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June 2, 2000

SVP-00-073

U. S. Nuclear Regulatory Commission
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Washington, D C 20555

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Summary Report of Changes, Tests, and Experiments Completed

In accordance with 10 CFR 50.59 and 10 CFR 50.71(e), we are forwarding Quad Cities Nuclear Power Station's Quarterly Summary Safety Evaluation Report. These safety evaluations cover the period of February 1, 2000 through April 30, 2000.

Should you have any questions concerning this letter, please contact Mr. C. C. Peterson at (309) 654-2241, extension 3609.

Respectfully,

Joel P. Dimmette, Jr.
Site Vice President
Quad Cities Nuclear Power Station

Attachment:
Summary Report of Changes, Tests, and Experiments Completed

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

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ATTACHMENT A

SUMMARY REPORT OF CHANGES, TESTS, AND
EXPERIMENTS COMPLETED

FEBRUARY 1, 2000 to APRIL 30, 2000

SVP-00-073

SAFETY EVALUATION INDEX

SE-98-081
SE-99-026
SE-99-029
SE-99-032
SE-99-050
SE-99-053
SE-99-066
SE-99-070
SE-99-074
SE-99-076
SE-99-080
SE-99-094
SE-99-103
SE-99-106
SE-99-109
SE-99-120
SE-99-126
SE-00-006
SE-00-008
SE-00-009
SE-00-016
SE-00-020
SE-00-021
SE-00-022
SE-00-023
SE-00-024
SE-00-025
SE-00-026
SE-00-027
SE-00-028
SE-00-029
SE-00-030
SE-00-032
SE-00-034
SE-00-037
SE-00-040

SS-H-99-0002
SS-H-99-0117
SS-H-99-0126
SS-H-99-0134
SS-H-99-0161
SS-H-99-0170
SS-H-99-0200
SS-H-99-0213
SS-H-99-0224
SS-H-99-0228
SS-H-99-0239
SS-H-00-0005
SS-H-00-0006
SS-H-00-0009
SS-H-00-0015
SS-H-00-0016
SS-H-00-0019
SS-H-00-0020
SS-H-00-0027
SS-H-00-0031
SS-H-00-0032
SS-H-00-0033
SS-H-00-0034
SS-H-00-0036
SS-H-00-0042
SS-H-00-0045
SS-H-00-0048
SS-H-00-0069

Tracking No. SE-98-081
Activity No. DCP 9700147

DESCRIPTION:

Change the span of the Feed Water (FW) flow transmitters to provide proper flow signal as determined by test results.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new Feedwater (FW) flow span calculation and uncertainty analyses were performed to ensure the FW flow input to the core thermal power meets accuracy requirements. Since the flow input will be accurately measured, the assumed starting point of the accident/transient analyses will be within the limits of this analysis. Therefore, this change will not increase the likelihood, predicted frequency, or consequences of an accident or malfunction previously evaluated in the SAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function and the configuration of the transmitters will remain the same during all operating modes and accident conditions. Re-spanning of the transmitters only changes the output of the transmitters for a given input and does not change the method of operation or function of the transmitters. The failure mode of the transmitter is not affected by this change and no new failure mode is introduced.

3. The margin of safety, as described in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety.

Tracking No. SE-99-026
Activity No. DCP 9900022

DESCRIPTION:

This Safety Evaluation is prepared for DCP 9900022 Revision 1. This Safety Evaluation supersedes Safety Evaluation SE-99-007.

The following changes are provided for the Unit 2 generator protective-relaying scheme in DCP 9900022. These changes will result in an increased reliability & stability of Quad Cities Units operations and interconnected transmission lines.

This DCP will install a new control switch in the relay house at Quad Cities Station for locally arming the Unit 2 Stability trips. This switch will be placed in the on position at the direction of bulk power operations based on breaker and/or disconnect switch positions at Nelson (TSS 155) and

Northwestern Steel and Wire. This control switch is referred to as the MAST on the schematic diagram.

Description of the trip scheme for the Unit 2 Stability Trips:

Revise the existing protection scheme, which trips Unit 2 as follows:

1) Line 0404 is open and Line 0403 is open and Quad Cities Station power output is above 1250 MWe and both Unit 1 and Unit 2 are in service.

2) Line 15504 at Nelson (TSS 155) is open and Line 0403 is open and Quad Cities Station power output is above 1250 MWe and both Unit 1 and Unit 2 are in service.

3) The MAST switch is in the On position and Line 0403 is open and Quad Cities Station power output is above 1250 MWe and both Unit 1 and Unit 2 are in service.

4) Line 15501 at Nelson (TSS 155) is open and Line 15502 at Nelson (TSS 155) is open and both Quad Cities Unit 1 and Unit 2 are in service.

5) Line 0404 is open and a multiphase fault occurs on Line 0403 and Quad Cities Unit 2 is in service.

6) Line 15504 at Nelson (TSS 155) is open and a multiphase fault occurs on Line 0403 and Quad Cities Unit 2 is in service.

7) The MAST switch is in the On position and a multiphase fault occurs on Line 0403 and Quad Cities Unit 2 is in service.

The trip scheme is also being revised by DCP 9900022, revision 1 to trip Unit 2 if both Units are in service and Line 0402 or Line 0403 are open and a multiphase fault occurs on Line 0405. This trip scheme was originally to trip Unit 1 but is being revised to trip Unit 2.

Other changes:

Test switches will be installed in the generator protective-relaying scheme of Unit 2. The purpose of the test switch is to isolate the logic circuits from the generator protective relaying scheme. This will facilitate testing during power operation, and testing of the logic circuits for adequacy prior to return to service/op authorized. The test switch also facilitates the installation of the logic when the generator is on line.

The affect of these changes is that it will maintain offsite power stability by tripping Unit 2 for analyzed conditions that would have otherwise resulted in an unstable offsite power system that might result in the loss of both Quad Cities Units along with other Units in the area resulting in a wide scale blackout.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the activity will not cause the generator protective scheme to operate outside its design or testing limits. Addition of the logic to the existing generator protective

schemes will not result in a change to the generator protective scheme interface in a way that would increase the likelihood of an accident. The trip scheme is being modified to protect the integrity of offsite power and therefore, maintain a reliable source of offsite power to the plant. Therefore, the probability of occurrence of a malfunction of equipment important to safety has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the activity does not change the failure mode or create new failure modes for the generator protective schemes. Additional logic has been added to the existing generator protective schemes. The failure modes associated with the new logic have been encompassed by the failure modes of the existing logic within the generator protective schemes. There are no changes of the failure modes, frequency, class or acceptance criteria of the accidents. There are no new accident initiators or failure modes as a result of this change. Hence, the possibility of a different type of malfunction of equipment important to safety than previously evaluated has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no reduction in Margin of Safety. The activity is being implemented to maintain the integrity of the offsite distribution system and thus ensure the availability of two independent sources of power are available. The activity will support the Technical Specification basis of ensuring that no anticipated single event can cause a simultaneous outage of all the offsite power sources during units operation, accident, or adverse environmental conditions.

Tracking No. SE-99-029
Activity No. DCP 9600436; UFSAR-99-R6-001

DESCRIPTION:

The activity being evaluated is the installation and operation of new Reactor Water Cleanup (RWCU) Isolation Actuation Instrumentation for High Area temperatures in the vicinity of existing RWCU high-energy piping. This design provides additional redundancy to the existing RWCU Isolation logic. The redundancy is provided by incorporating room temperature trip setpoints into the Group III Primary Containment Isolation System (PCIS) where RWCU water temperature exceeds 200 degrees F and pressure is 275 psig. The UFSAR has been revised to reflect this isolation design.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this modification installs an additional automatic isolation system on the Unit 2 Reactor Water Cleanup (RWCU) System which will close RWCU isolation valves MO 2-1201-2, MO 2-1201-5, and MO 2-1201-80 on direct indication of high temperature in areas containing high energy RWCU piping outside of primary containment. Portions of the existing RWCU leak detection system will be utilized in the new automatic isolation

logic including area Resistance Temperature Detectors (RTDs) and some power and instrumentation cables.

The basic functions of the RWCU Automatic Isolation System are to initiate an automatic isolation of RWCU and to provide alarm indications in the Main Control Room (MCR) of high temperatures and system isolation. The areas containing high energy (greater than 200 degrees F and 275 psig) RWCU piping outside of primary containment include the RWCU Heat Exchanger Rooms, the Phase Separator Tank Area, the "D" Heater Bay Area, and the Main Steam Isolation Valve (MSIV) room. This modification provides a safety-related one-out-of-two automatic isolation logic. The RWCU Automatic Isolation System is classified as safety-related, Seismic Category I. The leak detection system will be upgraded to a safety-related, environmentally qualified, and seismic system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the design does not alter piping system flow paths and does not have an impact on the integrity of the RWCU piping system. The function of the RWCU system to isolate in the event of a loss of coolant accident causing the low reactor level setpoint (+8") to be reached has not been modified by this plant design change. Adding an automatic isolation of the RWCU system based on direct indication of high temperature in the RWCU phase separator room, heat exchanger room, "D" Heater Bay Area, and MSIV room does not have an impact on the integrity of the RWCU piping system. As a result, the probability of an accident to occur is not increased by this plant design change.

   The RWCU area high temperature isolation system has been designed to be reliable, fail safe, single failure proof, and will function independently from other plant controls and instrumentation. The system has been designed with two redundant trains of instrumentation, both of which will cause isolation of the RWCU system if high temperature isolation setpoints are reached.

   This facility change does not change redundancy and performance requirements of any existing equipment important to safety. Therefore, design is still single failure proof and malfunctions would still have the same result. Therefore, there is no increase in the probability of a malfunction of equipment important to safety previously evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because TS 3/4.2.A and corresponding tables contain Isolation Actuation Instrumentation which function to isolate systems to prevent or limit the release of fission products from the reactor coolant system boundary. This design change employs an existing Group 1 Main Steam Tunnel Temperature isolation signal to provide an RWCU (Group 3) isolation. The existing Group I Main Steam Tunnel signal/setpoint is not changed. The existing Group 3 RWCU System Isolation signals and setpoints are not changed. This modification also installs safety-related instrumentation and logic for detecting RWCU pipe breaks in areas containing RWCU high-energy piping. The addition of these isolation signals makes the plant more conservative than previously designed and are not in conflict with the existing requirements.

Tracking No. SE-99-032
Activity No. DCP 9700390

DESCRIPTION:

DCP 9700390 will add an additional means of disconnection for MO 2-1001-47 to meet Appendix R requirements. The disconnect switch, junction box, conduits/supports, and cables will be installed in the turbine building, outside of the Unit 2 D heater bay to allow the power to MO 2-1001-47 to be disconnected to prevent a post-fire spurious opening of this MOV to maintain the high/low pressure interface.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because installation of the new disconnect switch, junction box, terminal blocks, conduits, supports, and cables cannot cause any of the accidents or transients analyzed in the UFSAR. A failure associated with any of the new components would only affect MO 2-1001-47. Should the components fail such that power is lost to the MOV, the valve's PCI function would be lost. This, however, is enveloped by the loss of the 250 VDC battery and is addressed in the UFSAR accident analysis (i.e. upstream MO 2-1001-50 fed from a 480 VAC MCC would still be available to close). Should the components fail such that the valve would open spuriously, the inboard SDC valve would still be closed to prevent low pressure piping from being exposed to damaging high reactor pressures. The addition of this equipment does not affect initiating events for the accidents/transients discussed in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the power circuit of MO 2-1001-47 will be modified per applicable safety-related and seismic requirements. In the unlikely event that these components fail and cause a short circuit, the feed breaker to MO 2-1001-47 would trip making that valve inoperable. However, MO 2-1001-50 (fed from 480 VAC MCC 28-1B), which is upstream of MO 2-1001-47, would still be available to close upon a Group 2 isolation signal. Should the modified power circuit fail while MO 2-1001-47 is closed, it could still be manually opened to allow the use of shutdown cooling. The opening of this valve for shut down cooling is not an automatic action and is not considered a safety function. The worst case scenario associated with the modified power circuit is a loss of function of MO 2-1001-47. Since this is enveloped by the loss of the Unit 2 250 VDC battery, the possibility of an accident/transient, or a malfunction of a different type than previously evaluated will not be created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specifications require that MO 2-1001-47 automatically isolates upon a Group 2 isolation signal. When the reactor pressure is less than 100 psig, the disconnect switch will be closed and power will be available at the MOV motor to perform this function. At pressures above 100 psig, the valve will be closed and the disconnect switch opened. Since the valve is already closed, the PCI requirement is met.

The Technical Specifications also require that MO 2-1001-47 be able to be opened to allow the use of the shutdown cooling mode of RHR either remotely or locally. Since the valve can always be opened with the handwheel regardless of power availability, this requirement is also met.

Tracking No. SE-99-050
Activity No. DCP 9700350

DESCRIPTION:

Modify the Turbine Trip Logic for the Thrust Bearing Wear Detector (TBWD)/Low Bearing Oil Header Pressure from a one-out-of-one-logic to a two-out-of-two-logic. New pressure switches, isolation valves and calibration tees have been installed on the TBWD junction box located on the Unit 2 Main Turbine.

Procedures will be revised to document the addition of the pressure switches. The DCP will change the Equipment Part Numbers (EPN) from PS 2-5600-11 and PS 2-5600-12 to PS 2-5600-11A(B) and PS 2-5600-12A(B). Additionally procedures QCIPM 5600-1, QCIPM-2, QCIPM 5600-3 and QCIPM 5610-39 will be revised to reflect the changes to the Turbine TBWD component location drawings.

"For Record" changes will be incorporated on Drawing M-2022, Sheet 5 & 6. These drawings contain pressure switch logic and numbering which was revised by DCPs 9700345, 9700346, 9700347 and 9700348, but not incorporated onto the drawings. The pressure switch logic for the "for record" changes has already been evaluated under Safety Evaluation SE-98-100 and SE-98-147.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because reduction of turbine trips caused by false actuation is the objective of this design change, with the ultimate goal of reducing reactor scrams. By placing two pressure switches in series (electrically) to monitor TBWD/Low Bearing Oil Header pressure, the probability of a false activation of a turbine trip is reduced.

   The turbine trip does not have any safety consequences that are directly related to the off-site dose. One of the consequences of a turbine trip is the Reactor SCRAM. This is designed to minimize the release of effluent off-site. Since this design change does not alter any system or component that is designed to mitigate the consequences of the turbine trip (such as Reactor SCRAM), consequences of the turbine trip will remain unaffected. The installation/connection of the new pressure switches in series (electrically) with the existing pressure switches for the TBWD/Low Bearing Oil Header pressure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because pressure switch installation and control logic change initiated by this design change is all within the Turbine Trip logic. A review has found these changes to be within the boundaries of the existing turbine trip

component's failure modes. Therefore, no new accidents are being introduced by this design change that have not been previously analyzed.

The failure modes of the pressure switches have been addressed and shown that there are no adverse impacts to the Turbine trip logic. This is due to the independence of the Turbine Trip logic signals and the reliability of the pressure switches.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety.

Tracking No. SE-99-053
Activity No. UFSAR-99-R6-083 and UFSAR-99-R6-088

DESCRIPTION:

The UFSAR incorrectly states the design pressure of the Off-Gas system and Krypton hold up time for the charcoal absorbers in the Off-Gas system. The UFSAR is being revised to state the correct design pressure and Krypton hold up time as specified in the Off-Gas system design documents. The design pressure is being changed from 300 psig to 350 psig. And the krypton holdup time is being changed from 10.4 hours to 19.4 hours.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the changes to the UFSAR are being made to reflect the original design requirements to the system. These changes cannot cause any of the initiating events for any accident or transient scenarios or increase the probability of failure of equipment. No new Off-Gas system interfaces or leakage paths are created that could increase the consequences of an accident. Containment barriers are not compromised.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the design functions, design configuration, interfaces and isolation capabilities of the Off-Gas system are not changed. Therefore, no accidents or malfunctions of a different type are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes to the UFSAR will not change the maximum limit of the sum of activities of the noble gases measured prior to the offgas holdup line specified in Technical Specification 3 / 4.8.I, Main Condenser Off-Gas.

Tracking No. SE-99-066
Activity No. DCP 9900131; UFSAR-99-R6-012

DESCRIPTION:

DCP 9900131 will reconfigure the Hydrogen Water Chemistry Monitoring Skid (2202-85B) and associated equipment to provide for long-term monitoring of future noble metal injections. The new skid will have a constant flow of primary system fluid from the Reactor Water Cleanup System (RWCU).

Another activity in this design change is to remove/disable the recirculating water dissolved oxygen (O_2) concentration High/Low annunciator at the 902-53 panel in the control room. The scope of the required work will include de-terminating and removing wiring from the 902-53 panel and replacing the engraved tile on the annunciator display with a blank tile. No system trips or other alarm functions are affected.

Also being evaluated are the changes required for the current revisions of station procedures, which will be revised to reflect the removal of the O_2 concentration alarm at the 902-53 panel. Other required editorial or format changes may also be incorporated into these procedures.

