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U. S. Nuclear Regulatory Commission
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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, "Changes, tests, and experiments," subpart (d)(2), we are forwarding Quad Cities Nuclear Power Station's Summary Report of Changes, Tests, and Experiments Completed. This submittal contains 10CFR 50.59 evaluations completed between August 1, 2000 through March 13, 2001. In addition, this submittal contains evaluations completed subsequent to that period which support UFSAR revisions in the station's upcoming UFSAR biennial update.

Should you have any questions concerning this letter, please contact Mr. W. J. Beck at 309-227-2800.

Respectfully,

Wally J. Beck for.

Tim Tulon
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
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

IE47

SAFETY EVALUATION INDEX

FULL 50.59 EVALUATIONS

SE-96-161
SE-98-157
SE-98-160
SE-99-040
SE-99-050
SE-99-072
SE-99-080
SE-99-082
SE-99-110
SE-99-116
SE-99-118
SE-99-122
SE-00-018
SE-00-019
SE-00-031
SE-00-038
SE-00-039
SE-00-043
SE-00-052
SE-00-053
SE-00-054
SE-00-055
SE-00-061
SE-00-062
SE-00-063
SE-00-065
SE-00-066
SE-00-067
SE-00-069
SE-00-070
SE-00-071
SE-00-072
SE-00-073
SE-00-075
SE-00-076
SE-00-077
SE-00-078
SE-00-079
SE-00-080
SE-00-081
SE-00-082
SE-00-083
SE-00-084
SE-00-088
SE-00-089
SE-00-090
SE-00-091
SE-00-092
SE-00-093
SE-00-094
SE-00-095
SE-00-096
SE-00-097
SE-00-098
SE-00-099
SE-00-100
SE-00-101
SE-00-105
SE-00-106
SE-01-001
SE-01-007
SE-01-009

QC-E-2001-005
QC-E-2001-006
QC-E-2001-007

VALIDATIONS OF 50.59 EVALUATIONS

SS-H-99-0212
SS-H-99-0227
SS-H-99-0235
SS-H-00-0059
SS-H-00-0067
SS-H-00-0071
SS-H-00-0072
SS-H-00-0074
SS-H-00-0104
SS-H-00-0124
SS-H-00-0127
SS-H-00-0129
SS-H-00-0130
SS-H-00-0132

ATTACHMENT A

SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

AUGUST 1, 2000 to MARCH 13, 2001

SVP-01-101

Tracking No. SE-96-161
Activity No. DCP 9600344

DESCRIPTION:

Implement permanent plant changes DCP 9600344. Remove the Abandoned Temporary Security Lighting Poles and Sidewalk Heaters South of the Service Building

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 security lighting system and the sidewalk heaters do not have any interfaces with any equipment important to safety. This change to the security lighting system and to the side walk heaters (sidewalk is outside the protected area) will be performed outside of the power block. There are no scenarios that a malfunction of the removed abandon security lighting poles or the removed sidewalk heaters would cause a malfunction of systems that are important to safety or the prevention of off-site dose.

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 removal of the sidewalk heaters will eliminate the additional load on the security diesel and the removal of abandoned security lighting poles will not affect any equipment in the power block. Because there are no interfaces with the safety systems no new malfunctions will be induced on the safety systems.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the security system is separate from the plant systems. There are no interfaces with the plant safety systems and therefore, no changes to the margin or tolerance of any safety equipment or controls.

Tracking No. SE-98-157
Activity No. DCP 9800299

DESCRIPTION:

Cable 12867, which is used as a control cable for MO 1-202-5B may not meet EQ requirements. A replacement of the same size (12/C #16 AWG) that does meet EQ requirements is currently not available. 12/C #14 AWG cable is available, but, will not fit into the existing 1" conduit. A 9/C #14 AWG cable will therefore, be used as a replacement for CA 12867. Wiring changes at the MOV's limit and torque switches will be necessary to allow the use of 9 conductors as opposed to the 12 conductors in the original cable.

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 purpose of the design change is to replace the currently installed cable with a environmentally qualified larger conductor 9/C #14 AWG cable. The change

will eliminate one of two conductors connected to both low and high side of the transformer without changing the electrical logic or function of the circuit. This is accomplished by adding jumpers to the limit/torque switches at each location. In addition, the change will reduce the voltage drop in the circuit because the new cable has a lower resistance value than the installed cable.

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 DCP will not change the circuit logic or function. Since an Appendix R fire is not postulated for the drywell, and due to the location of the conductors in the control circuit, the new cable configuration is not susceptible to "Hot Shorts". Since the new cable will follow the existing routing points, no new fire zones will be entered. The new cable utilizes larger conductors, which have a lower resistance than the existing cable. This produces a slight improvement in voltage at the control components. In addition, the new cable is environmentally qualified which ensures that the circuit will perform its intended design function. The new cable will be procured and installed Safety-Related. The applicable EQ terminations will be performed per approved plant EQ procedures. Based on this, equipment failures and their effects remain unchanged and no new failure modes will be created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the MOV operational characteristics have not been changed. The new cable produces less voltage drop, which is in a conservative direction.

Tracking No. SE-98-160
Activity No. DCP 9800294

DESCRIPTION:

Cracking discovered in welds 02BS-14, 02-AD-F8, 02BS-F7, 02BS-F4, and 02AS-F9 in the Unit 1 Recirc System will be repaired by a weld overlay repair method. Prior weld overlays were designed in accordance with NUREG 0313, Rev. 2 and the NRC approved the overlays on a case by case basis. However, since the previous installation, NRC has adopted Code Case N-504 in Regulatory Guide 1.147 Revision 11. The weld overlays for Q1R15 are designed in accordance with ASME Section XI, IWB-03640 and ASME Code Case N-504. This safety evaluation included Weld #02-AS-F9 (DCP 9800294) which was later evaluated as acceptable for continued operation. The associated DCP was not installed.

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 planned repairs will restore the affected piping back to its full design capabilities. The repair works in two ways. First, the overlay metal has a much lower susceptibility to cracking as compared to the base metal and thus, crack propagation will essentially be "arrested" at the overlay metal boundary. Secondly, the process of the deposition of molten metal in the weld overlay will place the region of the crack into compression when the overlay cools to the ambient temperature of the pipe. Placing the crack into compression has the affect of suppressing crack propagation. The crack; therefore, will be severely inhibited in its ability to grow to the point where it could cause

piping failure. With the mechanism for crack propagation inhibited, the probability of failure/accident is greatly reduced.

The consequences of an accident or of a malfunction of equipment important to safety are not increased because nothing in the planned repair has any mechanism to affect the magnitude of any radioactivity released from the ruptured piping. The operating characteristics of the reactor and the reactor coolant system are not changed; therefore, the motive force for a release remains unchanged. These repairs in no way affect the source term available for release. The consequences of an accident are therefore, unaffected.

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 are passive in nature and are changes only in the sense that additional metal is weld deposited over the affected areas to reinforce the cracked area of the piping and establish a boundary beyond which the crack will not be able to propagate. The function of the piping remains unchanged. The way the piping performs its function is unchanged. Because the system is unchanged except for the deposition of additional metal on the external surfaces of the pipe, there is no possibility of creating an accident or malfunction of a type different from those previously evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change restores the effected piping to full design capabilities and does not affect any parameters upon which Technical Specifications are based.

Tracking No. SE-99-040
Activity No. DCP 9900079, 9900080, UFSAR-99-R6-134

DESCRIPTION:

These DCPs will modify the control logic associated with the Unit 1 and 2 HPCI Auxiliary Oil Pump (AOP), the Emergency Oil Pump (EOP), and the Motor Speed Changer (MSC). Time delay relays will be incorporated into the control circuits of the AOP and EOP to prevent the AOP from tripping unexpectedly, and to prevent the EOP from starting during a HPCI initiation. The Motor Speed Changer (MSC) control circuits will be modified to trip the motor when the MSC reaches its High Speed Stop (HSS). These DCPs will also incorporate a seal-in function to allow the AOP and MSC control circuits to remain energized after receipt of a LOCA signal regardless of how long the LOCA signal is present, and will install a banana jack adapter on the MSC control circuit terminal boards. A change to the UFSAR will be performed under UFSAR-99-R6-134 to incorporate the above modifications.

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 that will be made by these DCPs are to the AOP, EOP, and MSC control circuits. These control circuits require a HPCI initiation to start the oil pumps or operate the motor speed changer. Due to the arrangement of these control circuits, they cannot produce a HPCI initiation. Since the AOP and EOP also have no

interfaces with the safety or relief valves, the changes cannot affect the probability of an inadvertent opening of these valves. There is also no credible scenario where the AOP, EOP or the MSC could cause a LOCA.

The changes will not adversely affect HCPI operation or any other system/component required to mitigate the consequences of an accident or transient. The logic changes will prevent premature tripping of the AOP during HPCI startup. Both the AOP and EOP will deliver the proper oil pressure to the system when required. The activity does not affect the analyzed sizing or capacity of the 250 VDC batteries.

The changes made to the MSC circuits will ensure that the HPCI turbine achieves rated speed under LOCA conditions regardless of how long the LOCA signal is present. The reconfiguration of the MSC control circuit will also ensure that the motor is not damaged when the MSC reaches its HSS. These measures will not adversely affect HPCI or safety/relief valve operation.

Since the changes made to the AOP, EOP and MSC will enhance HPCI operation and will not affect it in an adverse manner, both the HPCI system and the safety/relief valves will function as required to mitigate the consequences of the accidents/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 the changes made by these DCPs only affect the AOP, EOP and MSC control circuits. A failure of these control circuits could adversely affect operation of the HPCI oil system, which could eventually cause a failure of HPCI itself. However, as stated in the previous questions, these failures have been previously evaluated (where the HPCI system is backed up by the ADS system). There are no other credible failures that could create a malfunction 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 changes will have no adverse affects on the operation of the HPCI system. HPCI will still meet its design requirement of delivering water at 5000 gpm within 45 seconds. The changes also will not adversely affect the 250 VDC system under plant operating or shutdown conditions.

Tracking No. SE-99-050
Activity No. DCP 9700349

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 1 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 1-5600-11 and PS 1-5600-12 to PS 1-5600-11A and PS 2-5600-12A. 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.

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-99-072
Activity No. Fire Protection Report Change 99-08

DESCRIPTION:

Combustible Load Calculation, QDC-4100-M-0691 Rev.1, revises the methodology for computing combustible loading. The following are changes from the previous combustible loading evaluation:
1. The heat release potential of materials has been updated using recent information.
2. Oil filled transformers that were previously assumed to not contribute to the combustible loading of a fire zone has been added into the combustible loading.

3. The combustible loading due to oil systems was changed to account for the possibility of pressurized oil piping breaks.
4. Combustible loading that has been added to the plant through the modification process and individual evaluations and tracked as transients have been incorporated into the base combustible loading.
5. The method for determining the amount of transient combustible in a fire zone has been changed to provide a more realistic determination of the possible combustible loading in a fire zone due transient combustibles. Plant personnel who perform work in the plant provided input as to the combustible material required for typical jobs that occurred in the plant. The combustible materials were converted into Heat Release Potential and all possible jobs that may occur a given fire zone were then summed to provide a bounding transient load.

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 a design basis fire has not been changed by the additional combustible loading. The probability of the failure or malfunction of a fire barrier has been evaluated by fire protection engineers and determined to not have increased significantly. The consequences of the accident or failure of equipment have remained unchanged by the change in combustible loading.

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 amount of combustible material is not a precursor of any accident other than a fire; therefore, the are no new accidents or transient created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the increase in the combustible loading does not effect any of the equipment required to achieve and maintain safe shutdown. The evaluations in Section 4.3 of the UFHA provide justification that the fire will not spread from one fire area to another. Therefore, the ability to safely shutdown has not been effected by this change.

Tracking No. SE-99-080
Activity No. DCP 9900090; 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-99-082
Activity No. DCP 9900169

DESCRIPTION:

This modification supports the Quad Cities Safe Shutdown Analysis (SSA) optimization program.

Cables 14216 and 14217 provided the normal feed to Unit 1 125Vdc Bus 1B-1 from Unit 2 Bus 2A. The modification will perform the following: The existing feed will be disconnected at Unit 2, 125 Vdc Bus 2A in the 2A Battery Charger Room, cut back and left abandoned-in-place. New cables will be connected to Bus 2A and routed in tray and a new dedicated conduit. The conduit will be wrapped with Fire Protective Wrap in designated areas. A new splice box will be installed in Fire Area TB-III. Existing cables 14216 and 14217 will be rerouted to the new splice box and the existing cable will be spliced with the newly routed cables. To facilitate the reroute, penetrations will be opened in five existing fire barriers. These penetrations will be fire sealed to the rating of the barrier using station-approved methods and details following installation. As the new cables will be spliced to the reused portion of the existing cables, the new cables will be assigned the same numbers as the existing cables. To allow for tracking of the abandoned cables, the abandoned-in-place cables will be identified as cables 13993 and 13994.

