup, accident scenario, or equipment failure mode was identified. As such, the probability of occurrence of an accident or malfunction of equipment important to safety has not been increased. As explained in the SHC associated with the Power Uprate LAR, some of the radiological consequences in the USAR were calculated to increase when operating at the higher power level. However, this effect on the radiological consequences was reviewed and approved by the NRC as part of License Amendment 112 which is now part of the Perry Licensing Basis. As such, the radiological consequences of an accident or a malfunction of equipment important to safety previously evaluated is not increased by this change.

II. No. The proposed testing activity has been addressed in Amendment 112. As explained in the SHC associated with the PU LAR, the full spectrum of accident considerations defined in RG 1.70 has been evaluated and no new or different kind of accident has been identified. The testing activity is the result of Power Uprate, which uses existing technology and applies it within the capabilities of already existing plant equipment in accordance with existing regulatory criteria. No new power dependent accidents have been identified with Boiling Water Reactors (BWR) designed for higher power. No new operating mode, safety related equipment lineup, accident scenario or equipment failure mode was identified. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety previously evaluated has not been created.

III. No. The proposed testing activity has been addressed in Amendment 112. As explained in the SHC associated with that LAR, Power Uprate does not involve a significant reduction in margin of safety. Thus, Power Uprate is considered to represent a slight reduction in the margin of safety. However, since this effect on the margin of safety was reviewed and approved by the NRC as part of License Amendment 112 which is now part of the Perry Licensing Basis. As such, the margin of safety as defined in the basis for technical specifications has not been reduced.

Safety Evaluation: 00-0040
Source Document: Simple Modification Request Form (SMRF) 00-5010, Revision 0

Description of Change:

This design change modifies the Primary Access Control Point (PACP) to provide for office space and a lunchroom area for plant personnel. A review of the available PACP floor plan and building functions has determined that a series of building changes could provide the required office space for the In Process Center (IPC) personnel and the employee lunchroom. This modification will change the eight entrance turnstiles on the South side of the building to be four (4) entrance and four (4) exit turnstiles.

Summary:

I. No. There is no equipment in the PACP that is important to safety. The proposed modifications are limited to architectural and electrical changes in the PACP building. No plant systems that would be used for safe shutdown are impacted by this design change. As stated in USAR Section 9A.4.20, the PACP building does not contain safe shutdown equipment. The proposed PACP design changes will maintain personnel access to the protected area as required by the Plant Security Plan. The integrity of the security system will not be compromised by this design change. The design requirements as identified in the Security Plan will be maintained by the new configuration.

II. No. The modification to the PACP building has been in accordance with the original building codes and standards and in accordance with the requirements of the Security Plan. The modification does not change the function or reliability of any equipment important to safety or required for safe shutdown. As stated in USAR Section 9A.4.20, the PACP building does not contain safe shutdown equipment and therefore changes to the building cannot create the possibility of an accident of a different type than those previously evaluated in the USAR.

This design change does not introduce any new failure modes. The modifications made to the PACP building do not inhibit or change the function of any system that contains equipment important to safety. As stated in USAR Section 9A.4.20, there is no safe shutdown equipment located in the building. In addition, the PACP is adequately separated from the buildings containing safety related equipment. A fire in the PACP will not jeopardize the operation or response of equipment important to safety or a safe shutdown response. Therefore, the proposed design changes will not create the possibility of an accident of a different type than any previously evaluated in the USAR.

III. No. The ability of the PACP building to process personnel into the Protected Area has not been affected by this design change. The building will continue to provide entrance and exit access control functions. This design change neither impacts the ability of a safety related system to perform the intended function nor does it reduce the margin of safety of any equipment important to safety.

This design change will not impact the Technical Specifications, Technical Specifications Bases, or Surveillance Requirements. Consequently, there is no affect or change on the performance of any safety systems. Therefore, this modification does not reduce the margin of safety as defined in the basis for any technical specification.

Safety Evaluation: 00-0041
Source Document: Simple Modification Request Form (SMRF) 00-5013, Revision 3

Description of Change:

Simple Modification Request Form (SMRF) 00-5013, Revision 3, installs a primary and back-up air supply source for the sluice gate seals. The primary air supply source for the seals will be the non-safety related P52 (Instrument Air) system. A back-up source of supply air for each sluice gate seal will be provided by a non-safety related compressed air bottle located in the northeast corner of the Emergency Service Water (ESW) pump house on the 620' elevation. The valve stands that support sluice gate seal inflation/deflation will be modified to accommodate the air supply sources.

Summary:

I. No. The ESW system is an accident mitigating system that provides a reliable source of cooling water during accident conditions and is not an accident initiator. Consequently, the inflation/deflation system for the inflatable sluice gate seals, which is part of the ESW system, is likewise not an accident initiator. The proposed modification does not interface with any plant systems, structures, or components in such a manner as to increase the probability of occurrence of a previously evaluated accident. Installation of the primary and back-up air supply sources for the inflatable sluice gate seals, and installation of the components to support operation of the seal inflation/deflation system do not create any interactions with any systems, structures, or components important to safety such that the accident mitigating function of the ESW system or any other system is compromised. These modifications do not affect the sluice gates ability to provide isolation between the discharge tunnel and ESW forebay and therefore the ESW inlet temperature will not exceed its maximum allowable value thereby ensuring the continued operability of the ESW system for accident mitigation.

Installation of the primary and back-up air supply sources for the inflatable sluice gate seals, and installation of the new components to support operation of the seal inflation/deflation system do not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. Installation of these components does not change the functional configuration of the plant and the presence of the new components in the ESW system does not create any interactions or interfaces with other structures, systems, or components in the ESW pump house or in the plant such that the probability of occurrence of malfunction of any equipment important to safety would be increased. The installation of the primary and back-up air supply sources for the inflatable sluice gate seals, and installation of the new components to support operation of the seal inflation/deflation system do not affect any modes of operation of the ESW system and therefore the accident mitigating capability of the ESW system is not compromised. Subsequent to the modifications, the sluice gates will still provide the required isolation between the discharge tunnel and ESW forebay thus maintaining the operability of the ESW system.

II. No. The ESW system is an accident mitigating system; it is not an accident initiator. Consequently, the inflation/deflation system for the inflatable sluice gate seals, which is part of the ESW system, is likewise not an accident initiator and therefore cannot create the possibility of an accident of a different type than previously evaluated. The tie-in to the P52 instrument air lines, the back-up air supply bottles, and the modifications to the valve stand for sluice gate seal inflation/deflation do not 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. Installation of the primary and back-up air supply sources for the inflatable sluice gate seals, and installation of the new components to support the operation of the seal inflation/deflation system cannot create the possibility of a different type of

malfunction of equipment important to safety. The new components do not interact with any systems, structures, or components important to safety in such a manner as to create a different type of malfunction of equipment important to safety. These components do not introduce any new failure modes of equipment important to safety nor result in any new or adverse failure effects.

III. No. Installation of the primary and back-up air supply sources for the inflatable sluice gate seals, and installation of the new components to support operation of the seal inflation/deflation system do not change the current operational modes of the sluice gates. The sluice gates will still be capable of automatic opening upon receipt of a low forebay water level signal and therefore the margin of safety in regard to the availability of cooling water for the ESW pumps is not reduced after installation of the new components. After installation of the new components, the sluice gates will still be able to perform their isolation function in the closed position and therefore the margin of safety associated with the ESW pump inlet temperature will be retained.

Safety Evaluation: 00-0042
Source Document: USAR Change Request (CR) 00-063

Description of Change:

The USAR Change Request (CR) removes duplicate and obsolete information from USAR Section 5.3, "Reactor Vessel", and also proposes editorial and clarifying changes. The USAR provides a text description together with selected data associated with Unit 2. This material is being removed from the USAR because it is obsolete information. The construction of Unit 2 has been terminated.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade systems, structures, or component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplements to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0043
Source Document: USAR Change Request (CR) 00-065

Description of Change:

The USAR Change Request (CR) removes excessively detailed, duplicate and obsolete information from USAR Section 9.5, "Other Auxiliary Systems." Editorial changes to improve text clarity are also included.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade System, Structure, or Component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplements to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0044
Source Document: USAR Change Request (CR) 00-066

Description of Change:

This USAR Change Request (CR) revises the discussion and tabulation of codes and standards in Tables 3.2-1, 3.2-6, and 6.1-1 to clarify applicability, to reflect as-built plant records, and to maintain General Electric Standard Safety Analysis Report II (GESSAR II) consistency. In addition, it revises the discussion of shift staffing consistent with regulatory codes and standards.

Summary:

I. No. The proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the proposed changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The proposed changes will not degrade System, Structure, or Component (SSC) reliability. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The proposed changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The proposed changes do not change any radiological consequences to the public or onsite personnel.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety, nor do they unbound any previously bounded event. The proposed changes are not related to any malfunction of equipment installed in the plant. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report (SER)/Supplements to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design basis of any SSC. The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0045
Source Document: Simple Modification Request Form (SMRF) 98-5011, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 98-5011 was prepared to abandon the construction water pump, the construction fire protection service water pumps, and associated components (including the rubber fire protection water storage tank, fire protection service water pumps, storage tank hot water heater and pump, and construction water booster pump). The non-safety related electrical supplies to these pumps will also be removed or de-energized.

Summary:

I. No. The proposed change does not alter the function or create any new failure modes for the existing plant fire protection system. The existing isolation valves will be maintained except that they will now be locked closed. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed change, no new equipment malfunctions are postulated. The proposed modification does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0046
Source Document: Design Change Notice (DCN 5871), Revision 0

Description of Change:

This DCN involves removing 91 non-functional, non-installed 'Unit 2' and 12 'Common' Maintenance and Calibration jack station (R52 system) asset numbers from plant drawings. None of the 91 Unit 2 jack station cables were installed, six of the twelve jack stations and their associated cables in the 'Common' chart were never installed in their Unit 2 panels, and the other six were removed during renovation work in the Unit 2 Diesel Generator rooms.

Summary:

I. No. The R52 system is an independent system that is not functionally connected to any other system, which eliminates the probability of occurrence of an accident previously evaluated in the USAR. Removal and/or abandonment of these jack stations will not affect any system or subsystem that is required for safe operation and shut down of the plant. This change will not directly or indirectly impact the ability of structures, systems or components to perform either their safety or design functions, nor will it impact the original design and operation of the Maintenance and Calibration system. The R52 system is not associated with any fission product barriers. Implementation of this change will not cause or lead to an increase in dose to the public or on-site. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. Removal of these Maintenance and Calibration jack stations will not create an accident of a different type than previously evaluated in the USAR. None of the jack stations affected by this change could become initiators of an accident of a different type; the scope and impact of this change preclude the creation of a different accident type than those previously evaluated. No new failure modes of equipment are created. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. These changes have no affect on the Technical Specifications, Operational Requirements Manual or the basis for NRC approval as documented by the Safety Evaluation Report. The Technical Specification that references the Maintenance and Calibration jacks specifically is Section 3.6, which describes requirements for establishing communication between the field and the Control Room during certain evolutions. It does not prescribe the means of communication, nor does this change prevent establishing communication via the Maintenance and Calibration system if desired since the jack stations detailed in this evaluation are for Unit 2. Therefore, the proposed activity will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0047
Source Document: Drawing Change Notice (DCN) 5301, Revision 0

Description of Change:

This proposed change will eliminate the demineralized wet lay-up requirements of the Emergency Service Water (ESW) portion of the Residual Heat Removal (RHR) System Heat Exchangers. In addition, this DCN will provide the basis for deleting ESW valves 1P45-F0014A/B and 1P45-F0068A/B from Generic Letter (GL) 89-10, Safety Related Motor Operated Valve Testing and Surveillance, since the valves will no longer be required to perform an active safety function. The System Operating Instruction (SOI) states that in the event of valve closure for longer than momentarily, the affected RHR loop will be placed in secured status.

Summary:

I. No. Based on the frequency that the ESW System is chemically treated by chlorination, stagnant water will not cause degradation beyond design corrosion allowables of the RHR heat exchangers. Additionally, this periodic operation of the system would also remove small amounts of silt that may be present before it could become harmful. Testing (performance testing and eddy current examination) and water box/tube inspections confirm that the RHR heat exchangers still meet their design requirement to remove heat for all modes of operation. The elimination of wet lay-up cannot cause a loss of shutdown cooling capability and cannot cause a Loss Of Coolant Accident (LOCA) event to occur. Overall system performance of the ESW and RHR systems is maintained. The RHR and ESW system will not be operated any differently than the way the systems have been operated in the past. Water is still present (Lake Erie water versus demineralized water) in the ESW side of the RHR Heat Exchanger, thus a water hammer event will not occur. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. No hardware is being added or modified by this change. Motor Operated Valves (MOV) 1P45-F0014A/B and 1P45-F0068A/B will be required to be open for system operability and do not need to satisfy the GL 89-10 program because this change eliminates the need for the valves to change positions under accident conditions. The motor operated valves will be maintained in the required position for accident conditions (open) for safety related non-active components. Future waterbox (ESW side) inspections and testing will further ensure that the heat exchanger will meet its design requirements with respect to wall thickness. The RHR and ESW systems will not be operated differently than the way the systems have been operated in the past. Valve positions of the two MOVs are not credited in the Safe Shutdown Capability Report (SSCR). Changing the valve position does not affect the SSCR. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The margin of safety is not reduced since the RHR Heat Exchanger remains capable of performing its design functions. Cooling capacity (ESW) for safety related equipment is not reduced during normal and accident conditions. Residual heat removal capability for removing decay heat remains unchanged. The proposed change allows for the elimination of a subsystem that is unnecessary based on the way the ESW/RHR systems are operated and maintained at Perry.

Safety Evaluation: 00-0049
Source Document: USAR Change Request (CR) 00-073; Plant Administrative Procedure (PAP) 0230, Revision 3

Description of Change:

The USAR description and governing procedure for the performance of independent safety engineering is being revised to differentiate between the Independent Safety Engineering (ISE) function and the Independent Safety Engineering Group (ISEG), including recognition that ISE can be performed by personnel other than those assigned to the ISEG. The change eliminates the required number of members of the ISEG and relaxes the USAR qualification requirements to perform ISE. Clarification is made as to the ISE role in the operating experience program and performance of independent verification that plant activities are performed correctly. Other minor administrative and editorial changes are also included.

Summary:

I. No. The changes have no impact on the operation of the plant nor any system, structure or component. The administration, composition and responsibilities of the ISEG have no affect on any USAR accident initiators. The ISE function is being retained. The ISEG provides an oversight function to help assure plant nuclear safety. Changes to an administrative oversight function cannot have any impact on the accident analyses described in the USAR. The administration, composition and functions of the ISEG are not in any way related to precursors to equipment malfunction. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The changes have no impact on the operation of the plant nor any system, structure or component. Changes in ISEG administration, composition or function or the failure of the ISEG to perform its oversight role cannot impact any accident precursor. The ISE function is being retained. A secondary argument that a smaller ISEG is less capable of identifying problems or issues and that could lead to less safe operation of the plant is beyond the scope of a safety evaluation. In order for a "less safe" situation to occur some other primary failure must also occur such as an inappropriate design or procedure change. Adequate controls are in place such as the Quality Assurance Program, Corrective Action Program, and Self-Assessment Program, to assure the quality of changes that have the potential to affect operation of the plant and they are not affected by this change. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The margin of safety is established by the difference between the failure point and the licensing acceptance limit for a given system, structure or component. Because neither the ISEG nor the ISE function have any impact on the operation of the plant nor any system, structure or component, they cannot reduce any margin of safety from an equipment prospective.

Safety Evaluation: 00-0050
Source Document: USAR Change Request (CR) 00-076

Description of Change:

This Change Request incorporated the First Energy Nuclear Operating Company (FENOC) Nuclear Operating Procedure (NOP) to USAR Section 13.5 as another form of an administrative level document.

Summary:

I. No. The addition of a new class of administrative procedures does not increase the probability or consequences of an accident or malfunction. Each facility utilizing NOPs is still required to verify compliance with their individual plant license basis, design basis, and commitments.

II. No. The addition of a new type of administrative procedure neither increases the probability of an accident nor increases the possibility of a new accident not previously analyzed. The contents of these documents are required to be analyzed on a case by case basis to verify compliance with all rules applying to each facility.

III. No. Since each facility reviews each NOP for compliance with all applicable requirements, the addition of a new type of administrative document by itself cannot change the margin of safety as defined in the Technical Specifications or their bases.

Safety Evaluation: 00-0051
Source Document: Temporary Modification to Install Sealing Mechanisms on the ESW Sluice Gates

Description of Change:

This Temporary Modification is a contingency to provide an alternate means to seal the leak path through the Emergency Service Water (ESW) sluice gates in the event the inflatable seals installed under Simple Modification Request Form (SMRF) 00-5013 fail to perform their isolation function after initial inflation. The Temporary Modification consists of a sealing mechanism that will be installed around the perimeter of the sluice gate on the ESW forebay side of the gate. The sealing mechanism will be installed into the gaps existing around the perimeter of the sluice gate and stationary gate frame. Divers will be required to install the sealing mechanisms. The sluice gate automatic and manual opening feature must be disabled when the temporary sealing mechanism is installed.

Summary:

I. No. The ESW system is an accident mitigating system that provides a reliable source of cooling water during accident conditions and is not an accident initiator. Consequently, the temporary sluice gate sealing mechanisms, which would be part of the ESW system when installed, are likewise not accident initiators. The sluice gate sealing mechanisms provide a barrier between the plant discharge tunnel and the ESW forebay. Performance of this isolation function does not result in any interfaces with any plant Systems, Structures, or Components (SSC) in such a manner as to increase the probability of occurrence of a previously evaluated accident. The isolation function between the discharge tunnel and ESW forebay provided by the sluice gates and sluice gate sealing mechanisms assures that the ESW system can provide adequate cooling to safety related components necessary for accident mitigation. The isolation capability of the sluice gates will not be affected by the sealing mechanisms and the sealing mechanisms can also be relied upon to prevent recirculation of discharge water during all modes of ESW operation. Therefore, the Temporary Modification does not increase the radiological consequences of any previously analyzed accidents since use of the sealing mechanisms maintains the ESW inlet temperature at its allowable value, and thus does not compromise the accident mitigating capability of the ESW system.