Finally, this safety evaluation also revises UFSAR Section 5.4.3.4 so that the new method of monitoring HWC performance on Unit Two is included in the UFSAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the intent of the Hydrogen Water Chemistry (HWC) system is to prevent and/or retard Inter Granular Stress Corrosion Cracking (IGSCC) in pressure bounding vessels and components such as the Recirc system piping. The installation of the Nobel Metal Injection system and the associated monitoring equipment will enhance the HWC system by providing more effective utilization of injected hydrogen and providing a more accurate method of measuring the effectiveness of the HWC system. The new method of measuring the effectiveness of the HWC system does not utilize the current O_2 concentration limits. Therefore, the removal of the O_2 concentration alarm and the changes to the Hydrogen Water Chemistry Monitoring Skid do not affect the ability of the HWC system to combat IGSCC.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the skid and its components are not located near any equipment important to safety. The components do not need to be seismically restrained and the design change does not have any EQ concerns. The pressure and temperature ratings of the new piping are appropriate for the application thus there is no increase in failure probability. Also should a failure occur, the new piping can be isolated by closing the RWCU containment isolation valves. Thus, no new un-isolable leak path is created.

The function of the O_2 concentration alarm was to alert control room operators of a concentration level outside the specified band of operation. This alarm would prompt Operations to monitor and assess the HWC system and make adjustments as required to ensure HWC system was operating within the established parameters to effectively combat IGSCC. Since the installation of the Noble Metal Injection system, this alarm function is no longer required to assess the effectiveness of the HWC system. Since this alarm function only notified operators when the HWC system was operating outside parameters and these parameters are no longer required, the removal of this alarm function will not create the possibility of an accident or transient of a different type than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification requirements, associated action items, associated surveillances, or bases are affected by this design change.

Tracking No. SE-99-070
Activity No. DCP 9900061; UFSAR-99-R6-013; CRN 99-07

DESCRIPTION:

The DCP revises the interface point between the Safe Shutdown Makeup Pump (SSMP) system and the High Pressure Coolant Injection (HPCI) system for Unit 2. The SSMP tie-in point is currently between HPCI valves 2-2301-7 and 2-2301-8. Valve 2-2301-7 is a check valve and valve 2-2301-8 is motor operated. The revised location will be downstream of valve 2-2301-7 and before the HPCI injection piping connection to the feedwater piping. The UFSAR has been revised to reflect the Unit 2 SSMP tie-in to HPCI. The Safe Shutdown Report has been revised to remove the requirement to close (or verify close) valve 2-2301-8 or 2-2301-9 prior to using the SSMP system to mitigate certain Appendix R fire scenarios.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because relocating the SSMP tie-in point from the upstream side of check valve 2-2301-7 to the downstream side will not adversely alter any starting (precursor) conditions required for an Appendix R fire. This DCP does not add any combustible materials to the plant. The SSMP system is used to mitigate the consequences of certain Appendix R fire scenarios. Relocating the SSMP tie-in location to the downstream side of check valve 2-2301-7 will remove the burden on the operations staff to close (or verify closed) valve 2-2301-8 or 2-2301-9 prior to using the SSMP system. Thus, the operations staff will be more effective in controlling the plant during certain Appendix R fire scenarios.

   A LOCA inside containment is not affected. The HPCI system mitigates the accident by injecting water into the vessel via the feedwater piping located outside of containment. The HPCI system is used to mitigate the consequences of a LOCA inside containment. The DCP relocates the SSMP tie-point on the HPCI system. While this adds a "tee" to the HPCI system, the overall affect on HPCI's hydraulic performance is negligible. The HPCI system will still perform as designed and thus, will not adversely impact the consequences associated with a LOCA inside containment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the piping reroute is minor in that the new tie-in location is a few feet from the existing location. The SSMP and HPCI systems will perform the same function with the same flows, pressures and temperatures as before. The valves used to isolate flow between systems are more than capable of safely handling the design pressures, flows and temperatures associated with the revised configuration. Therefore, a different type of equipment malfunction will not be created by this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the additional tie-in location on the HPCI system will not adversely impact the system's hydraulic characteristics. The added head loss is negligible compared to the overall system characteristics. Thus, margin of safety is not reduced.

   The minor piping reroute has been evaluated for its effect on system hydraulics. It has been determined that the SSMP system can still deliver at least 400 gpm against a head pressure (reactor vessel pressure) greater than 1120 psig. Therefore, the margin of safety is not reduced.

   The purpose of this DCP is to eliminate the dependency on operator action under certain Appendix R fire scenarios. By reconfiguring the Unit 2 SSMP tie-in to the HPCI system, plant personnel will no longer be required to close (or verify closed) valve 2-2301-8 or 2-2301-9 prior to using the SSMP system to mitigate consequences of certain Appendix R fire scenarios. Therefore, margin of safety is not reduced.

Tracking No. SE-99-074
Activity No. UFSAR-99-R6-017; DCP 9700262, Phase 1

DESCRIPTION:

The activity being evaluated is the installation of an Oscillation Power Range Monitor (OPRM), Class 1E, microprocessor-based system and required additional supporting instrumentation. The OPRM system utilizes the existing Local Power Range Monitor (LPRM) signals to detect reactor core thermal hydraulic instabilities using Period Based, Amplitude Based, and Rate of Growth algorithms. The OPRM system monitors the output of all available LPRMs, in parallel with the existing Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) functions. The OPRM system is designed to initiate a reactor scram via existing Reactor Protection System (RPS) trip logic on detection of core power oscillations in response to the thermal hydraulic instability that may occur under high power, low core flow conditions. The existing RPS system functions will be augmented by addition of the suppress function of thermal-hydraulic oscillations in the reactor core. The OPRM system, when fully operational, will be automatically enabled at high power and low recirculation flow. However, the Reactor Protection System (RPS) trip inputs from the OPRM modules will be bypassed during the tune-up and monitoring phase of the OPRM operation. The UFSAR has been revised which adds a description of this OPRM system Phase 1 installation.

The scope of this safety evaluation is limited to the first phase. The second phase will be implemented under a separate DCN and safety evaluation. The system will function as a real-time monitor of the core stability for an anticipated duration of one full operating cycle with the output to the RPS bypassed. The OPRM will provide indication of core stability while undergoing system tune-up and startup trials as appropriate for a first time use component for a safety-related application. The Oscillation Power Range Monitor (OPRM)'s display outputs can be used to enhance the operators' awareness and ability to detect the onset of core instabilities. No credit is taken for this monitoring function while the plant continues to be operated under the current Technical Specifications (TS), procedures, and operating limitations committed to in response to NRC and industry concerns. The OPRM System consists of four redundant OPRM trip channels, two per RPS Trip System. Each channel consists of two OPRM modules, each module pair providing trip outputs to the existing corresponding RPS channel. Each OPRM module receives input from the (20 or 21) LPRMs from the associated APRM page or LPRM Group page and the additional 20 or 21 LPRMs' inputs from the companion OPRM module.

Each OPRM module also receives input from the APRM power and Reactor Recirculation Drive flow signals to automatically enable the trip function of the OPRM module. The enabled region is conservatively large to provide reasonable assurance that no oscillations will occur outside this region under expected operating conditions.

The OPRM Modules are being installed in the Power Range Neutron Monitoring System (PRNMS) Panel in the Control Room. The modules are being inserted into card slots that are now vacant or will be vacated by installation of dual output Voltage Regulator cards, one in place of the current two, in the APRM pages. In the APRM and LPRM Group pages, the OPRM receives signals from the LPRM cards. In addition, it receives an average power signal from an APRM and a Reactor Recirculation total flow signal from the Flow Unit, which is used to enable the OPRM trip functions when the APRM power is high and core flow is low.

The OPRM equipment receives its source of power from redesigned replacement power supplies that are associated with the APRM or LPRM Group page where the OPRM is mounted. Additional components (i.e. - annunciator relays, trip relays, and digital isolators) are being mounted on DIN rails in the back of the Ion Chamber Power Supply (ICPS) page associated with the mounting location of the OPRM.

The hardware changes being made in phase 1 of this modification are as follows:

a. Remove two existing voltage regulators in each APRM page (total 12 voltage regulators) of Panel 902-37. Replace six of these voltage regulators, one in each APRM page of panel 902-37, with new dual voltage regulators (2-0756-VR-1A through 6A). Install eight OPRM signal processing modules (2-0756-0PRM 1 through 8) in the location of the other six voltage regulators that were removed and in two spare locations in the LPRM Group Pages.

b. Install an Automatic Suppression Function (ASF) Trip Relay Assembly, an OPRM Annunciator Relay Assembly and two Digital Isolation Blocks in each APRM and LPRM Group Page in Panel 902-37.

c. Replace eight existing power supplies powering RBM, APRM, and LPRM Group Pages with new bulk power supplies (2-0756-PS 11 through 14 and 2-0756-PS 17 through 20) in Panel 902-37.

d. Install four new analog signal isolators (2-0756-AI-3 & 4 and 2-0756-AI-3T & 4T) in Panel 902-37. (The 2 isolators with the "T" designation will be removed as part of phase 2 of the installation when the recirc flow units are replaced.)

e. Install new instrumentation and control cables inside Panel 902-37 and new control cables between 902-34 and 902-37. These cables will be placed in RPS divisional instrumentation and control bundles as required. The cables from different RPS divisions will be isolated by using qualified sleeving or conduit to protect them from failures in other RPS divisions.

f. Replace the 12A, 120VAC Bulk Power Supply input fuses with 5A fuses. Replace the two existing 20A Bussmann type MIN fuses for the incoming 120VAC RPS power at panel 902-37 with 20A Gould type ATM fuses. Add RBM 7 and RBM 8 isolation fuses at panel 902-37.

g. Remove two Flow Converter/RMCS interposing relays (2-0756-101A & 101B).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the OPRM is being installed to provide automatic defense-in-depth for conformance with the reactor core protection General Design Criteria 10 and 12 as required per NRC Bulletin 88-07 Supplement 1 and Generic Letter 94-02. The operation and efficacy of the OPRM system are documented in Generic Topical Report, CENPD-400-P-A (Rev. 1) and Licensing Topical Report NEDO-32465-A. The NRC has prepared an SER and provided a letter of acceptance to the BWROG for each of these topical reports. Operation in the interim period, with the system installed, but not fully functional, is covered under the existing Interim Corrective Actions. The procedures that implement these corrective actions will be reviewed and modified as determined appropriate when the OPRM automatic suppression function is enabled at the end of the functional tune-up period.

   All the accidents listed in this safety evaluation are similar in that they rely on the APRM System function for RPS actuation (i.e., high neutron flux scram). The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux changes. The APRM Neutron FluxHigh Function is capable of generating a trip signal to prevent fuel damage. The decrease in feedwater temperature event would cause an increase in reactor power at a moderate rate, resulting in a scram if operator action was not taken to keep the power below the scram setpoint. For the inadvertent MSIV closure event, reactor scram is initiated on 10% closure of the main steam isolation valves (MSIVs) and the APRM Neutron FluxHigh Function is relied upon as a back-up to initiate the scram. For sizing the main steam line safety valves, it is conservatively assumed that the direct reactor scram (based on MSIV position switches) fails, and the back-up scram due to high neutron flux shuts down the reactor. The high flux trip, along with the safety/relief valves, limit the peak reactor vessel pressure to less than the ASME Code limits. The Recirculation Loop Flow Controller Failure Event (pump runup) is terminated by the high neutron flux trip. The control rod drop accident (CRDA) analysis in Chapter 15 takes credit for the APRM Fixed Neutron FluxHigh Function to terminate the CRDA.

   The OPRM installation does not cause a change to the existing APRM and RPS design or trip philosophy but only augments the existing APRM trip outputs (after installation of phase

2 of the modification) such that the OPRM trip will logically function in the same manner as the existing APRM trips. Impact to the loading of the LPRM, APRM power and Reactor Recirculation flow circuit interfaces have been evaluated to ensure the OPRM does not load down the existing circuits and the power sources can handle the additional load of the OPRM modules. The OPRM system is designed to detect core power oscillations in response to the thermal hydraulic instability that can occur under high power, low core flow during any condition of normal operation and initiate a scram via the existing RPS trip circuit (input to RPS trip logic disabled during the tune-up phase). The installation of the OPRM does not cause a change to the APRM or RPS design or trip philosophy, and as a result, there is no impact on the plant-specific design basis accident analyses, and conclusions from those analyses remain valid. Based on a review of the SAR Sections associated with the all accidents/transients listed in this safety evaluation, these accidents can not be initiated by the equipment involved in the modification.

This modification does not degrade the performance or operation of APRM equipment associated with the mitigation of these accidents. The single failure tolerant design of the APRM assures that the APRM protective function is not affected by a worst-case OPRM failure.

Since the addition of the OPRM equipment into the PRNMS has not increased the equipment malfunction probability or consequences of the PRNMS equipment, there has been no change to the probability of occurrence or consequences of any accident or transient, as previously evaluated in the SAR.

The OPRM is designed with signal isolation and buffering to ensure there are no safety impacts to existing plant systems. The OPRM function does not require an upgrade to any interfacing or associated systems. Impact to the loading of the LPRM, APRM Power, and Reactor Recirculation flow circuit interfaces have been evaluated to ensure the OPRM does not load down the existing circuits and the power sources can handle the additional load of the OPRM modules. However, electrical faults in the OPRM module may affect interfacing components associated with inputs and outputs of the OPRM. But, due to the single failure tolerant design of the APRM channels, the APRM protective function is not affected by a worst-case OPRM failure. The worst possible outcome of a serious common failure is APRM channel trip resulting in an RPS half-scram or loss of no more than one APRM channel. The impact of electrical faults in other components installed by this design change on the APRM channels has not changed or does not affect their protective capabilities.

This change does not increase the probability of a malfunction of equipment important to safety as previously evaluated in the SAR. The new OPRM equipment is designed and installed to not degrade the existing APRM, LPRM, and RPS systems. These systems will still perform all of their intended functions. The new equipment is tested and installed to the same or better environmental and seismic envelopes as the existing systems. The new equipment has been designed and tested for EMI requirements which further assures correct operation of the existing equipment. The new system has been designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The electrical loading is within the capability of the existing power sources and the heat loads are within the capability of existing cooling systems. With the OPRM's trip output to the RPS deactivated, any inadvertent trip of the OPRM during the initial tune-up period will not impact the RPS functions.

Since the OPRM is a stand-alone system, the consequences of an APRM malfunction will not be increased due to the installation or operation of the OPRM system, and the plant safety and protection of the reactor core will be improved overall.

Therefore, there is no increase in the probability of occurrence or consequences of a malfunction of equipment important to safety previously evaluated in the SAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the proper operation of the OPRMs requires extensive functional interfacing with the existing systems such as PRNMS, RPS and the main annunciator. Electrical faults in the OPRM module may affect interfacing components associated with inputs and outputs of the OPRMs. However, the single failure tolerant design of the APRM assures that the APRM protective function is not affected by a worst-case OPRM failure. The worst possible outcome of a serious common failure of any LPRM group is an APRM channel trip, resulting in an RPS half-scram. In other cases, the impact of electrical faults from the OPRM on associated circuits has not changed or will cause loss of no more than one APRM channel. Therefore, there is no failure generated in the OPRM system that can prevent the APRM or RPS circuits from responding to the possible accidents evaluated in the SAR. The installation of the OPRM equipment does not create the possibility of a different type of malfunction of equipment important to safety than previously evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:

TS 2.0	Safety Limits and Limiting Safety System Settings
TS 2.2.A	Reactor Protection System Instrumentation Setpoints
TS 3/4.11.C	Minimum Critical Power Ratio

There has been no reduction in the margin of safety as defined in the basis for the TS. The OPRM system does not negatively impact the existing APRM system. As a result, the margins in the TS for the APRM system are not impacted by this addition. In addition, the existing interim corrective actions for thermal hydraulic stability will continue to be relied upon until the TS change for the OPRM to be placed in full functional service (i.e., trips not bypassed) has been implemented. Current operation under the interim corrective actions provides an acceptable margin of safety in the event of an instability event as the result of preventative actions and TS controlled response by the control room operators. Once the OPRM system is fully functional, prudent operating guidance will continue to be followed but the OPRM will be capable of automatically detecting and suppressing oscillations within the defined region of potential instability.

Tracking No. SE-99-076
Activity No. DCP 9700408; DCN 0016981

DESCRIPTION:

This safety evaluation addresses a modification to add high and low current limiting potentiometers to the manual output circuit of the master feedwater regulating valve controller LC 2-0640-18, and to replace as "like-for-like" the current limiting potentiometers in the automatic output circuits of

LC 2-0640-18 and low flow controller FC 2-0640-20.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because there is no change to the function of the automatic circuits of controller LC 2-0640-18. There are two possible failure states for the new current limiting potentiometers added to the manual output of controller LC 2-0640-18; either fail open or short. If the potentiometers failed open, the result would be no controller output, so the FRVs would fail "as is". Any conditions resulting from FRV failure "as is" are bounded by the worst case extremes of fail open or fail closed, which are already analyzed as Increase in Feedwater Flow (Section 15.1.2) and Loss of Normal Feedwater Flow (Section 15.2.7). A short across the potentiometers is the equivalent of the controller circuit as it is now, without the potentiometers. The output of the controller will be the same as it is now. Therefore, there is no increase in the probability of controller failure, the possible types of controller failures, nor the consequences of any failure. This modification has no effect on any other SSC.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because there is no change to the function of the automatic circuits of controller LC 2-0640-18. There are two possible failure states for the new current limiting potentiometers added to the manual output of controller LC 2-0640-18: either fail open or short. If the potentiometers failed open, the result would be no controller output, so the FRVs would fail "as is". Any conditions resulting from FRV failure "as is" are bounded by the worst case extremes of fail open or fail closed, which are already analyzed as Increase in Feedwater Flow (Section 15.1.2) and Loss of Normal Feedwater Flow (Section 15.2.7). A short across the potentiometers is the equivalent of the controller circuit as it is now, without the potentiometers. The output of the controller will be the same as it is now. No new interactions are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modified controllers do not form the basis for any Technical Specification, and cannot affect any SSC that does. Therefore, there can be no effect on the margin of safety.