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 supports the SSA optimization program. The effect of this modification will not alter the existing plant 125 Vdc system operating breaker configuration, any protective devices or the SSA. The power cables being rerouted and/or protected are currently not credited for post fire operation in Fire Zones TB-I and TB-II. A portion of the relocated cable will be covered with fire barrier material. Vendor data provided indicates that the Darmatt KM-1 fire barrier material used is designed to meet the requirements of NRC Generic Letter 86-10, Supplement 1. Derating factors for the protected cables has been provided to SLICE to properly evaluate the current load and any future load additions to these cables. The additional weight of the fire wrap has been accounted for in the design of the new conduit and conduit supports. Cables that are abandoned-in-place by this design change are identified with new cable numbers to allow the abandoned cables to be tracked by SLICE. The abandoned-in-place cables cannot increase the probability of occurrence of an accident, as they are de-energized. The impact of this modification on all affected SSCs has been determined and evaluated and is acceptable. Therefore, the change cannot increase the probability of an accident.

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 cable loading with its reduced ampacity is still within the design requirements of the existing loads. The revised ampacity is provided to SLICE to properly evaluate future load additions to these cables. The additional weight of the fire wrap has been accounted for in the design of the new conduit and conduit supports. All aspects of the installation have been evaluated and the installation will not adversely impact any Structures, Systems or Components (SSCs). Therefore, there can be no 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 bases to Technical Specifications 3/4.9.E & F provide no specific limits but rather discuss the availability of ac and dc busses which provide power to ECCS components. Following installation, the modification will not affect the reliability, availability or operability of any ECCS component, because the function of Cables 14216 and 14217 is unchanged. During installation, loss of the normal power feed to Bus 1B-1 will occur as a result of the cable determination, re-routing, splicing and re-termination of Cables 14216 and 14217. The appropriate actions as described in Technical Specification Sections 3/4.9.E & F will be implemented when the normal feed to Unit 1 Bus 1B-1 is out of service. Following the installation of the change all SSCs will perform their designed safety functions, meet all of their design requirements and no margins of safety will be reduced with respect to the affected systems.

Tracking No. SE-99-110
Activity No. DCP 9700395

DESCRIPTION:

The activity will modify the power feed circuit for MO 1-1001-47, "Residual Heat Removal (RHR) Shutdown Cooling (SDC) Outboard Isolation Valve" to provide a disconnection means for the MOV in the Unit 1 D heater bay. The 4 conductor power cable, which extends from the 250 VDC MCC to the MOV, will be disconnected at the valve motor. The cable will then be pulled back and terminated on a new 3 pole disconnect switch. The switch will be mounted adjacent to the valve motor. Three of the four conductors coming from the MCC will be terminated at the incoming terminals of the switch. The fourth conductor will be connected to a terminal strip mounted inside the disconnect switch. A new cable will be installed and connected to the other side of the disconnect switch. The other end of the new cable will be terminated at the valve motor. New conduit and supports will be required to facilitate installation of the disconnect switch, and interconnecting cable.

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 addition of the new disconnect switch only changes the method of removing power from the valve during reactor operation above 100 psig. This is presently accomplished by lifting the motor leads. There are no new interfaces of equipment, affects on the operation of the plant, changes to the valve control circuit, affects to the valve's power supply, structural effects, affects to accident/transient initiating events, or affects to offsite radiation doses.

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 only effect of this modification is to change the method by which electrical power is removed from MO 1-1001-47 during reactor power operation above 100 psig with the valve in the closed position. The same purpose is achieved currently by lifting motor leads. Operating procedures will be revised to reflect the different method of disconnecting power from the valve and there will be no change to the current function of the valve. Because there are no new equipment interfaces created by the modification and the valve will operate as before, the modification does not create the possibility of an 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 no Technical Specification is affected by the change. No margin of safety as described in the basis for any Technical Specification is reduced.

Tracking No. SE-99-116
Activity No. DCP 9900093

DESCRIPTION:

The activity is to replace six recorders on the main control room panels currently in use by the station. The equipment to be replaced are: a Torus Water Temperature Recorder, TR 1-1640-08 (GEMAC Model 521 2-pen); a Main Generator Gross Megawatts Recorder, JR 1-6040-13 (GEMAC Model 50-520 1-pen); two Containment Hydrogen/Oxygen Concentrations Recorders, UR 1-2406-A/B (L&N/SPDMX M 2-pen); and two Drywell and Torus Gamma Radiation Monitor Recorders, RR 1-2420-A/B (Baily Model 732 2-pen). Respectively, recorders will be replaced by a Torus Water Temperature Recorder (Yokogawa RS1000 Model 436502 2-pen); a Main Generator Gross Megawatts Recorder (Yokogawa RS1000 Model 436501 l-pen); two Containment Hydrogen/Oxygen Concentrations Recorders (Yokogawa RS1000 Model 436502 2-pen); and two Drywell and Torus Gamma Radiation Monitor Recorders (Yokogawa RS1000 Model 436502 2-pen). The installation of these new recorders will enhance the capability of recording and/or indicating the various system variables.

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 replaced recorders do not connect (mechanically) directly to the pressure boundary of any systems or provide automatic control of any system components. Any connections to safety-related power sources are provided with appropriate isolation protection devices in the event of an electrical fault. Failure of the replacement recorders will not affect the operation of any system. Therefore, the activity will not increase the probability of occurrence of any accident or transient.

   Replacement of the recorders is to improve equipment reliability and reduce maintenance. Therefore, the activity will not increase the consequence of any accidents or transients.

   The probability of recorder failure is less with the new recorders than the existing ones. Because no new interfaces with safety-related equipment are being created by this design change, the probability of a 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 the impact of any recorder failure is no different with the new recorders than the existing ones. The initial conditions or consequences of an accident are not adversely affected. Release paths and containment systems that can affect the off-site or control room dose consequences are unaffected.

   Installation of replacement recorders has been evaluated. The evaluation ensures that the new recorders will perform their intended functions within design limits. There are no new interfaces or system interactions created by this modification. The ability of each of the overall systems affected by this change to achieve its specific design function is not altered by this design change. Therefore, the activity 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 there is no Technical Specification whose margin of safety is affected by the recorders being replaced by this modification. Therefore, replacing the recorders can not reduce the margin of safety as described in the Technical Specification bases. There is no adverse impact on surveillance intervals identified in Table 4.2.K-1 because the drift specification for the replacement recorders is equal to or better than that of the recorders being replaced.

Tracking No. SE-99-118
Activity No. DCP 9500009

DESCRIPTION:

This DCP will enhance plant operation by eliminating the potential for a reactor scram caused by failure of the Main Turbine Stop Valve (MSV) MSV-2 open limit switch. In the present configuration, if the MSV-2 open limit switch (SVOS-2) fails closed, the non-controlling, or following, MSV-1, 3 & 4 would close. Closure of these valves will lead to a reactor scram. The

design intent is for all valves to go closed based only on turbine speed or as selected, "all valves closed", to stop steam flow to the turbine to effect turbine shutdown. General Electric (GE) recommends removing the master/slave relationship between MSV-2 and the non-controlling MSVs as a design enhancement in accordance with TIL-1212-2.

The purpose of this DCP is to revise the MSV logic such that the non-controlling MSVs will no longer respond to an inadvertent change in the MSV-2 position. This DCP also removes the control lockout feature that was part of the master/slave relationship, provided to ensure that MSV-1, 3, and 4 could not be reopened unless MSV-2 was at least 90% open. These enhancements help eliminate needless turbine trips, particularly during testing evolutions.

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 removal of the MSV master/slave relationship has no affect on any of the assumed initiating events for evaluated accidents/transients. Removal of the master/slave relationship will decrease the probability of a single component failure causing an unnecessary reactor scram and turbine trip. The turbine speed circuitry is not adversely affected because this change adds only a relay to the control circuit designed to be energized after "all valves closed" is initiated by turbine speed or selected. This relay is energized after the logic affecting turbine speed, via this circuitry, is already satisfied. This activity has no affect on the response of the plant to accidents. After the DCP is installed, the reactor scram functions will be accomplished just as they were before the change. Turbine trip functions that play a part in the response to the listed accidents are unaffected by the modification.

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 introduced and there is no effect on the response of the plant to accidents. Turbine speed circuitry is not adversely affected because this change adds only a relay to the control circuit designed to be energized after "all valves closed" is initiated by turbine speed or as selected. This relay is energized after the logic affecting turbine speed, via this circuitry, is already satisfied. After the DCP is installed, the reactor scram functions are unchanged. Turbine trip functions that occur as a result of the listed accidents are unaffected by this DCP. This DCP affects the non safety-related MSV control circuitry. The activity will enhance plant operation by eliminating the potential for a reactor scram caused by failure of the MSV-2 open limit switch. This is a recommended action in accordance with GE TIL 1212-2. The design intent is for all valves to go closed based only on turbine speed or all valves closed selected. The turbine speed circuit is also non safety-related and is not adversely affected since failure of the relay contacts will not affect the relay coil installed in that circuit. This change uses control relays and minor wiring changes that do not introduce new modes of failure.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no technical specification is affected by the change. No margin of safety as described in the basis for any technical specification is reduced.

DESCRIPTION:

1. Install an S&C Bankgard type LUC over-voltage relay and adjustable shunt resistors, all in parallel with the existing main generator backup ground relay (ABB type CV-8) located in the broken delta auxiliary potential circuit of the main generator relay and metering potential transformer circuit. A new enclosure equipped with a louver kit for ventilation will be installed to house the LUC relay and adjustable shunt resistors. The enclosure will be mounted to Unistruts anchored to the floor and ceiling in the Auxiliary Electric Equipment Room (AEER). New cable, conduit, raceways will be run inside the AEER as necessary to connect the new equipment. These changes will prevent an unnecessary main generator trip when minor problems occur in the potential transformer circuit.

2. Modify and relocate the existing warm-up SV relay circuitry in the tripping circuit. The modified tripping circuit will trip the generator primary system lockout relay instead of the generator backup system lockout relay.

3. Install a second SV warm-up ground relay connected to monitor the residual voltage at the broken delta auxiliary potential transformers on the potential circuit that will trip the backup system generator lockout relay.

4. Install new auxiliary relays GOL1X and GOL2X to provide interlocking capability for both the sensing and tripping circuits of the SV relays.

This Safety Evaluation was also used for DCP 9900129, 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 modifications are to the circuits, which provide fault protection to the main generator by tripping it. There is no affect on the probability of occurrence of the initiating events for any of the related accidents/transients. The modifications do not affect the consequences (offsite dose) of a generator trip, they ensure that a trip will occur when required. The plant's response to a trip remains unchanged. Therefore, offsite doses are not increased.

   The activity has no affect on malfunction of equipment important to safety. There are no newly created interfaces with any equipment important to safety. The components altered or added by this modification are not safety-related, and are not required to function in order to mitigate any accident.

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 equipment being added is not important to safety. There are no newly created interfaces with any equipment important to safety. The effects of the modification on normal structural loads, supports,

seismic, Category II over I, electrical, instrumentation & control, and fire hazards have been considered. The modifications cannot 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 no Technical Specifications are affected by this modification nor is the operation of any SSC, as addressed by the Technical Specifications, altered. No margin of safety involved with any Technical Specification is changed.

Tracking No. SE-00-018
Activity No. DCPs 9900275, 9900276, and 9900277; UFSAR-99-R6-122

DESCRIPTION:

The activities are modifications to incorporate a one second time delay into one of the initiation signals for the Emergency Diesel Generator (EDG) Automatic Start Relay (ASR). This delay will be associated with the signal from the Main and Reserve Feed breakers at the 4 KV Bus. The purpose of this modification is to prevent the inadvertent start of the EDG during a successful automatic transfer of the Main and Reserve Feed breakers.

A permanent change to the UFSAR is required. Wording will be incorporated into Section 8.3.1.6.4 of the UFSAR to describe the 1 second time delay associated with the automatic start of the EDG due to the Main and Reserve Feed breakers both being open. This revision will be tracked under change number UFSAR-99-R6-122.

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 only affects the automatic start logic of the EDG. The automatic start logic only responds to accident/transient conditions. Therefore, the probability of occurrence probability of occurrence of an accident is not increased.

   The EDG automatic start logic will still function in the same manner to mitigate the consequences of an accident. Therefore, the consequences of an accident have not been increased.

   The new time delay relays being installed are rated for the application. The same types of relays have already been qualified for use in the EDG start logic and have been proven suitable and reliable. Therefore, the probability of a malfunction has not increased.