Installation of the temporary sluice gate sealing mechanisms does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR. The presence of these devices in the ESW system does not create any interactions or interfaces with any structures, systems, or components in the ESW pump house or in the plant such that the probability of occurrence of a malfunction of any equipment important to safety would be increased. Installation of the sealing mechanisms does not affect any modes of operation of the ESW system, and thus the accident mitigating capability of the ESW system will not be degraded, and therefore the radiological consequences of a malfunction of equipment important to safety cannot be increased. The sluice gates with the sealing mechanisms installed will still provide the required isolation between the discharge tunnel and ESW forebay, thus maintaining the operability of the ESW system.

II. No. The ESW system is an accident mitigating system; it is not an accident initiator. Consequently, the temporary sluice gate sealing mechanisms, which would be part of the ESW system when installed, are likewise not accident initiators, and therefore cannot create the possibility of an accident of a different type than previously evaluated. Installation of the Temporary Modification does not result in any interactions or interfaces with other plant systems, structures, or components that could create the possibility of an accident of a different type. Installation of the temporary sluice gate sealing mechanisms cannot create the possibility of a different type of malfunction of equipment important to safety. The new components do not interact with any systems, structures, or components important to safety in such a manner

as to create a different type of malfunction of equipment important to safety. Installation of the Temporary Modification does not adversely affect the satisfactory operation of the ESW system, and thus no new malfunctions can be created.

III. No. The margin of safety in regard to the cooling water supplied by the ESW system will be retained after implementation of this Temporary Modification since elimination of sluice gate leakage during the time period when the sealing mechanisms are installed will maintain the ESW forebay at or below its design limit of 85°F. This Temporary Modification will ensure that adequate safe shutdown margin is available since the inlet temperature to the heat exchangers cooled by ESW will not exceed its maximum allowable value due to sluice gate leakage.

Safety Evaluation: 00-0052
Source Document: Design Change Package (DCP) 00-6009, Revision 0

Description of Change:

This Design Change Package (DCP) replaces the existing Testable Rupture Disc (TRD), Asset Number 1E22-D0012, Division 3, High Pressure Core Spray (HPCS), Emergency Diesel Generator (EDG), with an actual rupture disc.

Summary:

I. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. American Society of Mechanical Engineers (ASME) code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD. The accidents and transients evaluated in the USAR were reviewed. The Division 3 EDG is not an initiator of any of these events. Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. ASME code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD. Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The same failure modes exist with the rupture disc that exists with the TRD. No new failure effects are created by the installation of a rupture disc. Failure of the new rupture disc is already bounded by the existing accidents evaluated in the USAR. Failure of the new rupture disc is not an accident initiator. The same modification is planned to be made to all three TRDs. Because of this, common mode failure needs to be considered. The Failure Modes Analysis concluded that the possibility of a common mode failure is not increased since the rupture discs are designed and tested in accordance with ASME code requirements. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The TRD is classified Safety Class 3 and is built to ASME Code, Section III, Subsection ND (Class 3). Thus one margin of safety limit is the ASME Code stress allowables. The replacement rupture disc is designed and manufactured in accordance with ASME Code, Section III, Subsection ND (Class 3).

The 7 day inventory values listed in the Technical Specifications are dependent upon the fuel consumption rate of each diesel. Since the new rupture disc does not change the existing exhaust flowpath or flow restrictions, operating conditions for the associated diesel are not changed. The margin of safety associated with diesel fuel inventory is not reduced.

By assuring that the new rupture disc bursts at the specified pressure when required (to relieve excessive exhaust back pressure), there will be no change to the associated diesel's ability to supply sufficient power such that it will remain capable of generating the required electrical output. Thus, the margin of safety associated with output of the EDG is not reduced.

Safety Evaluation: 00-0053
Source Document: Simple Modification Request Forms (SMRF) 99-6005, Revision 0; SMRF 99-6006, Revision 0

Description of Change:

Simple Modification Request Forms (SMRFs) 99-6005 and 99-6006 replace fuel oil pressure regulating valves 1R45-F0562A and B for the Division 1 and 2 Emergency Diesel Generators (EDGs). The fuel oil pressure regulating valves were originally supplied by the EDG vendor, Delaval, as part of the diesel engine system. The existing valves were identified through the corrective action program as not having documentation including material compliance and seismic documentation as they were classified as non-safety. The function of the valves is to regulate fuel oil pressure. By supplying backpressure on the engine's fuel distribution header, sufficient fuel is forced into the engine's injector pumps. The replacement valves have been procured from the original vendor as safety related following internal guidelines for commercial dedication.

Summary:

I. No. The change is consistent with the standards defined for "engine mounted piping and components" provided as part of the EDG package and the installation satisfies the pressure boundary capabilities specified by the original system design requirements. System and plant operation, availability, and response to transients remain the same as described in the USAR. The change does not affect any system interface in a way that would increase the likelihood of an accident. The change does not affect existing accident scenarios, including accident initiators or assumptions. No changes are made to any assumptions or inputs previously made to assess radiological consequences, and no fission product barriers are affected. Thus, the radiological consequences of accidents discussed in the USAR are not increased by this change. The replacement valves are functionally equivalent to the existing valves and will be installed in the existing location. The replacement valves do not adversely affect the seismic qualification of the system. The proposed change does not add any new failure modes or effects. The change does not affect system reliability or performance by imposing transients not analyzed in the design basis or degrade equipment protective features, redundancy or independence. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The replacement fuel oil pressure regulating valves are designed and procured consistent with the currently defined quality and seismic standards for the EDG Fuel Oil System. The proposed change does not initiate any sequence of events resulting in any type of accident. The change does not affect any existing initiators and does not produce a failure mode not previously considered in the USAR. The proposed change is limited to the diesel generator rooms and will not adversely impact any important to safety equipment or diesel generator subsystems. The change does not result in the system operating outside design or test limits. Redundancy of the plant systems is maintained. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. There is no specific margin of safety defined for the EDG Fuel Oil Supply and Transfer System that would be affected by the change. Technical Specifications provide values for the fuel oil storage capacity, regulatory guidance for fuel oil quality and requirements for redundancy of components in the fuel oil transfer system. The proposed change has no impact on fuel oil capacity, quality, or redundancy. The replacement of the fuel oil pressure regulating valves in the EDG Fuel Oil System with safety related/seismically qualified components is

consistent with previously defined standards and has no impact on safety or non-safety structures, systems, or components and has no operational impact. Therefore, the margin of safety is not reduced.

Safety Evaluation: 00-0054
Source Document: USAR Change Request (CR) 00-083

Description of Change:

This USAR Change Request (CR) removes obsolete information from the USAR. In addition, the USAR CR also makes editorial changes. A large number of changes and deletions are made to USAR Section 8.3 to remove references to Unit 2, excessive detail, and to make general editorial/clarification changes.

Summary:

I. No. This activity does not increase the probability of occurrence, or the radiological consequences, of an accident or a malfunction of equipment important to safety as previously evaluated in the USAR. The proposed change does not impose increased testing requirements on systems or equipment important to safety. The proposed changes do not create any new failure modes or failure effects for equipment important to safety; do not degrade the reliability of any plant system, and do not introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of protective systems are not affected by the proposed changes. The proposed changes do not change any radiological consequence to the public or onsite personnel.

II. No. The USAR CR makes a number of changes to text and tables in Sections 8.3 to remove the reference to Unit 2, to clarify various sections, and to remove excessive detail. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident; will not cause or facilitate the occurrence of any known accident initiators or contributors; will not increase the probability of an accident previously thought to be incredible; do not make a previously non-credible event credible; and do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not affect the design and operation of any plant SSCs; do not affect how SSCs react to normal and abnormal transients; do not degrade any equipment; and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR.

III. No. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. The proposed changes will not degrade the capability of SSCs to mitigate the effects of postulated transients and accidents.

Safety Evaluation: 00-0055
Source Document: USAR Change Request (CR) 00-084

Description of Change:

USAR Section 10.4.6.2 will be revised to remove the requirement to measure pH and chloride concentration in the Condensate Cleanup System. The pH and chloride concentrations will only be measured in the Condensate Cleanup System when the reactor water level pH and concentrations are elevated.

USAR Section 5.4.8.2 will be revised to remove the requirement to measure pH on the Reactor Water Cleanup System (RWCU) when the conductivity of the reactor water cleanup effluent is ≤ 1.0 μmho/cm.

USAR Section 9.1.3.2.1 will be revised to remove the requirement to monitor fuel pool demineralized effluent to ensure gross gamma levels are less than 2,000 cpm/ml and replace it with a requirement to perform isotopic analysis.

Summary:

I. No. The proposed change does not authorize any changes to the plant. The Regulatory Guide 1.56 requirements to control water chemistry are not being change by the revision to the USAR. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not eliminate or revise any existing water chemistry criteria that could increase the corrosion rate for piping and components. The Regulatory Guide 1.56 requirements will be maintained through the appropriate plant procedures. Therefore, the proposed USAR change will have no impact on the physical plant. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0056
Source Document: USAR Change Request (CR) 00-090

Description of Change:

The proposed change revises USAR Sections 9.1.3.2.2, 9.1.4.2.10.11, and Table 9.1-1a to adjust the specified discharge cycle, operating cycle time periods and fuel burn-up consistent with a corresponding pending license amendment change to support the implementation of a 24-month operating cycle.

Summary:

I. No. The proposed change does not involve initiators to any accident previously evaluated. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change does not result in an increase in the frequency of the occurrence of a fuel handling accident and does not increase the maximum normal heat load from the spent fuel. The ability to provide long-term cooling to the spent fuel is not adversely impacted. A new decay heat load was calculated for the Spent Fuel Pool and found to be bounded by the existing design decay heat load identified in the USAR. Thus, the probability of an accident or malfunction of equipment important to safety has not increased. The affect on radiological source terms for the spent fuel based on 24-month fuel cycle was evaluated and concluded that the affect was negligible. Thus, the radiological consequences of an accident or malfunction of equipment has not increased.

II. No. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. There are no new failure modes identified as a result of this change. The proposed change does not increase the maximum normal heat load from the spent fuel. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created.

III. No. The proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification. The proposed change does not increase the maximum normal heat load from the spent fuel. Moreover, the change in the cycle frequency has been evaluated to ensure that it does not impact or is a contribution to the decay heat. The methodology used for development of the decay heat curves uses bounding assumptions that render them insensitive to variations in parameters that are secondary contributors to the decay heat. As a result, the proposed change does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0057
Source Document: USAR Change Request (CR) 00-089

Description of Change:

The proposed change removes excessive details from the USAR section 11.5.2.3.3 with respect to the time interval for performing maintenance on the process and effluent radiological monitoring and sampling systems.

Summary:

I. No. The proposed changes remove non-essential information from the USAR. No changes are being made to the plant. These proposed changes do not modify the design bases or the safety analyses. The material remaining in the USAR is sufficient to permit understanding the design bases, the safety analyses, and facility operation. Since there is no adverse change to the facility, the proposed changes can not, in any manner, affect any analysis contained in the USAR. It has been determined that the proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. No equipment protection features are being deleted or modified by the proposed changes. System/equipment redundancy and independence will not be reduced by the proposed changes. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. Based on this, it has been determined the proposed change will not increase the probability of occurrence of an accident or a malfunction of equipment important to safety. It has been determined that the proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. The equipment will still be tested, maintained, and required to be operable and capable of performing accident mitigation functions assumed in the accident analysis. The proposed changes do not change the radiological consequences of an accident or a malfunction of equipment important to safety.

II. No. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible. The proposed changes do not make a previously non-credible event credible. The proposed changes do not create any new failure modes or failure effects for equipment important to safety. The proposed changes do not affect any system important to safety, and do not affect the way any of these systems react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to any malfunction of equipment installed in the plant. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created.

III. No. The associated equipment will continue to be tested and maintained at frequencies that give confidence that the it can perform its assumed safety function when required. Since the proposed changes will not affect the function or operation of the equipment, the margin of safety and availability of the equipment will not be reduced. Therefore, the proposed changes will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0058
Source Document: USAR Change Request (CR) 00-088

Description of Change:

The proposed change adjusts the specified frequency of the turbine rotor inspection consistent with a pending license amendment change to support the implementation of a 24-month operating cycle and a corresponding Missile Probability Analysis.

Summary:

I. No. The proposed change to USAR Section 10.2.3.6.1 has been developed based on maintaining a combined unit missile probability of less than 1.0E-5 rather than on a specified frequency interval (i.e., an 18-month cycle). Each low-pressure turbine wheel is inspected within its operating interval as required by probabilities of the Turbine Missile Probability Analysis described in USAR Section 10.2.3.6.1.1. The associated change does not involve initiators to any accident previously evaluated. Surveillance inspections and their corresponding frequencies are not initiators to any accident previously evaluated. The proposed change does not result in an increase in the frequency of the occurrence of an accident. Moreover, the change to establish inspection frequencies based upon probability analysis continues to ensure acceptable levels of equipment reliability as specified in the Turbine Missile Probability Analysis. Consequently, equipment that could initiate an accident previously evaluated will continue to operate as expected. As a result, the proposed change does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety. Additionally, the radiological consequences of an accident or a malfunction of equipment important to safety has not been increased.

II. No. The proposed change revises the specified frequency of the turbine rotor inspection consistent with a corresponding proposed license amendment change to support the implementation of a 24-month operating cycle. The proposed change to USAR Section 10.2.3.6.1 has been developed based on maintaining a combined unit missile probability of less than 1.0E-5 rather than on a specified frequency interval (i.e., an 18-month cycle). Each low-pressure turbine wheel is inspected within its operating interval as required by probabilities of the Turbine Missile Probability Analysis described in USAR Section 10.2.3.6.1.1. The surveillance frequencies are based upon a combined unit missile probability of less than 1.0E-5 consistent with the turbine missile generation analysis described in USAR Section 10.2.3.6.1.1. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created.

III. No. The proposed change revises the specified frequency of the turbine rotor inspection consistent with a corresponding proposed license amendment change to support the implementation of a 24-month operating cycle. The surveillance frequencies are based upon a combined unit missile probability of less than 1.0E-5 consistent with the turbine missile generation analysis described in USAR Section 10.2.3.6.1.1. The change in the frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at a frequency that gives confidence that the equipment can perform its assumed safety function when required. As a result, the proposed change does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0059
Source Document: USAR Change Request (CR) 00-091;
PDB-R0001, Revision 0, PIC24

Description of Change:

The surveillance test interval for the area radiation monitoring calibration test, as described in USAR Section 12.3.4.1.2, has been revised from "on an 18 month basis" to "in accordance with plant procedures". The proposed change involves extending various equipment channel calibration test intervals, specifically, the Control Room Area Radiation Monitor Channel Calibration test interval, per Operational Requirements Manual (ORM) Section 6.2.6.3.

Summary:

I. No. The relaxed frequencies have been established based upon achieving acceptable levels of equipment reliability for the associated function. A qualitative review and surveillance test history review was performed for the associated function utilizing guidance contained in NRC Generic Letter 91-04 to further justify the increase in frequency interval. The associated change does not involve initiators to any accident previously evaluated. Maintenance and surveillance frequency intervals are not initiators to any accident previously evaluated. The equipment being tested is still required to be operable and capable of performing any accident mitigation functions assumed in the accident analysis. The proposed change does not degrade the performance of a System, Structure or Component (SSC) below the design basis that was assumed in the accident analysis. The proposed change does not increase the challenges to system, structures, or components assumed to function in the accident analysis such that system performance is degraded below the design basis without compensatory measures. Consequently, equipment that could initiate an accident previously evaluated will continue to operate as expected. As a result, the proposed change does not increase the probability of occurrence or radiological consequences of an accident or a malfunction of equipment important to safety.

II. No. The proposed change revises the specified intervals consistent with a corresponding pending license amendment change to support the implementation of a 24-month operating cycle. The change in maintenance and surveillance frequency does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created.

III. No. The change in maintenance and surveillance frequency does not reduce the margin of safety as defined in the bases for any Technical Specification. The change in maintenance and surveillance frequency has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at frequencies that gives confidence that the equipment can perform its assumed safety function when required. As a result, the proposed change does not reduce the margin of safety as defined in the basis for any Technical Specification.

Safety Evaluation: 00-0060
Source Document: Design Change Package (DCP) 00-6007, Revision 0

Description of Change:

This Design Change Package (DCP) replaces the existing Testable Rupture Disc (TRD), Asset Number 1R48-D0014A, Division 1, Emergency Diesel Generator (EDG), with an actual rupture disc.

Summary:

I. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. American Society of Mechanical Engineers (ASME) code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD. Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. ASME code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD. Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The same failure modes exist with the rupture disc that exists with the TRD. No new failure effects are created by the installation of a rupture disc. Failure of the new rupture disc is already bounded by the existing accidents evaluated in the USAR. Failure of the new rupture disc is not an accident initiator. The same modification is being made to all three TRDs. Because of this, common mode failure needs to be considered. The Failure Modes Analysis concluded that the possibility of a common mode failure is not increased since the rupture discs are designed and tested in accordance with ASME code requirements. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The TRD is classified Safety Class 3 and is built to ASME Code, Section III, Subsection ND (Class 3). Thus one margin of safety limit is the ASME Code stress allowables. The replacement rupture disc is designed and manufactured in accordance with ASME Code, Section III, Subsection ND (Class 3).

The 7 day inventory values listed in the Technical Specifications are dependent upon the fuel consumption rate of each diesel. Since the new rupture disc does not change the existing exhaust flowpath or flow restrictions, operating conditions for the associated diesel are not changed. The margin of safety associated with diesel fuel inventory is not reduced.

By assuring that the new rupture disc bursts at the specified pressure when required (to relieve excessive exhaust back pressure), there will be no change to the associated diesel's ability to supply sufficient power such that it will remain capable of generating the required electrical output. Thus, the margin of safety associated with output of the EDG is not reduced.