Tracking No. SE-99-080
Activity No. DCP 9900091; UFSAR-99-R6-022

DESCRIPTION:

The existing Barksdale Reactor Vessel High Pressure scram switches will be replaced with Rosemount pressure transmitters that will utilize an analog trip unit and an Agastat trip relay to interface with the existing Reactor Protection System (RPS) logic. One transmitter, one trip unit, and one Agastat trip relay will be required for each channel. Wiring for each pressure transmitter will utilize spare conductors in existing cables. As required, these cables are routed in separate conduits for each channel.

The replacement of Barksdale pressure switches with Rosemount pressure transmitters in the RPS reactor high-pressure logic scheme conflicts with Anticipated Transient Without Scram (ATWS) rule 10CFR50.62. This rule requires an Alternate Rod Injection system (ARI) that is diverse from the RPS from sensor output to final actuation device. Therefore, the existing Rosemount trip units for ATWS high reactor pressure will be replaced with General Electric trip units. These four trip units are designated as 2-263-22A-D, and are located in the Auxiliary Electric Room in ATWS cabinet panels 2201(2)-70A and 70B.

The overall effect of this activity is to provide an identical function as the previous Reactor Vessel Pressure High RPS trip. The design will maintain compliance with the requirements identified in the UFSAR for RPS and Analog Trip System instrumentation. The change will provide increased reliability and better overall performance for trip function.

The UFSAR is being updated to reflect the replacement of these switches. A Technical Specification change is also required which involves specifying a different surveillance requirement due to component replacement from a pressure switch to a pressure transmitter.

The Safety Evaluation was also used for DCP 9900090 (Unit 1), which was not Op authorized during this report period. The summary will be included when the DCP becomes Op authorized.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the existing Barksdale pressure switches are extremely sensitive to vibration from local traffic in the area, which makes them unreliable. They are also difficult to calibrate and have a tendency to drift. The replacement Rosemount pressure transmitters have a higher reliability and thus will give a more accurate reading of reactor vessel pressure. Therefore, by replacing the existing configuration with one that is more reliable, the activity is actually decreasing the probability of equipment malfunction.

   The replacement GE trip units are considered comparable replacements for the Rosemount trip units. This has been identified by the NRC during discussions on trip unit diversity for ATWS rule 10CFR50.62. The intent of replacing the Rosemount trip units with GE trip units is to maintain diversity between RPS and ATWS. The reactor high-pressure sensors for RPS currently use Rosemount trip units. By employing GE trip units in ATWS, the possibility of propagating common mode failures to both RPS and ATWS will be avoided. Therefore, the activity will enhance the overall scram system reliability and decrease the probability of occurrence of a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because with respect to the overall RPS trip logic scheme, it is ultimately the trip logic contact that is required to function. A malfunction in either the existing pressure switches or the replacement transmitters can result in a failure of the trip logic contact to open. Therefore, the activity does not create a different type of malfunction that did not already exist.

   The new GE trip unit cards do not introduce any new failure modes or different types of malfunctions into the ATWS reactor high-pressure scram logic. These new trip units are comparable to the old Rosemount trip units and operate in a similar manner. A failure of either trip unit card (GE or Rosemount) would result in an alarm condition on that particular channel. A failure on both channels A & B would be required in order to prevent a high-pressure scram.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification change involves specifying a different surveillance requirement due to component replacement from a pressure switch to a pressure transmitter. This does not affect the margin of safety in the RPS system and therefore, does not reduce the margin of safety. The surveillance frequency requirements specified in the current Technical Specifications are conservative with respect to instrumentation upgrade.

Tracking No. SE-99-094
Activity No. DCR 960246

DESCRIPTION:

The as-built Drawing Change Request (DCR) is associated with the [1] Process Radiation Monitoring, [2] Containment Atmosphere Monitoring (CAM), & [3] Primary Containment Oxygen Analyzer Systems. The changes make corrections regarding component type, drawing references, Equipment Piece Numbers (EPN), switch actuation, relay contact configuration, and manual valve position. Additionally, the electronic equipment database is updated to provide information regarding instrument calibration & set points, switch actuation, and relay contact configuration.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the equipment impacted by this as-built DCR is not connected to the primary system boundary, but rather provides a monitoring function of the Main Chimney & RB Vent Stack release paths, and monitoring functions for drywell atmosphere radioactivity & percent hydrogen/oxygen. Therefore, this equipment can not cause a LOCA.

   There are no new failure modes introduced by these changes. These changes ensure proper instrument setpoints & switch actuation, and provides the proper sample flow path, such that alarms are received as designed for abnormal conditions and proper monitoring is accomplished. Therefore, since the affected equipment provides a monitoring function only and the changes ensure proper monitoring, there is no affect on the consequences of an accident or malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the affected instrumentation serve a monitoring function only, and can not impact equipment important to safety. There is no change in equipment function nor introduction of new failure modes as a result of this change. There are no changes to plant equipment, except to establish sample flow alarm set points (none were existing) for the Reactor Building Vent SPING to alert operators of abnormal flow conditions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there were no Technical Specifications or associated Bases impacted by these changes.

Tracking No. SE-99-103
Activity No. UFSAR-99-R6-045

DESCRIPTION:

The current UFSAR wording defines a high radiation area as an area in which the radiation levels could result in a major portion of the whole body and could receive a radiation exposure greater than 100 millirem in one hour at 30 centimeters from the source or any surface the source penetrates. This is also the 10CFR20 definition of a high radiation area. Station Technical Specifications are written more restrictive than this definition. Technical Specification 6.12.A defines a high radiation area as an area in which the radiation intensity is greater than 100 mr/hr at 30cm. This UFSAR revision will change the UFSAR to state a high radiation area will be posted and controlled per Station Technical Specifications.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the UFSAR definition revision of a high radiation area is administrative, it does not interface with any safety-related operating plant equipment either directly or indirectly. This UFSAR revision does not modify or direct operation of any safety-related plant systems, structures, or components; therefore, the consequences of any accident or transient will not be increased. All safety-related systems will continue to operate as currently stated in the UFSAR, as this UFSAR revision does not affect operating plant systems, structures or components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this UFSAR revision does not introduce any of the precursors or initiators for any accidents or transients; therefore, this UFSAR revision cannot increase the probability of occurrence for any of these accidents or transients.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the definition of a high radiation area is not used as a margin of safety for any Technical Specification.

Tracking No. SE-99-106
Activity No. DCP 9900212; UFSAR-99-R6-048

DESCRIPTION:

This modification revises the Reactor Protection System (RPS) Turbine Generator Load Rejection (40% Mismatch) SCRAM signal logic. This logic consists of an Electro Hydraulic Control (EHC) system fluid reservoir low pressure switch in series with a Turbine Control Valve Fast Closure pressure switch. Four separate logic trains are provided. These trains are shown on Drawings 4E-2465 Sheet 2 and 4E-2466 Sheet 3. This modification will revise the RPS logic to remove the four (4) turbine EHC system fluid reservoir low pressure scram switches, PS-2-5650-1, 2, 3 and 4, including instrument service lines back to the process header.

This modification is being performed in the Unit 2 Turbine Building, ground floor, elevation 595'-0", southeast side, along the turbine centerline in the EHC area.

The effect of the modification is to reduce spurious reactor SCRAMs by removing trip functions which are not credited in any accident analysis and have the possibility to cause spurious unit trips.

The UFSAR has been revised in Section 7.2 to state that the RPS trip signal for EHC low oil pressure occurs at the turbine control valves and not at the EHC actuator supply header for Unit 2.

This modification has been approved by the NRC in License Amendment Nos. 193/189.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the modification will have no effects on any plant operating mode or equipment that is credited to mitigate an accident. This modification has no direct interface with the plant equipment, which can initiate an accident. The RPS function is to provide a reactor trip signal to mitigate the consequences of an accident. The RPS is a fail safe on loss of power system and cannot, by itself, initiate an accident. The turbine EHC fluid reservoir low-pressure switches are being deleted from the plant. The cables associated with the switches will be spliced to maintain the Turbine Generator Load Reject (40% Mismatch) SCRAM signal. Instrument sensing lines, manifolds and valves associated with these devices will also be removed. Testing as described in the Modification Approval Letter will ensure that the modified physical and electrical systems function as designed. The RPS reactor scram formerly provided by PS-2-5650-1, 2, 3 and 4 will be initiated by the Turbine Control Valve Fast Closure switches (PS-2-5641-122, 123, 124 and 125). The fast closure switches are credited in the accident analysis and will provide adequate protection during a postulated loss of turbine EHC fluid event. No instrument setpoints or operational procedures required to mitigate transients or accidents are changed as a result of this modification. Because the RPS system cannot initiate an accident and all of the credited SSCs will continue to perform their desired function, there can be no increase in the probability of an accident or transient. Therefore, the removal of non-credited components can have no effect on inputs and can have no effect upon the UFSAR Chapter

15 Accident and Transient Analyses. Following the modification, all essential plant systems and credited equipment will function as assumed in the Accident and Transient Analyses. Therefore, offsite doses are not affected and remain unchanged as a result of this modification. Accordingly, the modification does not increase the consequences of any accident or transient evaluated previously in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this modification will delete the turbine EHC fluid reservoir low pressure switches and the associated RPS reactor scram function. The EHC fluid low-pressure response formerly provided by the removed switches is provided by pressure switches PS-2-5641-122, 123, 124 and 125. There will be no reduction in the capability of existing plant equipment to function as required during all operational and accident modes because the RPS reactor scram function will be initiated in accordance with all applicable accident and transient analyses by the turbine EHC low fluid pressure switches located at the turbine control valves. The changes have been evaluated and will not result in the degradation or failure of any SSC. All modified and interfacing components have been analyzed and will be tested following installation as indicated in the modification approval letter to ensure that they will continue to function exactly as before.

   There are no other events postulated as a result of this modification which would create the possibility of an accident of a different type than any evaluated previously in the UFSAR. As RPS is designed to be fail safe, a failure of the revised wiring will initiate the protective function. Likewise, failure of the EHC piping will be sensed by the remaining pressure switches, which will initiate the protective function. These failure modes are unchanged by the deletion of pressure switches PS-2-5650-1, 2, 3 and 4. Therefore, this modification, as previously described, will not create the possibility of an accident or transient of a different type than evaluated previously in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modification will not change any plant operation parameters, or any protective system actuation setpoints other than removal of the turbine EHC Control Oil Pressure-Low scram function. This function is not credited in any accident analysis. The SCRAM function associated with the Turbine Control Valve Fast Closure is credited in the accident analyses and provides adequate protection for events involving fast turbine control valve closure including the loss of turbine EHC control oil pressure. For this reason, eliminating the turbine EHC Control Oil Pressure-Low scram function, which is redundant to other protective instrumentation, does not reduce the margin of safety.

Tracking No. SE-99-109
Activity No. DCP 9900120; UFSAR-99-R6-049

DESCRIPTION:

This modification includes (1) upgrading moisture separators 2-5605 A, B, C, D located at elevation 595' – 0" in the turbine building by removing existing internals and replacing them with new internals; (2) replacing the trim and actuators for feedwater heater C normal drain level control valves LCV-2-3504 A, B, C and feedwater heater C emergency drain level control valves LCV-2-

3505 A, B, C; (3) replacing feedwater heater D normal drain level control valves LCV 2-3506 A, B, C and feedwater heater B emergency drain level control valves LCV-2-3503 A, B, C; and (4) replacing the upstream isolation gate valves 2-3599-51, 2-3599-35 and 2-3599-5 for valves LCV-2-3503 A, B, C respectively. The drain valves are located in the feedwater heater bays at elevation 595' – 0" in the turbine building. Both the moisture separator and heater drain valve modifications are non-safety-related.

The changes to the moisture separator internals include removing and replacing the existing vanes with double hook vanes; adding boxing and seal buckets; replacing some of the existing downcomers with larger downcomers; replacing the existing seal buckets with new seal buckets. These upgrades will improve the moisture removal efficiency of the moisture separators from approximately 82% to 95% resulting in drier steam being sent to the LP turbine and increasing drain flow to the feedwater heater system by approximately 5% to 20%.

The existing feedwater heater drain level control valves described above are inadequately sized to accommodate the increased drain flow. Replacing either the valves or trim and the actuators of the feedwater heater drain level control valves listed above is necessary to make them capable of passing more flow so they can handle the increased drain flow from the moisture separators.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the functions of the moisture separators, the moisture separator drain tanks, and the feedwater heater drain system, heaters, and valves are not being changed by this modification. Even though the moisture separators are more efficient as a result of this change, the drain system has been upgraded to handle the increased demand. The system and its components will function as required during accident or transient conditions because component failure modes are unchanged. No instrument setpoints or operational procedures required to mitigate accidents or transients are changed as a result of this modification. Therefore, there is no increase in the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because there will be no reduction in the capability of existing plant equipment to function as required during all operational and accident modes. The moisture separators, moisture separator drain tanks, and the feedwater heater drain system, heaters and valves will continue to perform their intended functions. Increasing the moisture removal efficiency of the moisture separators and increasing the capacity of undersized feedwater heater drain level control valves will not adversely affect any safety-related structures, systems, and components or equipment important to safety. This modification will not affect the operation of any plant equipment necessary to mitigate the consequences of accidents or transients. There will be no effect on equipment failure modes or malfunctions as a result of this modification. Therefore, the possibility for an accident or malfunction of a different type is not created by this modification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specifications relevant to or affected by this modification.

Tracking No. SE-99-120
Activity No. WR 990098458

DESCRIPTION:

During quarterly testing (QOS 5600-05 Turbine Generator Periodic Testing) of the Combined Intermediate Valves (CIV), PIF Q1999-03219 was initiated (9/25/99) because the U2 Intercept Control Valve (ICV) portion of the #3 CIV failed to fast close and its associated Intercept Stop Valve (ISV) failed to close at all. The Intercept Stop Valve (ISV) and the Intercept Control Valve (ICV) each share a common valve body and together are referred to as a Combined Intermediate Valve (CIV). The testing circuitry is designed to fast close the ICV on the actuation of its 90% closed limit switch. This limit switch also initiates the closure of its associated ISV. Preliminary troubleshooting data suggests that the failure to satisfactorily perform the surveillance was apparently caused by a faulty limit switch.

The activity will involve stroking the ICV twice utilizing QOS 5600-05. The first stroking of the ICV will address measuring the voltage drop across its 90% closed limit switch. This should determine the functionality of the limit switch. During the second stroking of the ICV an electrical jumper will be utilized, jumpering around the ICV's 90% limit switch. The jumper will contain a 5 amp inline fuse and also a single-throw switch. This jumper will be installed prior to performing the surveillance. The single-throw switch will be manually closed when main control room indication of ICV valve position indicates the valve is 90% closed. Closing the switch in this manner will simulate the actuation of the 90% limit switch. This should initiate the fast closure of the ICV and should also initiate the slow closure of it's associated ISV. This activity should prove the failure of the previous surveillance was solely due to a faulty limit switch.

SAFTY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the CIV's fast acting solenoid and its testing circuitry is non-safety-related and is not relied upon to function during any accident or transient referenced within the UFSAR. Furthermore, the function of the CIV is not impacted by the failure of the test circuit. The activity will be performed on the CIV testing circuitry only and will not have any affect on other plant components or any plant protective systems. A turbine trip is not anticipated during this test, but if one was to occur, the turbine will trip within its normal protective functions. A reactor scram would occur upon closure of the turbine control and stop valves. This sequence of events has been evaluated and will not result in the breach of any fission product barriers and will not result in a radiation release in excess of 10CFR100 limits. Therefore, the probability of occurrences or the consequences of an accident important to safety previously evaluated in the UFSAR have not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the activity will only be performed on the CIV test circuitry. This testing will have no affect on the protective functions of the turbine. The only postulated failure this activity could cause is the loss of control power to the Turbine valve test circuit. This loss of control power is not considered likely due to the use of a 5A fuse in the jumper. This 5A fuse would preferentially fail prior to causing the 20A, 115 VAC control power fuse to fail. Failure of the 5A fuse would have

the same affect as having not installed the jumper, which is the fail safe condition. In the unlikely event control power were to be lost, a turbine trip would not be anticipated. Even if a Turbine Trip were to occur, all Turbine equipment will remain within its normal protective functions. The turbine trip has been previously analyzed. Since the only postulated malfunction is bounded by the previously analyzed Turbine Trip, this activity will not create the possibility of an accident or transient of a different type than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the CIV's or the CIV's test circuitry is not referenced within the Technical Specifications.

Tracking No. SE-99-126
Activity No. DCP 9700344; DCN 001792I

DESCRIPTION:

A change is being made to the circuit of the Unit 2 Main Turbine Electro-Hydraulic Control System (EHC) Master Trip Solenoid Valve (MTSV) solenoids (MTSV-A and MTSV-B). During plant operations these solenoids are normally energized. The MTSV trips the turbine when both solenoids are de-energized. The relay contacts open in the MTSV solenoids circuit to de-energize both solenoids and initiate a turbine trip upon a trip signal from the EHC Master Trip Bus. As currently configured, the EHC MTSVs are normally maintained energized through test switch contacts which allow de-energization of one solenoid at a time, so that a turbine trip is not initiated. The test switch logic is being changed so that the switches energize new interposing relays, whose contacts will then open the circuit of the EHC MTSV solenoids, one at a time, for testing.