   The malfunctions associated with the new components are exactly the same as the malfunctions of the existing components. The failures associated with these malfunctions are not changed. Therefore, the consequences of a malfunction 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 malfunctions and failures associated with the new components are exactly the same as the existing components.

Therefore, the possibility for an accident of malfunction of a different type than any evaluated previously in not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification Bases associated or affected by the change. Therefore, there will be no reduction in any margin of safety.

Tracking No. SE-00-019
Activity No. DCP 9900119, Rev. 2

DESCRIPTION:

This Revision to the DCP is to move the scope of work associated with replacing the Moisture Separator (MS) internals (per DCN 001909M) from DCP 9900119 to DCP 9900590, which will be completed during a future outage. Because of dose and contamination problems discovered when the Low Pressure Heater Bay was entered during Refuel Outage Q1R16, the replacement of the MS internals has been deferred. The replacement of these internals was considered a system improvement, which would improve the moisture removal efficiency of the MS and increase the electrical output of Unit 1. There are no regulatory commitments to replace these internals. Therefore, operation with the current internals is acceptable, and the installation of the MS Internals may be deferred to DCP 9900590. The Heater Drain valve modifications (per DCN 001910M) will continue to be installed.

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 and feedwater heater drain system, heaters, and valves are not being changed by this modification. The system and its components will function as required during accident or transient conditions because component failure modes are unchanged. Therefore, there is no increase in the probability or consequences of an accident or 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 the existing plant equipment to function as required during all operational or accident modes. The moisture separators and heater drain system, heaters, and valves will continue to perform their intended functions. There will be no effect on equipment failures or malfunctions as a result of this modification. Therefore, the possibility of a different 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.

DESCRIPTION:

The purpose of this design change is to improve the heat removal capacity of the Train B Control Room (CR) HVAC Refrigeration Condensing Unit (RCU) by increasing the design cooling water flow rate from 120 gpm to 130 gpm, increasing the setpoint of the cooling water flow control valve to maintain a condenser refrigerant pressure of 285 psig, and modifying the cooling water piping connections to the RCU to provide a 6-pass configuration. This design change will also add a local pressure indicator, a differential pressure indicator across the cooling water connections to the RCU, and a temperature indicator on the RCU cooling water outlet piping. This design change will also replace the downstream flow indicator with a venturi-type that has a greater flow range. The addition of these instruments will improve the ability of the station to trend the performance of the RCU and will have no effect on the operation of the 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 this design change does not affect the integrity of the reactor coolant pressure boundary or of any system connected to the reactor pressure boundary or any steam system outside containment. The increased RCU condenser/compressor discharge pressure is less than the design pressure of the associated components and within the vendor recommended maximum operating range. The RCU has adequate capacity to provide the required cooling to the control room under design basis accident conditions with the increased condenser operating pressure and cooling water flow rate. This ensures that the control room will be maintained within the required environmental/temperature conditions following a design basis accident. The increased design cooling water flow rate will not prevent the RHRSW system from providing its required mitigating function (containment cooling) following a LOCA inside containment. Other than these two functions, this design change has no effect on any release barriers or accident mitigation systems or equipment. Therefore, the probability or consequences of any accidents will not be increased by this design 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 design change does not create any new failure modes or any new system interactions or dependencies. The new operating parameters for the condenser pressure and design cooling water flow rate are within the capabilities of the compressor and condenser and will have negligible effect on the service water and RHR service water systems. The addition of the new instrumentation is for trending purposes and does not affect the functions or failure modes of the system or affect any interactions with other systems. 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 this design change will improve the heat removal capacity of the CR HVAC RCU by increasing the cooling water design flow rate and the number of passes through the

condenser and by increasing the refrigerant pressure and temperature in the condenser. This change improves the ability of the CR HVAC system to meet the cooling requirements of Technical Specification 3/4.8.D, and will ensure that the RCU will not demand more than the design cooling water flow rate (130 gpm) from the RHRSW system. The increased design cooling water flow rate from 120 gpm to 130 gpm will have an insignificant impact on the discharge pressure of the RHRSW system (Technical Specification 3/4.8.A); however, the surveillance that verifies this requirement will be updated to set the cooling water flow rate to CR HVAC to 130 gpm while verifying the ability of a RHRSW to meet its discharge pressure requirement.

Tracking No. SE-00-038
Activity No. UFSAR-99-R6-098

DESCRIPTION:

1. Revise Table 15.6-7, Loss-of-Coolant Accident Input Parameters for Control Room Dose Analysis, to reflect the corrected Design Basis Assumptions form the most recent Control Room Dose Calculation.
2. Revise Table 15.6-8, Loss of Coolant Accident Control Room Radiological Effects, to reflect the corrected Thyroid doeses from MSIV leakage, stack release and total control room doses.
3. Revise Sections 6.4.4.1 and 15.6.5.5.3 to show that the design basis thyroid dose for Control Room personnel following a design basis Loss of Coolant Accident is 29.4 rem.
4. Revise Section 9.4.1.1 Control Room Area HVAC System section to refer to section 15.6.5.5 for information on radiation exposure to control room personnel.
5. Revise Endnotes for Section 15.6 to reference ComEd letter to NRC dated 1/8/93.
6. Revise Section 6.2.3.2.1 for editorial change to show the correct P&ID.

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 Control Room HVAC System is an accident mitigation/normal operation system that is not directly connected to reactor operation or reactor coolant system operation. Revising the Control Room Dose Analysis parameters and radiological effects will not affect the normal or emergency operation of the Control Room HVAC System, or any Reactor operation systems. The editorial change for the turbine building and reactor building interlock doors is simply to show the correct layout drawing instead of an incorrect piping drawing. Changing the drawing number will have no effect on creating the probability of an 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 revision to the Control Room Dose Analysis assumptions and Radiological effects will not create any new failure modes for the Control Room HVAC System, Standby Gas Treatment or Control Room instrumentation. The Control Room HVAC System does not interact with the reactor coolant system or any related system that could cause an accident or transient. Changing a drawing number related to the reactor/turbine building interlock doors will not affect how

those doors function and will not cause any equipment failures or malfunctions in either building.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Design basis thyroid dose for Control Room Personnel is 29.4, which is below the 30 rem thyroid does number put out in General Design Criteria 19. The increase from 21.88 to 22.8 rem does not exceed the design basis dose of 29.4 rem and does not reduce the margin of safety between 29.4 rem and 30 rem limit from GDC 19. The correction to the section for the interlock doors has no technical changes that allow it to operate differently than is described in the SAR. The change is editorial and will reduce the margin of safety.

Tracking No. SE-00-039
Activity No. DCPs 9900272, 9900273, and 9900274; UFSAR-99-R6-123

DESCRIPTION:

The activity is a modification to be performed on each of the three Emergency Diesel Generator (EDG) systems at Quad Cities station and the activity will accomplish two separate changes.

The first change will be to install a "Re-Crank" modification on the starting air system for the EDG. One solid state repeat cycle timer (Agastat Series SCR) and a pressure switch (Static-O-Ring) will be incorporated into the starting air system to retract and re-engage the pinion gears on the two air driven starting motors. The effect of this portion of the activity will be that during a diesel start command, if either air start motor pinion gear fails to fully insert to engage the fly wheel (condition referred to as an abutment), the pinion gears will automatically be retracted and then re-engaged. The new timer and pressure switch will be set such that the pinion gears could be cycled up to 5 times before the starting sequence is stopped by the existing Start Failure Relay.

The second change being implemented by the activity is a replacement of the existing electro-pneumatic time delay relay TD-1 with a solid state time delay relay (National Technical Systems Series 812). There will be no effect on the existing system as a result of the replacement of the relay.

Permanent changes will be made to Section 9.5.6 (Diesel Generator Starting Air System) and Figure 9.5-3 (Diagram of Service Air Piping Diesel Generator Air Start) of the UFSAR. Section 9.5.6 will be revised to incorporate a description of the Re-Crank function and Figure 9.5-3 will be revised to accurately depict the new pressure switch and time delay inputs for the Re-Crank function. This UFSAR change will be tracked under change number UFSAR-99-R6-123.

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 does not affect or interconnect with any system, structure, or component that can initiate any accident or transient other than the failure of one EDG to start. The activity will actually decrease the probability of this 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 malfunctions and failures associated with the new components are exactly the same as the existing components. Therefore, the possibility for an accident of malfunction of a different type than any evaluated previously in not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification Bases associated or affected by the change. Therefore, there will be no reduction in any margin of safety.

Tracking No. SE-00-043
Activity No. QCOP 5400-02, Rev. 6; QCOP 5400-01 Rev. 13; QCOP 5400-08 Rev. 5; QCGP 1-1, Rev. 35; QCOP 2-1 Rev. 29, QCGP 2-3 Rev. 33; UFSAR-99-R6-109

DESCRIPTION:

This Safety Evaluation is for a change to the UFSAR and a number of associated procedure changes. The UFSAR in its description of the Sparge Air system currently states that it is used to purge hydrogen from the system during startup and shutdown. The UFSAR is being changed to state that sparge air is used to purge hydrogen from the offgas system when necessary. Procedures will insure that it is done, when necessary.

The procedural changes are summarized as follows:

1) sparging of the Offgas train will not be required during Unit shutdown or after a SCRAM, if the Offgas train has been run at least 4 hours since the reactor was critical and 4 hours since hydrogen was injected into the feedwater process stream;

2) sparging the Offgas train is not required during Unit startup, as the shutdown of the Offgas System will insure that the system cannot have hydrogen concentrations above the detonation limits of 4% hydrogen; sparge air flow is also not needed to heatup the system when steam dilution is ON;

3) the condenser Mechanical Vacuum Pump will be shutoff during the startup of the Offgas System, instead of letting the pump run until it trips.

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:

Loss of Reactor Coolant Inventory
Since the systems are not in close proximity to or in any way attached to the primary system pressure boundary, they cannot cause a LOCA or other loss of inventory event.

Explosion in the Offgas System
The Offgas System shall be operated in a manner that minimizes to potential of an explosion hazard. Sparging the system with air is intended to purge and dilute the hydrogen concentration to the point where it will no longer be explosive. Insuring that the Offgas train remains running for at least 4 hours after all hydrogen generation has stopped insures that the system has purged itself of hydrogen and sparging is no longer necessary. If, for any reason, this condition is not met, it is expected that sparging of the system would still be performed. Therefore, this change does not make an explosion in the Offgas System more likely.

Loss of Vacuum SCRAM
The shutdown of the Mechanical Vacuum Pump cannot increase the potential of a loss of vacuum SCRAM, because this SCRAM is only possible when in the RUN mode (mode 1). The Offgas System is put online when the reactor pressure is about 130-300 psig and the vacuum pump would be OFF prior to reaching Mode 1 conditions. Therefore, whether using the original procedure or the new one, the Mechanical Vacuum Pump would be off prior to going to Mode 1.

The use of (or failure to use) sparge air to purge the Offgas System has no impact on maintaining a vacuum on the main condenser. The procedures for using sparge air at Quad Cities only involve Offgas System startup, swapping of trains, and shutdown of the system. The only activities performed in Mode 1, swapping trains, will still have Sparge Air used.

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 to the way the Offgas System is operated (i.e., eliminating the use of Sparge Air to purge the system under certain limited conditions and changing how the Mechanical Vacuum Pump is turned off) has no impact on the system in such a way as to make it more likely to fail or to not be available when needed. It also creates no new interfaces with other systems.

   All of the modified procedures affect Offgas and its function to maintain a vacuum in the condenser. The loss of vacuum and a potential fire/explosion in the Offgas System have already been considered in the design basis. There is no other consequences possible from making these changes. Therefore, there is no new failure modes, such as a new type of transient or accident, created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes do not impact the automatic isolation of Offgas or the ability to detect high hydrogen concentration downstream of the recombiners. Sparging Air is not required to prevent having high hydrogen concentration downstream of the recombiner provided the system has been online after the contributors of hydrogen have stopped generating the hydrogen. The system operation 150 CFM of gas and vapor going through it normally. After the reactor is no longer critical and the hydrogen injection has stopped, much of the makeup flow to the system would stop, but water vapor and air inleakage would still amount to about 50 CFM (Reference UFSAR Table 11.3-4). This is more flow than the Sparge Air blowers can provide through a 1" line to the Offgas Train. Therefore, the Offgas System would purge itself of hydrogen in a period of 4 hours.

Tracking No. SE-00-052
Activity No. UFSAR-99-R6-116

DESCRIPTION:

This safety evaluation is being performed to support UFSAR Change Request #99-R6-116.

UFSAR Section 11.3 will be changed to read; "The mechanical vacuum pump system establishes and maintains the main condenser vacuum when steam is not available".

UFSAR Section 11.3.2.3 will be changed to read: "The mechanical vacuum pump system establishes and maintains the main condenser vacuum at 20-25 inches of mercury. This system, which is used when steam to operate the air ejectors is not available, exhausts through a discharge silencing tank at about 2320 standard ft3/min of gas (air) at 15 in. Hg.