Safety Evaluation: 00-0061
Source Document: Design Change Package (DCP) 00-6008, Revision 0

Description of Change:

This Design Change Package (DCP) replaces the existing Testable Rupture Disc (TRD), Asset Number 1R48-D0014B, Division 2, Emergency Diesel Generator (EDG), with an actual rupture disc.

Summary:

I. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. American Society of Mechanical Engineers (ASME) code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The use of an actual rupture disc eliminates the moving parts and the possibility of increased frictional resistance to disc opening that are inherent to the TRDs. ASME code burst testing of the rupture disc validates the specified burst pressure. Thus, the rupture disc is more likely to open at the specified exhaust pressure than the existing TRD. Failure of a rupture disc is no different from a failure of the existing TRD and the rupture disc is more reliable than the TRD due to ASME code burst testing and the elimination of moving parts and associated frictional resistance. The same failure modes exist with the rupture disc that exists with the TRD. No new failure effects are created by the installation of a rupture disc. Failure of the new rupture disc is already bounded by the existing accidents evaluated in the USAR. Failure of the new rupture disc is not an accident initiator. The same modification is being made to all three TRDs. Because of this, common mode failure needs to be considered. The Failure Modes Analysis concluded that the possibility of a common mode failure is not increased since the rupture discs are designed and tested in accordance with ASME code requirements. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The TRD is classified Safety Class 3 and is built to ASME Code, Section III, Subsection ND (Class 3). Thus one margin of safety limit is the ASME Code stress allowables. The replacement rupture disc is designed and manufactured in accordance with ASME Code, Section III, Subsection ND (Class 3).

The 7 day inventory values listed in the Technical Specifications are dependent upon the fuel consumption rate of each diesel. Since the new rupture disc does not change the existing exhaust flowpath or flow restrictions, operating conditions for the associated diesel are not changed. The margin of safety associated with diesel fuel inventory is not reduced.

By assuring that the new rupture disc bursts at the specified pressure when required (to relieve excessive exhaust back pressure), there will be no change to the associated diesel's ability to supply sufficient power such that it will remain capable of generating the required electrical output. Thus, the margin of safety associated with output of the EDG is not reduced.

Safety Evaluation: 00-0063
Source Document: USAR Change Request (CR) 00-092

Description of Change:

The change involves, in part, testing of non-safety related filtration and adsorption units, in accordance with Regulatory Guide 1.140, that cannot be performed on line. Specifically, the testing interval for ventilation exhaust system air filtration and adsorption units as provided in USAR Table 12.3-3 is being revised from a verbatim compliance of the Regulatory Guide 1.140 Positions C.5.c, C.5.d, C.6.a(3), and C.6.b. Regulatory Guide 1.140 recommends testing at intervals of approximately 18 months and Table 12.3-3 will be revised to "testing frequency will meet the intent of the provision but may be based upon refueling outage intervals for Systems M14 and M38." M14 is the Containment Vessel and Drywell Purge System, M38 is the Auxiliary Building Ventilation System.

Summary:

I. No. The proposed change relaxes the Regulatory Guide (RG) 1.140 testing interval. The relaxed testing interval has been established based upon achieving acceptable levels of equipment reliability for the associated functions. A qualitative review and surveillance test history were performed for the associated functions using the guidance contained in NRC Generic Letter 91-04 to further justify the increase in test interval. The associated change does not involve initiators to any accident previously evaluated. The proposed change does not degrade the performance of a system, structure or component below the design basis that was assumed in the accident analysis. The proposed change does not increase the challenges to systems, structures, or components assumed to function in the accident analysis, such that system performance is degraded below the design basis without compensatory measures. As a result, the proposed change does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety previously evaluated in the USAR.

The equipment being tested is non-safety related filtration and adsorption equipment and is not required to perform any accident mitigation functions assumed in the accident analysis. The associated systems are utilized for minimization of normal and off-normal dose and associated releases. The filter systems are not relied upon for accident mitigation. In addition, the proposed change does not increase the consequences of an accident or malfunction of equipment important to safety.

II. No. The relaxed testing interval has been established based upon achieving acceptable levels of equipment reliability for the associated functions to further justify the increase in frequency interval. The change in testing intervals does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal or off-normal plant operation. As such, the possibility of an accident of a different type or the possibility of a different type of malfunction of equipment important to safety has not been created.

III. No. The relaxed testing interval has been established based upon achieving acceptable levels of equipment reliability for the associated functions to further justify the increase in frequency interval. The change in testing intervals does not reduce the margin of safety as defined in the bases for any Technical Specification. The change in testing intervals has been evaluated to ensure that it provides an acceptable level of equipment reliability. Thus, appropriate equipment continues to be tested at frequencies that gives confidence that the non-safety related equipment can perform its assumed function. Review of the NRC Safety Evaluation Report for Perry concluded that no reliance or commitment was made with regards to periodic testing intervals for RG 1.140. As a result, the proposed change does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0064
Source Document: USAR Change Request (CR) 00-094

Description of Change:

This USAR Change Request (CR) removes obsolete information from the USAR. In addition, the USAR Change Request also makes editorial changes.

Summary:

I. No. The USAR CR makes a number of minor changes to text and tables with regards to the reactor vessel and reactor design sections of the USAR. The effect of the proposed USAR deletions/editorial changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The proposed change does not impose increased testing requirements on system or equipment important to safety. The effect of the proposed changes does not create any new failure modes or failure effects for equipment important to safety; does not degrade the reliability of any plant system, nor does it introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of protective systems are not affected by the proposed changes. The affect of the proposed changes does not change any radiological consequence to the public or onsite personnel.

II. No. The USAR CR makes a number of minor changes to text and tables with regards to the reactor vessel and reactor design sections of the USAR. The effect of the proposed changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible; does not make a previously non-credible event credible, and does not create any new failure modes or failure effects for equipment important to safety. The effect of the proposed changes doesn't affect the design and operation of any plant Structures, Systems and Components (SSCs), and does not affect how SSCs react to normal and abnormal transients, does not degrade any equipment; and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR.

III. No. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. The proposed changes will not degrade the capability of SSCs to mitigate the effects of postulated transients and accidents. This activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0066
Source Document: Simple Modification Request Form (SMRF) 00-5014, Revision 0

Description of Change:

Simple Modification Request Form (SMRF) 00-5014 will replace the Circulating Water (CW) pump impellers, wear rings, impeller keys and nuts. The new CW pump impellers have been designed to improve pump performance by increasing both pressure and flow and decreasing the required Net Positive Suction Head (NPSH). As a result of the increased CW pump performance, only 2 CW pumps (1N71-C0001A/B/C) will be required to provide the necessary condenser cooling during the winter months.

Condition Report (CR) 00-2420 was written to document non-conservative assumptions/inputs in Calculation 0-DC-235-413-006, Revision 0 (renumbered as N71-30, Revision 0) which determines the service water makeup volume to the cooling tower basin. Based on the non-conservative assumptions/inputs used, the total flooding volume identified in USAR Section 2.4.13.5.2 is incorrect and will be revised.

Summary:

I. No. The proposed modification will replace the circulating water pump impellers. The failure of one or more circulating water pumps and the resultant loss of condenser vacuum accident scenario is already discussed in the USAR. The circulating water expansion joint failure accident scenarios that are discussed in the USAR are not adversely affected as documented in revisions to calculations N71-026 through N71-033. In addition, the circulating water yard piping failure accident is not adversely affected by the proposed modification. Since the proposed modification has been evaluated against all existing accident scenarios and found to be acceptable, the proposed change will not increase the probability of or radiological consequences associated with a previously identified accident or malfunction of equipment important to safety. The function and performance of all Engineered Safety Function (ESF) components is unaffected. The proposed modification does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The replacement circulating water pump impellers and associated hardware will be functionally equivalent to the existing impellers and hardware. The proposed change will not impact the ability of the Circulating Water System to perform its intended design function. The Loss of Condenser Vacuum event (USAR Section 15.2.5) has as an initiator the loss of one or more circulating water pumps. The operation of the system in either two or three pump operation does not invalidate the existing accident analysis or create a new accident. There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF systems, structure or components are affected by the proposed change. Since no new system interactions are being created by the proposed modification, no new equipment malfunctions are postulated. The proposed modification does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed modification will not create an accident of a different type or malfunction of equipment important to safety.

III. No. The change in plant building flooding water levels by the proposed modification does not decrease any of the margins of safety, but rather only changes the operating margin. There are no Technical Specifications associated with the proposed change. Therefore, the proposed modification can have no impact on the margins of safety associated with SSCs and will not reduce the margin of safety implied or specifically stated by any licensing documents, including the USAR, Safety Evaluation Report, Operational Requirements Manual (ORM), Core Operating Limits Report (COLR), Offsite Dose Calculation Manual (ODCM), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0067
Source Document: USAR Change Request (CR) 00-099

Description of Change:
This USAR Change Request (CR) removes obsolete information from the USAR. In addition, the USAR Change Request also makes editorial changes. The changes/deletions are made to USAR Section 9.4.

Summary:

I. No. The USAR CR makes a number of changes to text and tables in USAR Section 9.4 to remove the references to Unit 2 and to clarify the HVAC sizing/design approach for the original common HVAC systems. The affect of the proposed USAR changes does not impact any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The effect of the proposed changes does not change any radiological consequence to the public or onsite personnel.

II. No. This activity makes a number of changes to text and tables in Sections 9.4 to remove the reference to Unit 2 and to clarify the HVAC sizing/design approach for the original common HVAC systems. The effect of the proposed changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident previously thought to be incredible; does not make a previously non-credible event credible, and does not create any new failure modes or failure effects for equipment important to safety. The effect of the proposed changes doesn't affect the design and operation of any plant Structures, Systems and Components (SSCs), and does not affect how SSCs react to normal and abnormal transients, does not degrade any equipment; and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR.

III. No. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. The proposed changes will not degrade the capability of SSCs to mitigate the effects of postulated transients and accidents. This activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0069
Source Document: Simple Modification Request Form (SMRF) 00-5017, Revision 0

Description of Change:

The purpose of the modification is to remove four Plant Radiation Monitoring System (D17) portable airborne radiation monitors from Unit 1.. Additionally, minor administrative changes will be made to the USAR to include the removal of references to Unit 2 portable D17 monitors which are not required for Perry Unit 1 operation, and clarification of Derived Air Concentration (DAC) conversion factors.

Summary:

I. No. The D17 portable monitors to be removed are not associated with or included in the evaluation of any USAR accidents. The portable D17 airborne radiation monitors are not equipment important to safety and do not interface with or support equipment important to safety. The portable D17 airborne radiation monitors do not interact with any plant Structure, System or Component (SSC) except for drawing electric power from 480V alternating current (AC) wall receptacles. They do not provide remote indication of any kind or receive input from any other system. The portable D17 monitors to be removed do not provide benefits over and above other Engineered Safety Features (ESF) and normal administrative radiological protection methods. The airborne radiological monitoring system associated with the plant ventilation systems provide monitoring that is adequate to ensure that any major change in plant airborne radioactivity will be detected at levels low enough that the dose to workers will be far below the levels required by 10CFR20. A significant release of airborne radioactivity due to a malfunction of equipment in the plant will be monitored by the permanent airborne radiation monitors in containment or the drywell or exhaust ventilation system monitors and cause alarm indications to control room operators. Therefore, the proposed activity will not increase the probability or radiological consequences of an accident previously evaluated in the USAR and will not increase the probability of occurrence or the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR

II. No. The portable D17 airborne monitors do not interact with any SSC except to draw electric power from 480VAC wall sockets. Their removal will not cause a configuration change and no physical change to the plant will be required. Removal of the radiation monitors does not create any new failure modes or effects. Therefore, the proposed activity cannot create the possibility of an accident of a different type or a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The portable D17 airborne radiation monitors are not safety related equipment. They are not described, referenced or required by Technical Specifications, the Operational Requirements Manual, the Offsite Dose Calculation Manual or any other licensing documents other than the USAR. The portable D17 airborne radiation monitors that are described in USAR Chapter 12 do not provide beneficial monitoring services over and above the other engineered safety features of the plant and normal administrative radiological protection methods. Their removal does not negatively affect the margin of safety provided to ensure that the dose to workers is maintained ALARA and far below the levels specified by 10 CFR 20. Therefore, removal of the portable D17 airborne radiation monitors does not reduce the margin of safety as defined in the basis for any technical specification.

Safety Evaluation: 00-0070
Source Document: USAR Change Request (CR) 00-101

Description of Change:

This USAR Change Request evaluates a site organization change. Specifically, the reactor engineering and fuels group reporting point is being changed from the site Operations Section to the Corporate Fuels Organization.

Summary:

I. No. This USAR change is an organizational change. No functions have been eliminated, only the reporting point has changed. The change does not alter the design, function, or operation of the plant. USAR accident analysis remains unchanged. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This USAR change is an organizational change. No functions have been eliminated, only the reporting point has changed. The change does not alter the design, function, or operation of the plant. No new system interactions or new failure modes have been created. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This USAR change is an organizational change. No functions have been eliminated, only the reporting point has changed. The change does not alter the design, function, or operation of the plant. The changes do not affect the Technical Specifications, the Technical Specification Bases, or the Operational Requirements Manual. Therefore, no margins of safety have been reduced.

Safety Evaluation: 00-0071
Source Document: Simple Modification Request Form (SMRF) 00-5025, Revision 0

Description of Change:

The proposed change is to modify openings such as doors, seismic gaps and miscellaneous penetrations to eliminate Control Complex (CC) Building tornado depressurization vent paths. The non-safety heating and cooling system for the CC elevator equipment room is also being modified. This modification, along with modifications discussed in Safety Evaluation 00-0082 (SMRF 00-5026) will eliminate interim compensatory operator actions and ensure components important to safety within the Control Complex are not adversely affected by a Design Basis Tornado (DBT).

Summary:

I. No. The proposed door modifications, seismic gap seals, new penetration seals and new pressure barriers are all passive components that do not interface with any safety related plant operating system, structure, equipment or component. These modifications have been designed to meet the DBT requirements, satisfy control room envelope leak tightness requirements, eliminate any seismic interaction concerns and satisfy fire-rating requirements. The new heating and cooling system for the CC elevator equipment room is non-safety related and does not interface with any safety related operating system, component or equipment. Therefore, the proposed modifications cannot initiate, and are not associated with, any accidents or malfunctions that would challenge safe plant operation. They do not affect the initiation of any of the accidents or malfunctions of equipment important to safety already described in the USAR, nor do they associate with any accident or malfunction that could result in radiation dose.

II. No. The design change does not add new equipment or components that would reduce or impact the capability of any safety system to perform its intended design function. The proposed door modifications, seismic gap seals, new penetration seals and new pressure barriers are all passive components that do not interface with or adversely affect any safety related plant operating system, structure, equipment or component. The CC-511 access control changes do not constitute a new failure mode or compromise the integrity of any equipment important to safety. Therefore, this design change does not create the possibility of a different type of malfunction of equipment important to safety and does not create the possibility of a different type of accident than any previously evaluated in the USAR.

III. No. The design change does not add new equipment or components that would reduce or impact the capability of any safety system to perform its intended design function. The integrity of the control room boundary will not be compromised. The design changes will not impact the Operating License, Technical Specifications, Technical Specifications Bases, Surveillance Requirements, or the Operational Requirements Manual (ORM). The modifications have been designed and installed to the appropriate safety classification in accordance with USAR design and licensing bases. The modifications do not create a new accident, or adversely effect any important to safety equipment. Therefore, this modification does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0072
Source Document: Safe Shutdown Capabilities Report Change Request 00-097

Description of Change:

This Safe Shutdown Capabilities Report (SSCR) change makes an editorial revision to the SSCR. The change revises or corrects various editorial issues within the SSCR, e.g., correcting typos, revising page numbers, correcting punctuation. These change do not affect any technical information in the SSCR.

Summary:

I. No. This SSCR change is an editorial revision. The change does not alter the design, function, or operation of the plant. USAR accident analysis remains unchanged. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety have not changed.

II. No. This SSCR change is an editorial revision. The change does not alter the design, function, or operation of the plant. No new system interactions or new failure modes have been created. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This SSCR change is an editorial revision. The change does not alter the design, function, or operation of the plant. The changes do not affect the Technical Specifications, the Technical Specification Bases, or the Operational Requirements Manual. Therefore, no margins of safety have been reduced.

Safety Evaluation: 00-0074
Source Document: USAR Change Request (CR) 00-105

Description of Change:

This Change Request revises the USAR text describing the processing of turbine lube oil and laydown area sump drainage for consistency with plant procedures, which provide additional control over the release of radioactivity compared to the originally licensed processing as industrial waste.

The proposed revisions will not change installed plant Structures, Systems, Components (SSC) or how they are operated, but only the description of their operation in the USAR consistent with the original design basis. The revised text is consistent with the NRC Standard Review Plan (SRP) and does not materially alter the basis for prior NRC review of applicable sections of the USAR.

Summary:

I. No. The changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. Additionally, the changes do not play a direct role in mitigating the radiological consequences of an accident described in the USAR. The changes will not degrade SSC reliability. No equipment protection features are being deleted or modified by the changes. System/equipment redundancy and independence will not be reduced by the changes. The changes do not create any new failure modes or failure effects for equipment important to safety. The changes do not alter, degrade or prevent actions described or assumed in any analysis discussed in the USAR. The changes do not directly or indirectly affect mitigation of the radiological consequences of a malfunction of equipment important to safety. The changes do not change any radiological consequences to the public or onsite personnel. Therefore, the proposed changes will not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR.