The new interposing relays will function to reset the MTSV solenoids at the end of on-line testing, reset a turbine trip and also maintain the MTSV solenoids energized during normal operation. The MTSV is the interface point between EHC electrical trip functions and the mechanical features on the main turbine-generator, which execute turbine generator trips. The design change includes the following scope:

1. The test switch contacts are removed from the MTSV solenoid circuits.

2. The new interposing relays will be energized through the test switch only during testing of the MTSV. The power supply that maintains each MTSV solenoid normally energized is routed through a normally closed contact of its associated interposing relay.

The subject DCP 9700344 installs the two 24 VDC interposing relays (one for each MTSV solenoid) in the main turbine EHC cabinet 902-31.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because reconfiguration of the turbine electrical trip test and reset function makes no change in the probability of occurrence or consequence of any of the evaluated accidents. There is no change to accident initiation factors for a turbine trip. There is no

change to the connection of the MTR to the MTSV, which is the mechanism for valid electrical trip function interfaces with the mechanical/hydraulic processes that actually trip the turbine.

The subject DCP 9700344 has no effect on the methods of detection of abnormal turbine operating parameters, or the method of performing the trip. The mechanical overspeed trip remains actuated through the MTV, and the electrical overspeed trip remains actuated through the MTR and MTSV. Therefore, there is no effect on the performance of the turbine overspeed trip functions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the changes made to the EHC system have been evaluated for operational impact and do not create any new interaction with any other SSCs that could result in any different kind of accident. Prior to the change, turbine trip solenoid testing was performed by actuating the test switch at Panel 902-7. Implementation of DCP 9700344 does not change the method of performing the test or reset. Any failure of the test/reset circuit still cannot result in anything more than a turbine trip, which is an analyzed condition. This change does not create the possibility of any previously unanalyzed accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes to the turbine electrical trip and reset logic do not change any parameters affecting Technical Specification requirements, actions or bases.

Tracking No SE-00-006
Activity No. DCP 9900252

DESCRIPTION:

DCP 9900252 implements a high radiation alarm setpoint change for Unit 1 Area Radiation Monitors (ARM) #32 & #33, Offgas Recombiner Area Level 1 & Level 2 ARM's. The alarm setpoints for ARM #32 & #33 are being changed from 1 mR/hr to 5 mR/hr, and from 4 mR/hr to 5 mR/hr, respectively. Normal background radiation at the detectors for ARM's #32 & #33 has increased slightly recently, due to the 1A Offgas Recombiner Train being placed in service for the first time in over 12 years, resulting in ARM #32 being continually in alarm.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because ARM's provide monitoring function only, and have no interaction with the reactor coolant pressure boundary. The affected ARM's are not used to mitigate an accident/transient described in the UFSAR. There are no new failure modes introduced. The ARM alarm setpoint value has no impact on the probability of a malfunction of the ARM. The consequences of a malfunction of an ARM remain the same, which are either failure to provide information or providing of erroneous information. The new alarm setpoint value is low enough to provide timely warning of abnormal conditions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because these changes affect an alarm setpoint only, such that spurious or invalid alarms during normal expected radiological conditions at the detectors do not occur, but low enough such that a timely alarm would be provided under abnormal radiological conditions. The ARM's provide a monitoring function only that is independent of other plant equipment. The alarm setpoint changes and the UFSAR expected background value changes do not impact the functions provided by the ARM's, nor is any new failure mode introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the ARM System is not required by Technical Specifications.

Tracking No. SE-00-008
Activity No. DCP 9900225; 9900226; UFSAR-99-R6-069

DESCRIPTION:

These modifications will change the instrument sensitivity, which corresponds to a response time increase of the SCRAM Discharge Volume (SDV) Instrument Volume (IV) thermal level switches. This change will be accomplished using site procedure QCIS 0300-01. The current time delay is approximately as calibrated is 1 - 2 seconds. The sensitivity adjustment will be limited to a maximum of 5 seconds time delay.

Increasing the sensitivity time delay setpoint of the SDV IV FCI Level Switches (EPN's LS 1-0302-82A, LS 1-0302-82B, LS 1-0302-82G, 1-LS-0302-82H, LS 2-0302-82A, LS 2-0302-82B, LS 2-0302-82G, 2-LS-0302-82H) provides greater reliability. It will reduce the affects caused by the previously more sensitive instrument settings sensing thermal transients. This will reduce the possibility of spurious SCRAMs.

The UFSAR has been revised to reflect this design change. A discussion has been included of the High Level Thermal switch sensitivity setting.

It should be noted that this UFSAR change contains an administrative error in that it was not made effective at the time the DCPs were Op Authorized. A PIF has been written to document and correct the deficiency.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the SDV and IV function to allow the Reactor Protective SCRAM to be completed properly when required. These components do not affect operating conditions or any parameters, which could lead to an accident or transient. The affected level switches will now have time to react to transients that would have caused a spurious SDV high level SCRAMs prior to this change. No new components were added and no parameters affecting reactor operation were changed.

The activity will not increase the consequences of the any accident or transient. The SCRAM function is not being affected. The operability of the switch is not being altered. The SDV will have sufficient volume for a SCRAM. The increased volume of water in the SDV IV will not create a hydraulic lock on the CRDs. This has been analyzed in Calculation QDC-0300-M-712 Rev. 0. This calculation shows the SDV IV volume is not required for the SDV to perform its function.

The only malfunction of concern is a malfunction that could impede a proper SCRAM. This setpoint change does not affect the degree of a CRD SCRAM malfunction. There is ample margin in the SDV and IV when the SDV high level initiates a SCRAM. This ensures other SDV malfunctions do not put the Reactor Protection in an unsafe condition. The switch actually operates more effectively at 4 -5 seconds delay than at 1 -2 seconds. The switch at one second cannot discern between splashes, bursts, or steam excursions. At 4 - 5 seconds, the switch has time to prevent over response to these transients.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because these components do not affect operating conditions or any parameters, which could lead to an accident or transient. No new components were added and no parameters affecting reactor operation were changed.

   The thermal switch reacts to changes in the thermal conductivity of its environment. Since we are not changing the environment, no new failures or malfunctions are being introduced. A Differential Pressure (DP) transmitter also measures the level of the IV. If the switch were to fail (highly unlikely), the DP transmitter backup would still activate the SCRAM. The reason for this sensitivity time delay change is due to transients and upset conditions. The switch has activated on spurious steam upsets in the drain piping.

   The activity will not affect the structures, loads, pipe stress of the IV. The IV was constructed to accommodate approximately 110 gallons of process water. The change, therefore, will not infringe on this part of the design. Per section 4.6.4.6 of the UFSAR the SDV was redesigned to include dual instrument Volumes and diverse level switches to minimize the malfunctions and equipment failures. This design does not affect these improvements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the switch will Sense level at the 35" setpoint, which is less than 40 gallons. The switch is permanently mounted on a flange on the IV. This cannot be altered without disassembling the IV. The sensitivity (time delay) feature of this thermal switch will SCRAM the reactor if actual volume is being accumulated in the instrument volume. As stated in the Technical Specification basis no credit is taken for the Instrument Volume.

   The basis states that the setting for the anticipatory SCRAM has been chosen on the basis of providing sufficient volume remaining to accommodate a SCRAM even with 5 gpm leakage per drive into the SDV. The new settings will still meet that requirement without using up the Instrument Volume and therefore, does not reduce the safety margin.

Tracking No. SE-00-009
Activity No. UFSAR-99-R6-068

DESCRIPTION:

Systems that cool the fuel pools can also be used as an alternate method of decay heat removal from the reactor cavity during refueling outages when the reactor cavity is flooded above a level of 23 feet (above the vessel flange). When the gates between the reactor cavity and the fuel pool and between the two fuel pools are removed, a natural circulation develops between the reactor cavity and spent fuel pools due to the temperature and density differences between the three bodies of water. To qualify this alternate method of decay heat removal, an analysis is performed prior to the refueling outage to evaluate the heat load in the reactor vessel and spent fuel pools that will be unique to each refueling outage. The heat load is calculated using the methodology described in NRC Branch Technical Position ASB 9-2. From the heat load, the required number of Fuel Pool Cooling (FPC) system trains and Residual Heat Removal (RHR) loops aligned to fuel pool assist (FPA) are determined. It may be necessary to route a portion of the cooling flow directly to the refueling cavity instead of the fuel pool. Conservative values for the RHR service water temperature and Reactor Building Closed Cooling Water (RBCCW) are determined based on the time of year during which the refueling outage occurs. This analysis demonstrates that the temperature of the water in the reactor cavity will not exceed Technical Specification limits if specified FPC and/or RHR-FPA system flow rates and cooling water temperatures are maintained. Requirements for fuel pool cooling as described in UFSAR Section 9.1.3.1 must also be satisfied. Furthermore, analysis is performed to show that no local boiling will occur on the surface of the fuel rods. Administrative controls are procedurally implemented and the water temperature in the reactor cavity and the fuel pools is monitored to ensure compliance with the analytical assumptions and results such as time, flow, and temperature limits. A description of this alternate decay heat removal procedure has been added to Section 5.4 of the UFSAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this activity has no affect upon the initiators of the flooding or dam break scenarios. This procedure does not control the movement of fuel or degrade the equipment used to move fuel. This procedure will not affect the administrative pool temperature limit at which fuel movement is ceased. The temperature elements will be adequately restrained to ensure they do not fall down into or inadvertently move around the reactor cavity or fuel pool. This procedure controls the operation of the fuel pool cooling system; however, it does not affect the reliability of the equipment in the fuel pool cooling system. Operating with the gates open and the fuel pools connected to the reactor cavity does not increase the likelihood of a malfunction of any equipment in the fuel pool cooling system. Engineering analysis is performed, and the reactor cavity is monitored to ensure that the water temperature is maintained within the previously established acceptance limits. Additional analyses demonstrate that acceptance limits for time to boil and boiloff rate are met under a loss of fuel pool cooling scenario. Additionally, none of the assumptions or parameters for analyzing the consequences of a cask drop accident or a Design Basis Fuel Handling Accidents During Refueling, such as cask weight, height from which it is dropped, fuel and bundle characteristics, or structural features of the spent fuel

pool are changed. None of the barriers or mitigation systems for a dam break or flooding scenario are affected by this alternate decay heat removal method.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because no new failure modes are created. No new or different types of hazards are introduced. The addition of the temperature elements to the reactor cavity and/or fuel pools do not create any new credible failure modes to the fuel pool or fuel pool cooling systems. The procedure will utilize existing channels between the reactor cavity and the fuel pool that are normally open during a refueling scenario and will credit the fuel pool cooling system which is normally in operation. The performance of the fuel pool cooling system and the spent fuel storage system is not degraded by this change. Additionally, this activity does not create a new interaction between the two fuel pools if the FPC system should fail on one of the units. Per QCOA 1900-02 and 1900-03, on a loss of FPC or on a high temperature alarm, the gates can be removed between the two fuel pools such that the other fuel pool and FPC system can be used to provide cooling to the other pool. This is consistent with the 1982 SER which stated that it is possible to "use the cooling system in one unit to assist cooling the pool water in the adjacent unit pool. This could be accomplished by opening the two gates in the transfer canal and allowing an interchange of water between the two pools." Therefore, the change does not create the possibility of any accident or transient of a different type than previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the purpose of this change is to allow the planned use of an alternate method capable of decay heat removal as allowed by TS 3/4.10.K. Engineering analysis is performed and the water temperature in the reactor cavity and fuel pools is monitored to ensure that the temperature is maintained within the acceptance limits defined in the Quad Cities Licensing Basis and the engineering analysis. This includes maintaining the reactor cavity temperature below 140° F as required by plant Technical Specifications during REFUELING (Mode 5). The heat load is calculated using the methodology described in NRC Branch Technical Position ASB 9-2, and the temperature behavior of the reactor cavity is predicted using additional conservative assumptions that result in a conservatively high temperature prediction. The performance of similar alternate decay heat removal analyses and procedures at Dresden and LaSalle stations has verified that the actual temperature behavior is significantly less than the temperature predicted by the engineering analysis. Therefore, the analysis performed for the alternate decay heat removal method and the temperature monitoring of the reactor cavity water as part of the procedure will ensure that the Technical Specifications temperature limit is maintained; therefore, this activity does not reduce the margin of safety associated with any Technical Specification.

Tracking No. SE-00-016
Activity No. DCR 990401; UFSAR-99-R6-074

DESCRIPTION:

This change revises UFSAR Figures 3.11-1 Sheets 1, 3, 5, 7, and 9, and drawings M-4A sheets 1-5, (all sheets at revision B). This change revised EQ Zone 7 LOCA temperature with room cooler to less than 120 degrees F, added supporting note 12 and reference 31, changed EQ Zones 1, 9, 20, 28, and 36 temperature and pressure to 294 degrees F and 62 psia and a added note 13 to clarify basis for temperature and pressure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because EQ Binders EQ-01Q, EQ15Q, EQ-17Q, EQ-39Q, EQ40Q, EQ-41Q, Q-43D/Q, EQ-45D/Q, EQ-65Q, EQ-75Q, EQ-GEN017, EQ-GEN018, and EQ-GEN036 demonstrate environmental qualification of the affected SSCs under the revised conditions. The HPCI components required to mitigate a LOCA are not and were not required to be Environmentally Qualified. Their original design specification and system surveillances ensures proper operation of the effected SSCs. Hence, the changes do not affect the functionality of the components required to mitigate this accident. Consequently, accidents, and malfunctions are not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because EQ Binders EQ-01Q, EQ15Q, EQ-17Q, EQ-39Q, EQ40Q, EQ-41Q, Q-43D/Q, EQ-45D/Q, EQ-65Q, EQ-75Q, EQ-GEN017, EQ-GEN018, and EQ-GEN036 demonstrate environmental qualification of the affected SSCs under the revised conditions. The HPCI components required to mitigate a LOCA are not and were not required to be Environmentally Qualified. Their original design specification ensures proper operation of these SSCs. The SSCs EQ and original design documentation demonstrates their ability to function under the revised environmental conditions. Hence, the changes do not affect the functionality of the components. Based on the above, accidents and equipment malfunctions are not affected by this change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because EQ Binders EQ-01Q, EQ15Q, EQ-17Q, EQ-39Q, EQ-40Q, EQ-41Q, Q-43D/Q, EQ-45D/Q, EQ-65Q, EQ-75Q, EQ-GEN017, EQ-GEN018, and EQ-GEN036 demonstrate environmental qualification of the affected SSCs under the revised conditions. The HPCI components required to mitigate a LOCA are not and were not required to be Environmentally Qualified. Their original design specification and system surveillances ensures proper operation of these SSCs. The SSCs EQ and original design documentation demonstrates their ability to function under the revised environmental conditions. Hence, the changes do not affect the functionality of the components. Therefore, the margin of safety has not been reduced.

Tracking No. SE-00-020
Activity No. UFSAR-99-R6-076

DESCRIPTION:

This change is the Q2C16 core design, which contains a reload of fresh ATRIUM-9B offset fuel. This will be the second ATRIUM-9B offset reload at Quad Cities Unit 2. The UFSAR has been updated to reflect Quad Cities Unit 2 Cycle 16 Reload Licensing Package, dated January 14, 2000.

This 10CFR50.59 safety evaluation addresses the Q2C16 reload core design and the associated UFSAR changes to support Q2C16 operation. The previous Q1C16 and Q2C15 10CFR50.59 Reload Evaluations contain generic evaluations of ATRIUM-9B offset fuel at Quad Cities Station. These evaluations conclude that ATRIUM-9B offset fuel is acceptable for use. Therefore, this 10CFR50.59 evaluation addresses only the changes applicable to the Q2C16 reload core design.

The Q2C16 core design consists of a total of 724 fuel assemblies, including 125 previously loaded GE 9B assemblies,143 previously loaded GE10 assemblies, 216 previously loaded ATRIUM-9B offset assemblies and 240 fresh ATRIUM-9B offset assemblies. The fresh ATRIUM-9B fuel consists of 136 SPC ATRIUM-9B High Gd assemblies and 104 SPC ATRIUM-9B Low Gd assemblies. All fresh fuel will be channeled with SPC advanced channels as was the Q2C15 fresh ATRIUM-9B fuel. The Q2C16 core is designed for approximately 2 years of operation with a 97.75% operating capacity factor. The SPC licensing basis Loss of Full Power Capability core average exposure for Cycle 16 is 31,467 MWd/MT.

The transient analyses performed for Q2C16 are based on input parameters that include a conservative increase of 5% to each Analytical Limit for nuclear instrumentation Limiting Safety System Settings in the Technical Specifications and an increase to +/- 3% to all the main steam safety valve set-point allowance. All Q2C16 transient analysis results reflect these conservative input parameters. Transients that assume these analytical setpoints have been re-evaluated (Reference EMF-2222(P)). However, FDLRC continues to use the allowable value of 120%. This change resulted in a conservative Rod Block Monitor (RBM) setpoint and conservative MCPRf curves for Q2C16. These conservative input parameters were used in anticipation of the implementation of a commitment to the NRC to control all of its Limiting Safety System Settings as defined by the Technical Specifications according to Reg Guide 1.97 Category A and ISA Standard S67.04 during Q2C16 which will be covered by a separate evaluation.