This change to the UFSAR eliminates the reference that the mechanical vacuum pump is only used for startup in preparation for condenser operation, and adds a reference to "maintain condenser vacuum" which will allow operation of the vacuum pump during a unit shutdown when steam pressure is inadequate to operate the steam jet air ejectors.

The reason for this change is to allow operation of the mechanical vacuum pump during a unit shutdown in operational modes 2 and 3 to maintain condenser vacuum adequate to provide a heat sink for the reactor decay heat and cooldown prior to clearing the RHR Shutdown Cooling subsystem (SDC) low pressure isolation interlocks. Recent unit shutdowns have involved injection of noble metals and/or manual insertion of all control rods (soft shutdown) which lengthens the amount of time that condenser vacuum must be maintained as a heat sink prior to clearing the SDC interlocks. Steam Jet Air Ejector operation is degraded during periods of low steam pressure due to the inadequate driving force required for proper operation; therefore, the mechanical vacuum pump could be used as necessary to remove non-condensibles from the main condenser.

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 steam line radiation monitor isolation logic will be operable any time the mechanical vacuum pump is operated as required by Technical Specification 3.2.L. The monitors would detect the increase in radiation levels associated with a control rod drop accident, and would automatically initiate signals to isolate the main steam line isolation valves, the steam jet air ejector and mechanical vacuum pump suction valves, and trip the mechanical vacuum pump. Isolations initiated from main steam line high radiation prevent offsite release rates from exceeding 10CFR100 limits.

   During an analyzed Control Rod Drop Accident, the mechanical vacuum pump is assumed to be in operation due to its high flow rate and the minimal gaseous radioactive decay holdup time. This would create the worst possible release rate. Operation of the mechanical vacuum pump during unit shutdowns would have no effect on the probability of a control rod drop accident. Existing isolations would remain operable to minimize the offsite release associated with an accident.

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 method of operation of the mechanical vacuum pump will remain unchanged. The main steam line high radiation trip of the vacuum pump feed breaker is required to remain operable in Mode 2. In Mode 3, all control rods are fully inserted and the probability of a control rod drop accident is nil. Therefore, operation of the mechanical vacuum pump during a unit shutdown is bounded by the current rod drop accident analysis, and does not create the possibility of an accident or malfunction 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 worst case Control Rod Drop Accident would occur with the reactor in a hot standby condition with the mechanical vacuum pump in operation. The main steam line (MSL) radiation monitors would detect the increase in radiation levels associated with a control rod drop accident, and would automatically initiate signals to isolate the main steam line isolation valves, the steam jet air ejector and mechanical vacuum pump suction valves, and trip the mechanical vacuum pump. Operation of the mechanical vacuum pump during a unit shutdown would increase the frequency of operation in this worst case condition; however, the existing isolation scheme is required to remain operable in this case. Isolations initiated from main steam line high radiation have been previously analyzed to limit offsite release rates well within 10CFR100 limits.

Tracking No. SE-00-053
Activity No. QGA-21 Revision 1 and QOA 6900-07, Revisions 0 through 9
(Manual load shedding performed for a Loss of AC Power to the 125 VDC Battery Chargers with Simultaneous Loss of Auxiliary Power); UFSAR 99-R6-136.

DESCRIPTION:

The activity is the manual load shedding performed in the noted procedures. These station procedures direct operators to remove 125 VDC loads from the battery by opening specific circuit breakers at distribution panels. The effect of the activity is that the associated equipment will be without 125 VDC power.

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 125 VDC loads which are being shed will have already performed their safety-related function to mitigate the consequences of a LOOP with or without a concurrent LOCA before they are de-energized. The affected systems and/or components can not contribute to any initiating event for any accident 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 activity de-energizes loads by opening the 125 VDC feed breakers to these loads. All of the equipment important to safety will perform their required functions for the applicable operating modes and accidents. There are no new failures or malfunctions introduced by the activity, which are not previously analyzed. This activity is accomplished after any required functions

important to safety have already occurred or the malfunction of the affected systems is previously analyzed. Other loads, which are de-energized, are not required to function for the applicable operating modes and applicable accident or transient conditions.

The activity de-energizes equipment important to safety after the affected equipment has performed its safety function for the applicable operating modes and accidents. The activity does not alter or change any system, structure, or component and this evaluation has not identified any new malfunction. Therefore, the possibility of a different type of malfunction than any previously evaluated has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the activity does not affect or change any basis for any Technical Specification. Therefore, there will be no increase in any margin of safety.

Tracking No. SE-00-054
Activity No. UFSAR 99-R6-110

DESCRIPTION:

This activity makes several changes to the CRD System description (UFSAR Section 4.6.3). The CRD coupling spud surface treatment is specified. The text describing the CRD pump suction flow path, CRD cooling water flow rate and pressure, and directional control valve DP and flow is changed. The test on control rod scram force and rod insertion resisting force is changed. A historical description of rod scram timing development is removed. The method for prevention of a scram failure mode (overtightening of scram valve stem packing, Section 4.6.4.6) was revised.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important o safety previously evaluated in the safety analysis report is not increased because two accidents/transients were reviewed as a result of this change: rod withdrawal at power, control rod drop. Non of the revisions cited in the description above will have any effect on the likelihood of these accidents occurring. The consequences of these accidents/transients will not change since the operation of the CRD system is unaffected by these revisions.

2. The possibility for an accident or malfunction of a different type that any evaluated previously in the safety analysis report is not created because the revisions cited above do not impact the design, function or operation of the CRD system. Also, no new failure modes are created. Therefore, creation of an unanalyzed accident/malfunction cannot occur.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the bases for Technical Specification Sections 3.3/4.3C thru H and 3.10/4.10C, I & J were reviewed and determined to not be impacted by this activity. UFSAR text changes performed by this activity have no effect the performance or operation of the CRD system, and thus can have no effect on any Technical Specification margin of safety.

Tracking No. SE-00-055
Activity No. DCP 9900373; QCAP 1500-01

DESCRIPTION:

This DCP is removing the hydrant houses and cabinets in the exterior portions of the plant property. The DCP will require that a "fire truck" be on site equipped with twice the equipment of a hydrant house to take the fire fighting equipment to the fire scene. This will improve the fire brigade's response to a fire outside of the power block.

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 changing the storage location or transportation of fire fighting equipment will not effect the probability of a fire. The probability of a fire is based upon the amount of combustible loading and its proximity to an ignition source. Since neither of these parameters are changing the probability will not change. This change only affects fires that are outside of the power block and an external fire does not impact safe shutdown. Therefore, the consequences are not changed.

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 fire hose is used to mitigate (fight) a design basis fire and is not a precursor to an accident (fire). This change will require that the truck be equipped with twice the NFPA 24 recommended equipment which will preclude any affect of an equipment failures. Further, if a malfunction of a hydrant should occur then there is adequate equipment available to use the next hydrant. Since this equipment is available in all operating modes and there are adequate backups available, this activity will not affect equipment failures

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications are affected by this change. Therefore, margins defined by the technical specifications are not changed. Administrative controls required by the technical specifications have been adequately updated for this change.

Tracking No. SE-00-061
Activity No. UFSAR-99-R6-115

DESCRIPTION:

1. Revise UFSAR Section 11.3.3.1.1 to show that the data in Table 11.3-3 is design information.
2. Revise Table 11.3-3 to show the design flow, exit velocity, and heat rate control for the Main Chimney and the Reactor Building Vertical Stack.
3. Revise Figure 11.3-1 to reflect the design air flows for the ventilation system that feed into the Main Chimney.

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 no accident or transient has been identified that is affected by this activity. This UFSAR revision involves reflecting the as-built and design conditions of the chimney and Rx Bldg. Vent Stack flow for non-accident conditions. The Main Chimney and Rx Bldg. Vent Stack are components used to mitigate the consequences of the accident are not directly connected to reactor operation. Effluents from the stacks do not initiate any analyzed transients and also do not increase the consequences of the 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 Main Chimney and Rx Building Vent Stack are components used to mitigate the consequences of the accident are not directly connected to reactor operation. Effluents from the stack do not initiate any different type accident or malfunctions than those previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification Sections have been identified that are affected by this change. Therefore, this UFSAR revision to Section 11.3, Table 11.3-3 and Figure 11.3-1 will not reduce the margin of safety as described in the Technical Specifications.

Tracking No. SE-00-062
Activity No. DCP 9900162; UFSAR-99-R6-135

DESCRIPTION:

Permanent clamp will be installed on Unit 1at the cracked RS-1 weld between the thermal sleeve and riser elbow of Jet Pump Pair 19/20. The installed clamp will provide redundant load path for the riser pipe elbow to thermal sleeve junction for all operating conditions and will minimize the increase in stress at a partially cracked weld which can accelerate crack growth.

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 extensive analysis of jet pump assembly with installed clamp has been performed and it is concluded that structural integrity of jet pump assembly is maintained in the repaired condition. Therefore, the probability of a jet pump malfunction and its consequences is not increased by this design change. It does not affect any Reactor Recirculating piping or equipment outside the vessel. Therefore, does not increase the probability and its consequences of a Design Basis LOCA or a Recirculation Pump Trip. The small additional leakage does not affect calculated maximum peak clad temperature following a LOCA and will have negligible effect on jet pump flows during a transient scenario.

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 clamp is installed as structural replacement of cracked weld. It does not increase challenges to or create new

challenge to any equipment. The amount of metal particles added to water in the vessel is insignificant and acceptable. No new failure initiators are created by the installation of clamp. Thus, no new accident scenarios will be created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the design change does not affect the Technical Specification requirements of the LPCI system or the jet pumps and does not adversely affect the ability of either system to meet design requirements.

Tracking No. SE-00-063
Activity no. UFSAR-99-R6-128; DCR 9900533

DESCRIPTION:

1. UFSAR change to Section 9.4.4 to add a description of the supply ductwork to the Reactor Feedwater Regulatory Valve (FWRV) station and hydraulic power units.

2. Document change request to the Reactor Feed Pump (RFP) Ventilation drawings to add the ductwork to the feedwater regulating valve station and Hydraulic Power Units (HPU) {also called hydraulic actuators}.

3. Document change request to add the EPN's for the balancing dampers on the ductwork to the FWRV station and HPU.

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 revising drawings and UFSAR sections for the Reactor Feed Pump Ventilation System to show the supply ductwork to the regulating valve station and hydraulic actuator will not increase the probability of equipment malfunction. Failure of the ductwork that feeds the Feedwater Reg Valve Station and hydraulic controller is the same failure mode as the addressed in accidents – Loss of Normal Feedwater Flow and increase in Feedwater Flow. Therefore, the consequences of the accident are the same as those addressed in the accident analyses and therefore, do 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 Loss of the Reactor Feed Pump Ventilation system may cause a malfunction of the feedwater regulating valve. The malfunction of the feedwater regulating valve has been addressed in two evaluated transients – Loss of Normal Feedwater Flow and Increase in Feedwater Flow.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Reactor Feed Pump Ventilation System or the Reactor Feedwater System is not described or mentioned in the Technical Specification bases.

Tracking No. SE-00-065
Activity No. DCP 9800238; 9800239; UFSAR 99-R6-132

DESCRIPTION:

These DCPs will Install a permissive relay in the HPCI initiation control circuit to inhibit a second start due to high drywell pressure. The new relay will use existing relay contacts from the HPCI 2330-112A relay and the 2330-144 relay as input signals. The new relay will use one of its own contacts as a seal in contact. The existing HPCI turbine trip reset button will be used to reset this relay. UFSAR change UFSAR-99-R6-132 will update the HPCI System description with the logic changes described by DCP 9800238 and DCP 9800239. Procedure changes will describe the changed logic and implement all required maintenance due to the installation of DCP 9800238 and DCP 9800239.