II. No. The proposed changes do not implement any physical changes to the plant, or change the safety related function of any SCC. The proposed change will not initiate or facilitate the occurrence of any known accident initiators or contributors. The proposed changes do not impact any system interface in a w away that would increase the likelihood of an accident or transient, or introduce a new accident. The proposed changes do not create any new potential failure modes, interactions or operation sequences that could result in degradation or failure of components important to safety. The proposed changes do not alter the failure effects associated with the malfunction of equipment important to safety. Therefore, the proposed changes will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The Technical Specifications, Operational Requirements Manual, and the Safety Evaluation Report/Supplements to the SER are not affected by the changes. The changes are not related to Technical Specification Bases. The changes will not affect the design basis of any SSC. The changes will not affect the ability of any SSC to perform as designed. Since the changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 00-0075
Source Document: Equivalency Change Package (ECP) 99-8055, 8056, both Revision 2; Setpoint Change Requests (SCR) 1-99-1108 through 1113, all Revision 0

Description of Change:

The purpose of this modification is to replace the obsolete Ultra-Sonics Level detectors located on Division 1, 2, and 3 Fuel Oil Storage Tanks (FOST) 1R45-A002A(B) and 1R45-A004. The existing transmitters 1R44-N0188A/B and 1R45-N008 monitor the fuel oil inventories and alarm at the 7 day and 24 hour inventory levels. The transmitter also provides a 4-20 milli-amp signal to the control room for indication and alarming purposes. Revision 2 of the ECPs adds a spool piece to the FOST level instrument as a result of conditions noted during the implementation of ECP 99-8057, Division 3 FOST Level Instrumentation. During the installation of ECP 99-8057, it was noted that the tornado barrier displayed signs of flooding above the existing level of the instrument. The additional spool piece elevates the instrument above the indicated flood level.

Summary:

I. No. The level measurement and instrumentation are of the same design and type as the originally approved capacitance level measurement system licensed for plant startup. The installation, including the spool piece, conforms to the same design codes and standards as the originally installed instrumentation. The Standby Diesel Generator Fuel Oil (R45) and Standby Diesel Generator Start Air (R44) instrumentation being changed by this modification is not an accident initiator. Neither the loss of Alternating Current (AC) Power or the probability of Station Blackout increases due to the change in this level instrumentation. The changes do not reflect any changes in the R44 or R45 System that will cause it to operate outside of applicable design or testing limits. The change does not result in any changes to system interfaces. The failure analysis indicates that this design modification will not increase the probability of a diesel system failure or transient. This design modification does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously described in the USAR.

II. No. The modification to the R45 system is performed using the same codes and standards as the original design. This modification will not change the function of the R45 System, and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator, or part of any initiating event, for any of the accidents/transients evaluated in the USAR. This change also does not cause any event evaluated in the USAR to be incredible to become credible. This design change does not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. Performance of the R45 System has not been affected by this modification since the system will continue to provide FOST level monitoring using the existing methods of indication and alarming. Consequently, there is no negative affect on the performance or reliability of the R45 system. The margin of safety as described in the Technical Specifications is: "The 31 day Frequency (verification of fuel oil inventory) is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms (7 day and 24 hour) are provided and unit operators would be aware of any large uses of fuel oil during this period." Operator action to dipstick the FOST is part of the original license basis and not a new

requirement. This proposed change does not eliminate or alter these alarms or indication, or compromise the ability to independently verify FOST level using the dipstick. Therefore, these changes do not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0076
Source Document: Simple Modification Request Form (SMRF) 00-5030, Revision 0

Description of Change:

The purpose of this modification is to install an extension pipe spool piece between the instrument connection located on the diesel generator fuel oil storage tank 1R45-A004 and its associated fuel oil level transmitter 1R45-N008. The design intent of this modification is to elevate the level transmitter above water indications on the interior of the missile barrier walls created by water intrusion into the protective missile barrier above the tank. Condition Report (CR) 00-1527 identified that the water level internal to the missile barrier has, in the past, been above the current level of the transmitter. The level transmitter is a watertight unit but is not classified as submergible. A conservative approach is to elevate the level transmitter above those water indications.

Summary:

I. No. The installation design of the level measurement instrumentation is of the same reliability and function as originally approved mounting as licensed for plant startup. The installation conforms to the same design codes and standards as the originally installed instrumentation. The accuracy and dependability of the instrumentation remains equal or better. The R45 instrumentation mounting being changed by this modification is not an accident initiator. Neither the loss of Alternating Current (AC) Power or probability of Station Blackout increases due to the change in this level instrumentation elevation. The changes do not reflect any changes in the R45 System that will cause it to operate outside of applicable design or testing limits. The change does not result in any changes to system interfaces. The failure analysis as described above indicates that this design modification will not increase the probability of a diesel system failure or transient. This design modification does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR.

II. No. This design change increases the reliability of the R45 system by adding reliability to the Fuel Oil Storage Tank (FOST) level monitoring instrumentation. The failure effects of the new design are the same as the current design. The change incorporates materials and equipment of the same design standards and quality of the existing equipment. The elevation of the existing level detection instrumentation provides no additional probability of failure than the original design as described in the Failure modes and Effects section in the safety analysis. Operator sounding of the tanks by way of dip-sticking method provides the verifiable means to determine fuel oil inventory. Failure of the level instrumentation to function does not preclude determination of FOST inventory. No additional system loads, increased frequency of operation of any equipment or reduction of redundancy to the plant have been introduced by this design modification. This instrumentation is not credited for any post accident system functions. The changes made to the R45 instrumentation mounting methods do not affect any equipment important to safety as defined in the USAR, either directly or indirectly. The failure modes and effects of this instrumentation will not impact any equipment important to safety. This design change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the USAR.

III. No. This change has no affect on the Technical Specifications (3.8.3.1), the Operational Requirements Manual, NRC Safety Evaluation Report, and Standard Review Plan. Performance of the R45 System has not been affected by this modification since the system will continue to provide FOST level monitoring using the existing methods of indication and alarming. Consequently, there is no negative effect or change to the performance or reliability of the R45 system. The margin of safety as described in the Technical Specifications is: "The

31 day Frequency (verification of fuel oil inventory) is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms (7 day and 24 hour) are provided and unit operators would be aware of any large uses of fuel oil during this period." Operator action to dipstick the FOST is part of the original license basis and not a new requirement. This proposed change does not eliminate or alter these alarms or indication, or compromise the ability to independently verify FOST level using the dipstick. Therefore, these changes do not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0077
Source Document: Engineering Change Package s(ECP) 00-8014 and ECP 00-8015, both Revision 0

Description of Change:

An entrance door (ECP 00-8014), two wire gates and two wire partitions (ECP 00-8015) are being added to the plant. The entrance door provides a barrier against weather and rodent intrusion into the Condensate Storage Tank (CST) Level Instrument Room via the missile shield. The room is a Restricted Radiation Area (RRA) Area. The gates and partitions prevent casual or accidental access into high radiation areas located in the Radwaste Building 623'-6" elevation, in the abandoned radwaste evaporator rooms. These rooms are currently barricaded with scaffold gates. Both changes require a revision of drawing E-013-0005.

Summary:

I. No. The wire gates, partitions and door subject to this evaluation are not initiators of accidents or transients evaluated in the USAR. The function of the gates and partitions are to prevent casual or accidental entrance into a High Radiation Area. The door functions as a barrier preventing intrusion of weather and rodents and as an egress control into an external RRA. There are no evaluated accidents in the USAR requiring activities involving operator interactions with the proposed gates/partitions and door. Therefore, accident and transient probability will not be affected by control of access through the gates and door. The gates, partitions, and door and their respective installations will not raise the probability of occurrence of an accident previously evaluation in the USAR. The public or an increase in onsite does that would impede actions necessary to mitigate the consequences of a loss-of-coolant accident or fuel handling accident. The gates, partitions, and door are non-safety related and do not interact with equipment important to safety. The gates, partitions and door are installed by fastening them to building concrete. The gates, partitions, and door are constructed and installed in accordance with Installation Standard Specification (ISS) SP-2156. Although the gates, partitions, and door are adequately constructed and installed to prevent impact to equipment during seismic or accident events, it is not necessary as the location of these items do not cause them to interface with other active plant components.

II. No. There are no accident initiators or failures created by the installation of the subject gates, partitions, or door. The identified installation locations do not cause any interactions of the subject items with other plant components. There are no new important to safety equipment failure modes or effects introduced as a result of the installation of the gates, partitions, and door.

III. No. Technical Specification 5.7 addresses Administrative Controls in High Radiation Areas. The addition of the subject gates complies with the requirements of this section. The replacement of existing scaffold gates improves the capability of compliance with the technical specifications. No functional changes were made to any plant or equipment systems required for safety as a result of the installation of the gates, partitions, and door. Therefore, the modifications to install the gates, partitions, and door will not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0078
Source Document: Drawing Change Notice 5907, Revision 0

Description of Change:

The purpose of this Drawing Change Notice (DCN) is to show the correct installation configuration of the 1N21-N0410 pressure switch (Condensate System).

Summary:

I. No. The proposed drawing change of the N21 pressure switch is not an accident initiator. Neither the probability of loss of the N21 system, nor the loss of pressure boundaries of the High Pressure (HP) Condenser, increases due to the location change of this pressure instrumentation. The change does not reflect any changes in the N21 System that will cause it to operate outside of applicable design or testing limits. The change does not result in any changes to system interfaces. The failure analysis as described above indicates that this drawing change will not increase the probability of a N21 system failure or transient. The change does not have any adverse affects to onsite doses. Therefore, this drawing change does not increase the probability of occurrence of an accident or a malfunction of equipment important to safety, or increase the radiological consequences previously evaluated in the USAR

II. No. This instrument relocation on the drawing will not change the function of the N21 System and there will not be any impact to systems required for safe shutdown or safe plant operation. These changes do not result in any increase in the probability of the failure of any equipment that is considered an initiator or part of any initiating event for any of the accidents/transients evaluated in the USAR. This change also does not cause any event evaluated in the USAR to be incredible to become credible or any event that was previously bounded to become bounding. This design change does not create the possibility of an accident or malfunction of a different type than any previously evaluated in the USAR.

III. No. This change has no effect on the Technical Specifications, the Operational Requirements Manual, NRC Safety Evaluation Report, and Standard Review Plan. Performance of the N21 System has not been affected by this drawing modification since the system will continue to provide the HP Condenser pressure monitoring using the existing method of pressure switch monitoring. Consequently, there is no negative affect on, change to the performance or reliability of, the N21 system. There is no margin of safety described in the Technical Specifications. This proposed change does not eliminate or alter the pressure monitoring, or compromise the pressure switch's ability to perform its intended design function. Therefore, these changes do not reduce the margin of safety as defined in the bases for the Technical Specifications.

Safety Evaluation: 00-0079
Source Document: USAR Change Request (CR) 00-111

Description of Change:

This USAR Change Request will remove the specific reference to the Safe Shutdown Capability Report from USAR Appendix 9A, Section 9A.2. The statement is made in the USAR that the details of the safe shutdown analysis/evaluation are contained in the Safe Shutdown Capability Report and the results are summarized in the USAR. The wording will be changed to eliminate the specific title of the Safe Shutdown Capability Report. This will support the revision of Plant Administrative Procedure (PAP) - 0520, "Changes to the Updated Safety Analysis Report and Other Licensing Documents", to eliminate the control of the Safe Shutdown Capability Report under this procedure.

Summary:

I. No. This change only impacts the description of the safe shutdown analysis in USAR Appendix 9A. It does not affect the Safe Shutdown Capability Report or the basis for any Appendix R exemption request. Safe shutdown equipment and equipment important to safety is protected in the same manner as previously evaluated and the possibility of a fire or the potential for adverse impact of the fire or fire fighting activities on is unchanged. In addition, the expected fire hazards and the associated worst-case fire scenarios are unchanged and will continue to be provided with an adequate level of protection. Likewise, the change the description or control of the Safe Shutdown Capability Report will not change the function of any equipment important to safety or the consequences of the failure of this equipment. Therefore, the change will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The proposed change only impacts the administrative control of the safe shutdown analysis and is not functionally related to any known failure mechanism for plant features important to safety. The fire hazards and the associated worst-case fire scenario as well as the potential impact of a fire on safe shutdown equipment will remain as previously evaluated. In addition failure modes for the fire protection system or consequence of malfunction of the fire protection system on equipment important to safety is unchanged. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The change to the Fire Protection Program is permissible under the Perry Nuclear Power Plant (PNPP) Fire Protection License Condition to the extent that the change does not adversely impact the credited post-fire safe-shutdown capability. The margin of safety, as it applies to this modification, is based on maintaining one train of equipment and circuits required to achieve and maintain safe shutdown free of fire damage. The separation and protection of redundant trains credited to support the post-fire shutdown capability is unaffected by the change. Therefore, equipment and circuits required to achieve and maintain safe shutdown will remain free of fire damage and the margin of safety established by the Fire Protection Program as reviewed and approved by the NRC is maintained.

Safety Evaluation: 00-0080
Source Document: USAR Change Request (CR) 00-112

Description of Change:

The objective of the USAR Change Request (CR) evaluation is to change the discussion and tabulation of liquid radwaste source terms and design information, to clearly describe the as-built plant and to clearly describe the conformance with the licensing basis. The types of changes include correcting source term equations presented in the USAR, correcting the carry-over factor to the secondary system, revising component nomenclature, inserting the Perry Nuclear Power Plant (PNPP) specific source terms, revising the stated pump shutoff head for the waste collector pumps, etc.

Summary:

I. No. This activity does not increase the probability of occurrence, or the radiological consequences, of an accident or a malfunction of equipment important to safety as previously evaluated in the USAR. The USAR CR makes a number of changes to text and tables in Chapter 11. The affect of the proposed USAR change does not impact any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The proposed change does not impose increased testing requirements on systems or equipment important to safety. The affect of the proposed changes does not create any new failure modes or failure effects for equipment important to safety; does not degrade the reliability of any plant system; and does not introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of protective systems are not affected by the proposed changes. The effect of the proposed changes does not change any radiological consequence to the public or onsite personnel.

II. No. The USAR CR makes a number of changes to text and tables in Chapter 11. The affect of the proposed changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, will not increase the probability of an accident previously thought to be incredible, does not make a previously non-credible event credible, and does not create any new failure modes or failure effects for equipment important to safety. The affect of the proposed changes does not affect the design and operation of any SSCs, does not affect how SSCs react to normal and abnormal transients, does not degrade any equipment; and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR. This activity does not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than any previously evaluated in the USAR.

III. No. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. This activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0081
Source Document: USAR Change Request (CR) 00-113

Description of Change:

This USAR Change Request (CR) removes obsolete information from the USAR. In addition, the USAR Change Request also makes editorial changes. The types of changes include removing the implied description of more than one unit on the site, such as noting the presence of only three diesels as opposed to six diesels, etc.

Summary:

I. No. The USAR CR makes a number of changes to text and tables in Sections 9.2, 9.3, and 9.5 to clarify that there is only one unit on the site and hence one set of diesel generators. The effect of the proposed USAR administrative/editorial changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR; will not adversely affect system or plant performance in a manner that would increase the occurrence probability of an accident; will not change, degrade, or prevent actions described or assumed in any accident evaluation discussed in the USAR; will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR; and will not adversely affect any fission product barriers. The proposed change does not impose increased testing requirements on system or equipment important to safety. The affect of the proposed changes does not create any new failure modes or failure effects for equipment important to safety; does not degrade the reliability of any plant system, and does not introduce any new failure mechanisms for any plant system. System redundancy and independence are not reduced. The current operation, function, performance, and expected response of protective systems are not affected by the proposed changes. The affect of the proposed changes does not impact any radiological consequence to the public or onsite personnel. This activity does not increase the probability of occurrence, or the radiological consequences, of an accident or a malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The USAR CR makes a number of changes to text and tables in Sections 9.2, 9.3, and 9.5 to clarify that there is only one unit on the site and hence one set of diesel generators. The affect of the proposed changes will not create any new initiators or contributors for an event that could be considered a new accident, will not cause or facilitate the occurrence of any known accident initiators or contributors, will not increase the probability of an accident previously thought to be incredible, does not make a previously non-credible event credible, and does not create any new failure modes or failure effects for equipment important to safety. The affect of the proposed changes does not impact the design and operation of any SSCs, does not affect how SSCs react to normal and abnormal transients, does not degrade any equipment, and will not create an initiator or contributor to a malfunction of equipment installed in the plant not previously evaluated in the USAR. This activity does not create the possibility of an accident, or a malfunction of equipment important to safety, of a different type than any previously evaluated in the USAR.

III. No. The proposed changes will not adversely affect the design basis of any SSC. The proposed changes will not adversely affect the ability of any SSC to perform as designed. Since the proposed changes will not adversely affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced. This activity does not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0082
Source Document: Simple Modification Request Form (SMRF) 00-5026, Revision 0

Description of Change:

The proposed modifications to the Heating Ventilation and Air-Conditioning (HVAC) openings and/or HVAC components ensure that the Design Basis Tornado (DBT) does not adversely affect the interior architectural walls in the Control Complex. Tornado dampers are being installed in specific HVAC openings. Specific HVAC components (ductwork) were modified or evaluated to withstand tornado depressurization affects in order to prevent depressurization of the Control Complex. This modification, along with modifications discussed in Safety Evaluation 00-0071 (SMRF 00-5025) will eliminate interim compensatory operator actions and ensure components important to safety within the Control Complex are not adversely affected by a DBT. The proposed changes were evaluated for the capability to provide protection against tornado depressurization and to ensure no adverse affects on HVAC system operation during non-tornado conditions.

Summary:

I. No. A tornado is a design basis event and not a design basis accident. By protecting safety related equipment, the probability of occurrence of an accident or malfunction of equipment important to safety from natural events, such as tornadoes, is minimized. The tornado protection is provided to allow for safe shutdown and is not a design criterion that needs to be applied coincident with a design basis event that results in the release of radioactive material. To avoid an increase in the probability of malfunction of equipment important to safety, the reliability of the tornado protection capability of the plant cannot be decreased. The only means of protecting the plant from a DBT was described in the USAR as structures designed to withstand the depressurization effect. The tornado protection capability was shown to be highly reliable and comparable to passive structural capabilities. The installation of tornado dampers and the ductwork modifications were shown to not adversely affect HVAC system operation during non-tornado conditions. The modifications do not initiate or increase the probability of an accident or malfunction to equipment important to safety. There are no increases to radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. SMRF 00-5026 interfaces with HVAC systems that are accident-mitigating systems; but are not accident initiators. This activity does not affect any system, structures, or components such that the possibility of a different type of accident could be created. This change is introducing a failure mode of a different type (failure of a tornado damper). However the effect of the malfunction or failure of a single tornado damper does not alter the ability to protect the Control Complex from a DBT or adversely affect the plant operation during non-tornado conditions. The change will not create any new systems, or add any new equipment that can compromise the function of any systems, structures, or components. Therefore, these modifications do not create possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The safety functions of the HVAC systems have been shown to be unaffected by the modifications. Design and safety margins that existed have not been changed or compromised. The plant changes will not degrade the capability of the HVAC systems to mitigate the effects of postulated transients and accidents. Thus the margin of safety provided through the system design as discussed in the USAR, Technical Specifications, Safety Evaluation Report, Operational Requirements Manual, and other related documents and respective bases is maintained.