The neutronics licensing and transient analysis were performed and evaluated in this 50.59 for the initial loading pattern for Q2C16. Since that loading pattern was developed, Quad Cities Unit 2 developed a leaking fuel assembly. It is assumed that this leaking fuel assembly is in the current Q2C15rev1 Control Cell M8. In order to eradicate the leaker from the Q2C16 core, these four fuel assemblies in Control Cell M8 will be discharged and replaced in the Q2C16 loading pattern with other fuel assemblies that were previously slated for discharge this refueling outage. This modification to the core-loading pattern introduces no changes to the fuel assembly types. There are only minor changes in assembly average exposure of a few assemblies near the periphery of the core. The total number of assemblies of each type is unchanged from the licensing loading pattern. Therefore, the modifications are expected to have insignificant impact on the results of the analyses and this safety evaluation remains unchanged for Q2C16 as long as the assumption remains true that the leaking fuel assembly is in Q2C15rev1 control cell M8.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. These changes do not affect the operability of plant systems, nor do they compromise any fuel performance limits. Therefore, no current precursors are changed.

An increased frequency of accident precursors may be created by modifications to the plant configuration, including changes in to allowable modes of operation. The Q2C16 reload core design does not involve any modifications to the plant configuration or to allowable modes of operation. No new precursors of an accident are created and no new or different kinds of accidents are created.

The Q2C16 reload core design does not physically alter the systems designed to prevent an accident from occurring. Specifically the Q2C16 reload core design does not alter:

Pressurization Control Equipment
* Relief valves
* Safety valves
* Main steam isolation valves
* Bypass valves
* Turbine stop valves
* Turbine control valves

Systems Utilized in Analyzing for MCPR Protection:
* Feedwater heaters or steam extraction lines to the heaters
* RBM logic
* CRD system
* Jet pumps or recirculation system

Fuel Handling Equipment or Procedures:
* Fuel handling equipment
* Fuel handling processes and procedures

Equipment and Input Parameters for Fuel Heat Generation Removal:
* CRD system
* Any ECCS system
* Fuel pool cooling and shutdown cooling system

Design Parameters and Equipment Ensuring Energy Deposition Protection (CRDA protection):
* Rod Worth Minimizer
* CRD system
* Control rod sequences established for Q2C16 (Q2C16 CRDA analysis was performed with bounding sequences
* Control Rods

Therefore, since
a.) the change does not alter any equipment necessary to prevent transient or accident conditions from occurring
b.) these analyses have been used to establish operating limits which protect the core within Technical Specification and 10CFR limits
the Q2C16 reload core design will not increase the probability of occurrence of any accident or transient.

The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences.

The accident analyses supporting the Q2C16 core design have been performed using NRC-approved methodologies. These analyses account for the reload fuel type and the Q2C16 core design. The analyses establish limits, which protect the core from the potential consequences of these accidents. Operation is bounded within these limits as established by the Core Operating Limits Report (COLR), Reference 17.

Protection Against the Onset of Transition Boiling
Specifically the dual and single recirculation loop Minimum Critical Power Ratio Safety Limit (MCPRSL) has been established for Q2C16 to ensure that less than 0.1% of the rods are predicted to experience boiling transition. MCPR Operating Limits (OL) have been established to protect the MCPR SL during AOOs . The MCPR OLs are established using the results of transient analyses, of which the limiting event was determined to be the Feedwater Controller Failure (FWCF). Specifically, the Q2C16 MCPR Operating Limits, including flow and exposure dependent limits, have been established to protect all fuel types in the Q2C16 core against transition boiling in 99.9% of the fuel rods in the core during:

* Feedwater Controller Failure (FWCF)
* Loss of Feedwater Heating (LFWH)
* Load Rejection (Generator Trip) Without Bypass (LRNB)
* Turbine Trip with no Bypass (TTNB) (Bounded by LRNB)
* Rod Withdrawal Error (at Power) (RWE)
* Fuel Loading Error ; Mislocated/Misoriented Fuel Assembly
* Recirc Pump Runup
* Events during EOD/EOOS Conditions

Heat Generation Rate Protection
Linear Heat Generation Rate (LHGR) limits (transient for ATRIUM-9B and steady state for both GE9/10 and ATRIUM9B) have been established to ensure that the fuel is protected from centerline melt and 1% plastic strain of the cladding for up to 120% of core power.

Maximum Average Planar Linear Heat Generation Rates (MAPLHGR) limits have been established to protect 10CFR50.46 and fuel design requirements for the new and existing fuel in Q2C16. 10CFR50.46 requirements protect the fuel from exceeding 2200 degrees F peak clad temperature, 1% core wide hydrogen generation, and 17% local clad oxidation thickness, and ensure that a coolable geometry and long term cooling are provided for the core. Specifically, analyses have been performed to demonstrate that APLHGR limits are protected during a decrease in reactor coolant inventory. All heat generation rate protection limits are fuel bundle design and exposure dependent and not core design dependent.

Overpressurization Protection
Analyses demonstrate that the ASME vessel and Technical Specification 2.1.C steam dome pressure limits of 110% design pressure and 1345 psig, respectively, are protected during pressurization transients.

Fuel Energy Deposition Protection
The control rod drop accident (CRDA) analyses (UFSAR 15.4.10) demonstrate that the Q2C16 core design meets the requirement that the maximum deposited energy into the fuel during a reactivity excursion is not projected to exceed 280 cal/g. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal is predicted to occur. Confirmation of meeting this limit is performed each cycle; therefore, it remains applicable for Q2C16.

Anticipated Transient Without Scram
An assessment was provided for the initial transition to ATRIUM-9B fuel at Quad Cities Station concluding that the change in core characteristics introduced by ATRIUM-9B in the Quad Cities cores does not have a significant impact on the current ATWS analysis results documented in References 31, 32 and 33. In addition, Standby Liquid Control, which would mitigate an ATWS event, is not affected by this reload.

Fuel Handling Accident
The consequences of a fuel handling accident are not impacted by the Q2C16 reload core design. The use of ATRIUM-9B regarding the fuel handling equipment has previously been evaluated.

Provided that plant operation is maintained within the established limits as specified by the Core Operating Limits Report, the consequences of an accident are not increased by the Q2C16 reload core design.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident or require a modification to the plant configuration, including changes in allowable modes of operation. The Q2C16 reload core design has been analyzed using NRC-approved methodologies and is supported in all allowable modes of operation. The Q2C16 reload core design does not involve any modifications to the plant configuration or allowable modes of operation. The reload fuel design has been NRC-approved and the reload core design utilizes the same geometry as previous cycles. Thus, no new precursors of an accident are created and no new or different kinds of accidents are created. Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The reload fuel is an NRC-approved design and the Q2C16 core design has been evaluated using NRC-approved methodologies. The Q2C16 reload analyses incorporated conservative input parameters.

   No plant systems, structures or components are physically changed or adversely affected; therefore, no new accidents or malfunctions of a type different than those previously evaluated are anticipated. The equipment assumed out of service in the Q2C16 reload analyses is consistent with previous cycles.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Q2C16 core design was generated using NRC approved methods. All required thermal limits have been established using NRC approved methodologies to protect the Q2C16 core during all anticipated operational occurrences. Therefore, since

the Q2C16 core is designed within all necessary criteria and operational limits have been established to protect the core, the margin of safety as described in the Technical Specifications is not reduced.

Tracking No. SE-00-021
Activity No. TS BASES 3 /4.9

DESCRIPTION:

Changed the reference in Technical Specification B3 /4.9 for test method of diesel generator fuel oil particulate contamination from ASTM D2276 to ASTM D5452.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the changed method for diesel fuel particulate contamination assures the same or higher quality of fuel oil to the diesel, which results in no negative impact on diesel operation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the changed method for diesel fuel particulate contamination assures the same or higher quality of fuel oil to the diesel, which results in no change to any plant system or structure.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changed test method does not change the performance of the diesels; therefore, it does not reduce the margin of safety.

Tracking No. SE-00-022
Activity No. UFSAR-99-R6-075

DESCRIPTION:

Revisions have been made to UFSAR Section 8.3.1.8 Analysis of Station Voltages, to correct and clarify the requirements for motor starting voltage for motors connected to 480-V buses. In the first paragraph on page 8.3-15, "To provide adequate torque for motor starting and to prevent contactors from dropping out at 480-V motor control centers, the starting voltage should be limited to 75% of motor rated voltage."

Will be replaced by
"To provide adequate torque for starting safety-related motors and to prevent contactors from dropping out at 480-V buses, the minimum starting voltage for safety-related low voltage motors shall be 75% of motor rated voltage; otherwise, lower starting voltage shall be supported by an analysis to justify the starting at lower voltage."

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the UFSAR revision does not alter the operation of any SSC. No new components are added. No actual change is made to any SSC or procedure. No equipment is modified. Because all SSCs will continue to perform their required design function as they do now, there can be no increase in the probability of occurrence or consequences of any accident or transient.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the UFSAR revision does not alter the operation of any SSC, nor does it add any new SSCs. No actual change is made to any SSC or procedure. No existing equipment failures or malfunctions are altered and no new equipment is added. All equipment will continue to function exactly as it does now.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because these changes do not affect Technical Specifications. There are no changes to any setpoint, surveillance, or bases in the Technical Specifications. Therefore, the Technical Specification margin of safety is not reduced.

Tracking No. SE-00-023
Activity No. DCP 9900311, UFSAR-99-R6-077

DESCRIPTION:

The activity is the implementation of Design Change Package (DCP) 9900311. This activity modifies the closing function of Motor Operated Valve (MOV) MO 2-2301-8. The closing logic for this MOV will be modified to incorporate an "open" limit switch in parallel with the existing torque switch and existing open limit switch. The existing open limit switch contact is in parallel with the torque switch contact and is only closed while the valve is only slightly closed. Consequently, in the existing circuit, the torque switch contact is the only maintaining contact for the closure of the valve for a majority of the closing sequence. The newly incorporated intermediate open (IO) limit switch contact will bypass the Torque Switch Close (TSC) while the valve position is within 0 to 97% closed. This will have no effect to the operation of High Pressure Coolant Injection (HPCI) for Emergency Core Cooling System (ECCS). It only effects when the torque switch can cutout the motor operator while a valid close demand is present. The limit switch contact that will be used to bypass the TSC is currently in use in this control circuit. This contact will be rewired such that it is in parallel with the TSC and a spare limit switch contact will be added to the circuit to replace the function of the relocated contact. A change to the UFSAR will also be necessary as a result of this DCP and will be performed under UFSAR-99-R6-077 which will revise Figure 6.3-15 to reflect the closing function of MO 2-2301-8.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because HPCI is a system that is maintained in a standby mode unless routine testing or maintenance is being performed. The TSC / TSC switch bypass logic for a HPCI Pump Discharge isolation valve does not interface with other systems. The changed operation of this circuit does not change any operating parameters or conditions; therefore, this change will not create the possibility of an accident/ transient of a different type than was previously analyzed.

   The new IO contact could fail in either the open position or the closed position. If it were to fail open, the torque switch would still be operational to stop valve travel once the valve was closed. Since the open IO contact would be in parallel with the existing torque switch contact, a failure in the open position produces that same circuit that currently exists. Should this contact fail in the closed position, the control circuit would be unable to stop valve travel once the valve was closed. This would damage the valve and possibly prevent HPCI from operating. However, the existing torque switch bypass could also fail in the closed position producing the same result. The New IO contact could also fail such that it would cause a short circuit which could possibly prevent the HPCI system from operating. However, there are other contacts in the control circuit for MO 2-2301-8, which could fail in the same manner producing the same result. The use of the existing spare contact to replace the contact that was relocated will not affect equipment malfunctions or failures. The spare contact is of the same type and part of the same limit switch arrangement as the relocated contact and will behave in a similar manner.

   Based on the above, the addition of the 97% close TSC bypass IO contact will improve the reliability of MO 2-2301-8. A failure of the added contacts will produce the same results as a failure of any other contact in the close circuit, which would be an inoperable HPCI system. Since the loss of HPCI is an accident/transient that has been previously analyzed equipment failures or malfunctions will be unaffected by this change. There will also be no new failure modes as a result of this DCP.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because as discussed in the preceding steps, the addition of the contacts will not affect HPCI operation under any conditions. The control circuit changes are being made for reliability purposes to prevent the torque switch from spuriously stopping valve travel. A failure of the contacts can only affect the HPCI system and its power source (250 VDC MCC 2A). The isolation demand logic remains unchanged and the SSMP operation is not adversely affected. A failure of the contacts cannot affect Automatic Depressurization system (ADS) or the relief valves themselves since they are supplied from the 125 VDC system. This change will not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the contacts will have no effect on the way HPCI is maintained operational or tested. It will also have no effect on equipment failures associated with the HPCI system or its supply bus. The above Technical Specification is not affected.

Tracking No. SE-00-024
Activity No. UFSAR-99-R6-073/078; QCOA 3700-01 Rev. 3; QCOA 3700-03 Rev.2;
QCOP 4700-05, Rev. 6.

DESCRIPTION:

This UFSAR revision is to correct heat transfer rate of the Reactor Building Closed Cooling Water (RBCCW) heat exchangers on page 9.2-8 to match the vendor data. UFSAR section 9.2.3.2 will be amended by the change to accurately describe the location of a radiation monitor in the RBCCW piping. The change will amend UFSAR section 9.2.3.2 to include the Unit 2 Drywell Pneumatic Compressor Heat Exchanger in the list of loads that are remotely isolated by MO 1(2)-3701. The change will revise UFSAR Figure 9.2-3 to add missing cooling loads, add Unit specific information relative to the cooling loads, correct the indicated direction of flow for the 1/2 RBCCW heat exchanger and correct the valve numbers for the 1/2 RBCCW pump. UFSAR Table 9.2-3 will be revised by the change to remove the Environs Sample Rack from the list of RBCCW loads and to incorporate two editorial changes. An editorial change is for UFSAR section 6.4.2.2 to omit the specified room numbers for areas in the service building which are provided with ventilation from the Train A HVAC system. The change will delete a sentence in UFSAR section 11.2.2 that describes Radwaste system alarms that are no longer installed in the main control room. The change will reword a paragraph describing the In-Plant Cement Solid Waste system in UFSAR section 11.4.2.1 and omit reference to this system in UFSAR section 11.4.2.2. The change will revise procedures QCOA 3700-01, QCOA 3700-03 and QCOP 4700-05 to identify correct interactions with the RBCCW system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because a loss of RBCCW could affect the probability of a Recirculation Pump Shaft break to occur. However, the changes do not cause the RBCCW system to be overloaded, there are no new loads being added to the system, the change in the heat exchanger specification increases the heat removal capacity of the system and the changes do not degrade the pressure boundary of the system. Therefore, the change does not increase the probability of an accident to occur. A failure of a Recirculation Pump would cause a rapid reduction in core cooling and an increase in reactor reactivity. The changes do not affect any components or interactions with equipment required to scram the reactor and do not affect any radiation release barriers or paths for a release. The change in description of the Drywell Pneumatic Compressors could impact the availability of air to the inboard Main Steam Isolation valves. However, a loss of air to these valves would cause them to fail closed which would not adversely affect primary containment. The change in the description of the HVAC systems is merely editorial which does not degrade or reduce the reliability of the system. The change in the description of Radwaste alarms was previously reviewed under 10CFR50.59 and the current method of Radwaste processing is already described in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the UFSAR change clarifies the possibility of isolating RBCCW to the Drywell Pneumatic Compressor, which could

cause a closure of the inboard Main Steam Isolation Valves and a loss of the normal heat sink. However, there are automatic and manual systems in place to prevent this from occurring, since a backup air supply is present. The operating procedures are being revised by this change to direct the steps necessary to ensure the backup supply is available should the RBCCW supply be isolated. The editorial changes to the UFSAR descriptions do not create any new system interactions. All of the loads added or modified in the tables, figures and text of the UFSAR are identified elsewhere in section 9.2.3. These changes are merely providing consistency and improving the accuracy of the descriptions. The change in the description of the HVAC systems is merely editorial which does not degrade or reduce the reliability of the system. The change in the description of Radwaste alarms was previously reviewed under 10CFR50.59 and the current method of Radwaste processing is already described in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the UFSAR or procedure revisions do not affect the Technical Specifications. The changes are enhancements to the descriptions of several systems, which are not addressed by the Technical Specifications. There are no changes to setpoints, surveillances or bases in the Technical Specifications as a result of the changes.

Tracking No. SE-00-025
Activity No. TC-0068

DESCRIPTION:

This procedure, "Feedwater Level Control System Test Procedure," is designed to demonstrate and provide documentation of observable level set point changes. This allows fine-tuning of the 3-element control. This is a one-time use of this procedure for Unit 2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because Feedwater controller failure during maximum flow demand - UFSAR 15.1.2 - the size of the level step changes will be limited during the procedure, single element and manual control will remain available to the operator at all times for their use, the runout flow control (ROFC) will remain available and will limit the run out or maximum flow condition, and the high reactor water level trip function will remain to mitigate the maximum flow transient. Therefore, the accident that is evaluated in the UFSAR will not be challenged.

   Feedwater controller malfunction demanding closure of the feedwater control valves UFSAR 15.2.7 - the size of the level step changes will be limited during the procedure, single element and manual control will remain available to the operator at all times for their use, and the low low reactor water level trip function will remain to mitigate the loss of flow transient. Therefore, the accident that is evaluated in the UFSAR will not be challenged.