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 that will be made by this DCP are to the HPCI initiation control circuits. These control circuit changes require the HPCI system to initiate and fill the reactor to +48 inches. Due to the arrangement of these control circuits, they cannot produce a HPCI initiation. Since these circuits have no interfaces with the safety or relief valves, the changes cannot affect the probability of an inadvertent opening of these valves. There is also no credible scenario where these circuits could cause any of the described transients or 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 these circuit logic changes can only affect HPCI operation. As stated in the previous questions, the changes made by these DCP will enhance HPCI operation and cannot affect it in an adverse manner. In the unlikely event that the modified control circuits function in an unexpected manner, the HPCI system and its power supply would be the only system/components affected. This loss of the HPCI system would be mitigated by the use of the automatic depressurization system (ADS) which has been previously analyzed. These DCPs will not create the possibility of an accident/transient of a different type than was previously analyzed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes will have no adverse affects on the operation of the HPCI system. It will still meet its design requirement of delivering water at 5000 gpm within 45 seconds. The changes also will not adversely affect the 125 VDC system under plant operating or shutdown conditions. The new relay will only require power when it de-energizes three (3) other relays. The high drywell pressure HPCI initiation signal is not effected. This signal is a one-time initiation signal as the drywell pressure would continue to be in excess of 2.5 psig during the period that the HPCI system is required. The drywell pressure response to a 0.01 square foot line break is analyzed in General Electric Document NESO-52-0682. This analysis clearly shows the drywell pressure remaining above 10 psig even after the drywell spray has been initiated

Tracking No. SE-00-066
Activity No. DCP 9900398

DESCRIPTION:

This design change modifies the existing moisture separator high-level turbine trip circuits. Each unit has four moisture separators (EPNs 1(2)-5605A,B,C,D), each of which has a high level switch (EPNs 1(2)-3541-37A,B,C,D) that actuates on increasing level to close the turbine stop valves, which trips the turbine. Actuation on high level of any one of the four switches will trip that unit's turbine. The existing high level switch on each of the moisture separator tanks will be replaced, and an additional high level switch will be added. The outputs from each new pair of switches will be wired in series, replacing the existing single switch contact in the high-level trip circuit. The new 2-out-of-2 logic will trip the turbine only if both switches on any moisture separator actuate. A total of eight new switches will be installed in each unit, two on each of the four moisture separators. The installation of each new pair of switches requires the addition of a new terminal box and rerouting of conduit at each moisture separator tank.

On each moisture separator tank, the existing high level switch also provides an input to a high level alarm that is annunciated in main control room panel 901(2)-6. This annunciation will be maintained in the new configuration by combining the outputs of each pair of switches in parallel to the existing annunciator circuit. This way, the alarm is initiated if either switch of the pair actuates, for a 1-out-of-2 logic.

The Safety Evaluation was also used for DCP 9900399, 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 moisture separator high level switches are one of several inputs that can initiate a Turbine Trip. Using a pair of high level switches on each moisture separator instead of a single high level switch does not increase the likelihood of a high level occurring. Therefore, there is no increase in the probability of occurrence of the Turbine Trip.

   The modified moisture separator high level turbine trip circuits are not required to function in order to mitigate any accident or transient. No changes are made to any SSC that can result in a change in off-site dose. Therefore, there cannot be any increase in the consequences of the Turbine Trip.

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 each new pair of switches will perform the function required of the existing switch. The effects of failure of one of the new switches is encompassed by the effects of failure of one of the existing switches. The moisture separators will continue to function exactly as before. Because no new system interactions are created, no new types of accidents or transients are created.

   The addition of the new components including switches with isolation and drain valves, terminal boxes, conduit, and cable has been evaluated and will not result in the

degradation or failure of any SSC listed previously. The installation will be performed using approved methods, materials and procedures. All modified components will be tested to ensure that they function in accordance with their design requirements. No new failures have been introduced. Therefore, the modification cannot 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 moisture separator level switches are not a factor in the basis for any Technical Specification, therefore, their replacement can have no effect on the margin of safety.

Tracking No. SE-00-067 (Supersedes SE-00-002)
Activity No. DCP 9900027, Rev. 1

DESCRIPTION:

This modification upgrades the fire zone boundaries for the Battery Charger Room (Fire Zone 6.1.B), 125 Vdc Panel Room (Fire Zone 6.1.A), and the Battery Room (Fire Zone 7.1) to provide one separate 3-hour fire rated barrier for these zones. It provides details to seal mechanical and electrical openings through floors, ceilings and walls in each fire zone as applicable. It also installs three (3) new 3-hour fire rated dampers in the Battery Room HVAC system and replaces the existing Battery Room non-rated louvered door assembly (EPN 1-0030-203) with a new 3-hour fire rated door.

This modification also updates procedures and fire zone descriptions in the FPR and corrects various editorial discrepancies in documentation discovered during the preparation of this modification (i.e. revise directional notes to give correct locations for details).

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 no additional fire loading or ignition sources are added to any fire area. There is no effect on the probability of occurrence or the consequences of a fire. The effect of a postulated failure of the new dampers is bounded by the effects of other previously analyzed conditions such as a loss of power or a loss of battery room HVAC. There are no physical or operational changes to the 250, 125, or 24/48 Vdc systems.

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 existing failure modes of the modified fire barriers, door and seals are unchanged and no new failure modes are introduced. The new fire dampers are passive components except when required to close in response to a fire. The effect of failure of a new battery room fire damper is significantly less than and bounded by other previously defined malfunctions. The existing failure mode of the HVAC system remains unchanged. There are no physical or operational changes to the 250, 125, or 24/48 Vdc systems. Therefore, no new system interactions are created, and systems will continue to function in accordance with their design requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the bases to Technical Specifications 3/4.9.C, D, E and F provide no specific limits but rather discuss the availability of the dc power distribution system. Following installation, the modification will not affect the reliability, availability or operability of any component in the dc system. Fire, fire doors and fire barriers are not discussed in the Quad Cities Technical Specifications. These items are discussed in the FPR. This modification revises the FPR to document the changes performed. There are no changes to any setpoint, surveillances or bases in the Technical Specifications.

Tracking No. SE-00-069 (Supersedes SE-00-058)
Activity No. DCP 9900175, Rev. 1

DESCRIPTION:

This modification upgrades the fire zone boundaries for the Battery Charger Room (Fire Zone 6.2.B), 125 Vdc Panel Room (Fire Zone 6.2.A), and the Battery Room (Fire Zone 7.2) to provide one separate 3-hour fire rated barrier for the combination of these zones. It provides details to seal mechanical and electrical openings through floors, ceilings and walls in each fire zone as applicable. It also installs three (3) new 3-hour fire rated dampers in the Battery Room HVAC system and replaces the existing Battery Room non-rated louvered door assembly (EPN 2-0030-224) with a new 3-hour fire rated door.

This modification also updates procedures and fire zone descriptions in the FPR and corrects various editorial discrepancies in documentation discovered during the preparation of this modification (i.e. revise directional notes to give correct locations for details).

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 no additional fire loading or ignition sources are added to any fire area. There is no effect on the probability of occurrence or the consequences of a fire. The effect of a postulated failure of the new dampers is bounded by the effects of other previously analyzed conditions such as a loss of power or a loss of battery room HVAC. There are no physical or operational changes to the 250, 125, or 24/48 Vdc systems.

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 existing failure modes of the modified fire barriers, door and seals are unchanged and no new failure modes are introduced. The new fire dampers are passive components except when required to close in response to a fire. The effect of failure of a new battery room fire damper is significantly less than and bounded by other previously defined malfunctions. The existing failure mode of the HVAC system remains unchanged. There are no physical or operational changes to the 250, 125, or 24/48 Vdc systems. Therefore, no new system interactions are created, and systems will continue to function in accordance with their design requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the bases to Technical Specifications 3/4.9.C, D, E and F provide no specific limits but rather discuss the availability of the dc power distribution system. The

modification will not affect the reliability, availability or operability of any component in the dc system. Fire, fire doors and fire barriers are not discussed in the Quad Cities Technical Specifications. These items are discussed in the FPR. This modification revises the FPR to document the changes performed. There are no changes to any setpoint, surveillances or bases in the Technical Specifications.

Tracking No. SE-00-070
Activity No. QCOA 6900-05; UFSAR R6-99-137

DESCRIPTION:

The activity is the manual shedding of the 250V DC loads in procedure QCOA 6900-05. If the 250V DC battery chargers cannot be energized concurrent with a design basis accident, this procedure directs the operator to trip the Recirc MG Set Emergency Lube Oil Pumps within 30 minutes and, if AC power is not available to the Generator H2 Main Seal Oil pump, the operator is directed to trip the Emergency H2 Seal Oil pump within 2 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 this activity will only take place after the accident has already occurred and the 250V DC battery will remain available to the required isolation valves and mitigating systems. This activity consists of simple manual actions to remove power from several nonessential 250V DC loads using equipment design for that purpose. The actions required are within the capability of an operator considering the allowable time to perform the actions and the accessibility of the 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 the 250V DC battery and the switches used to shed loads will be used as designed. The loads that are shed support nonessential pieces of equipment that are no longer operating due to a loss of offsite power.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does not affect the basis for any Technical Specifications.

Tracking No. SE-00-071 (Supersedes SE-99-100)
Activity No. DCP 9900059

DESCRIPTION:

This modification upgrades non-rated Unit 1 Fire Doors 300 and 302, which are part of the boundary between Fire Zones 8.2.4 and 8.2.6.A, by replacing the existing non-rated door assemblies with new three-hour rated fire doors. The new replacement fire doors have been tested in accordance with ASTM E119 and NFPA 251 for floor/ceiling assemblies and were also

subjected to hose stream testing per NFPA 252. These "A" labeled doors are installed in accordance with the as tested configuration. Supplemental steel is required to mount the new doors. A 3-hour rated fire barrier (Pyrocrete) will be provided for all exposed surfaces of supplemental steel. A non-sealed pipe penetrating the fire barrier will also be capped by the modification.

The fire protection "F" drawing has been revised to incorporate penetrations not previously shown on the drawing. These penetrations will be added to the "Electrical Penetrations Seal Fire Testing and Installation program".

The effect of this design change is to support the Appendix R Fire Protection Enhancement Program by uprating the existing non-rated barriers between Fire Zone 8.2.4 and Fire Zones 8.2.6.A North of Column 25 to a three-hour rating. Fire Protection Report (FPR) Volume 1 indicates that the remaining Fire Zone 8.2.4 boundaries are acceptable and do not require upgrading.

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 upgrades fire barriers for Fire Zone 8.2.4 to a three-hour rating by replacing existing non-rated penetrations with three-hour rated seals. An open ended pipe penetrating the barrier will be capped and existing non-rated access hatches will be replaced with three hour rated fire doors. None of these improvements add additional fire loading to any fire area. In addition, no new ignition sources are added by the modification. None of these improvements have any effect on the probability of occurrence of a fire. Therefore, there can be no increase in the probability of occurrence of a fire as a result of this modification. As the installation improves the ability of the fire barriers to perform their function which is to limit the spread of fire by confining the fire to one side of the barrier. Therefore, there can be no increase in the consequences of a fire.

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 installation of the replacement doors and the repair of other areas of the fire zone boundary by this modification enhances the ability of the fire barrier to perform its duties. The changes result in no increase in the fire loading in the subject fire zones. The installation will be performed using approved methods, materials and procedures. All modified components have been analyzed and will continue to perform their design function as required. No new system interactions are created. Therefore, the modification cannot 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 fire, fire doors, fire barrier penetration seals and fire barriers are not discussed in the Quad Cities Technical Specifications. They are discussed in the Fire Protection Report (FPR). This modification revises the FPR to document the changes performed. There are no changes to any setpoint, surveillances or bases in the Technical Specifications. Quad Cites License Condition h.3.F allows these revisions to the FPR without prior NRC approval as the changes do not adversely affect the ability of the station to achieve and maintain safe shutdown in the event of a fire.

Tracking No. SE-00-072 (Supersedes SE-99-101)
Activity No. DCP 9900174

DESCRIPTION:

This modification upgrades non-rated Fire Doors 301, 303, 304, 305 and 306, which are part of the Unit 2 Cable Tunnel (Fire Zone 8.2.5) boundary, by replacing the existing non-rated door assemblies with new three hour rated fire doors. These "A" labeled doors are installed in accordance with the as tested configuration. The new replacement fire doors have been tested in accordance with ASTM E119 and NFPA 251 for floor/ceiling assemblies and were also subjected to hose stream testing per NFPA 252. Supplemental steel is required to mount the new doors. A 3-hour rated fire barrier (Pyrocrete) will be provided for all exposed surfaces of supplemental steel.

Additionally, gaps/cracks in the 3-hour walls of Fire Zone 8.2.5 will be sealed using approved station details and procedures to provide a three-hour barrier for this area. Non-sealed conduit penetrations will also be sealed using station approved sealing details. During construction, the sealing will be inspected per station procedures to ensure that the seals meet the construction details specified.

The fire protection "F" drawing has been revised to incorporate penetrations not previously shown on the drawing. These penetrations will be added to the "Electrical Penetrations Seal Fire Testing and Installation program".

The effect of this design change is to support the Appendix R Fire Protection Enhancement Program by uprating the existing non-rated barriers between Fire Zone 8.2.5 and Fire Zones 8.2.6.A, 8.2.6. C & 8.2.6.E to a three-hour rating. Fire Protection Report (FPR) Volume 1 indicates that the remaining Fire Zone 8.2.5 boundaries are acceptable and do not require upgrading.