Safety Evaluation: 00-0083
Source Document: USAR Change Request (CR) 00-116

Description of Change:

USAR Change Request 00-116 was initiated to revise note 7 on USAR Figures 3.11-20, -21, -22, -23, -24, -25, and -26 (and the associated B-022 series environmental drawings) to more accurately describe the operation of the containment spray system.

Summary:

I. No. The proposed USAR change does not authorize any changes to the plant. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed USAR change will have no impact on the physical plant. The proposed change is limited to the clarification of a note on the B-022 series environmental drawings and the associated USAR figures. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed modification does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0084
Source Document: Design Change Package (DCP) 00-5018, Revision 0

Description of Change:

The Emergency Closed Cooling Water (ECCW) surge tanks have a safety function to assure an adequate suction head for the ECCW pumps and provide makeup water for expected leakage from the system during all modes of operation. This modification raises the ECCW surge tank water levels. Currently there is enough water in each loop of the Emergency Closed Cooling system's surge tanks to tolerate a 0.5 gallon per hour (gph) leak (per loop) over a seven day period without any system make up. With this change, the water level in each loop's surge tank is increased such that a 2.7 gph leakage from the system is acceptable over a seven day period. The modification adds new tank level instruments as well as a permanent site glass to be used for system leak rate testing. The instruments are safety related whereas the site glass is non-safety and only used for testing of a single loop at one time.

Summary:

I. No. The materials chosen and their pressure retaining capability are appropriate for the application. The system and plant operation, availability, and response to transients remain the same as described in the USAR. The proposed change will not cause a change to any system interface in a way that would increase the likelihood or frequency of an accident. Therefore, the probability of occurrence of an accident previously evaluated in the USAR is not increased. No changes are being made to any assumptions or inputs previously made to assess radiological consequences and no fission product barriers are being affected. Therefore, there is no increase in the radiological consequences of an accident previously evaluated in the USAR. The proposed modification does not change, degrade or prevent actions described or assumed in the USAR. The replacement level switches are functionally and physically equivalent to the existing switches. The replacement switches, sight glass and connecting piping will not impact the seismic qualification of the system. The proposed change does not increase the probability of system failure and does not result in an impact on any other safe shutdown components. Therefore, there is no increase in the probability of occurrence of a malfunction of equipment important to safety previously evaluated. Radiological dose rates in plant buildings will be unaffected by the proposed change. Therefore, previous evaluations of the radiological consequences of a malfunction of equipment important to safety remain unchanged.

II. No. The replacement switches are designed and procured consistent with currently defined quality and seismic standards for the ECCW. The replacement switches will be functionally equivalent to the existing switches and installed in the same location as existing switches. The site glass is utilized for testing only, and is isolated from the ECCW system during all modes of system operation. The change does not introduce any new or different components which could act as initiators of an accident of a different type. The change does not have the potential to adversely impact any important to safety systems or components and will not result in the system being operated or tested outside of its existing design or test limits. Therefore, this modification does not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The replacement level switches with safety related /seismically qualified components are consistent with previously defined standards and have no impact on any structure, system, or component (SSC), safety or non-safety. The licensing commitment to maintain a seven day water supply is not changed. Therefore, the margin of safety discussed in the USAR, Technical Specifications, Operational Requirements Manual, and Safety Evaluation Report, with respect to the available water supply in the ECCW system, is unchanged.

Safety Evaluation: 00-0085
Source Document: Simple Modification Request Form (SMRF) 00-5013, Revision 4

Description of Change:

Simple Modification Request Form (SMRF) 00-5013, Revision 4 will upgrade the Emergency Service Water (ESW) sluice gate inflatable seals and associated inflation system, inclusive of the check valves in the P52 instrument air supply line, to safety grade in accordance with 10CFR50 Appendix B requirements for components important to safety prior to relying on the seals during the summer of 2001. This upgrade is in accordance with the commitment associated with License Amendment 114. All components will be seismically qualified and installed. The function and basic operation of the sealing system will be unchanged from its current design.

Summary:

I. No. The ESW System is an accident mitigating system that provides a reliable source of cooling water during accident conditions and is not an accident initiator. Consequently, the upgraded sluice gate seals and associated inflation system, which are part of the ESW System, are likewise not accident initiators. The sluice gate seals and associated inflation system are essentially independent components, and thus do not interface with any plant systems, structures, or components in such a manner as to increase the probability of occurrence of a previously evaluated accident. Installation of this modification does not create any interactions with any systems, structures, or components important to safety such that the accident mitigating function of the ESW System or any other system is compromised. This modification does not affect the sluice gates ability to provide a more leak-tight isolation between the discharge tunnel and ESW forebay, and therefore the ESW inlet temperature will not exceed its maximum allowable value thereby ensuring the continued operability of the ESW System for accident mitigation.

Implementation of this modification does not change the function of the sluice gate sealing system and it does not affect the current operational modes of the sluice gates. The sluice gates will still be capable of opening on a low forebay water level signal (when the seals are deflated) and the gates will still perform their isolation function while in the closed position. The seals, when inflated, will be capable of performing their safety function of eliminating the leakage of discharge water to the ESW forebay. Thus, the activity cannot increase the probability of malfunction of the sluice gates. This modification does not create any adverse interactions with any other plant systems, structures, or components, and therefore it cannot increase the probability of malfunction of any other equipment important to safety. Upgrade of the sluice gate seals and associated sealing system to safety related does not affect any modes of operation of the ESW system, and therefore the accident mitigating capability of the ESW System is not compromised. The sluice gate seals and supporting inflation system components do not interface or interact with any other accident mitigating structures, systems, or components such that their accident mitigating capability is compromised, and therefore the radiological consequences of a malfunction of equipment important to safety cannot be increased.

II. No. This modification to piece/parts of the ESW System does not introduce any new accident initiators and consequently cannot create the possibility of an accident of a different type than previously evaluated. The upgraded components installed via this modification do not 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. The safety related upgrade of the sluice gate seals and inflation system cannot create the possibility of a different type of malfunction of equipment important to safety. The sluice gate seals and

inflation equipment do not interact with any systems, structures, or components important to safety in such a manner as to create a different type of malfunction of equipment important to safety. All components will be seismically qualified and seismically installed, and thus these components will not interfere with the safety related functions of any nearby safety related components.

III. No. Subsequent to implementation of this modification, the sluice gates in the closed position and the sluice gate seals when inflated will still be able to perform their isolation function of preventing hot discharge water from entering the ESW forebay and therefore the margin of safety associated with the ESW pump inlet temperature will be retained. Preservation of this margin of safety supports the Technical Specification margin of safety related to providing and maintaining adequate cooling to the Emergency Core Cooling Systems and components.

Safety Evaluation: 00-0086
Source Document: Emergency Plan (E-Plan), Revision 15

Description of Change:

Revision 15 of the E-Plan was prepared to incorporate extensive administrative/editorial changes. In addition to the administrative changes, the Environmental Protection Agency (EPA) guidelines for the ingestion of food was incorporated into the E-Plan; the upgraded meteorological tower information was revised to reflect current plant configuration; and the frequency of the emergency planning audits was revised from annually to biennially.

Summary:

I. No. The proposed change does not authorize any changes to the plant. The proposed changes to the E-Plan do not have any affect on the probability of occurrence of an accident. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change will have no impact on the physical plant. The proposed changes do not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed changes do not impact the operation of any other plant system and do not create any new failure modes. Therefore, the proposed changes will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0087
Source Document: USAR Change Request (CR) 00-124

Description of Change:

USAR Change Request 00-124 was initiated to revise USAR Figures 5.4-9 (Sheets 1 and 2) and 5.4-13 (Sheets 1, 2, and 3) to remove excessive detail from the drawings as noted in Condition Report (CR) 00-0650. The USAR figures correspond to P&IDs 302-631, -632, -641, -642, and -643. In addition, some minor drawing discrepancies were also corrected on the USAR figures as noted in CR 00-0650.

Summary:

I. No. The proposed change does not authorize any changes to the plant. The proposed change is limited to removing notes which are considered to be excessive detail or correcting minor drawing discrepancies. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change is limited to removing notes which are considered to be excessive detail or correcting minor drawing discrepancies. The proposed USAR change will have no impact on the physical plant. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed USAR change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed USAR change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed USAR changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed USAR changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0088
Source Document: USAR Change Request (CR) 00-123

Description of Change:

USAR Change Request 00-123 implements an organization change to the Perry Nuclear Power Plant (PNPP) management structure. Under this organization change, the Quality Assurance Section (QAS) is renamed Nuclear Quality Assessment, Perry and will transfer its reporting point to a newly established organization, the First Energy Nuclear Operating Company (FENOC) Oversight and Process Improvement Department. This organization reports to the President and Chief Nuclear Officer, FENOC and is responsible to perform oversight activities, namely quality assurance, for the FENOC operated nuclear power plants. Quality control inspection activities and the administration of the corrective action program, both currently assigned to QAS, will not transfer and remain in the Perry Nuclear Services Department. The Independent Safety Engineering Group (ISEG) will be incorporated into the QAS assessment staff and will no longer function as a separate, dedicated group. The most significant aspect of this organizational change is that assessment activities (i.e., quality assurance audits, surveillances, and selected independent safety engineering functions) will be performed by a FENOC element external to the plant organization.

Summary:

I. No. This USAR Change Request implements a site organization change, which is being reflected in USAR Chapters 13 and 17.2. The organizational change involves re-assigning the performance of Perry quality assurance functions to a new FENOC organization. There are no quality assurance requirements or operational functions changed in the process. The organizational changes made in USAR Change Request 00-123 do not impact the design, function, or operation of the plant. The change does not involve any hardware or operational changes to the plant. The operational quality assurance program does not input into the accident analysis or initiating event as described in the USAR Chapter 15. The quality assurance organization alignment also does not affect the accident analysis. Since the accident analysis is not affected by this change, the probability of occurrence of an accident previously evaluated in the USAR is not increased. Similarly, the radiological consequences for any accident previously evaluated in the USAR are not increased since the accident analysis is not being affected.

   The organizational change, as described, makes no change to plant equipment, systems or operating procedures. The operational quality assurance program establishes quality assurance requirements and controls for performing safety related and augmented quality processes and does not direct the design or operation of plant equipment. As a result, the quality assurance organizational changes made do not increase the probability of occurrence of a malfunction of equipment previously evaluated in the USAR, nor do they increase the radiological consequences of a malfunction of equipment important to safety previously evaluated.

II. No. This USAR Change Request is applicable to the Perry quality assurance organization and operational quality assurance program description. It does not involve or impact any plant systems, equipment, or operating procedures. There are no hardware changes to structures, systems, or components. The quality assurance program does not factor into the accident analysis as contained in the USAR Chapter 15. Also, it cannot be postulated that this organizational change could impact an initiator for any other accidents. As a result, this USAR Change Request does not create the possibility of an accident of a different type than previously evaluated in the USAR. The organizational changes do not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated.

III. No. This USAR Change Request is applicable to the Perry quality assurance organization and operational quality assurance program description. There are no hardware changes or system operation changes being made with this change. The quality assurance program description provides quality assurance controls for safety related and augmented quality systems and processes. It has no direct impact on the operation or design function of any structures, systems or components or their applicable technical specification limits. There is no specific margin of safety associated with the quality assurance program. As a result, the organizational changes and quality assurance program changes being made do not reduce the margin of safety for any technical specification item.

The changes evaluated by this Safety Evaluation have been determined to not include reductions in quality assurance program commitment per 10CFR50.54(a). The new organizational alignment, as described, for quality assurance maintains independence from cost and production influences. Quality assurance functions will continue to be performed in accordance with current requirements. The organization changes have no impact on the Technical Specifications, Operating License, Environmental Protection Plan, or the Operational Requirements Manual.

Safety Evaluation: 00-0089
Source Document: Temporary Modification (TM) 1-00-0004

Description of Change:

This Temporary Modification utilizes the application of a freeze seal on two 3/4" Two-Bed Demineralizer Water System (P21) service drops in order to support the maintenance activities to repair or replace valves 0P2F0686 and 1P2F0833. Valve 0P2F0686 is located in the Radwaste Building on elevation 574'. Valve 1P21-F0833 is located in the Turbine Building on elevation 577'. The freeze seals will be placed on the ¾" pipe drop located above these valves to allow these valves to be replaced while the balance of the P21 system remains in service.

Summary:

I. No. The P21 system is not relied upon for the safe shutdown of the plant. A failure of either freeze seal will not prevent the P21 system from performing its design function, nor will it cause the plant to go through an unanalyzed transient. Any leakage from a failed freeze seal will not create any flooding concerns. This is because the current flooding analysis bounds the anticipated flow of 50 to 60 gpm. Administrative controls per Generic Maintenance Instruction (GMI)-0024 minimize the probability of the freeze seal failing. There is no vital equipment located in the area of either freeze seal that could be affected by the failure of the freeze seal. There is no increase of the dose consequences of any accident described in the USAR. No adverse system interactions are created by the implementation of these freeze seals. This activity makes no structural, system or component changes to the plant that would impact the performance of equipment important to safety. No new failure modes or effects are created by this activity. No radionuclides, release rates, release mechanisms or impact to radiological barriers are affected by these activities such that operator actions necessary to mitigate the consequences of any accidents or malfunctions or equipment important to safety are impeded. A review of the USAR determined that there are no relevant USAR accidents that could result form the failure of the either of these freeze seals. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety as previously evaluated in the USAR.

II. No. The use of these freeze seals will not create any new systems or adversely affect the function of any operating system. These activities do not increase the effects of any event that was previously bounded by other accidents to such that it would become bounding. The use of these freeze seals is not an accident mitigator, nor can there use cause any new accidents to occur. These activities do 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. No equipment has been removed or altered that would affect equipment important to safety from functioning and interacting with the plant. No new failure modes or effects are created by this activity. Therefore, there will be no possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. The Technical Specifications and its Bases, Operating License and the Operational Requirements Manual are not affected by the application of these freeze seals to the P21 system. There is no change to any accident analysis or margin of safety by the use of these freeze seals as discussed in the USAR, Technical Specifications and its Bases or the applicable NRC Safety Evaluation Report. These freeze seals will not introduce any new radiological or environmental concerns. Therefore, by applying these freeze seals by the current standards and procedures, the margin of safety as defined in the bases for any Technical Specifications is not reduced.

Safety Evaluation: 00-0090
Source Document: USAR Change Request (CR) 00-129; Plant Administrative Procedure (PAP) 0802, Revision 6, PIC 5

Description of Change:

This safety evaluation evaluates the change of the maximum fuel bundle lattice enrichment allowed to be stored in Perry's high density fuel storage racks located in the Fuel Handling Building from 4.5% to 4.9 weight percent U235. Global Nuclear Fuels [GNF, formerly General Electric (GE)] re-performed the criticality analysis for the high density racks and found that the higher enriched fuel bundles do not exceed the Technical Specification limit of less than or equal to 0.95 K-effective. The analysis also demonstrated that any fuel with a peak lattice reactivity of less than or equal to 1.3746 (in-core K-infinity) can be safely stored. The change request is limited to the storage of unirradiated fuel bundles until such time the Cycle 9 Core Design Safety Evaluation is approved. The methodology of determining the in-core K-infinity to confirm the compliance with the in-rack K-effective limits is reviewed and approved in GE Standard Application for Reactor Fuel (GESTAR II, GNF's licensing topical for reload cores) and the GE14 Compliance with Amendment 22 to GESTAR II.

Summary:

I. No. The analysis has shown that fuel with in-core K-infinities of less than or equal to 1.3746 can be safely stored in Perry's high density storage racks located in the Fuel Handling Building. This analysis was completed with qualified codes and included the required uncertainty analysis. As such a criticality accident in these racks has been shown not to be possible since the calculated K-effective for this fuel stored in these racks is 0.9280 (including uncertainties). The calculated K-effective is less than the licensing limit of less than or equal to 0.95. The method of using the in-core K-infinity to confirm compliance with the in-rack K-effective is reviewed and approved in GESTAR II and as documented within the General Electric GE14 compliance document. Since criticality is not a credible event in Perry's high density storage racks located in the Fuel Handling Building, there was not an accident analyzed with radiological consequences. This continues to be the case. A criticality accident in these racks has been shown not to be possible since the calculated K-effective for this fuel stored in these racks is 0.9280 (including uncertainties). Since this Change Request involves the storage of unirradiated bundles, there is no increase to the source term used in the accident analysis. For the Fuel Handling Accident (FHA) the assumption is that some pins are damaged in the dropped bundle and in the impacted bundles. A FHA involving unirradiated higher enriched bundles will not contribute to the source term. Also the GE14 bundle is identical in size and weight as the previously analyzed GE12 bundles. Thus, the consequences of the FHA will not increase as a result of this change. This change affects the analysis basis of Perry's high density storage racks located in the Fuel Handling Building. The racks have been shown to maintain the stored fuel sub-critical. No new failure modes of equipment important to safety are introduced by this change. The change will not create any new systems, or add any new equipment that can compromise the function of any system, structures, or components (SSC). As a result, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. This change affects the analysis basis of Perry's high density storage racks located in the Fuel Handling Building. The racks have been shown to maintain the stored fuel sub-critical. This activity does not affect any SSC such that the possibility of a new accident could be created. The change will not create any new systems, or add any new equipment that can compromise the function of any SSC. This change will not result in any new equipment failures, and therefore this change will not create any new initiators or contributors for an event that could be considered a new accident. This change 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. This change affects the analysis basis of Perry's high density storage racks located in the Fuel Handling Building. The racks have been shown to maintain the stored fuel sub-critical. No new failure modes of equipment important to safety are introduced by this change. As a result, the possibility for an accident or malfunction of a different type than any evaluated previously in the USAR is not created.