   Feedwater controller failure during maximum flow demand - UFSAR 15.1.2 -This procedure doesn't contain any activities that will increase off site dose. Additionally, the feedwater system is not used to mitigate the consequences of any accident. Therefore, the consequences are not increased.

Feedwater controller malfunction demanding closure of the feedwater control valves UFSAR 15.2.7 - This procedure doesn't contain any activities that will increase off site dose. Additionally, the feedwater system is not used to mitigate the consequences of any accident. Therefore, the consequences are not increased.

The procedure will not cause any malfunctions. If a malfunction were to occur concurrently, the failure of the feedwater level control system has been previously evaluated.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because all operation is within the bounds of the feedwater system and the feedwater system is not used to mitigate any accident. Therefore, the procedure will not create the possibility of a new accident or transient.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the technical specifications do not rely on the feedwater system for any safety function, the margin of safety is not affected.

Tracking No. SE-00-026
Activity No. UFSAR-99-R6-085

DESCRIPTION:

This UFSAR Change corrects the location of and seismic requirements for the Main Steam Line Flow Switch given in Table 3.10-4.

These switches initiate main steam line isolation valve closure and subsequent reactor scram upon excess flow in the main steam lines. Main steam line high flow could indicate a break in a main steam line. A control room HVAC isolation would also occur.

This change will reflect the correct location of the switches in the basement of the Reactor Building, rather than at the ground level floor. From a seismic perspective, the basement has lower seismic requirements.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the UFSAR change reduces the required seismic level for the main steam line flow switch by reflecting its correct location lower in the plant. Consequently, the switch's seismic capability is 2.08 times required, rather than 1.67 times, and the probability of a main steam line isolation valve closure due to a seismic event is decreased by this change. There is no other change in the functional capability of the switches.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the location of the Main Steam Line Flow Switches in the basement of the Reactor Building means seismic requirements are lower than at ground level. Consequently, the switches perform the same function as before this change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because reducing the seismic requirements for the flow switch does not negatively affect the function of the switch.

Tracking No. SE-00-027
Activity No. UFSAR-99-R6-084

DESCRIPTION:

Correct UFSAR Table 6.3-11B, 60% Break Size LOCA SLO PCT (degrees F) value for Siemens ATRIUM-9B fuel from 1623 to 1602, as specified in the base document EMF-96-184(P), Table 9.2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this change does not impact the operation of any plant equipment or plant design. The change is to correct the reported value, the end result of an analytical calculation, and is not related to plant SSCs.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the change corrects the reported value (non-bounding LOCA value) in Table 6.3-11B. As this is the reporting of an analytical value, and nothing is physically changed, no new accident or transient is possible.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not impact the operation of any plant equipment nor plant design and system response. Further, no actual LOCA analysis value is changing, only the UFSAR is being corrected to match the analysis values contained in the EMF-96-184(P) document.

Tracking No. SE-00-028
UFSAR-99-R6-053

DESCRIPTION:

The UFSAR change to Section 3.9.1.1.1 and Section 5.3.1.7 incorporates the results of the latest General Electric RPV closure studs fatigue evaluation to demonstrate that the cumulative fatigue usage factor (CFUF) will remain below 1.0 for the forty year design life. The results of this analysis show that the CFUF for the vessel closure studs to be 0.73 for both Units 1 & 2 at the end of the forty-year design life. This value is well below the allowable CFUF limit of 1.0 established by the ASME Section III Code and as a result justifies at least forty years of operation.

The UFSAR change supports a revision to procedure QCTP 0500-10. This procedure revision eliminates the need to track a limited and specific cycle count for the RPV closure studs. The concern with the studs reaching CFUF of 1.0 in the year 2002 has been eliminated by the updated fatigue analysis.

Also, the UFSAR change to Section 5.3.1.6 where the specific reference to Section II.C.3.A of 10 CFR 50 Appendix H is being deleted, is judged to be editorial and will not be discussed further in this evaluation. Note that the general reference to 10 CFR 50 Appendix H will remain.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increase because the majority of accidents/transients in the UFSAR require the structural integrity of the reactor coolant pressure boundary to be maintained. The accidents/transients include the predominant pressurization and reactor coolant leakage events. The CFUF limit of 1.0 has not changed and the updated fatigue analysis meets ASME Code Section III, affording the same fatigue protection.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the function of the RPV closure studs remains unchanged. The updated fatigue analysis meets ASME Code Section III requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the closure studs updated fatigue analysis meets ASME Code Section III requirements and their function remains unchanged. Therefore, there is no reduction in the margin of safety as described in the Technical Specification Basis.

Tracking No. SE-00-029
Activity No. DCP 9900341

DESCRIPTION:

This activity will install an electrical jumper in Aux Electrical Equipment Room (AEER) panel 902-28 between terminal point BB-21 and BB-23. The existing electrical circuit configuration between these terminal points contains a grounded conductor in a portion of the circuit that provides an interlock/permissive function. This has caused the associated circuit supply fuse to blow causing a rod block signal to be generated. The installation of the jumper eliminates this false signal and permits the Reactor Manual Control System (RMCS) to operate as designed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the activity replaces a defective portion of electrical control wiring and eliminates needless lengths of conductor. The original basis for this extensive length of conductor was to provide a method of assuring a rod block control capability from the service platform during re-fueling operations. Use of this platform at Quad Cities has been physically & procedurally eliminated. Eliminating needless lengths of conductor while providing an electrically equivalent circuit that maintains the original design intent of this branch of the RMCS logic cannot affect the probability of occurrence for any of the accidents/transients listed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changing the overall length of a conductor cannot introduce any additional failure modes. By lifting leads to isolate the defective portion of the affected circuit and installing the equivalent jumper, the original design intent of the circuit is maintained. When considered electrically, the affected portion of the RMCS circuit adds no resistive load & provides only a lengthy jumper. Because the net change in circuit design intent and functionality is "no change", the activity cannot create a possibility of a new of different type of malfunction of equipment important to safety beyond those already evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:

   Section 3/4.2.E "Control Rod Block Actuation Instrumentation"

   The margins of safety as described in the bases for this section are unaffected by the change. The Rod Block Monitor (RBM), Average Power Range Monitor (APRM), Intermediate Range Monitor (IRM) and Source Range Range Monitor operability requirements are unchanged. The change does not affect their method of operation of circuit interfaces. Change does not affect surveillance or operability requirements. Therefore, there is no adverse affect on any margins of safety.

   Section 3/4.10.A "Refueling Operations / Reactor Mode Switch"

   The margins of safety as described in the bases for this section are un-affected by the change. The change does not affect the switch when placed to SHUTDOWN or REFUEL because placing switch in these positions bypasses the affected portion of the circuit. Change does not affect surveillance or operability requirements. Therefore, there is no adverse affect on any margins of safety.

Tracking No. SE-00-030
Activity No. UFSAR-99-R6-092

DESCRIPTION:

This change revises UFSAR Section 13.2.1.1.4 (Radwaste shipments training) to reflect training periodicity requirements of 49CFR172.704 (Department of Transportation training requirements for Hazmat employees) 49CFR172.704 is also being added as a reference to section 13.2.1.1.4.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this procedure does not affect equipment, and has no affect on operating conditions or equipment operation, or introduces new failure modes.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this procedure does not affect equipment, has no affect on operating conditions or equipment operation, and therefore, creates no new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this UFSAR change does not affect equipment, and has no affect on operating conditions or equipment operation.

Tracking No. SE-00-032
Activity No. TC-0086

DESCRIPTION:

This activity will implement a temporary procedure that will test the 2D RHR pump to determine if a bypass flow path exists such that total pump flow is not being measured by the flow element. This activity will close several manual valves and one motor operated valve at various points in the procedure and will make the 2D RHR Pump, the 2C RHR Pump, and the LPCI Mode of RHR inoperable at separate times in the procedure. An operator will be available to return this equipment to a fully operable status should it be required.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because none of the revisions (testing of an RHR pump, valving out the pump minimum flow lines, administratively declaring inoperable the tested RHR pump and making LPCI inoperable by closing the Loop Cross-tie valve) are related to accident initiators. The testing of an RHR pump and administratively declaring inoperable the effected equipment will not increase the consequences of the accident. The condition of

having one RHR pump Out of Service is currently addressed by the Technical Specifications (30 day LCO). This test will only make the RHR pump or LPCI Mode unavailable for the short period of time that it takes to perform the testing. This is well less than the Technical Specification allowable LCO duration. The systems will be returned to their normal status at the completion of the testing and the LCO exited. In consideration of making one RHR pump or the LPCI Mode inoperable for a short period of time, the LOCA analysis/Appendix K analysis is performed with this equipment not available. All reactor cooling parameter limits are satisfied in this condition.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because while the testing is being performed on the RHR pump, the pump or the affected equipment will be administratively declared inoperable. Therefore, the equipment being tested will not be relied upon as equipment that will be expected to perform any function during an accident while the test is in progress. The redundancy required by the Technical Specifications will insure that all other equipment important to safety that performs these functions are operable. No new failure modes are created by this evolution. Adequate minimum flow will be provided by the main flow path of the test loop.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification limit is to allow one RHR pump to be inoperable for up to 30 days and the LPCI mode to be inoperable for 7 days. This testing will make one RHR pump inoperable for well less than 30 days and the LPCI mode inoperable for well less than 7 days. Therefore, the margin of safety is not reduced.

Tracking No. SE-00-034
Activity No. UFSAR-99-R6-091

DESCRIPTION:

This Safety Evaluation addresses the correction of the point at which the refuel bridge auxiliary hoist raising motion is blocked by a drum revolution counter switch. UFSAR section 9.1.4.2.1.2 stated that the hoist raising motion is stopped "about 11 feet below the personnel walkway deck." The limit switch is set to stop the hoist raising motion at a point about 8 feet below the refuel bridge rails (per Vendor drawing W-23260-02-D, Rev. 2). The UFSAR has been revised to reflect this. No physical work has been performed to make this change.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the change made to the UFSAR text describes the point at which the Refuel Bridge auxiliary hoist raising motion is stopped. This does not add any equipment or components. The only type of accident which the Refuel Bridge could be a factor in is the fuel handling accident. To drop a fuel assembly, either the assembly bale, the fuel grapple, or the grapple cable would have to break. These components are not affected by this change, and the probability of occurrence of this accident in not increased. This

activity does not change the probability of occurrence of a malfunction of the Refuel Bridge, and has no effect on any other equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this activity does not add any additional equipment of components. No physical changes are made to the crane, or any of its' components. Therefore, no factor is introduced which could lead to the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the auxiliary hoist normal up limit is not described in the Technical Specifications or any of its bases. The safety features associated with the auxiliary hoist normal up limit switch does not affect the was safety-related equipment will respond to an accident or transient, and has no direct or indirect effect on any margin of safety expressed or implied in the basis for any technical specification.

Tracking No. SE-00-037
Activity No. UFSAR-99-R6-095

DESCRIPTION:

Revise UFSAR Table 6.2-7, PENETRATIONS OF PRIMARY CONTAINMENT AND ASSOCIATED ISOLATION VALVES to accurately reflect the correct closure timing criteria for the 1-1601-61 and 2-1601-61 valves. Also, a change is being made to update UFSAR Table 6.2-7 to correctly indicate the appropriate Equipment Piece Numbers (EPN's) for penetration X-106B for Unit 1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because there is no change in any accident initiators.

   This activity will change the stroke timing listed in the UFSAR back to the value supported by the analysis. These changes do not affect any of the previous analyses for the release of radioactive materials.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because no new components will be added and no deletions will be made as a result of this activity; therefore, there can be no possibility of an accident or transient of a different type than previously evaluated nor increased consequences.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Primary Containment Integrity is maintained and is not changed due to this activity.

Tracking No. SE-00-040
Activity No. UFSAR-99-R6-106; QAP 0300-03

DESCRIPTION:

UFSAR change to Section 13 and associated administrative procedure (QAP 0300-03) revision. Delete the term "Equipment Attendant" and modify or delete those related statements to reflect the updade of the Equipment Attendants into the Equipment Operator position.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this issue does not affect or impact an existing accident or transient; therefore, the consequences or probability of occurrence cannot change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this issue is specifically an individual qualification issue and as such does not impact the possibility of creating an accident, transient, or malfunction. The total number of qualified individuals per shift is not being reduced and the qualification requirements are remaining equal to the ANSI requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the revision of these documents brings the station requirement equal to the requirement stated in the ANSI document; therefore, the margin cannot be reduced.

Tracking No. SS-H-99-0002
Activity No. DCP 9800306; SE-98-138

DESCRIPTION:

DCP 9800306 replaces the EHC system time delay dropout relay board D29 with a like-for-like board, SI# 147849 and changes the time delay settings of two relays in the Main Turbine Trip Reset circuits to provide more time to ensure that reset proceeds to completion when actuated. The EHC system relay D1-D29 time delay setting changes from 12 seconds to 25 seconds (+/- 1 second). The EHC system relay D2-D29 time delay setting changes from 10 seconds to 20 seconds (+/- 1 second).

The purpose of this change is to increase the overall reliability of plant operation by eliminating possible inadvertent trips during the Main Turbine Trip Testing performed weekly.

The change provides additional delay time into the trip reset circuits that block various Main Turbine Trip functions during the weekly trip testing. Thus when testing is complete and the resets are actuated to "unblock" the trip functions, the additional time delay will prevent spurious trips until all trips are unblocked and operational.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the Main Turbine EHC system is equipment important to safety because a malfunction of this equipment could cause a reactor scram. However, as discussed previously, the effect of DCP 9800306 time delay relay settings is to increase the reliability of operation with respect to an inadvertent trip during Main Turbine Overspeed Testing reset. Failure modes related to the changes in the Turbine Trip Reset logic do not result in any new type of failures that could result in a Turbine Trip. There are no new system interactions added which would degrade other equipment to cause an accident. DCP 9800306 has no effect on any other function in the Turbine EHC system and thus has no effect on any SSCs important to safety. Therefore, DCP 9800306 will not result in degrading any equipment important to safety and the probability of a malfunction of equipment important to safety does not increase.

   Increasing the relay's delay time setting causes a slight increase in the time when some of the Turbine Trip functions are not available compared to a total operating time. Thus, a Turbine Trip (valid or spurious) is less likely to occur than before the installation of the subject DCP. Because a Turbine Trip is the initiating mechanism of the accidents addressed in UFSAR Sections 15.2.2.1 and 15.2.3.1, the probability of occurrence of the accidents as well as the consequences of the accidents will not be increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the changes being made to the Main Turbine EHC system do not result in any new interaction with safety-related SSCs or plant equipment important to safety. The changes made to the system have been evaluated for operational impact and do not create any new adverse interaction with any other SSCs that would result in any different kind of accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification is not impacted by this DCP 9800306.

Tracking No. SS-H-99-117
Activity No. NWR 980110255; SE-98-114

DESCRIPTION:

Authorizes the use of a freeze seal as an OOS boundary for Personnel and Reactor Safety to repack valve 2-0305-101-38-27. WR 980110255 will be used to accomplish the valve repack and establish the freeze seal.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because based on the administrative and technical controls associated with the

installation of the freeze seals and the availability of proven contingency measures to prevent leakage, the probability of an accident or malfunction of equipment important to safety is not increased. In addition this activity will be accomplished with the CRD Out of Service and the reactor subcritical, depressurized and subcooled. These conditions are much less severe than those evaluated in the UFSAR Accident Analysis.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the CRD Insert/Withdrawl lines will not be in service during this evolution and does not provide support to any system that is required to be operable during this evolution except for the requirement to maintain primary coolant pressure boundary.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the piping code, USAS B31.1.0, provides minimum requirements applicable to piping maintaining the reactor coolant pressure boundary. The procedural and technical requirements applied to the installation of the freeze seal provide a high degree of confidence that the seal will be reliable. In addition, installation procedures, the contingencies and the reliability of the design of the freeze seal meets or exceeds those expected to be encountered in the plant. These considerations ensure that the reliability of the reactor coolant pressure boundary will be maintained at a level comparable to that of the code for the duration of this repair evolution and the margin of safety is not reduced.

Tracking No. SS-H-99-0126
Activity No. QOP 6400-01; QOA 912-2 A-1 Rev. 2
QOP 6400-03 Rev. 9; QOA 912-2 A-4 Rev. 2
QCOP 6400-08 Rev. 1; QOA 912-2 C-5 Rev. 2
QOS 6400-01 Rev. 17; QCAN 912-2 D-4 Rev. 0
QOS 6400-S01 Rev. 14; SE-99-026

DESCRIPTION:

Revision to the Unit 2 Generator protective relaying scheme installed in accordance with DCP 9900022.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the activities only reference use of a drawing that reflects the relay scheme that has been implemented via DCP 9900022. These activities do not result in a change to how components are operated or when they are operated.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the activities only reference use of a drawing that reflects the relay scheme that has been implemented via DCP 9900022. These activities do not result in a change to how components are operated or when they are operated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the activities are to maintain the integrity of the offsite distribution system and thus ensure the availability of two independent sources of power are available. These support the Technical Specification basis of ensuring that no anticipated single event can cause a simultaneous outage of all the offsite power sources during units operation, accident, or adverse environmental conditions.