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 upgrades fire barriers for Fire Zone 8.2.5 to a three-hour rating by sealing gaps and cracks and installing or replacing existing non-rated penetrations with three-hour rated seals. The sealing will be performed in accordance with approved station sealing details and procedures. In addition existing non-rated access hatches will be replaced with three hour rated fire doors. None of these improvements add additional fire loading to any fire area. In addition, no new ignition sources are added by the modification. None of these improvements have any effect on the probability of occurrence of a fire. Therefore, there can be no increase in the probability of occurrence of a fire as a result of this modification. As the installation improves the ability of the fire barriers to perform their function which is to limit the spread of fire by confining the fire to one side of the barrier. Therefore, there can be no increase in the consequences of a fire.

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 installation of the replacement doors and the repair of other areas of the fire zone boundary by this modification enhances the ability of the fire barrier to perform its duties. The changes

result in no increase in the fire loading in the subject fire zones. The installation will be performed using approved methods, materials and procedures. All modified components have been analyzed and will continue to perform their design function as required. No new system interactions are created. Therefore, the modification cannot 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 fire, fire doors, fire barrier penetration seals and fire barriers are not discussed in the Quad Cities Technical Specifications. They are discussed in the Fire Protection Report (FPR). This modification revises the FPR to document the changes performed. There are no changes to any setpoint, surveillances or bases in the Technical Specifications. Quad Cites License Condition h.3.F allows these revisions to the FPR without prior NRC approval as the changes do not adversely affect the ability of the station to achieve and maintain safe shutdown in the event of a fire.

Tracking No. SE-00-073 (Supersedes SE-99-119)
Activity No. DCP 9900063

DESCRIPTION:

This Modification is provided to ensure that cable 80220 remains intact and undamaged during a postulated Appendix R Fire, in the TB-II or 13-1 Fire Area of the Unit 1 Turbine Building. In this design change, the existing cable 80220 will be de-terminated at Unit 1 control room console 901-74, pulled back into tray, spared, coiled, and re-identified as cable 80224. The Bus 14-1 end of the cable will be de-terminated and spared, coiled, and re-identified as spared cable 80224.

A new cable 80220 will be installed and routed from Control room console 901-74 through existing raceways and a new conduit to the underside of Switchgear 14-1. The reroute will route the cable through Fire Area TB-III thereby avoiding fire Areas TB-II and 13-1

The new cable will be terminated in the control room console 901-74 and identified with the existing cable number 80220. The other end of the new cable will be terminated in the 14-1 switchgear located in the turbine building and tagged with the existing cable number 80220. The length of the new cable 80220 has been determined to be shorter than the existing spared cable, therefore, there is no adverse impact upon the control circuit voltage as a result of the reroute.

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 impacts the Station Blackout (SBO) system and the Onsite power system. It allows remote control of Bus 14-1 to allow the SBO system to provide power to Bus 14-1 for a fire in Fire Areas TB-II or 13-1. This method of control is currently not credited for fires in this fire area. The impact of the modification on all affected SSCs has been evaluated and is acceptable. Testing will verify that the SSCs affected will operate as designed. Therefore, the probability of failure of the affected SSCs is no different than before they were modified. The cable reroute does not change the function or operational characteristics of the affected system. Therefore, there will be no increase in 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.

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 cable and its failure modes are unchanged by the modification. The affect of the modification on the SSCs identified has been evaluated and will not result in any new failures. Testing will verify that the affected systems logic operates as designed. As the function of the cable is unchanged and all of the impacted SSCs will continue to function as before, there is no possibility of creating a different type of malfunction of equipment important to safety than previously analyzed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modification does not change the function or method of operation of any component. It enhances the current fire protection program as, following installation, remote control of the Bus 14-1 SBO Diesel Generator feeder breaker will be available in the event of a fire in Fire Zones TB-II and 13-1. Following the installation of the change all SSCs will perform their designed safety functions, meet all of their design requirements and no margins of safety will be reduced with respect to the affected systems. There are no changes to any setpoint, surveillance or bases listed in the Technical Specifications.

Tracking No. SE-00-075
Activity No. DCP 9900367

DESCRIPTION:

DCP 9900367 will connect monitoring instrumentation within the Turbine Supervisory Instrumentation (TSI) Cabinet. This instrumentation will monitor inputs into the Turbine High Vibration Trip circuitry using existing test points on the vibration amplifier circuit boards. Additionally, the output of the reference detector for the vibration phase angle will be monitored. All instrumentation test leads will be fused and routed to a Teac RD-200T PCM Data Recorder (or equivalent). The recorder is buffered and designed to prevent feedback into the circuitry being monitored. A ground phase isolation transformer will be utilized to prevent the development of electrical ground circulating currents. This recorder has a tape deck, which will be used to collect data continuously during turbine coastdown and start-up (or at the direction of the vibration analyst). The recorder will also output to a data collector, which can be used to periodically collect information during normal operation.

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 steps have been included in this DCP to provide a high level of assurance that the monitoring equipment will not impact the turbine supervisory circuitry. These measures include the selection of appropriate test equipment and the use of fused test leads. The process of installing the monitoring equipment introduces the potential for improper installation of the test leads or unanticipated impact on output of the turbine vibration amplifier circuit board to the high vibration trip circuitry. Therefore, during the installation, testing and removal processes, the U1 turbine supervisory trip relay will be

disabled to prevent spurious trips. While the turbine high vibration trip is disabled, increased monitoring of turbine parameters will be performed and the turbine will be tripped manually if required. Prior to re-enabling turbine supervisory trip relay, checks will be performed to verify that the operation of the monitoring equipment does not adversely affect the turbine high vibration trip circuitry. Based on these actions, the probability of an accident or the occurrence of a malfunction of equipment important to safety have not increased.

The change will not increase the consequences of a turbine trip because the plant response and the equipment used to mitigate the affects of a turbine trip are not affected by this DCP. The fuel cladding integrity safety limit would not be violated.

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 actual turbine vibration levels will not be affected by this DCP. The addition of the monitoring equipment has been evaluated and should not affect the output of the vibration amplifier circuit board to the high vibration trip circuitry. It is possible that the monitoring equipment could fail and affect the output signal of the vibration amplifier circuit board. If the output signal failed high, the turbine would trip. If the output signal failed low, the turbine trip due to high vibration would not occur as designed. These are not new failure modes because a failure of the existing circuitry could lead to the same results, therefore, the possibility of an accident or transient of a different type than previously evaluated does not exist.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification requirements associated with this temporary design change.

Tracking No. SE-00-076
Activity No. UFSAR-99-R6-133

DESCRIPTION:

Update UFSAR section 3.5.3 to describe the Brown Boveri Company (BBC) low-pressure turbine rotors currently installed in Quad Cities Units 1 and 2. The low-pressure turbine rotors were replaced under modifications M-4-1-87-013 and M-4-2-88-023 to reduce the probably of stress corrosion cracking in high stress areas of the rotors. The BBC rotors have already been described correctly in UFSAR Section 10.2.3.

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 a turbine disk failure or a subsequent loss of condenser vacuum has been reduced by the installation of the BBC low pressure turbine rotors by reducing the susceptibility to stress corrosion cracking as compared to the original GE low pressure turbine rotors. With the BBC turbine rotors installed, the same protective barriers and separation of equipment will be in place that were in place with the GE turbine rotors to safely shutdown the plant in the event of a failure of a turbine rotor.

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 BBC low-pressure turbine rotors are basically a like for like replacement for the original GE rotors. The difference in the design of the BBC low pressure turbine rotors which were assembled by welding forged sections together instead of using a shrink fit has been shown to reduce the probability of a turbine disk failure. The analytical basis which determined the reduced probability of failure of the BBC low-pressure rotors is described in modifications M-4-1-87-013 and M-4-2-88-023. The BBC rotors are dimensionally the same, will have the same steam path and will operate the same as the original GE rotors. The failure mechanism for the Turbine Generator is not being changed by the installation of the BBC low-pressure turbine rotor.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no changes to setpoints, surveillances or bases in the Technical Specifications as a result of the UFSAR revision. The probability of failure of the Turbine Generator and potential loss of the main condenser was reduced by the installation of the BBC low-pressure turbine rotors. However, the reliability of the Turbine Generator does not form the basis for any Technical Specification. Therefore, the margin of safety is unchanged.

Tracking No. SE-00-077 (Supercedes SE-99-104)
Activity No. DCP 9900067 Rev. 1

DESCRIPTION:

This design change will provide a redundant power feed to the Safe Shutdown Makeup Pump (SSMP) flow indicating controller (FIC) 1/2-2940-07. This controller is located in the control room and currently receives its power from Unit 2 120Vac Essential Service Distribution Center Panel 902-49. The controller modulates motor operated valve 1/2-2901-6 to provide the required flow. Both local and control room controllers are provided. The valve control circuit and the local FIC (1/2-2940-06) are not affected by the modification. The existing SSMP control room FIC feed from 902-49 will be disconnected at 902-49 and rerouted to a new manual transfer switch (0-2940-7). This switch is a three-position "Normal-Off-Alternate" SBM switch located in a new junction box in the Auxiliary Electrical Equipment Room. The switch is procured and installed as Safety-Related. A new feed from Unit 1 120Vac Essential Service Distribution Center 901-49 is routed to the transfer switch. The new transfer switch will allow the operator to select either Unit 2 (Normal) or Unit 1 (Alternate) as the source of power to the FIC. The SSMP FIC is currently powered from a 20A breaker at 902-49. A 20A feeder breaker at 902-49 will provide the main feed to the FIC. A spare 20A breaker at 901-49 will provide the alternate feed. The new and rerouted cables are routed in existing tray and a new seismically installed conduit.

The effect of the modification is to enhance the existing capabilities of the SSMP system by providing FIC 1/2-2940-07 with more than one power supply. This provides a greater probability that the SSMP can be operated from the control room in the event of a fire.

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 impact of the modification on all affected SSCs has been evaluated and is acceptable. Testing as described in the MAL will verify that the modified SSCs, including the SSMP FIC circuit, will operate as designed. The cable reroute does not change the function or operational characteristics of the affected systems.

The addition of the new components including conduit, conduit supports and cable has been evaluated and will not result in the degradation or failure of any SSC listed previously. The installation will be performed using approved methods, materials and procedures. All modified components will be tested to ensure that they continue to function exactly as before. No changes are made to either primary or secondary containment or to any systems that maintain primary and secondary containment. No new failures have been introduced.

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 normal state of the modified circuit will be that the FIC is powered from 902-49, just as before the modification. The modification adds the ability to power the FIC from 901-49, as an alternate. There is no possibility created of a failure of both 901-49 and 902-49 due to a fault at the new transfer switch (0-2940-7). The new transfer switch is procured and installed as safety-related and all new components are seismically supported. There is no change to existing failure modes, and no new failure modes are created.

   The addition of the new components including conduit, conduit supports and cable has been evaluated and will not result in the degradation or failure of any SSC listed previously. The installation will be performed using approved methods, materials and procedures. All modified components will be tested to ensure that they continue to function in accordance with their design requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduce because the bases to Technical Specification 3/4.8.J provides no specific limits but rather discusses SSMP availability. Following installation, the modification will not affect the reliability, availability or operability of the SSMP, because the modification does not change the function of any SSMP component. During installation, loss of the SSMP control room FIC (1/2-2940-07) will occur as a result of construction activities. During this time, the SSMP can be operated locally using local FIC (1/2-2940-06). The appropriate actions as described in the section listed above will be implemented when the SSMP is out of service. Following the installation of the change all SSCs will perform their designed safety functions and meet all of their design requirements. There is no change to any setpoint, surveillance, or margin of safety as described in the basis for any technical specification.

Tracking No. SE-00-078
Activity No. UFSAR-99-R6-139

DESCRIPTION:

The principle changes being evaluated are the connection of the Cordova Energy Center (CEC) switchyard to Line 0402 and the connection of the TSS 940 Substation switchyard (near the CEC) to Line 0403. The tie-in of the CEC switchyard to Line 0402 is expected to occur in Mid-

September,2000. The tie-in of the TSS 940 Substation switchyard to line 0403 is expected to be implemented in Mid-October 2000.

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 connections of the Cordova Energy Center to Line 0402 and of TSS 940 Substation to Line 0403 are external to Quad Cities Station. Loss of offsite power from Lines 0402 and 0403 in the past has resulted from line faults or loss of transmission through Substations at Barstow and Nelson. The probability of a line fault or loss of transmission from Lines 0402 and 0403 with the tie-ins to the CEC plant and TSS 940 Substation is reduced because of the shortened line distances to Quad Cities. Shorter incoming transmission lines are subject to fewer lightning strikes and component failures or damage. The addition of the CEC plant introduces a remote possibility of a line loss in the event that the protective relaying fails to trip the generator during instability or loss of excitation. However, the failure of both primary and back-up protective relays to isolate the generator is unlikely. The probability of a loss of offsite AC power transient to Quad Cities Station overall is not increased.