III. No. Technical Specification 4.3.1.1.a requires a rack K-effective less than or equal to 0.95 if fully flooded with unborated water including allowance for uncertainties as described in Section 9.1.2 of the USAR. The analysis has shown that fuel with in-core K-infinities of less than or equal to 1.3746 can be safely stored in Perry's high density storage racks located in the Fuel Handling Building. This analysis was completed with qualified codes and included the required uncertainty analysis. As such a criticality accident in these racks has been shown not to be possible since the calculated K-effective for this fuel stored in these racks is 0.9280 (including uncertainties). The calculated K-effective is less than the licensing limit of less than or equal to 0.95. As a result, the margin of safety as defined in the bases for any Technical Specification is not reduced.

Safety Evaluation: 00-0091
Source Document: Equivalent Change Package (ECP) 00-8010, Revision 1; Simple Modification Request Form (SMRF) 00-5028, Revision 0

Description of Change:

ECP 00-8010 installs additional communication headset jacks, 0R52-M013B and 1R52-M094D, on the fuel handling platform, 0F11-E0014, and refueling platform, 1F15-E0003 to aid refueling personnel in communication between the platforms and the control room.

SMRF 00-5028 replaces the power cable for the auxiliary platform, 1F15-E0005, to address cable feed problems associated with the platform cable reel. The current power cable is a 27-conductor cable, 12 are spare conductors, along with a communication conductors, for headset jack station 1R52-M094C on the platform. The replacement power cable is more flexible and has a smaller diameter, containing four conductors, but does not contain communication cable. 1R52-M094C is being abandoned in place.

Summary:

I. No. Accidents evaluated in the USAR that may be affected by this modification include fuel handling accidents and misplaced bundle accident described in USAR Sections 15.4.7, 15.7.4.1 and 15.7.6. However, the replacement of the power cable meets the same functional requirements to provide power to the 1F15-E0005 platform. Abandonment of 1R52-M094C and removal of the associated communication cable on the auxiliary bridge does not affect communications in performance of any normal or emergency actions, procedures or surveillances. The addition of 0R52-M013B and 1R52-M094D, on 0F11-E0014 and 1F15-E0003 provides an alternate location for local intra-plant communication, paralleling the existing local communication stations local to the platforms. The new control cables and jack stations are constructed to the same quality standards as the original communication equipment and are fully compatible with all existing system components. Therefore, the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. The replacement cable has been sized to carry the maximum current required to operate the platform, and meets all applicable codes and standards. The addition of 0R52-M013B and 1R52-M094 parallels existing intra-plant communication circuitry, failure of the jack stations could only affect one of the five selectable page/party stations as discussed in USAR Section 9.5.2.2.2. Safety Evaluation Report (SER) Section 9.6.1.2 has accepted that a complete failure of the communication system (plant PA, telephones, radios, etc.) would be adequate to meet the NRC acceptance criteria in NUREG-0800. The abandonment of 1R52-M094C and removal of the associated communication cable on the auxiliary bridge does not affect performance of any normal or emergency actions, procedures or surveillances, as originally designed, no head set communication equipment is permanently installed at that location for general use. Therefore, the possibility of an accident or malfunction of equipment of a different type than any evaluated previously in the USAR is not created.

III. No. Upon completion of the modifications, functional testing of the components will be completed to ensure that the platform and communication system are properly operating. A review of the Technical Specifications, bases for the Technical Specifications, the operational requirements manual and the USAR has shown that there is no clear trend toward a reduction in the margin of safety as a result of these changes.

Safety Evaluation: 00-0092
Source Document: USAR Change Request (CR) 00-130

Description of Change:

This USAR Change Request was prepared to bring the Beaver Valley Power Station under the scope of the First Energy Nuclear Operating Company (FENOC) Quality Assurance Program Manual (QAPM). All three FENOC operated nuclear power plants will now function under a single quality assurance program. The FENOC QAPM is the operational quality assurance program description for Perry Nuclear Power Plant (PNPP) and is incorporated by reference in Chapter 17.2 of the USAR.

Summary:

I. No. The proposed change does not affect any plant equipment. The reductions in Quality Assurance (QA) program commitments will not negatively affect the operation of plant equipment. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. There is no specific margin of safety that is associated with the QA program. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 00-0093
Source Document: Design Change Package (DCP) 97-5079, Revision 0

Description of Change:

This design change package will modify communications cabinet 1H13-P5001 to remove the existing Communications Modules and associated hardware to make room for new communication termination equipment for existing fiber optic, telephone and category-5 cables.

Summary:

I. No. The communications modules are an independent device that is not functionally connected to any other system, outside of communications in the plant and as such it has been determined that the proposed changes do not affect any of the initiators or contributors to the accidents previously evaluated in the USAR. The proposed changes will not alter any assumptions previously made in evaluating the radiological consequences of an accident described in the USAR. No equipment protection features are being deleted or modified by the proposed changes. The proposed changes do not create any new failure effects for equipment important to safety. The proposed changes will not alter, degrade or prevent actions described or assumed in any accident described in the USAR. Therefore, the proposed changes will not increase the probability of occurrence or radiological consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The proposed changes will not create any new initiators or contributors for an event that could be considered a new accident. The proposed changes will not cause or facilitate the occurrence of any known accident initiators or contributors, and therefore will not increase the probability of an accident. The proposed changes do not make a previously non-credible event credible. The proposed changes do not affect any system important to safety, and does not affect the way the system will react to normal and abnormal transients. The proposed changes will not be an initiator or contributor to the malfunction of any equipment installed in the plant. Therefore, the removal of these non-applicable and/or outdated notes will not create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR.

III. No. It has been determined that the Technical Specifications, Operational Requirements Manual and the Safety Evaluation Report (SER)/Supplements to SER are not affected by the proposed changes. The proposed changes are not related to Technical Specification Bases. The proposed changes will not affect the design bases of any System, Structure, or Component (SSC). The proposed changes will not affect the ability of any SSC to perform as designed. Since the proposed changes will not affect the function or operation of SSCs, the margin of safety and availability of the SSCs will not be reduced.

Safety Evaluation: 01-0001
Source Document: Temporary Instruction TXI 0321, Revision 0

Description of Change:

This TXI incorporates the NobleChem™ process into the plant. The NobleChem™ process injects noble metal compounds into the reactor vessel as part of an overall process to prevent crack initiation and to mitigate Intergranular Stress Corrosion Cracking (IGSCC). NobleChem™ employs the reactor coolant as the transport medium to deposit minute amounts of noble metal platinum (Pt) and rhodium (Rh) on all wetted Reactor Pressure Vessel (RPV) surfaces and related reactor and piping components. NobleChem™ creates catalytic surfaces to maintain oxygen deficient water in contact with reactor components when a small amount of hydrogen is injected. In doing so, protection from IGSCC can be achieved with small amounts of injected hydrogen.

Summary:

I. No. The application, and the resulting condition after the application, has no affect upon plant operations. No accident frequencies are increased because the result of the process is a microscopic deposition into an internal oxide film that already exists. No radiological consequences are increased because the process and resulting condition does not affect any equipment or instrumentation that is used for detection or mitigation of radiological accident consequences. The process, being a passive, microscopic addition to reactor internal corrosion film, does not interfere with any equipment required for accident mitigation or safe shutdown of the plant. The application process and the deposition of platinum and rhodium does not have an impact upon safety related equipment or components, equipment, systems or structures important to safety.

II. No. The application adds no equipment to the plant. It does not change any assumed failure modes or effects of failures. No new equipment interaction is created with the process nor any new operating scenario or sequence of events that can cause any unanalyzed failure mode or effect of failure. Because the process results in a passive, microscopic addition of metal to an existing oxide film, the process has no effect upon the plant's ability to function within the license basis.

III. No. The application of NobleChem™ and operating after the application has no affect upon plant operations. All operations will remain within Technical Specification limits and allowables. The process does not require a change to the Technical Specifications nor to the basis for any Technical Specification. The process is applied to mitigate IGSCC. In this fashion, it maintains the margin of safety associated with protecting the reactor vessel, internals and associated piping from IGSCC.

Safety Evaluation: 01-0005
Source Document: Plant Administrative Procedure (PAP) 0806, Revision 4, PIC 1

Description of Change:

The "Spill Prevention Control and Countermeasure Plan" (SPCC), Attachment F to Plant Administrative Procedure (PAP) 0806, is being changed to reflect the condition of the oil skimmer plate and the clogged sediment standpipe for the sediment control dam at the major stream impoundment. Credit will no longer be taken for the skimmer plate and sediment removal capability of the sediment control dam at the major stream.

Summary:

I. No. The proposed change will no longer take credit for the oil skimmer plate and sediment removal capability of the control dam. The major stream is unrelated to plant activity. It is a stream originating offsite, south of Perry Village. It enters the site from the south and flows along the south side of the closed construction landfill, under the main entrance road and then south of the Training and Education Facility. The sediment control dam is located just a few hundred feet from the Lake Erie shoreline, at the extreme northwestern corner of the Owner Controlled Area. The original purpose of the structure was to provide for a settling area to prevent sediment from entering Lake Erie during construction of Perry Nuclear Power Plant (PNPP). The oil skimmer plate was erected a few feet upstream of the dam to prevent spilled oil from entering Lake Erie. This structure is in place for environmental reasons and does not interface in any way with plant systems. This structure will not affect plant system performance in any manner and would not lead to an accident or cause an accident previously evaluated to shift to a higher frequency category. This activity will not increase the probability of occurrence of an accident, malfunction of equipment, or the radiological consequences of an accident previously evaluated in the USAR.

II. No. The proposed change will no longer take credit for the oil skimmer plate and sediment control dam. The major stream is unrelated to plant activity. This structure is in place for environmental reasons and does not interface in any way with plant systems. Based on this information the proposed activity will not create a different type of malfunction of equipment important to safety or a different type of accident than previously evaluated in the USAR.

III. No. The site remains in compliance with environmental regulations and the Environmental Protection Plan. The design or operation of the plant will not be affected. Hence, accident analysis, as described in the USAR, will not be impacted. Therefore, no margin of safety has been reduced.

Safety Evaluation: 01-0006
Source Document: Plant Data Book PDB-F0001, Revision 8; Plant Administrative Procedure (PAP) 0802, Revision 6, PIC 6; USAR Change Request (CR) 01-008

Description of Change:

The Core Operating Limits Report and USAR are changed to incorporate the Cycle 9 Core Design. The analysis performed for License Amendment 119 Safety Limit Minimum Critical Power Ratio (MCPR) was not repeated in this evaluation. PAP-0802 was changed to delete restrictions on using higher enriched fuels until this safety evaluation was complete (reference Safety Evaluation SE 00-0090).

General Electric Standard Application for Reactor Fuel (GESTAR II) is General Electric's topical report which applies NRC approved methodologies to the reload analysis. GESTAR II is referenced by Perry's USAR as the means for reload analysis.

Summary:

I. No. The reload design was performed in accordance with the requirements of the GESTAR II. The fuel failure mechanisms described in GESTAR II are also unchanged. The cycle specific Safety and Operating Limits will protect the fuel in accordance with the design basis. The introduction of the GE14 fuel has been analyzed per GESTAR II and has been found to remain intact for normal operations, transients, and accidents. The effects of increased bundle enrichment, bundle exposure, and bundle decay heat loading were also reviewed and were found to be bounded by existing analysis. As a result, the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR, is not increased.

II. No. The reload design was performed in accordance with GESTAR II. The essential components of the fuel are the same as previously analyzed. The function and operation of the fuel remains unchanged. The initiating sequence of events has not changed. The GESTAR II analysis has been accepted by the NRC as comprehensive for ensuring that the reload design will perform within acceptable bounds. The introduction of GE14 fuel has been analyzed per GESTAR II and has been found to remain intact for normal operations, transients, and accidents. As a result, the possibility for an accident or malfunction of a different type than previously evaluated in the USAR is not created.

III. No. The introduction of GE14 fuel bundles, higher enrichment loadings, and a new reference loading pattern do not alter the design or function of any plant system. The reload design was produced using NRC-approved methods described in GESTAR II. The cycle specific MCPR Safety Limit values do no alter the design or function of any plant system. The cycle specific MCPR Safety Limit values were produced using NRC-approved methods described in GESTAR II and Reference 2, USAR Sections 4.4.1 and 15.0.3.3.1, the NRC Safety Evaluation Report and its Supplements for Section 4.4.1, and the Technical Specification Bases (Section 2.1.1.2) for the MCPR Safety Limit. The design satisfies the acceptance criteria of the other fuel-related Technical Specifications (MCPR Operating Limits, Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Linear Heat Generation Rate (LHGR), Shutdown Margin, and Fuel Storage). As a result, the margin of safety, as defined in the bases for any Technical Specification, is not reduced.

Safety Evaluation: 01-0007
Source Document: Temporary Instruction (TXI) 0321, Revision 0, PIC 2

Description of Change:

PIC 2 to TXI-0321 changes the temperature control band for the application of NobleChem™. The lower temperature limit is changed from 250° to 225° F. Changing the lower temperature from 250° F to 225° F changes no previous assumptions.

NobleChem™ is a process that injects noble metal compounds into the Perry Nuclear Power Plant (PNPP) reactor vessel as part of an overall process to prevent crack initiation and to mitigate Intergranular Stress Corrosion Cracking (IGSCC). NobleChem™ employs the reactor coolant as the transport medium to deposit minute amounts of noble metal platinum (Pt) and rhodium (Rh) on all wetted Reactor Pressure Vessel (RPV) surfaces and related reactor and piping components. NobleChem™ creates catalytic surfaces to maintain oxygen deficient water in contact with reactor components when a small amount of hydrogen is injected. In doing so, protection from IGSCC can be achieved with small amounts of injected hydrogen.

Summary:

I. No. The application and the resulting condition after the application has no affect upon plant operations. No accident frequencies are increased because the result of the process is a microscopic deposition into an internal oxide film that already exists. No radiological consequences are increased because the process and resulting condition does not affect any equipment or instrumentation that is used for detection or mitigation of radiological accident consequences. The process, being a passive, microscopic addition to reactor internal corrosion film, does not interfere with any equipment required for accident mitigation or safe shutdown of the plant. The application process and the deposition of platinum and rhodium do not have an impact upon safety related equipment or components, equipment, systems or structures important to safety.

II. No. The application adds no equipment to the plant. It does not change any assumed failure modes or effects of failures. No new equipment interaction is created with the process, nor any new operating scenario or sequence of events that can cause any unanalyzed failure mode or effect of failure. Because the process results in a passive, microscopic addition of metal to an existing oxide film, the process has no effect upon the plant's ability to function within the license basis.

III. No. The application of NobleChem™ and operating after the application has no affect upon plant operations. All operations will remain within Technical Specification limits and allowables. The process does not require a change to the Technical Specifications nor to the bases for any Technical Specification. The process is applied to mitigate IGSCC. In this fashion, it maintains the margin of safety associated with protecting the reactor vessel, internals and associated piping from IGSCC.

Safety Evaluation: 01-0008
Source Document: Design Change Package (DCP) 98-0050, Revision 0

Description of Change:

This DCP performs blade modifications to the High Pressure (HP) turbine. The Turbine Control System (N32) and Electric Hydraulic Control System (EHC) will be recalibrated for partial arc operations. The stop rings in Turbine Control Valves (TCV) 1, 2 and 3 will be changed to increase their strokes. The stop ring in TCV 4 will be changed to shorten its stroke. The TCV low point drain will be changed. The drain valve for the #4 TCV will be changed to operate based on the #4 TCV position. General Electric will supply a new plant heat balance and thermal kit for partial arc operation.

Summary:

I. No. USAR Chapters 3, 10, and 15 describe turbine events and licensing basis accidents. The discussion of each issue and the accidents described in the USAR has been reviewed with respect to DCP 98-0050. It is concluded that this modification does not increase the probability of occurrence of an accident previously evaluated in the USAR. As such, the radiological consequences of postulated Chapter 15 accidents are not impacted by the proposed modification. These changes do not degrade the performance of any system, structure or component below the design basis that was assumed in the accident analysis. Also, these changes do not increase the challenges to systems, structures, or components assumed to function in the accident analysis such that system performance is degraded below the design basis without compensatory measures. This DCP does not have any effect on the established product fission barriers. None of the USAR Accidents or Transients are impacted as a result of this modification. The equipment being changed by this modification is not relied upon for accident mitigation.

II. No. The USAR addresses accident analysis of the reactor based on events such as turbine transients, including spurious trips, pressure regulator failures and turbine missiles. The change in turbine operation from full to partial arc will not create a new accident of any type. No new failure modes having a different effect are added as a result of either the change in turbine operation or the existing plant equipment. This modification is designed and installed to the same design requirements and standards as the existing equipment. No new turbine system failures have been identified that could cause a new malfunction of existing systems/equipment. There has been no change in the system interface with Rod Control and Reactor Protection Systems. There are no changes in any existing evaluated malfunction of these systems. These changes do not degrade the performance of any system, structure or component below the design basis that was assumed in the accident analysis. Also, these changes do not increase the challenges to systems, structures, or components assumed to function in the accident analysis such that system performance is degraded below the design basis without compensatory measures.