Tracking No.SS-H-99-0134
Activity No. DCP 9900041; UFSAR-99-R6-004; SE-98-111

DESCRIPTION:

A fire in the Main Control Room (MCR) can induce a short circuit, which could cause maloperation of the ADS following a reactor SCRAM, resulting in a reactor blowdown. To mitigate the potential of this occurrence, an ADS Inhibit Switch is provided in the MCR to isolate the affected circuit. However, in the event that a MCR operator is unable to inhibit ADS prior to evacuation during an MCR fire, a Remote ADS Inhibit Switch located in the AUXILIARY ELECTRIC ROOM (AEER) can be utilized to prevent blowdown. This modification adds a remote auto-blowdown inhibit switch in the auxiliary electric equipment room which will be used as a back-up to the inhibit switch located in the control room.

UFSAR Change 99-R6-004 revised the UFSAR to include Unit 2 for remote switch.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because there are no functional changes to the ADS logic. The new remote inhibit switch will serve as a backup to the inhibit switch located in the control room. The new switch is mechanically, electrically, and functionally the same as the existing inhibit switch and is seismically mounted. The contacts of the new inhibit switch are normally closed and connected in series with the normally closed contacts of the existing inhibit switch. The inhibit switches are passive components and failure of either the new or existing inhibit switch cannot cause inadvertent ADS actuation. The only credible failure is a failure of the contacts to open on demand, which is no different than the existing switch contacts failure mode. The new switch will not adversely impact any other plant systems or components that could affect the probability of any of the accidents described in the UFSAR nor will it change or alter any controls associated with mitigating these accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the functional logic of the ADS instrumentation and controls as described in the UFSAR is not affected by the addition of the remote ADS inhibit switch. The new switch is a passive component and its contacts will remain closed at all times except during performance of various tests or in the event of a control room fire. The switch addition has no impact on any other system or component important to safety and therefore, will not adversely impact any systems function so as to create the possibility of an accident or malfunction of a different type than previously analyzed in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which the Technical Specification are based. The new switch is a passive component and its normally closed contacts will be connected in series with the normally closed contacts of the existing ADS auto-blowdown inhibit switch. The functional logic of the ADS instrumentation and controls as described in the UFSAR is not affected by the addition of the remote ADS inhibit switch. Also, the switch addition has no impact on any other system or component important to safety and will not adversely impact the function of any other systems.

Tracking No. SS-H-99-0161
Activity No. DCP 9800284; SE-98-115

DESCRIPTION:

Replace the existing obsolete GEMAC Unit 2 Reactor Vessel Narrow Range Level transmitters LT 2-0646-A&B with new Rosemount transmitters. These transmitters provide input to the Feedwater Level Control system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the probability of failure of the new transmitters is not greater than the probability of failure of the old transmitters. The consequences of failure are the same. Both existing and replacement transmitters receive a differential pressure input induced by change in reactor vessel level and produce a corresponding 10 to 50 mA output.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because both existing and replacement transmitters receive a differential pressure input induced by change in reactor vessel level and produce a corresponding 10 to 50 mA ouput. The consequences of failure are the same. No new failure modes are introduced. The replacement transmitters will function to produce a reactor vessel level input to the Feedwater Level control system in the same way as the existing transmitters do.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not affect any parameters upon which the Technical Specification are based, therefore, there can be no effect on the margin of safety.

Tracking No.SS-H-99-0170
Activity No. QCOS 0010-03, Rev. 7; QCOS 0201-08, Rev. 17; QCOS 0203-06, Rev. 3; QCAN 901(2)-3 G-14, Rev. 1; QCTS 0310-04, Rev. 7; QCARP 0500-01, Rev. 2; SE-98-111

DESCRIPTION:

A fire in the Main Control Room (MCR) can induce a short circuit which could cause maloperation of the ADS following a reactor SCRAM, resulting in a reactor blowdown. To mitigate the potential of this occurrence, an ADS Inhibit Switch is provided in the MCR to isolate the affected circuit. However, in the event that a MCR operator is unable to inhibit ADS prior to evacuation during a MCR fire, a Remote ADS Inhibit Switch located in the AUXILIARY ELECTRIC ROOM (AEER) can be utilized to prevent blowdown. This modification adds a remote auto-blowdown inhibit switch in the auxiliary electric equipment room which will be used as a back-up to the inhibit switch located in the control room.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because there are no functional changes to the ADS logic. The new remote inhibit switch will serve as a backup to the inhibit switch located in the control room. The new switch is mechanically, electrically, and functionally the same as the existing inhibit switch and is seismically mounted. The contacts of the new inhibit switch are normally closed and connected in series with the normally closed contacts of the existing inhibit switch. The inhibit switches are passive components and failure of either the new or existing inhibit switch can not cause inadvertent ADS actuation. The only credible failure is a failure of the contacts to open on demand, which is no different than the existing switch contacts failure mode. The new switch will not adversely impact any other plant systems or components that could affect the probability of any of the accidents described in the UFSAR nor will it change or alter any controls associated with mitigating these accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the functional logic of the ADS instrumentation and controls as described in the UFSAR is not affected by the addition of the remote ADS inhibit switch. The new switch is a passive component and its contacts will remain closed at all times except during performance of various tests or in the event of a control room fire. The switch addition has no impact on any other system or component important to safety and therefore, will not adversely impact any systems function so as to create the possibility of an accident or malfunction of a different type than previously analyzed in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which the Technical Specification are based. The new switch is a passive component and its normally closed contacts will be connected in series with the normally closed contacts of the existing ADS auto-blowdown inhibit switch. The functional logic of the ADS instrumentation and controls as described in the UFSAR is not affected by the addition of the remote ADS inhibit switch. Also, the switch addition has no impact on any other system or component important to safety and will not adversely impact the function of any other systems.

Tracking No. SS-H-99-0200
Activity No. DCP 9600453 Rev. 3; DCN 001791M; SE-98-091

DESCRIPTION:

DCP 9600453 Revision 3 and DCN 001791M will relocate the existing Static-O Ring (SOR) pressure switch PS 2-4641-42A for the 2A Emergency Diesel Generator (EDG) starting air compressor (2-4609A) from the compressor skid to a location on the south wall of the Unit 2 EDG room. The existing instrument isolation valve will be relocated and the supporting tubing and control wiring will be reconfigured to accommodate the new location of the pressure switch. Seismically qualified supports will be installed to facilitate the rerouting of the tubing and control wiring. The pressure switch functions to control volume (pressure) in the air receiver tanks by controlling the starting and stopping of the air compressor.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the new components perform the same function as the existing components and are considered more reliable. The replacement components associated with the safety-related pressure boundary and/or safety-related function are purchased and installed safety-related to ensure the pressure boundary of the air system is maintained. The new pressure switch is tested to higher standards than the existing (original) switch. Also, the existing copper tubing is being replaced by stainless steel tubing which is considered an overall improvement due to increased strength. Thus, existing accident or malfunction scenarios are not increased by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the replacement components perform the same function and are considered more reliable than the existing components. No new system interfaces result from this activity. Therefore, new accident or malfunction scenarios are not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because system parameters and technical specification parameters are not altered by the component changes. The new components are considered more reliable than the existing components. Thus, margin of safety is not reduced by this activity.

Tracking No. SS-H-99-0213
Activity No. QCAN 901(2)-5 B-6 Rev 5, GROUP 3 ISOLATION NOT RESET;
QCAN 901(2)-5 B-8 Rev 5, RWCU ISOLATION CHANNEL A/B TRIPPED;
QCAN 901(2)-5 D-9 Rev 4, CHANNEL MAIN STEAM LINE TUNNEL HIGH TEMP;
QCAN 901(2)-5 D-16 Rev 4, CHANNEL B MAIN STEAM LINE TUNNEL HIGH TEMP;
QCOP 1200-07 Rev 15, RWCU SYSTEM COOLANT REJECT;
QCOP 1200-10 Rev 13, INJECTION OF BORON USING THE RWCU SYSTEM;
QCOP 1200-15 Rev 7, OPERATION OF DECAY HEAT REMOVAL MODE OF RWCU;
QOM 2-6800-T08 Rev 4, MCC 28-1A-1 208/110 VAC DISTRIBUTION PANEL;
QOM 2-6800-T11 Rev 6, MCC 19-1-1 208/110 VAC DISTRIBUTION PANEL;
DCP 96000436; SE-99-029

DESCRIPTION:

These procedures are being revised due to the installation and operation of new Reactor Water Cleanup (RWCU) isolation actuation instrumentation for high area temperatures in the vicinity of existing RWCU high-energy piping.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the basic functions of the RWCU automatic isolation system are to initiate an automatic isolation of RWCU and to provide alarm indications in the main control room of high local area temperatures and system isolation. The referenced modification provides the power feeds to the circuitry, a safety-related one-out-of-two automatic isolation logic, and various alarms indicating abnormal temperatures. The referenced procedures only reflect how the system will operate after the modification is installed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changes to the referenced procedures do not change how the RWCU system is operated. These changes only indicate plant configuration and response as a result of the modification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the plant has been made more conservative than previously designed. The referenced modification installs a safety-related local area temperature isolation function to the RWCU system. This isolation function was not part of the original design. As such, the margin of safety has not been reduced.

Tracking No. SS-H-99-0224
Activity No.: QCAN 901(2)-4 A-12 Rev. 4, RWCU AREA TEMP A HI;
QCAN 901-4 A-12 Rev. 2 (delete), RWCU AREA TEMP A HI;
QCAN 902-4 A-12 Rev. 2 (delete), RWCU LEAK DETECTION A HIGH TEMP;
QCAN 901(2)-4 A-13 Rev. 2, RWCU AREA TEMP B HI;
QCAN 901-4 A-13 Rev. 2 (delete), RWCU AREA TEMP B HI;
QCAN 902-4 A-13 Rev. 1 (delete), SPARE;
QCAN 901(2)-4 B-12 Rev. 3, RWCU HI TEMP PANEL A BYP/TEST/FAIL;
QCAN 901-4 B-12 Rev. 1 (delete), RWCU HI TEMP PANEL A BYP/TEST/FAIL;
QCAN 902-4 B-12 Rev. 2 (delete), RWCU LEAK DETECTION B HIGH TEMP;
QCAN 901(2)-4 B-13 Rev. 2, RWCU HI TEMP PANEL B BYP/TEST/FAIL;
QCAN 901-4 B-13 Rev. 1 (delete), RWCU HI TEMP PANEL B BYP/TEST/FAIL;
QCAN 902-4 B-13 Rev. 1 (delete), SPARE;
QCAN 901(2)-4 C-15 Rev. 2, MST HI TEMP RWCU INBD ISOL BYPASS;
QCAN 901-4 C-15 Rev. 1 (delete), MST HI TEMP RWCU INBD ISOL BYPASS;
QCAN 902-4 C-15 Rev. 1 (delete), SPARE;
QCAN 901(2)-4 C-16 revision 2, MST HI TEMP RWCU OUTBD ISOL BYPASS;
QCAN 901-4 C-16 Rev. 1 (delete), MST HI TEMP RWCU OUTBD ISOL BYPASS;
QCAN 902-4 C-16 Rev. 1 (delete), SPARE;
QCOA 0201-05 Rev/ 6, PRIMARY SYSTEM LEAKS (SLOW LEAKS) OUTSIDE PRIMARY CONTAINMENT;
QCOP 1200-08 Rev. 8, RWCU SYSTEM SHUTDOWN;
QCOP 6700-20 Rev. 6, DEENERGIZING MCC 29-1 FOR MAINTENANCE AND REENERGIZE;
QCOP 6700-23, Rev. 4, DEENERGIZING MCC 28-1A FOR MAINTENANCE AND REENERGIZING;
QCOS 1600-06 Rev. 6, ECCS AND PRIMARY CONTAINMENT ISOLATION TRIP INSTRUMENTS OUTAGE REPORT;
QOA 6800-03 Rev. 19, 120/240 VAC ESSENTIAL SERVICE BUS FAILURE;
SE-99-029

DESCRIPTION:

This safety evaluation supports the following revisions to the above listed procedures:

a. Indicate that a Unit 2 RWCU system isolation will result from high temperature in the RWCU Heat Exchanger Room, RWCU Phase Separator Tank area, D Heater Bay, or MSIV Room.
b. Correct annunciator tile wording at Panel 902-4 windows A-12, B-12 and B-13.
c. Indicate that the power supply for Unit 2 RWCU automatic isolation Panel 2202-77A is from MCC 28-1A-1.
d. Indicate that the power supply for Unit 2 RWCU automatic isolation Panel 2202-77B is from MCC 29-1-1.
e. Indicate that Panel 901(2)-5 annunciator B-6 will alarm when an Essential Service Bus failure occurs.
f. Indicate that Panel 902-4 annunciators A-12 and B-12, and Panel 902-5 annunciator B-6 will alarm when MCC 28-1A is de-energized.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the basic functions of the RWCU automatic isolation system are to initiate an automatic isolation of RWCU and to provide alarm indications in the main control room of high temperatures and system isolation. Modification DCP 9600436 provides the power feeds to the circuitry, a safety-related one-out-of-two automatic isolation logic, and various alarms indicating abnormal temperatures. The referenced procedures only reflect how the system will operate after the modification is installed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changes to the referenced procedures do not change how the RWCU system is operated. These changes only indicate plant configuration and response as a result of the modification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of a local area temperature isolation for the RWCU system has made the plant more conservative than previously designed.

Tracking No. SS-H-99-0228
Activity No. QCOP 0600-02 Rev. 7, PLACING MAIN FEEDWATER REGULATOR ON-LINE OR OFF-LINE; SE-99-076

DESCRIPTION:

Indicate that for Unit 1 only, if the Reactor Level Master Controller is in the manual mode of operation, a Feedwater Regulating Valve lock-up will occur when the manual pot is over-ranged fully open or fully closed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the change made to this procedure is informational only. This does not result in a change to any actions being performed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the change made to this procedure is informational only. This does not result in a change to any actions being performed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modified controllers are not in the basis for any Technical Specification, and can not effect any SSC that is. Therefore, there can be no effect on the margin of safety.

Tracking No. SS-H-99-0239
Activity No. QCOP 1200-11 Rev. 17, RWCU SYSTEM START-UP AND PUMP OPERATION;
QCOS 1600-36 Rev. 5, 18-MONTH PCI GROUP 3 AND TIP ISOLATION TEST;
QOS 6500-02 Rev. 31, 4 KV BUS 24-1 UNDERVOLTAGE FUNCTIONAL TEST;
QOS 6500-04 Rev. 15, 4 KV BUS 23-1 UNDERVOLTAGE FUNCTIONAL TEST; SE-99-029

DESCRIPTION:

a. Indicate that a Unit 2 RWCU system isolation will result from high temperature in the RWCU Heat Exchanger Room, RWCU Phase Separator Tank area, D Heater Bay, or MSIV Room.
b. Correct annunciator tile wording at Panel 902-5 windows B-6 and B-8.
c. Indicate that the power supply for Unit 2 RWCU automatic isolation Panel 2202-77A is from MCC 28-1A-1, and that de-energization of the MCC will initiate numerous RWCU system related alarms and an RWCU system isolation signal. Add action to subsequently reset tripped components.
d. Indicate that the power supply for Unit 2 RWCU automatic isolation Panel 2202-77B is from MCC 29-1-1, and that de-energization of the MCC will initiate numerous RWCU system related alarms and an RWCU system isolation signal. Add action to subsequently reset tripped components.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the basic functions of the RWCU automatic isolation system are to initiate an automatic isolation of RWCU and to provide alarm indications in the main control room of high temperatures and system isolation. Modification DCP 9600436 provides the power feeds to the circuitry, a safety-related one-out-of-two automatic isolation logic, and various alarms indicating abnormal temperatures. The referenced procedures only reflect how the system will operate after the modification is installed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changes to the referenced procedures do not change how the RWCU system is operated. These changes only indicate plant configuration and response as a result of the modification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the plant has been made more conservative than previously designed.

Tracking No. SS-H-00-0005
Activity No. QOM 2-2700-01, Rev. 9, U2 H2 WATER CHEMISTRY VALVE CHECKLIST;
SE-99-066

DESCRIPTION:

This mechanical valve checklist has been revised by adding three new valves to the system. These valves are a part of a modification, DCP #990013.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the intent of the Hydrogen Water Chemistry (HWC) system is to prevent and/or retard Inter-Granular Stress Corrosion Cracking (IGSCC) in pressure bounding vessels and components such as the Recirc system piping. The installation of the Noble Metal Injection system and the associated monitoring equipment (along with the added valves) will enhance the HWC system by providing more effective utilization of injected hydrogen and providing a more accurate method of measuring the effectiveness of the HWC system. This will in turn decrease the probability of occurrence or consequences of an accident or malfunction of equipment important to safety that has been previously evaluated.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the skid and its components are not located near any equipment important to safety. The components do not need to be seismically restrained and the design change does not have any EQ concerns. The pressure and temperature ratings of the new piping and valves are appropriate for the application. Thus, there is no increase in failure probability. Also should a failure occur, closing the RWCU containment isolation valves can isolate the new piping. Thus, no new unisolable leak path is created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification requirements, associated action items, associated surveillances, or bases are affected by this design change. Therefore, the margin of safety has not changed.