   The consequence of losing power from Lines 0402 or 0403 to Quad Cities Station remains unchanged. The loss of an offsite transmission line reduces the number of alternate power supplies to the Station.

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 accident/transient relevant to the addition of the CEC plant to the electrical grid near Quad Cities Station is the potential loss of offsite AC power. With 5 incoming transmission lines to Quad Cities 345KV switchyard as before, the risk from loss of offsite power is minimized. Furthermore, loss of offsite power does not jeopardize the safe shutdown of the Station, because onsite standby diesels are capable of supplying the necessary power required for safe shutdown and accident mitigation. No adverse impact on existing accident analyses are created, and no new accidents are created as a result of the changes.

   The introduction of another generator in the electrical grid creates a minor potential for a tripped transmission line due to CEC generator failures, such as loss of excitation to the generator or generator instability. Protection schemes are being installed at the Cordova Energy Center to trip its generator upon such failures. The loss of a transmission line to the Quad Cities 345KV switchyard as a result of a CEC generator failure is enveloped under a loss of offsite AC power transient. The transmission lines are also not safety-related. Therefore, the changes do not create the possibility of a different type of malfunction of equipment important to safety at Quad Cities Station.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the accident/transient relevant to the addition of the CEC plant to the electrical grid near Quad Cities Station is the potential loss of offsite AC power. With 5 incoming transmission lines to Quad Cities 345KV switchyard as before, the risk from loss of offsite power is minimized. Furthermore, loss of offsite power does not jeopardize the safe shutdown of the Station, because onsite standby diesels are capable of supplying the necessary power required for safe shutdown and accident mitigation. No adverse impact

on existing accident analyses are created, and no new accidents are created as a result of the changes.

Tracking No. SE-00-079
Activity No. DCP #9300330; UFSAR-99-R6-141

DESCRIPTION:

The existing low suction pressure trip switch for the Reactor Feed Pumps (RFP) will be replaced by two new pressure switches arranged in a two-out-of-two logic configuration. A time delay will also be installed to trip the RFPs on a suction pressure of <125 psi for three seconds. The switches will also have a second trip setpoint for a suction pressure of <50 psi to instantaneously trip the RFPs. The additions of the second switch and the relay will improve the reliability of the Feedwater 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 new switches will increase the reliability of the Feedwater system. By utilizing the two-out-of-two logic, the failure of a single pressure switch will not trip the RFPs. The three second time delay from when the suction pressure drops below 125 psi will prevent the loss of feedwater during a condensate/condensate booster pump trip. No changes are being made to any safety-related equipment or any equipment that is utilized to shutdown during a loss of feedwater. Therefore, the probability or consequences of an accident or malfunction of equipment important to safety is 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 closed failure of both switches or the time delay relay would initiate a loss of feedwater accident, which has been previously evaluated. A failure of a switch during a low suction pressure accident will also initiate a loss of feedwater accident. No other accident can result from the failure of this equipment.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification is affected by this change. No margin of safety as described in the basis for any Technical Specification is reduced.

Tracking No. SE-00-080 (Supercedes SE-00-057)
Activity No. DCP 9900381 & 9900382; UFSAR-99-R6-126; UFSAR-99-R6-127; FPR-00-06

DESCRIPTION:

The activity is to provide an automatically connecting alternate source of 125 VDC control power to 4kV buses 13-1, 14-1 (Unit 1, DCP 9900381) and 23-1, 24-1 (Unit 2, DCP 9900382) from the Unit 1 and Unit 2 Station Blackout (SBO) 125 VDC system. A new auto transfer switch will be installed

in the control power circuit of each of these 4kV (dash) buses. The switches will be classified as safety-related and will be qualified to withstand a seismic event. Each new transfer switch will be mounted in a new box in the vicinity of the associated 4 kV switchgear. A combination of existing spare and new cables will be used to accomplish wiring of the new transfer switch, along with new fuse boxes and junction boxes. The existing normal 125 VDC control power feed will be disconnected from the 4 kV bus inside its switchgear cabinet and rerouted to the terminals of the normally closed contacts on the line side of the transfer switch. The Unit's 125 VDC Station Blackout (SBO) Battery will be connected to the terminals of the normally open contacts on the line side of the transfer switch. The load side terminals of the transfer switch will be routed back to the 4 kV bus inside the switchgear cabinet.

The existing 30 Ampere control power reserve feed circuit breakers for 14-1 & 24-1 will be replaced with 100 ampere circuit breakers.

UFSAR Changes UFSAR-99-R6-126 and UFSAR-99-R6-127, and Fire Protection Report Change Request Tracking Control No. 00-06 are associated with this 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 loss of the 125 VDC control power is not an initiator of any accident; loss of power to the circuit is less probable following the modification. The addition of a qualified safety-related component (the transfer switch) does not increase the likelihood of loss of 125 VDC control power and malfunction of equipment important to safety. Consequences of loss of the new transfer scheme are no worse than the consequences of loss of the existing 125 VDC control power.

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 potential interactions between the safety-related and non-safety-related power sources is not possible. The transfer switch main contacts are interlocked to prevent the two power sources from interacting. The automatic repowering is an improvement over the present manual repowering method.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the LCOs for the 125 VDC control power system being modified are based on the operability of the reactor building distribution panels and the turbine building reserve buses; i.e., the buses must be energized with the appropriate voltage available. This activity does not affect these LCOs. The 125 VDC LCOs remain valid after this modification is implemented.

Tracking No. SE-00-081
Activity No. DCP 9900528, DCP 9900529 & TIC 0162

DESCRIPTION:

This activity will install automatic start logic for the 1-0202-47A(B) and 1-0202-48A(B) Fluid Coupler Lube Oil pumps for the 1A(B) Recirc MG Sets. The scoop tube control, 1-0202-49A(B) emergency lube oil pump and alarm circuits will also be modified.

This activity will also test the new configuration. The logic of the modified circuits will be tested and the response of the Recirc MG sets to simulated failures will be measured.

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 does not increase the probability of occurrence or consequences of an accident because it does not alter the initial conditions or sequence of events assumed in the SAR analysis. This activity does not increase the probability or consequences of a malfunction of equipment important to safety because the modification will be performed to industry standards and no new failure modes will be introduced to the Reactor Recirc or Reactor Recirc Flow Control systems.

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 create the possibility of an accident or transient different than previously analyzed because the rate and magnitude of recirculation flow changes are well within the bounds of existing analyses. The potential for failure or malfunction of the components is not affected by this activity and this activity does not introduce any new failure modes to the Reactor Recirc or Reactor Recirc Flow Control systems.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does not affect the basis for any Technical Specifications.

Tracking No. SE-00-082
Activity No. UFSAR-99-R6-144

DESCRIPTION:

The changes are:

1. The Q1C17 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 1.

2. The introduction of improved LHGR limits for the Unit 1GE fuel provides more operational flexibility. This is accomplished by removing the thermal mechanical restrictions from the APLHGR limits associated with Unit 1 GE fuel and using APLHGR to monitor LOCA based limits, and using improved methodology for determining the LHGR limits used to monitor thermal-mechanical based limits. This improvement results in increased LHGR limits at high exposure. A benefit of this improvement program is to extend the APLHGR xposure limits to 61.1 GWd/MT.

3. Add a new option of operating with uncalibrated LPRMs at startup (from BOC up to 500 MWd/MT) to increase operational flexibility during startup.

   This 10CFR50.59 safety evaluation addresses these changes and the associated UFSAR and Technical Specification Bases changes to support Q1C17 operation. Additional UFSAR changes are being implemented via this 50.59 to address Design Basis Initiative (DBI) concerns regarding the stability section (4.3) of the UFSAR. These UFSAR changes are purely editorial in nature and will not be addressed further in this 50.59.

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. The changes being evaluated in this 50.59 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 to allowable modes of operating. The Q1C17 reload core design does not involve any modifications to the plant configuration. No new precursors to an accident are created and no new or different kinds of accidents are created.

   The changes implemented addressed by this 50.59 do not physically alter the systems designed to prevent an accident from occurring.

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 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. The changes addressed by this 50.59 have been analyzed using NRC-approved methodologies and are supported in all allowable modes of operating. These changes do not involve any modifications to the plant configurations. 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.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Q1C17 core design was generated using NRC approved methods. All required thermal limits (including the Improved LHGR Limits for GE10 and uncertainty penalties for operating with uncalibrated LPRMs at startup) have been established using

NRC approved methodologies to protect the Q1C17 core during all anticipated operation occurrences and these limits are presented in the COLR. Therefore, since the Q1C17 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-00-083
Activity No. Interim Procedure - TIC-0163

DESCRIPTION:

During installation activities for DCP 9900169, Bus 1B-1 (Division II) will be disconnected from it's main feed at Bus 2A and re-powered from it's alternate feed at Bus 1A. This will be performed during Q1R16. This procedure will also disconnect various non-vital loads from the Unit 1 battery prior to swapping the power feeds to ensure that the existing battery capacity is maintained.

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 interim procedure will affect the loading of both the Unit 1 and Unit 2 batteries. With Bus 1B-1 on its alternate feed, the loading on the Unit 2 battery will decrease which is in a conservative direction. The increased loading on the Unit 1 battery is offset by the removal of non-vital loads such as plant lighting, Main Generator, HPCI, etc. The net effect of this bus alignment in regards to loading and capacity will remain unchanged. These changes cannot cause a LOOP/LOCA or any fires in regards to Appendix R.

   In regards to equipment/cable separation, there will be Division I and II equipment fed from the same battery with this arrangement. However, since Unit 1 will be shutdown during this time period, only one division is required to be operable. In regards to Appendix R, Bus 1B-1 will be fed from a different source (Bus 1A). However, since Buses 1A and 1B-1 are located in the same fire area, the consequences of a fire in that area remain unchanged.

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 battery loading will not be adversely affected by this bus arrangement. Since only one division is required to be operable on a shutdown unit, both divisions may be powered from the same battery. The added loads from Division II to the Unit 1 battery are the same type that exist on Unit 2 which the Unit 1 battery is currently feeding. Negative interactions associated with the added loads produce the same results as negative interactions from the Unit 2 equipment. In regards to Appendix R, both Bus 1B-1 and its new feed breaker are located in the same fire area. A fire associated with the new feed at Bus 1A produces the same results as a fire at Bus 1B-1. Fires in this area have been previously evaluated. The activity, therefore, will not create the possibility of an accident/transient or malfunction than previously analyzed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the use of the alternate feed for Bus 1B-1 is restricted to time periods when Unit 1

is shutdown. During this time, only one division of AC and DC power sources/buses are required. Since Unit 2 will be running during this time, both divisions of AC and DC power sources/buses are required to be operable. However, Unit 2 will be unaffected by the use of the Bus 1B-1 Alternate feed and will therefore, meet this requirement.

Tracking No. SE-00-084
Activity No. QCOS 0700-12 (Rev 0)

DESCRIPTION:

New procedure QCOS 0700-12, Controlling The Use Of Alternate SRMs, provides administrative controls and a tracking record for the use of special movable detectors. An SRM detector may be installed in an IRM location to provide monitoring of the core quadrant as discussed in Technical Specifications. Installation of an alternate SRM and restoration of the original SRM are verified within the procedure. The procedure has attachments for each SRM (quadrant) where an alternate SRM will be installed in an IRM location. The status sheets will allow for specific checks to be documented and provide a sign off by the Unit Supervisor verifying operability.

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 SAR documents describe that prior to and during the design base fuel handling accident the SRMs provide indication to the control room to verify the core is subcritical. The fuel handling accident states that a bundle is accidentally dropped on to the core. The movement of an SRM to a different location in the core does not affect the probability of a bundle being dropped. The alternate SRMs provide the identical information to the control room operator as the normal SRM to insure subcriticality is maintained.

   The consequences of a bundle drop accident are impacted by the weight, height of fall and the accumulated fission product activity in the bundle. The refueling accident is mitigated by the containment systems. These are independent of the function of the SRM 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 SRM detectors are designed to be installed in the same dry tube as the IRM. The original SRM cabling will be attached to the alternate SRM in the same core quadrant. Any failure mode of the detector would be identical to a failure mode of a normal SRM. Loss of an alternate SRM signal will result in the same action as the loss of the original SRM.

   The consequences of a failure of the alternate SRM would be the loss of control room indication. The action and results would not be changed. Affected core alterations would be halted until the detector and circuit were repaired.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications 3/4.10.B, Instrumentation, states that special movable detectors (use in place of the current SRMs) are permissible provided that they are connected to the normal circuit. The SRM detector will be moved to an IRM location in the

same quadrant and remain connected to the current SRM circuit. The position signal will still generate the required rod block using the IRM drive control. The basis section of the Technical Specifications describes that these special movable detectors may be used during core alterations in place of the normal SRM neutron detectors. These special detectors must be connected to the normal SRM circuit such that the applicable neutron flux indications, control rod blocks and scram signals can be generated. The detectors provide flexibility during fuel loading since they can be positioned anywhere within the core quadrant.