III. No. The design/criteria of the Turbine Overspeed Protective System, and how it relates to Operational Requirements Manual (ORM) Section 6.2.12, USAR Section 3.5.1.1, USAR Section 15.2.3, Perry SER 3.5.1.3, GDC-4, and Regulatory Guide 1.115, were evaluated to assess the impact to any margins of safety. The slight change in response time of the main turbine control valves does not affect the margin of safety as reflected in these documents. There is no change in the safety margin criteria associated with this change, as defined in the USAR or the bases for any Technical Specifications (TS), or the Operational Requirements Manual (ORM). Relative to the TS Bases 3.3.2.1, the setpoints that provide the Control Rod Block low power setpoint (LPSP) and high power setpoint (HPSP) trip functions have not been changed. The current settings related to these functions were reviewed and they will not be

affected based upon the operating margin available in the calculations that establish the setpoints and their tolerances. Relative to TS Bases 3.3.1.1.9 and 10, the setpoints that provide Turbine Stop Valve and Turbine Control Valve fast closure inputs to the Reactor Protection System RPS have not been changed. The current settings related to these functions were reviewed and they will not be affected based upon the operating margin available in the calculations that establish the setpoints and their tolerances. Thus, there is no reduction in the margin of safety for any Technical Specification Bases caused by this modification change.

Safety Evaluation: 01-0009
Source Document: Drawing Change Notice (DCN)-5909, Revision 0

Description of Change:

DCN 5909 was prepared to add a note to P&ID 302-861 (USAR Figure 2.4-71) to indicate that cleanout ports may be provided on the Underdrain System (P72) sump pump discharge piping. The cleanout ports may only be used during times when the P72 system is out of service for maintenance. This DCN does not authorize any changes to the physical plant.

Summary:

I. No. The proposed change does not alter the function or create any new failure modes for the existing Underdrain System. The underdrain pumps are non-safety related. The notation on the P&ID and associated USAR figure does not create any new failure modes. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system, and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs, and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0010
Source Document: Emergency Plan (E-Plan), Revision 15, PIC 1

Description of Change:

PIC 1 to the E-plan was prepared to revise the thermal power rating of the reactor. The reference to "FTS 2000 Emergency Telecommunication System" was replaced with "Emergency Telecommunications System (ETS)". In addition the communication network systems were further discussed and explained. Several typographical/editorial changes were also made to the E-Plan.

Summary:

I. No. The proposed change does not authorize any changes to the plant. The proposed change is limited to administrative items with the exception of the reference to the FTS (Federal Telecommunications System). Per NRC Regulatory Issue Summary: (RIS) 2000-11, the FTS is no longer required and a corporate emergency communications systems can be used. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed USAR change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change will have no impact on the physical plant. The proposed change does not impact the operation of any system, structure, or component. There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF Systems, Structures, or Components (SSCs) are affected by the proposed change. Since no new system interactions are being created by the proposed change, no new equipment malfunctions are postulated. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs and will not reduce the margin of safety implied or specifically stated by any licensing documents, including the USAR, SER, Operational Requirements Manual (ORM), Core Operating Limits Report (COLR), Offsite Dose Calculation Manual (ODCM), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0011
Source Document: Temporary Instructions (TXI) 0327, TXI 0328, TXI 0329, TXI 0330, TXI 0332, TXI 0335, all Revision 0

Description of Change:

The above listed temporary instructions will be utilized to install, test, and operate equipment for inspecting and reconstituting fuel during Refueling Outage 8 (RFO8). The procedures listed above will be used for the RFO8 fuel inspection activities.

Summary:

I. No. Installation of inspection and reconstitution equipment and performance of fuel inspection and reconstitution is not a safety function. Fuel assemblies will be handled and stored by plant equipment designated for these purposes. The fuel preparation machine is designed with interlocks ensuring adequate water coverage. Fuel rods outside of fuel bundles will be handled with equipment specifically designed for this purpose. Fuel inspection and reconstitution are bounded by the refueling accident analyzed in the USAR. The proposed activities do not increase the probability of a fuel bundle drop accident. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not adversely impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Any failure of the inspection or reconstitution equipment is bounded by the existing fuel handling accident analysis. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. All fuel inspection and/or reconstitution will be performed in accordance with General Electric (GE) approved procedures. The bases for Technical Specification 3.9.1 discuss restrictions on fuel movement. The GE approved procedures maintain adherence to the Technical Specification bases. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0012
Source Document: Temporary Instructions (TXI) 0338, Revision 0; TXI 0339, Revision 0

Description of Change:

The above listed temporary instructions will be utilized to install, test, operate and remove the ABB Telescope Fuel Sipping System. This system will be used to detect the presence of fuel defects during the Refueling Outage 8 (RFO8) fuel shuffle.

Summary:

I. No. Installing/removing, testing, or operating the Fuel Sipping System does not represent a safety function. Fuel assemblies will be handled and stored by plant equipment designated for these purposes. The fuel preparation machine is designed with interlocks ensuring adequate water coverage. The Fuel Sipping System does not add any significant weight to the fuel handling grapple. The proposed activities do not increase the probability of a fuel bundle drop accident. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised. Therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The proposed change does not adversely impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Any failure of the Fuel Sipping System equipment is bounded by the existing fuel handling accident analysis. The proposed change does not impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Refueling equipment interlocks of the refueling platform and hoist movement system are unaffected by the presence of the sipping equipment. All Technical Specification requirements will be maintained. Therefore, the proposed changes will have no impact on the margins of safety associated with SSCs and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0013
Source Document: Design Change Notice (DCN) 5906, Revision 0

Description of Change:

This DCN will remove relief valve 1R45-F0571B from drawing D-302-356. This valve is shown between the outlet of the High Pressure Core Spray (HPCS) diesel generator engine driven fuel oil pump filter manifold and the injector header on the engine. This valve was never installed.

Portions of the HPCS diesel generator fuel oil supply lines are carbon steel piping and portions are stainless steel tubing. Drawing D-302-356 shows all lines as line specification G18-8 which is stainless steel tubing. Therefore, drawing D-302-356 will be revised to indicate the referenced line specification breaks.

The configuration of drawing D-320-356 will be changed to accurately depict the line specification breaks and the as-built configuration of the fuel oil piping and tubing.

Summary:

I. No. There are no accidents identified in the USAR that are affected by the proposed change. Therefore, the proposed change will not increase the probability of an accident previously evaluated in the USAR.

No changes are being made to any assumptions or inputs previously made to assess dose consequences and no fission product barriers are being affected. The proposed change does not change, degrade or prevent actions described or assumed for any accident discussed in the USAR. Therefore, the proposed change will not increase the radiological consequences of any accident evaluated in the USAR. The proposed change does not increase the probability of the HPCS diesel generator fuel oil system failure and does not result in an impact on any other Structures, Systems or Components (SSCs). Therefore, the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

Radiological dose rates in plant buildings will be unaffected by the proposed change. No changes are being made to any assumptions or inputs previously made to assess dose consequences and no fission product barriers are being affected. The proposed change will not alter any assumptions previously made in evaluating the radiological consequences of a malfunction of equipment important to safety. Therefore, the proposed change will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The proposed change does not add any credible failure modes or failure mechanisms for any SSC. The proposed change does not initiate any credible sequence of events resulting in any type of accident. Therefore, the proposed change does not create the possibility of an accident of a different type than any previously evaluated in the USAR.

The proposed change will not degrade or prevent acceptable SSC performance, will not create any new failure modes or effects, and will not impact any existing failure modes or effects for any SSC. Therefore, the proposed change will not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. Since the proposed change does not affect fuel oil consumption, the margin of safety associated with diesel fuel inventory is not reduced. Fuel oil delivery and operation of the diesel, including the ability to supply sufficient electrical output, is not affected by the proposed change. Therefore, the margin of safety associated with output of the diesel generator is not reduced. The proposed change is consistent with previously defined standards and will have no

impact on any SSC, safety or non-safety, and has no operational plant impact. Therefore, the proposed change will have no impact on the margin of safety implied or specifically stated by any licensing documents.

Safety Evaluation: 01-0014
Source Document: Design Change Package (DCP) 99-5051, Revision 1

Description of Change:

This modification makes active the trip function of each of the eight Oscillation Power Range Monitor (OPRM) modules. One OPRM module is located in each of the Average Power Range Monitor (APRM) chassis which is a subsystem of the Neutron Monitoring System (NMS). Thus, there are two OPRM modules installed in each NMS trip channel. A contact is wired from an existing Automatic Suppression Function (ASF) relay associated with each OPRM module into the Reactor Protection System (RPS) trip logic chain. The contacts are added in the NMS cabinets (1H13-P0669, 1H13-P0670, 1H13-P0671 and 1H13-P0672) where the NMS scram trip contacts are located. This Safety Evaluation is superseded Safety Evaluation 01-0003

Summary:

I. No. The analysis of the anticipated operational transients and design basis accidents described in the USAR for NMS were evaluated with respect to the OPRM System in Safety Evaluation 98-0033. Because the ASF trip and OPRM bypass switch contacts interface to RPS with the existing NMS trip contacts, evaluation of other design basis accidents or anticipated operational transients are not necessary for making active the OPRM trip function in RPS. It is concluded that the installation of this OPRM System does not increase the probability of occurrence of an accident previously evaluated in the USAR. No accidents are found impacted by the installation of this modification and thus the likelihood of these accidents occurring would not be increased as a result of this modification. Making active the OPRM ASF trip function with bypass capability does not alter any assumptions previously made in evaluating the radiological consequences of accidents described in the USAR. The OPRM equipment being added is designed to not degrade the existing Neutron Monitoring System and the Reactor Protection System. The systems will still perform their intended safety functions and will respond in the same manner/time frame as the previous design. The OPRM equipment is designed to single failure criteria and is electrically isolated from equipment of different electrical divisions or non-1E equipment. The OPRM does not impact the protection features of the APRM or RPS that limit the radiological consequences of an accident. Therefore, the activation of the OPRM ASF trip function does not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The OPRM is considered to be a Protection System per NEDO-31960, Supplement 1. These systems are not accident initiators and are not degraded as a result of activation of the OPRM trip function. No new failure modes having a different effect are added as a result of either the OPRM equipment or the existing APRM and RPS equipment. The OPRM System does not provide any control function that could create a new type of accident. The ASF trip bypass switch provides the same conservative failsafe action that would occur for the failure of an existing component in a protective system channel.

III. No. The NRC has reviewed the ABB™ Option III OPRM Topical Report CENPD-400-P-A and found it acceptable for applications to the extent specified, and under the limitations delineated in the Topical Report and the associated NRC Safety Evaluation Report. The SER states that the staff found the ABB-CE OPRM design, as described in the Topical Report, acceptable. The staff does not intend to repeat its review of the matters found acceptable in CENPD-400-P-A when the report is referenced in license amendment submittals, except to ensure that the plant-specific issues identified in the NRC SER have been properly addressed. Plant specific design considerations, as it relates to the OPRM installation at Perry Nuclear Power Plant (PNPP), have been evaluated in this Safety Evaluation (SE) and found to be

acceptable and will be addressed in the license amendment submittal. As a result, the OPRM System, fully functional with an active ASF trip interfaced to the RPS, is not considered an Unreviewed Safety Question, and the ASF trip contact with bypass capability can be installed into the plant under the provisions of 10CFR50.59.

Safety Evaluation: 01-0016
Source Document: Condition Report (CR) 01-1017

Description of Change:

CR 01-1017 evaluates the "use-as-is" disposition for the newly discovered 1" crack on the top convolution of the Southeast 18" extraction steam metal expansion joint (bellows) in the 1N61-B0001A Main Condenser.

Summary:

I. No. The "use-as-is" disposition does not alter the function or create any new failure modes for the existing plant systems. The crack has been analyzed and found to be acceptable. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised. Therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is adversely affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. The catastrophic failure of the 18" expansion bellows within the Main Condenser is bounded by the more severe "Loss of Condenser Vacuum" and "Loss of Feedwater Heating" accidents. No new failures or new initiators are created as a result of this Use-As-Is disposition. This disposition does not result in any interface with any plant system, structure, or component in such a manner as to create the possibility of an accident or malfunction of equipment important to safety of a different type than any previously evaluated in the USAR. This activity 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. This disposition will not result in any new equipment failures and therefore this disposition will not create any new initiators or contributors for an event that could be considered a new accident. This disposition 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, this change will not create the possibility of a different type of accident or malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. Margins of safety associated with the design and operation of expansion bellows within the condensers is not explicitly addressed in the Perry Nuclear Power Plant (PNPP) Technical Specifications, USAR, Safety Evaluation Report (SER) or other related basis documents. However, the approximately 1" long crack, as found on the 18" diameter expansion bellows within the condenser, as well as the other pre-existing cracks on other bellows identified and evaluated for CR 94-0391, have been judged to have a marginal effect on the overall capabilities of the expansion bellows in the convolutions region such that it cannot be reasonably concluded that any margin has been changed (i.e., reduced). Hence, no safety margins are reduced as the result of the continued use of these expansion bellows. This disposition does not interface with or affect any plant systems, structures, or components in such a manner as to reduce any margins of safety. Consequently, this activity does not reduce the margin of safety, defined or implied, in the bases for any Technical Specification.

Safety Evaluation: 01-0017
Source Document: Simple Modification Request Form (SMRF) 01-5003, Revision 0

Description of Change:

SMRF 01-5003 removes the ASME test connection lines for 1N36-F0526B and C between the condenser shell coupling and the turbine shell coupling. A pipe cap will be welded onto the pipe stub at the turbine shell coupling.

Summary:

I. No. The design function of these ASME test lines is to perform the ASME turbine acceptance test. The ASME Test has not been performed since initial plant start-up and has no function during normal operation. The turbine acceptance test will not be performed in the future. There are no accidents identified in the USAR that are affected by the proposed change. Therefore, the proposed change will not increase the probability of an accident previously evaluated in the USAR.

No changes are being made to any assumptions or inputs previously made to assess dose consequences and no fission product barriers are being affected. The proposed change does not alter, degrade or prevent actions described or assumed for any accident discussed in the USAR. Therefore, the proposed change will not increase the radiological consequences of any accident evaluated in the USAR. The proposed change does not result in an impact on any other Structures, Systems or Components (SSCs). Therefore, the proposed change will not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the USAR.

Radiological dose rates in plant buildings will be unaffected by the proposed change. No changes are being made to any assumptions or inputs previously made to assess dose consequences and no fission product barriers are being affected. The proposed change will not alter any assumptions previously made in evaluating the radiological consequences of a malfunction of equipment important to safety. Therefore, the proposed change will not increase the radiological consequences of a malfunction of equipment important to safety previously evaluated in the USAR.

II. No. The design integrity of the Condensate/Main Steam systems will be increased as a result of removing/plugging these test lines. The proposed change does not add any credible failure modes or failure mechanisms for any SSC. The proposed change does not initiate any credible sequence of events resulting in any type of accident. Therefore, the proposed change does not create the possibility of an accident of a different type than any previously evaluated in the USAR. The proposed change will not degrade or prevent acceptable SSC performance, will not create any new failure modes or effects, and will not impact any existing failure modes or effects for any SSC. Therefore, the proposed change will not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the USAR.

III. No. The affected test lines are not described within the Technical Specifications and their removal and plugging will not decrease the integrity of the Condensate/Main Steam Systems. This change does not change the operation of the turbine, condenser, or related systems. Implementation of this change will not reduce the margin of safety associated with any plant system, structure, or component as defined in the bases for any Technical Specification. The affected test connections do not interface with, or affect any plant systems, structures, or components in such a manner as to reduce any margins of safety. Consequently, this activity does not reduce the margin of safety, defined or implied, in the bases for any Technical Specification.

Safety Evaluation: 01-0018
Source Document: Design Change Notice (DCN) 5910, Revision 0

Description of Change:

DCN 5910, Revision 0 documents the fact that the Division 3 High Pressure Core Spray (HPCS) Emergency Diesel Generator (EDG) rupture disc 1E22-D0012 is in the burst open condition as identified in Condition Report 00-3320.

Summary:

I. No. The proposed activity involves the rupture disc for the Div. 3 EDG. The rupture disc, EDG, and HPCS system are used to mitigate accidents/transients and are not considered initiators of any accidents/transients. In its current configuration, the open rupture disc is performing its design function of providing a safety related exhaust path for the Division 3 EDG. The Division 3 EDG will still meet its requirements for start times and power output. As such, the Division 3 EDG will perform its design function without degradation in response to any event. Since Division 3 EDG can perform its safety function, HPCS will be able to perform its design function of reactor vessel depressurization and water level inventory control such that the consequences of postulated accidents/transients remain unchanged. A failure modes and effects evaluation concluded that the open rupture disc, in its current configuration, does not increase the probability of malfunction of the Division 3 diesel generator or any other Structures, Systems or Components (SSCs) important to safety. Thus, the probability of occurrence or radiological consequences of accidents or equipment malfunctions has not been increased.

II. No. A failure modes and effects evaluation concluded that no new credible failure effects are created by the continued use of the rupture disc in the failed open position. All equipment important to safety will continue to function as it has previously and no new accident initiators have been created. Malfunctions of other equipment important to safety (including the diesel generator itself) were considered in the failure modes and effects discussion. With the rupture disc in its current configuration and the diesel operating, the possibility of a different type of a malfunction of the Division 3 EDG or other equipment important to safety is not credible. The original plant design had already considered the long-term operation of the EDG during an accident with the rupture disc in the open position. As such, no new type of failure mechanism has been created with the diesel operating in the proposed configuration. During standby conditions, opening of the rupture disc does create a hole in the exhaust piping. However, this hole does not create any new security risks to the plant than what previously existed. Furthermore, Foreign Material Exclusion (FME) concerns with the exhaust piping have been previously evaluated. As such, the addition of the rupture disc opening does not create a new type of FME concern to the diesel. Thus, the possibility of a different type of accident or equipment malfunction has not been created.

III. No. The rupture disc is described in the USAR and the SER. In this description, the rupture disc is required to open when the exhaust line is sufficiently deformed to restrict engine exhaust. Since the proposed activity will have the rupture disc already in the open position, it will be open if the exhaust line is deformed. The rupture disc in the open position was evaluated against the fuel consumption rate for the Division 3 diesel engine and required electrical output for both accident and normal operating conditions. The margin of safety associated with diesel fuel inventory and the EDG's ability to generate the required electrical output were not adversely effected. Thus, the margin of safety is not reduced.