Tracking No. SS-H-00-0006
Activity No. QCAN 902-53 B-4 Rev. 1, REACTOR O2 CONC HI/LO;
QCOP 2700-01 Rev. 11, HYDROGEN WATER CHEMISTRY SYSTENM STARTUP AND OPERATION;
QCOP 2700-05 Rev. 6, HYDROGEN INJECTION FLOW CONTROLLER OPERATION;
SE-99-066

DESCRIPTION:

a. Revise QCAN 902-53 B-4 to indicate that it is now a spare annunciator.
b. For QCOP 2700-01 revise direction for adjustment of the Hydrogen Demand Adjust controller to indicate that the amount of Hydrogen required is less than currently needed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the intent of the Hydrogen Water Chemistry (HWC) system is to prevent/retard Inter Granular Stress Corrosion Cracking (IGSCC) in pressure bounding vessels and components such as the Reactor Recirculation system piping. The new method enhances the HWC system by providing more effective utilization of injected hydrogen and providing a more accurate method of measuring the effectiveness of the HWC system. Therefore, the changes do not affect the ability of the HWC system to combat IGSCC.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because should a failure of the associated piping occur, the piping can be isolated by closing the RWCU containment isolation valves. Thus, no new un-isolable leak path is created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not affect any parameters upon which Technical Specifications or safety functions are based.

Tracking No. SS-H-00-0009
Activity No. QCOP 1800-01, Rev. 9, OPERATION OF ARM INDICATOR/TRIP UNITS; SE-00-006

DESCRIPTION:

This procedure is being revised to reflect the setpoint change of two Unit 1 Area Radiation Monitors, #32 & 33. The setpoint change was directed by DCP # 9900252.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not

increased because ARMs provide monitoring function only and have no interaction with the operation of the Reactor. The affected ARMs are not used to mitigate any accident or transient described in the UFSAR. Operation of the units remains the same so there is no increase in occurrence of an accident or malfunction. The new setpoint value is low enough to provide timely warning of abnormal conditions. Therefore, the consequences of an accident or malfunction remain the same.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this change affects the alarm setpoint only so that spurious or invalid alarms do not occur but low enough so that an alarm would be provided from abnormal radiological conditions. The setpoint changes do not impact functions from the ARMs since they provide a monitoring function only.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the ARM system is not required by Technical Specifications and therefore, does not affect any margin of safety.

Tracking No. SS-H-00-0015
Activity No. QOS 0005-01 Rev. 70, OPERATIONS DEPARTMENT WEEKLY SUMMARY OF DAILY SURVEILLANCE;
QOS 0005-S01 Rev. 94, OPERATIONS DEPARTMENT WEEKLY SUMMARY OF DAILY SURVEILLANCE UNIT; SE-99-0080

DESCRIPTION:

Once per day in Operational Modes 1, 2 and prior to entering required Modes, record Reactor Vessel pressure and perform a channel check.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the replacement transmitters have a higher reliability (less sensitive to vibration, easier to calibrate, less tendency to drift) and thus will give a more accurate reading of Reactor vessel pressure. Therefore, the probability of equipment malfunction is decreased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because a malfunction in either the pressure switches or transmitters is the same malfunction that could have occurred previously. Therefore, a different type of accident or malfunction has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because a corresponding Technical Specification Amendment has been approved and made effective. In accordance with the Amendment, the procedure changes do not affect the margin of safety and therefore, do not reduce the margin of safety.

Tracking No. SS-H-00-0016
Activity No. QCIS 0700-09 Rev.. 16; QCIS 0700-11 Rev. 5; SE-99-074

DESCRIPTION:

Revise QCIS 0700-9 Rev. 16, PRIOR TO STARTUP NEUTRON MONITORING FUNCTIONAL TEST and QCIS 0700-11 Rev. 5, PRIOR TO STARTUP APRM/RBM DOWNSCALE CONTROL ROD BLOCK FUNCTIONAL TEST. Revisions include precautions, references, diagram steps to bypass and steps to unbypass all associated with the new Unit 2 Oscillation Power Range Monitors (OPRM's).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because this evaluation is limited to Phase I of the Unit 2 OPRM modification when it is not connected to RPS. It is functioning as a monitor only for the interim tune-up period. Bypassing the OPRM's in these procedures is to ensure inaccurate alarms are not received in the control room. These changes do not change the intent or basic function of the procedures.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because on Unit 2, APRM's/OPRM's share power supplies in companion pairs. They are bypassed in these companion pairs ensuring only one scram circuit is effected at a time. Because the OPRM's are not connected to RPS in Phase I, no new accident or malfunctions are introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because OPRM's are not required by Technical Specification in Phase I.

Tracking No. SS-H-00-0019
Activity No. QCIS 0700-15 TC-0044; SE-99-074

DESCRIPTION:

Add steps to LIMITATION AND ACTIONS section clarifying when steps may be NA'ed. Also, added notes not to perform two steps that verify maintenance display messages. The note also states, these two steps "will be grayed out until DCN 001720I is implemented during Q2R16."

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the maintenance display function is not activated in Phase I of the OPRM mod.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because one requirement to N/A a step is that it is NOT Technical Specification related. This is annotated by (TS) next to the step. This ensures that all required testing is performed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification related steps may not be N/A'ed.

Tracking No. SS-H-00-0020
Activity No. QCEM 600-12; SE-00-023

DESCRIPTION:

Changes to QCEM 600-12, Rev 10, are necessary to implement DCP 9900311. The 9-9C contact of the MO 2-2301-8 limit switch, which is currently spare, will be reconfigured and added to the control circuit. Attachment E of QCEM 0600-12 must be revised to reflect this change. The procedure will also be revised to include a cautionary note to prevent maintenance personnel from pre-conditioning the MOVs until all required inspections and testing are complete.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the changes made to QCEM 0600-12 are enveloped in SE-00-023 for DCP 9900311. The inputs and assumptions used for SE-00-023 are valid for these procedure changes. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety will not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because as stated in step 1, the changes made to the procedure are enveloped by the previously performed safety evaluation. The revision to the procedure to incorporate changes made by DCP 9900311 will not create the possibility of an accident or malfunction of a different type.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the procedure changes are enveloped by SE-00-023 for DCP 9900311. Safety Evaluation SE-00-023 determined that no changes to the Technical Specifications are required.

Tracking No. SS-H-00-0027
Activity No. QCAN 901(2)-5 C-13 Rev. 5, CHANNEL A/B REACTOR HIGH PRESSURE;
SE-99-080

DESCRIPTION:

Revise Unit 2 Reactor Vessel high pressure setpoint from 1033 +5 psig to 1026 +5 psig. This change is due to a setpoint change performed in conjunction with DCP 9900091.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the replacement transmitters (DCP 9900091) have a higher reliability (less sensitive to vibration, easier to calibrate, less tendency to drift) and thus will give a more accurate reading of Reactor vessel pressure. Therefore, the probability of equipment malfunction is decreased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because a malfunction in either the pressure switches or transmitters is the same malfunction that could have occurred previously. Therefore, a different type of accident or malfunction has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because a corresponding Technical Specification has already been incorporated in conjunction with this change. This change does not affect the margin of safety and therefore, does not reduce the margin of safety.

Tracking No. SS-H-00-0031
Activity No. QCIS 0300-01 Rev. 8; SE-00-008

DESCRIPTION:

Change QCIS 0300-01 to incorporate time delay setpoint changes resultant from DCP 9900225 and DCP 9900226.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because QCIS 0300-01 Rev. 8 incorporates the time delay setpoint change evaluated by SE-00-008. The thermal level switches receive an increasing voltage signal, that increases to a plateau over time, when water hits their sensors. Curves of this time response are provided by the vendor for each sensor. The voltages reached after four seconds are used as the trip points in this procedure. The purpose of this time lag is to

eliminate spurious scrams caused by steam hitting the sensors instead of actual water level.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because all other changes to the procedure are editorial in nature (e.g. Changing personnel titles, removing document retention notes, removing approval signature block, changing reference to stores item number and adding DCP references).

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because QCIS 0300-01 Rev. 8 incorporates the time delay setpoint change evaluated by SE-00-008 and meets the requirements of UFSAR-99-R6-069.

Tracking No. SS-H-00-0032
Activity No. QOS 0005-S18 Rev. 46, OPERATORS' SURVEILLANCE /TURNOVER SHEETS U-2 EQUIPMENT OPERATOR; SE-99-0080

DESCRIPTION:

The NSO daily surveillance currently documents readings from PIS 2-263-191A/B/C/D which are located in the Cable Spreading Room. The NSO relies upon the U-2 EO to obtain these readings. Consistent with other readings that the NSOs are required to document that are obtained from areas outside the Control Room, the U-S EO will record these readings and the data will then be transferred to the NSO surveillance sheets.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the replacement transmitters have a higher reliability (less sensitive to vibration, easier to calibrate, less tendency to drift) and thus will give a more accurate reading of Reactor vessel pressure. Therefore, the probability of equipment malfunction is decreased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because a malfunction in either the pressure switches or transmitters is the same malfunction that could have occurred previously. Therefore, a different type of accident or malfunction has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because a corresponding Technical Specification Amendment has been approved and made effective. In accordance with the Amendment, the procedure changes do not affect the margin of safety and therefore, do not reduce the margin of safety.

Tracking No. SS-H-00-0033
Activity No. DCR-990424 to revise drawing 4E-2318 to incorporate changes made by modification M-4-2-81-12

DESCRIPTION:

This DCP is to change drawing 4E-2318 so that the breaker size matches that installed in the plant, and shown on drawing 4E-2685A. Modification M-4-2-81-12 changed the location of the feed breakers, it did not change the size of the breakers. Drawing 4E-2318 was missed when updating the drawings during the mod process.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the probability and consequences of an accident or malfunction are not changed because this drawing change makes no change to the physical plant, it only brings the drawings into conformance with the plant. It has been determined that the plant configuration is the correct configuration after reviewing modification M-4-2-81-12. That modification moved both the breaker and that breakers load from one location in the distribution panel to another location. Therefore, as far as the electrical circuit is concerned, there was no change; therefore, an accident or malfunction is not changed. Drawing 4E-2318 was missed when updating the drawings during the mod process.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because neither this DCR or the associated modification made any change to this circuit, other than the physical location in the breaker panel. With no changes made to the schematic, there will be no change in the plant operation; therefore, no change in the types of accidents or malfunctions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not changed because this drawing change makes no change to the physical plant, it only brings the drawings into conformance with the plant. It has been determined that the plant configuration is the correct configuration after reviewing modification M-4-2-81-12. That modification moved both the breaker and that breakers load from one location in the distribution panel to another location. Therefore, as far as the electrical circuit is concerned, there was no change; therefore, the margin of safety is not changed. Drawing 4E-2318 was missed when updating the drawings during the mod process.

Tracking No. SS-H-00-0034
Activity No. QOM 2-3500-02 Rev. 2, U2 FEEDWATER HEATER DRAIN VALVE CHECKLIST; SE-99-109

DESCRIPTION:

Added instrument air isolation valves for various Feedwater Heater and Off Gas components to the valve checklist. Added instrument air accumulator drain valves to the valve checklist.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because DCP 9900120 has upgraded the Moisture Separators. The Feedwater Drain system has been upgraded to handle the increased demand developed by the new Moisture Separators' efficiency. The replacement components are standard for these types of valves and are at least equivalent to the existing components in terms of reliability. This DCP has no adverse impact on existing plant equipment, nor will it affect the operation of any plant equipment necessary to mitigate the consequences of accidents or transients. The systems and components will function as required during a transient condition.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because this modification does not impact any existing plant equipment important to safety. All plant equipment remains available to mitigate the consequences of evaluated accidents or transients.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not affect any parameters upon which Technical Specifications or safety functions are based.

Tracking No. SS-H-00-0036
Activity No. QOP 6800-03 Rev. 15, ESSENTIAL SERVICE SYSTEM; SE-99-029

DESCRIPTION:

Indicate that a Unit 2 RWCU system isolation will occur on ESS load transfers between UPS and MCC 28-2 due to loss of power to the ESS Bus.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the basic functions of the RWCU automatic isolation system are to initiate an automatic isolation of RWCU and to provide alarm indications in the main control room of high temperatures and system isolation. Modification DCP 9600436 provides the

power feeds to the circuitry, a safety-related one-out-of-two automatic isolation logic, and various alarms indicating abnormal temperatures. The referenced procedures only reflect how the system will operate after the modification is installed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changes to the referenced procedures do not change how the RWCU system is operated. These changes only indicate plant configuration and response as a result of the modification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the plant has been made more conservative than previously designed.

Tracking No. SS-H-00-0042
Activity No. UFSAR-99-R-080; SE-99-035

DESCRIPTION:

Revise UFSAR Section 13.1.2.1.9 to delete "Radwaste Supervisor". This position was eliminated at Quad Cities and evaluated under a previous safety evaluation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the change was previously evaluated via a previous safety evaluation and no unreviewed safety question was identified.

2. The possibility for an accident of malfunction of a different type than any evaluated previously in the safety analysis report is not created because the change was previously evaluated via a previous safety evaluation and no unreviewed safety question was identified.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change was previously evaluated via a previous safety evaluation and no unreviewed safety question was identified.

Tracking No. SS-H-00-0045
Activity No. DCR 990433; SE-89-41

DESCRIPTION:

Revise Drawings 4E-2354, 4E-2362, and 4E-2610 to reflect changes made to Turbine Tuming Gear Logic by WR Q32640.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the Turbine Turning Gear is not used to mitigate any accidents evaluated in the SAR. It is for protection of the Turbine only.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the Turbine Turning Gear is not evaluated in the safety analysis. The addition of the contact in the start logic does not change the automatic start of the Turning Gear to protect the Turbine.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Turbine Turning Gear is not included in the Technical Specification.

Tracking No. SS-H-00-0048
Activity No. DCP 9900341; SE-00-029

DESCRIPTION:

This activity will install an electrical jumper in Aux Electrical Equipment Room (AEER) panel 902-28 between terminal point BB-21 and BB-23 and determ the WHT-BLK and GRN-BLK conductors of cable 25540. The existing electrical circuit configuration between these terminal points contains a grounded conductor in a portion of the circuit that provides an interlock/permissive function. This has caused the associated circuit supply fuse to blow causing a rod block signal to be generated. The installation of the jumper eliminate this false signal & permit the Reactor Manual Control System (RMCS) to operate as designed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the activity replaces a defective portion of electrical control wiring and eliminates needless lengths of conductor. The original basis for this extensive length of conductor was to provide a method of assuring a rod block control capability from the service platform during re-fueling operations. Use of this platform at Quad Cities has been physically & procedurally eliminated. Eliminating needless lengths of conductor while providing an electrically equivalent circuit that maintains the original design intent of this branch of the RMCS logic cannot affect the probability of occurrence for any of the accidents/transients listed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because changing the overall length of a conductor cannot introduce any additional failure modes. By lifting leads to isolate the defective portion of the affected circuit and installing the equivalent jumper, the original design intent of the circuit is maintained. When considered electrically, the affected portion of the RMCS circuit adds no resistive load & provides only a lengthy jumper. Because the

net change in circuit design intent and functionality is "no change", the activity cannot create a possibility of a new of different type of malfunction of equipment important to safety beyond those already evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Section 3/4.2.E "Control Rod Block Actuation Instrumentation"

   The margins of safety as described in the bases for this section are un-affected by the change. The Rod Block Monitor (RBM), Average Power Range Monitor (APRM), Intermediate Range Monitor (IRM) and Source Range Monitor operability requirements are unchanged. The change does not affect their method of operation of circuit interfaces. Change does not affect surveillance or operability requirements. Therefore, there is no adverse affect on any margins of safety.

   Section 3/4.10.A "Refueling Operations / Reactor Mode Switch"

   The margins of safety as described in the bases for this section are un-affected by the change. The change does not affect the switch when placed to SHUTDOWN or REFUEL because placing switch in these positions bypasses the affected portion of the circuit. Change does not affect surveillance or operability requirements. Therefore, there is no adverse affect on any margins of safety.

Tracking No. SS-H-00-0069
Activity No. QCOP 6100-12 Rev 10, MAIN POWER TRANSFORMER BACKFEED OPERATION; QCOP 6100-13 Rev 7,MAIN POWER TRANSFORMER RESTORATION FROM BACKFEED OPERATION; SE-99-026

DESCRIPTION:

Add direction to disable the Unit 2 Generator Stability trip for the time period that Unit 2 Main Power Transformer is backfed from the 345 kv switchyard. This is done by maintaining a test switch at Relay House Panel 326 in the open position until backfeed is discontinued.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the procedure change continues to ensure the Unit 2 Generator trip scheme functions as designed. The trip scheme does not cause the generator protective scheme to operate outside its design or testing limits. This does not result in a change to the generator protective scheme interface in a way that would increase the likelihood of an accident. There are no ties, changes, or direct interfaces to equipment required for safe shutdown of the plant. This does not change the acceptance criteria for accident classes described in the SAR or create a new accident initiator.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because the procedure change continues to ensure the Unit 2 Generator trip scheme functions as designed. The trip

scheme responds to a loss of major system components and the possible loss of portions of the Bulk Power System will reduce the probability of the loss of off-site power to Quad Cities Station by relieving local equipment overloads, thereby reducing the likelihood of further transmission trips. The trip scheme does not result in any change of frequency or type of accident described in the SAR and does not create a new accident initiator.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the procedure change continues to ensure the Unit 2 Generator trip scheme functions as designed. The trip scheme maintains integrity of the offsite distribution system and thus ensures the availability of two independent sources of power. This supports the Technical Specification basis of ensuring that no anticipated single event can cause a simultaneous outage of all the offsite power sources during Units operation, accident, or adverse environmental conditions. Therefore, there is no reduction in the margin of safety.