Safety Evaluation: 01-0019
Source Document: Simple Modification Request Form (SMRF) 00-5022, Revision 1

Description of Change:

SMRF 00-5022 installs permanent lead shielding inside the drywell at six locations around the Reactor Recirculation (B33) suction and discharge risers. Permanent lead shielding is also installed inside the drywell at four locations around the Reactor Water Cleanup (G33) suction risers, horizontal piping and valve 1G33-F0102, and horizontal piping and valves 1G33-F0101 and 1G33-F0103. This shielding consists of high temperature lead wool blankets manufactured by Lancs Industry using covers made from Alpha Meritex fiberglass fabric manufactured by Alpha Associates, Inc. The shielding is supported by new and existing steel structures.

Summary:

I. No. Evaluations were performed on the materials used in the lead blankets (Alpha Meritex fiberglass fabric, Styles 8459-2-SS and 3259-2-SS, lead wool, and Kevlar thread). The materials were found to be capable of withstanding the environmental operating and accident temperatures and radiation conditions that the lead blankets will see during the remaining life of the plant including a design basis loss-of-coolant accident. These evaluations used test results performed on the fabric materials for other plants as well as manufacture's and industry data. The results were that, conservatively, no more than a 25% reduction in strength could be expected. This reduction was factored into the design. Field tests were performed to show that the grommets used to suspend the blankets were capable of supporting the blankets with a significant margin of safety. The new and existing structures that support the lead blankets were designed for deadweight, seismic, dynamic and jet impingement loads imposed on them from the lead blankets.

Evaluations determined that, as a result of this change, there are no adverse effects on fire protection, the Safe Shutdown Capability Report, drywell free volume, suppression pool pH, combustible gas control, the potential for the presence of lead in the suppression pool and reactor coolant, the thermal effect on drywell environmental and equipment qualification, the performance of emergency closed cooling systems, and revised accident source terms. The blankets are classified as noncombustible, however for conservatism, the mass of the blanket material is being added to the combustible loading in the USAR. Therefore the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR is not increased.

II. No. The permanent lead shielding blankets and support steel have been designed for all accident loading conditions defined for the drywell environment. The change will not degrade or adversely affect any of the existing non-safety related or safety related equipment or structures that interface with the permanent lead shielding. The reactor coolant pressure boundary and the drywell pressure response will not be adversely affected by this change. The structural performance of existing plant systems will still meet the design and licensing basis and, as such, can not create the possibility of an accident or malfunction of a different type.

III. No. The addition of the permanent lead shielding will not degrade the capability of the interfacing systems to mitigate the effects of postulated transients and accidents. Therefore, the margin of safety provided by the system design, as required by the Technical Specifications and their bases, is not reduced.

Safety Evaluation: 01-0021
Source Document: Perry Specification Technical Guidelines (PSTG), Revision 5, PIC 5

Description of Change:

PSTG, Revision 5, PIC 5 incorporates the following:

- Updated Minimum Alternate Reactor Pressure Vessel Flooding Pressure (MARFP) values based on Cycle 9 fuel parameters.
- Updated Maximum Core Uncovery Time Limit (MCUTL) values based on Cycle 9 fuel parameters.
- Added new Perry specific caution #9 and associated deviation sheet based on operator training feedback. Caution #9 is referenced in steps C2-1.3 and C4-1.3.
- Added override to transfer from reactor pressure control (RC/P) to reactor pressure vessel (RPV) Flooding on loss of RPV level and removed associated deviation sheet.
- Deleted "is not stabilized" from last paragraph of step reactor level control (RC/L)-7.2. Direction is given to stabilize RPV pressure in first part of step RC/L-7.2. Added deviation sheet "Stabilized and increasing" to justify.
- Changed the entry condition for the maximum normal main steam tunnel radiation levels in the secondary containment control guideline. The maximum normal main steam tunnel radiation levels used as an entry condition to the secondary containment control guideline are defined as the high alarm setpoint value of the Main Steam Line Radiation Monitors.
- Minor editorial changes which affect page layout.

This Safety Evaluation supersedes Safety Evaluation 01-0020.

Summary:

I. No. The revisions to MARFP and MCUTL are based on Cycle 9 Core Reload and performed using the formulas and methodologies previously used which are in accordance with Emergency Plan Guidelines (EPG)/Safety Analysis Guidelines (SAG) and those previously used in PSTG Revision 5. The changes for radiation monitor setpoints were previously evaluated in accordance with Plant Administrative Procedure PAP 1403. Therefore, the revisions to MARFP, MCUTL and main steam line radiation monitor setpoints will not increase the consequences of an accident previously evaluated. The other changes are entirely editorial. Therefore, the probability of and consequences of accidents and of equipment malfunctions previously evaluated in the USAR are not affected.

II. No. The PSTG provides the licensing bases for the Plant Emergency Instructions (PEI). The PEIs provide symptom based actions to take in response to an accident or transient to shutdown the reactor, restore and maintain adequate core cooling, and maintain containment integrity. Operation of the systems and components, as directed by the PEIs, occurs after the accident or transient has begun, and therefore, does not affect the possible initiators of any accidents or transients. All systems operated by the PSTG/PEIs are operated within the design bases of the system. Therefore, this change does not create the possibility of an accident or a different type of malfunction of equipment important to safety than any previously evaluated in the USAR.

III. No. The revisions to MARFP and MCUTL are based on Cycle 9 Core Reload and performed using the formulas and methodologies previously used. The changes for the main steam line radiation monitor setpoints were previously evaluated. This change affects operator actions taken post-accident to restore the plant to a safe condition and to reestablish Technical

Specification (TS) assumptions. These actions are taken in response to an accident which is beyond the USAR design, Technical Specification, Operational Requirements Manual (ORM) and Offsite Dose Calculation Manual (ODCM) assumed bases. The margin of safety as defined in the bases for any Technical specification is not reduced.

Safety Evaluation: 01-0022
Source Document: Perry Security Plan, Revision 30

Description of Change:

These changes to the Security Plan pertain to personnel access control measures for the protected and vital areas. These changes are considered to be safeguard information and, as a result, are managed in accordance with 10CFR73.21.

Summary:

I. No. This change is administrative only. The design and operation of the plant is unchanged. Accident analysis is unaffected. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not changed.

II. No. This change is administrative only. The design and operation of the plant is unchanged. Accident analysis is unaffected. Therefore, the possibility of an accident or malfunction of equipment of a different type than previously evaluated has not been created.

III. No. This change is administrative only. The design and operation of the plant is unchanged. Accident analysis is unaffected. Therefore, no margins of safety have been reduced.

Safety Evaluation: 01-0023
Source Document: Engineering Change Package (ECP) 01-8010, Revision 0

Description of Change:

ECP 01-8010 removes a section of 3/8" Outside Diameter (OD) ASME Turbine Test tubing between 1N62-F0536B and 1N62R0490. 1N62-F0536B is the isolation valve from the "B" Auxiliary Condenser (1N61-B0002B) to the ASME Turbine Test pressure point (1N62-R0490). This tubing line was last used during startup testing to measure the pressure from the "B" Auxiliary Condenser as part of the ASME Turbine Test. This test will not be performed again and this tubing has been capped and abandoned in place. Pressure test point 1N62-R0490 has also been abandoned in place.

Summary:

I. No. The ASME Turbine Test tubing is no longer being used and has no interactions with any systems, structures or components that are important to safety. Removing tubing, capping tubing, or abandoning tubing from the "B" Auxiliary Condenser to the temporary test point pressure recording connector does not affect the design function or operations of the Auxiliary Condenser. The isolation valve, 1N62-F0536B, is normally closed and will continue to perform as a pressure isolation boundary from the "B" Auxiliary Condenser. Because this activity has no affect on equipment important to safety, the probability of an accident or malfunction of equipment will not increase.

II. No. This ASME Turbine Test tubing is no longer used and capping/abandoning in place the tubing cannot create an accident or malfunction of equipment important to safety. The "B" Auxiliary Condenser will continue to perform as designed and will be isolated from the removed section of tubing. No new accidents or new equipment malfunctions are created by removing/capping off unused tubing.

III. No. The ASME Turbine Test tubing and pressure test point for the "B" Auxiliary Condenser is not described in the Technical Specifications (TS), TS Bases or Operational Requirements Manual (ORM). No margin of safety is associated with the ASME Turbine Test tubing and temporary pressure recording connections. No system, structure or component important to safety is affected by removing/capping/abandoning the test tubing. No margin of safety as described in the TS, TS Bases or ORM is affected.

Safety Evaluation: 01-0024
Source Document: Temporary Instruction (TXI) 0313, Revision 1, PIC 3

Description of Change:

TXI 0313 Revision 1, PIC 3 temporarily disables the automatic high vibration turbine trip signal for performance of testing.

Summary:

I. No. The accidents described in the USAR have been reviewed with the temporary disabling of the turbine high vibration trip in order to perform the TXI. It is concluded that this modification does not increase the probability of occurrence of an accident previously evaluated in the USAR. The basis for this determination is that the main turbine high vibration trip circuit performs operational protection of the main turbine in a non-safety capacity. This circuit does not interfere with higher priority protection systems such as turbine overspeed protection, which is an important characteristic in the generation of missile hazards and is not impacted by this activity. The accident analysis of USAR Section 15.2.3, Turbine Trip, takes into account a number of spurious turbine trips. Temporarily disabling the automatic high vibration turbine trip will not result in a turbine trip that would ultimately result in a reactor scram. Thus, with the implementation of the temporary automatic turbine trip disabled, the boundaries of the accident analysis will be less challenged and will not result in a reactor scram.

No reliance is made upon the main turbine vibration trip circuit to prevent or mitigate the effects of an accident described in the USAR. There is no increase in the probability of a turbine missile event, and therefore there would be no increase or change in radiological consequences. Thus, there are no radiological consequences that are different than those described in the USAR analysis, in particular USAR Section 3.5.1.3, Turbine Missiles. In the unlikely event that operator action results in failure of the main turbine, there is no increase in the probability of a low trajectory missile damaging a safety related system, structure, or component. There are no new failure effects created as a result of temporarily disabling the automatic high vibration turbine trip that would increase the probability of malfunction of equipment important to safety.

II. No. The disabling of the automatic main turbine vibration trip circuit does not create an accident of a different type. The accident analysis for turbine trip and missile protection remains unchanged. The consequences of failure of the high vibration trip to perform its function are bounded by the accident analysis in Chapters 3 and 15 of the USAR. The consequences of operator error are bounded by the accident analysis in Chapters 3 and 15 of the USAR. The result of failure of the main turbine vibration trip circuit or operator error does not change the outcome of the accident analysis of Chapters 3 and 15. The turbine trip circuitry does not impact or interface with any equipment that is important to safety.

III. No. The temporary disabling of the main turbine high vibration trip does not affect the turbine overspeed trip function, which is a part of the Technical Specifications. There are no safety related interfaces with the high vibration trip circuit. The turbine trip protection feature, provided by the vendor, is to be used at the discretion of the utility. There are no Technical Specification margins of safety associated with this circuit. Therefore, there is no reduction of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0026
Source Document: Temporary Modification 01-002, Revision 0

Description of Change:

The Temporary Modification was installed as a compensatory measure to restore operability of the Division 1 Emergency Diesel Generator (EDG) room ventilation system based on a degraded condition which had been identified on Condition Report 01-1801.

The TM disengaged the EDG room cooling inlet air damper (1M43-F0220A) from its actuator and maintained it fully open with a series of temporary, non-safety related mechanical clamps. In order to ensure operability of the equipment within the Division 1 EDG room, administrative controls placed a minimum limitation on the outside ambient temperature to ensure operability of the Division 1 EDG to perform as required. An operator action was required to manually start/stop the EDG Room Ventilation (M43) System fans in the event that the EDG room temperature falls below a prescribed limit in order to further ensure operability of the EDG.

Summary:

I. No. The changes implemented by the TM were evaluated to show that the function and the long-term reliability of the Division 1 EDG were maintained. Maintaining the damper in the full open position was shown not to interfere with the automatic starting ability of the ventilation system in response to an accident. It had been determined that implementation did not degrade, alter, or prevent actions described or assumed in an accident. The ventilation does not function as a radiological filtration system and does not have an impact on any radiation release barriers or personnel exposure. The use of non-safety related components serves a passive structural function and had concluded that the installed method of fixing the damper blades provides an equivalent level of component reliability as compared to the normal design. A new reliance on operator action during cold outdoor temperatures was not considered to increase the probability of equipment malfunction as a result of operator error. The bases were that the timing of the action was not expected to be required at the onset of an accident and if required, would likely occur well before or well after the occurrence of an accident. Therefore, implementation of the TM does not increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

II. No. The changes implemented by the TM evaluated and determined that the use of non-safety material could not create a new credible failure mode for the M43 System. The TM did not compromise seismic or separation criteria or equipment qualification criteria within the EDG room. Removal of the modulating capability of one damper does not make any event credible that was previously considered to be incredible. There were no new effects identified resulting from previously evaluated failure modes. Therefore, it had been concluded that implementation of the Temp Mod did not create a possibility for an accident or malfunction of a different type than previously evaluated in the safety analysis report.

III. No. The function of the M43 System is a support system for the EDG and is required to be operable such that the EDG and other supporting components remain within the Equipment Qualification analyzed temperatures. The reliance on operator action in lieu of automatic controls during cold weather conditions was evaluated and concluded that no new operator action would be required within 10 minutes of the onset of an accident. Therefore, it had been concluded that implementation of the TM did not reduce the margin of safety as defined in the bases for any Technical Specification.

Safety Evaluation: 01-0027
Source Document: USAR Change Request (CR) 01-030

Description of Change:

This USAR Change Request revises the titles for plant operations personnel (Shift Supervisor to Shift Manager, Supervising Operator to Reactor Operator, Shift Technical Advisor to Shift Engineer, Plant Attendant to Plant Assistant) to be consistent with other First Energy Nuclear Operating Company (FENOC) plants. This change also relocates Operational Requirements Manual (ORM) administrative requirements 7.2.1 Unit Staff and 7.2.3 Shift Technical Advisor to USAR Chapter 13.

There is no change to the duties, job descriptions, or responsibilities of these individuals, only their titles are being changed. Required shift staffing and manning levels are not affected. Qualification and training for these positions is not changed. Required shift staffing and manning levels is not changed. Information on staffing and manning levels is being relocated to the USAR to consolidate this information in one location.

Summary:

I. No. The proposed change to the USAR titles for plant operations personnel and the relocation of staffing requirements has no affect on USAR evaluated accidents. This USAR change does not represent any physical or process change that could impact Systems, Structures, or Components (SSC) or their operating parameters, or could cause a change to any SSC that would increase the likelihood or frequency of an accident. The proposed change has no affect on the radiological consequences of USAR evaluated accidents. Control Room staffing levels will remain unchanged. The proposed change will continue to ensure the proper monitoring and control of plant systems and 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. This USAR change does not involve any credible failure modes or mechanisms for any SSC, therefore the change could not initiate any sequence of events resulting in any type of accident. The USAR change does not represent any physical or process change that could impact SSC operating parameters. Therefore the possibility of an accident of a different type than any previously evaluated in the USAR is not created as a result of this USAR change. Therefore, the possibility of a different type of malfunction of equipment important to safety, than any previously evaluated in the USAR, is not created as a result of this USAR change.

III. No. The proposed change to the USAR titles for plant operations personnel and the relocation of staffing requirements does not affect the monitoring of plant parameters or the response time to manipulate plant equipment controls when required. The revised titles are different than the generic titles for these positions listed in Technical Specifications (TS). However, TS 5.2.1.a states "These requirements, including the plant specific titles of the personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the USAR." Therefore, there is no change required to the TS and this revision to the position titles listed in the USAR is permissible. Therefore, the margin of safety, as defined in the TS and Bases, Operational Requirements Manual, Operating License, Safety Evaluation Report (SER), and the USAR, is not reduced.

Safety Evaluation: 01-0029
Source Document: Design Change Packages (DCP) 99-5057 and 00-5036, both Revision 0

Description of Change:

The purpose of these modifications is to replace the obsolete air monitor control systems, 0M31-K0030A/B, used in the Radwaste Building Ventilation (M31) System Supply trains "A" and "B". The manufacturer of the installed control system no longer manufactures or supports repair of their pneumatic controller systems. The air monitor panels and transmitters 0M31-K0030A/B and 0M31-N0140A/B, which are pneumatic, are replaced by single units that are electronic and perform both the sensing and controlling functions.

Summary:

I. No. The M31 instrumentation that is changed by this modification is not an accident initiator. There are no accidents or transients that are relevant to the Radwaste Building Ventilation System. The function and performance of all Engineered Safety Feature (ESF) components is unaffected. The proposed change does not create any additional system interactions. The operation and function of equipment important to safety is not being compromised, therefore, no additional radiological consequences will occur as a result of the proposed change. No changes are being made to any assumptions or inputs that have previously been used to assess dose consequences. No equipment important to safety is affected by the proposed change and the radiological consequences of a malfunction of equipment important to safety are unchanged.

II. No. All requirements for digital control systems have been met. The proposed change does not impact the operation of any System, Structure, or Component (SSC). There is no change in the ability of any important to safety component to perform its function and therefore, the change does not create the possibility of a different type of accident or malfunction. No ESF SSCs are affected by the proposed change. Since no new system interactions are being created by the proposed change, no new equipment malfunctions are postulated. The proposed change does not adversely impact the operation of any other plant system and does not create any new failure modes. Therefore, the proposed change will not create an accident or malfunction of equipment important to safety of a different type than evaluated in the USAR.

III. No. The proposed changes do not revise the current design and licensing basis definition of any SSC. SSC redundancy and reliability are unchanged, as are the methods and procedures with which any SSC is operated or tested. Since the proposed changes utilize digital equipment, all NRC Generic Letter 95-02 requirements have been met. Therefore, the proposed changes can have no impact on the margins of safety associated with SSCs and will 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), Plant Operating Procedures (POP), Plant Process Control Program (PCP), Technical Specifications or as defined in the bases for any Technical Specification.