Tracking No. SE-00-088
Activity No. DCR 99-0732; UFSAR-99-R6-145
QCOP 5370-04; Rev. 5; QOM 1-3900-01, Rev. 11; QOM 2-3900-01, Rev. 8

DESCRIPTION:

The 1(2)-3999-116, 2" Hydrogen Cooler TCV Bypass Valve will be returned to the normally closed position. This returns the system to the as-designed condition and brings Quad Cities into conformance with the industry's normal operation of the hydrogen cooling 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 this is a non-safety related system and does not affect any safety-related equipment. The 1(2)-3999-116, 2" Hydrogen Cooler TCV Bypass Valve will be returned to the normally closed position. This returns the system to the as designed condition and brings Quad Cities into the industry's normal operation of the hydrogen cooling system. There are no interfaces with fission product barriers or accident mitigation systems so there is no increase in dose consequence to the public.

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. This returns the system to the as-designed condition and brings Quad Cities into the industry's normal operation of the hydrogen cooling system. This is a non-safety-related system and does not affect any safety-related equipment.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are not any Technical Specifications affected by this activity.

Tracking No. SE-00-089
Activity No. QCOP 5600-08

DESCRIPTION:

Procedure QCOP 5600-08, "Unit 2 Operation with One Turbine Control Valve Closed" provides guidance to operate the plant with any one TCV closed. The procedure requires Reactor Thermal Power to be maintained less than or equal to 83%. It will also require that the reactor will be operated at a flow control line less than or equal to 93%. The turbine load limit set will be increased to assure maximum opening of the TCVs when necessary. The TCV fast closure scram will continue to be operable in accordance with the 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 no current accident analysis is predicated on the loss of one or more turbine control valves nor are any event initiators affected by operating with one control valve closed.

   The consequences are not increased because the plant is being limited to 83% power and 93% flow control line. This ensures that the current accidents and transients evaluated in the UFSAR remain bounding during 3-valve 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 original UFSAR safety evaluations remain bounding under the plant operating restrictions; therefore, no new accident scenarios are created. Also, if the closed valve fails open, the plant returns to normal 4-valve operation for which it was originally analyzed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because due to the operating restrictions placed on the plant under this configuration, the original bounding accident analyses remain so, and so the margins of safety are not affected.

Tracking No. SE-00-090
Activity No. UFSAR-99-R6-142

DESCRIPTION:

Section 5.2.5.4.1 is being revised to reflect and clarify the as-built configuration of the Primary Containment Particulate Sampling System (PCPSS). Specifically, sample line primary containment isolation is provided by either redundant manual or automatic isolation valves. Also, the air sample is collected at the local rack using a filter cartridge holder or by obtaining a grab sample for laboratory analysis.

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 PCPSS is a containment air sampling system, and is not used for accident mitigation. Both of the affected sampling points are isolated upon receipt of a PCI Group 2 isolation, and are isolated under normal plant operating conditions by a manual valve (and cap). Therefore, the affected equipment will have no impact on the reactor coolant pressure boundary nor the primary containment boundary.

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 equipment is outside the PCI Group 2 isolation boundary, and has no other interaction with other plant equipment except for providing primary containment air sampling points. The as-built configuration is less likely to fail (than a filter cartridge holder), since a cap provides a second isolation boundary to the manual valve.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification function is provided by a Continuous Air Monitor, which was unaffected by this change. The function of the affected PCPSS equipment is a supporting role, used to help locate and find the source of an identified leak. This supporting function is not required by Technical Specifications.

Tracking No. SE-00-091
Activity No. DCP 9900589 and DCP 9900594

DESCRIPTION:

A temporary source of 125 VDC control power will be provided to 4kV switchgear 23-1 and 24-1. A temporary cable will be installed between an available breaker at 125 VDC Turbine Building Main Bus 2A-1 and Switchgear 23-1. Once work associated with Switchgear 23-1 is complete the same cable will be used between an available breaker at 125 VDC Turbine Building Reserve Bus 2B-1 and 4kV Switchgear 24-1. The cable will run from the appropriate Unit 2Two Battery Charger Room elevation 615', out an open charger room door, up through the stairwell to the turbine deck elevation 639', south on elevation 639' to either 4kV Switchgear 23-1 or 24-1. The cable will be free-air for the entire route and secured at appropriate points.

The temporary 125 VDC power source will be paralleled with the existing 125 VDC main feed for each switchgear. The existing main feed will then be taken out-of-service. Prior to removing the temporary 125 VDC power source the main feed will be returned to service.

The feed breaker control circuit for the Unit Two SBO EDGs at Switchgear 23-1 and 24-1 will be utilized to back-feed the temporary power to the switchgear's 125 VDC control power bus. The SBO breakers at cubicle 3 (switchgear 24-1) and cubicle 8 (switchgear 23-1) will be removed from service while the temporary cable is connected to the breaker control circuit. Any control circuit wiring that is disturbed will be returned to its original configuration, and the breaker tested, prior to returning the breaker to service.

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 temporary 125 VDC power feed will be relied on for operation of safety-related 4kV breakers. These breakers close to provide power to loads important to mitigation of the accidents. The temporary circuit will provide this safety-related 125 VDC power as sufficiently and reliably as the existing circuit.

   Voltage drop, ampacity, divisional separation, fire-induced faults and the affects of a design basis seismic event have been evaluated. The temporary circuit will be equivalent to the existing circuit for all critical parameters. Precautionary measures have been taken to prevent a malfunction of the cable due to it being routed in free-air.

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 provides a temporary 125 VDC feed to Switchgears 23-1 and 24-1. The temporary feed is equivalent to the existing feed electrically and will withstand a design basis seismic event. Complete failure of the cable is bounded by failure of a division of AC power. The temporary cable interacts with the 125 VDC system in the same way as the existing cable. A breaker is provided to protect the cable and isolate it from the system should the cable fault.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Electrical Power System Technical Specification is based on operability of the AC and DC onsite power distribution systems (i.e. they must be energized with the appropriate voltage available). This activity does not affect the power distribution systems.

Tracking No. SE-00-092
Activity No. UFSAR-99-R6-143

DESCRIPTION:

This Safety Evaluation addressed the addition of information to UFSAR Section 6.2.5.2 to accurately describe the function of the ACAD Drywell Pressure instrumentation. This instrumentation is required for post-accident monitoring by Reg.Guide 1.97.

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 ACAD Drywell pressure instrumentation provides an indication and alarm function in the control room for post-accident monitoring. This activity does not affect the seismic qualifications or structural integrity of either the Drywell or the ACAD system. The instruments are not relied upon to provide for any automatic function to mitigate the consequences of an accident or transient, and do not affect the design, function or operation of any other 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 Operation of the ACAD Drywell pressure instrumentation does not affect the automatic actuation of any equipment important to safety. No single failure within either the ACAD Drywell pressure instrumentation, its auxiliary alarm feature, or its power supply concurrent with the failures that are a condition or a result of any specified accident will prevent the control room operators from being presented the required Drywell pressure indication necessary for them to determine the safety status of the plant, and to bring it to and maintain it in a safe condition following an accident. Therefore, this activity does not create the possibility of an accident or malfunction 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 ACAD Drywell pressure instrumentation is not described in the current Technical Specifications or bases. Upon implementation of the ITS Submittal, the ACAD Drywell pressure instrumentation will be required to meet Technical Specification requirements for Post-accident Monitoring Instrumentation. This UFSAR change does not affect these requirements, nor does it affect the system function, as described in UFSAR Section 7.5.1. Therefore, the margin of safety is not affected.

Tracking No. SE-00-093
Activity No. DCP 9800294

DESCRIPTION:

The weld overlay application during Q1R16 for weld 02BS-F4 in the Unit 1 Reactor Recirculation piping was stopped after three layers (approximately 0.2 inches) of weld material was deposited due to ALARA considerations. The purpose of Revision 1 to DCP 9800294 is to leave the overlay at a design thickness of 0.2 inches for one cycle.

The partial overlay does not meet the ASME Section XI requirements for a permanent repair. A full structural overlay of this weld will be completed under DCP 9900600 during Q1R17. Following completion of DCP 9900600, the weld overlay design will be in compliance with the requirements of NUREG 0313, Rev. 2, ASME Section XI - 1989, and Code Case N-504-1.

This evaluation concludes that NRC approval of the partial weld overlay and the supporting evaluations is required prior to Start-up from Q1R16.

The Safety was also used for DCP 9900600, 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 two engineering evaluations have been performed in support of this change. These evaluations are documented in Calculations QDC-0200-M-1029 Rev. 0 and QDC-0200-M-1030 Rev. 0.

Calculation QDC-0200-M-1029 evaluated the flaw based on the indication size measured during the 1998 inspection (27 inches long and 0.25 inches deep) and assumed crack growth until Q1R17 in 2002 per NUREG 0313, Rev. 2. This evaluation demonstrated that the indication can be qualified to the criteria of ASME Section XI (1989) without taking credit for the 0.2 inches of the weld overlay that were deposited during Q1R16.

The second evaluation was performed because difficulty conducting a UT examination on the 02BS-F4 weld following the overlay prevents confirmation of the evaluated flaw size. Although the UT procedure used was demonstrated to be capable of interrogating the entire pipe wall through the partial weld overlay, the Performance Demonstrated Initiative (PDI) qualified procedure has only been qualified to detect and size indications in the weld overlay and outer 25% of the base metal. Due to the lack of PDI qualification for interrogating the piping inside diameter, a second evaluation was performed for a postulated, worst case flaw.

QDC-0200-M-1030 postulated an initial flaw could be 75% through wall and crediting the additional wall thickness (0.2 inches) applied by the weld overlay, operation until Q1R17 was qualified to the criteria of ASME Section XI (1995 with 1996 addenda). The 1989 edition of Section XI imposes a maximum limit on the flaw depth for flux welds of 0.6 times the thickness. This limit was increased to 0.75 times the wall thickness in the later editions of Section XI. Both of the evaluations performed by GE were submitted to the NRC.

The planned repairs will restore the affected piping to its design. The joint design provides for a ferrite barrier, which ensures that the crack will not propagate into the structural weld overlay material. The process of the deposition of molten metal in the weld overlay will place the region of the crack into compression when the overlay cools to the ambient temperature of the pipe. Placing the crack into compression has the affect of suppressing crack propagation. The crack therefore, will be inhibited from growing to the point where it could cause piping failure. Additional layers of weld material are applied to provide additional structural strength to the weld design and meet code minimum requirements. With the mechanism for crack propagation inhibited and the structural strength restored, the probability of failure/accident is greatly reduced. Therefore, the partial overlay and deferred completion of the full overlay will not increase the probability of occurrence of any accident or transient.

The weld overlay does not have any mechanism to affect the magnitude of any radioactivity, or steam/water released from the ruptured piping. The operating characteristics of the reactor and the reactor coolant system are not changed; therefore, motive force for a release remains unchanged. These repairs in no way affect the source term available for release. Therefore, the consequences of an accident of malfunction of equipment important to safety is 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 instantaneous failure of this piping is already evaluated. The application of the weld overlays does not introduce any new failure modes to the Reactor Recirculation piping, or to any other component or equipment important to safety. Inspections were performed to verify that axial piping shrinkage due to the overlay has not affected supports on the piping system. No other equipment is affected by this activity. Therefore, the activity does not create the possibility of an accident or malfunction 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 Technical Specifications require that inservice inspection program for the piping associated with Generic Letter 88-01 be performed in accordance with NRC staff positions on schedule, methods, personnel, and sample expansion. The Technical Specification permits alternate measures to be used to evaluate this piping subject to NRC approval.

   The original intent of this design change was to provide a full structural weld overlay repair in accordance with ASME Sections IX and XI 1989 Edition, and Code Case N-504-1. Due to ALARA concerns, the weld overlay was only partially completed during Q1R16. Calculations QDC-0200-M-1029 Rev. 0 and QDC-0200-M-1030 Rev. 0 provide a technical justification of the acceptability of the partial weld overlay until the next refueling outage (Q1R17). The as left condition of the piping is not completely addressed by the existing NRC GL 88-01 staff positions, therefore, NRC approval of the partial weld overlay and the supporting evaluations is required prior to Start-up from Q1R16 by Generic Letter 88-01.

Tracking No. SE-00-094
Activity No. FPR 00-09

DESCRIPTION:

The change permanently incorporates the use of the SBO Diesel Generators (SBO DG's) in the upgraded safe shutdown methodology for certain fire areas. The original safe shutdown methodology only credited the Emergency Diesel Generators (EDG's) to provide on-site AC power during post-fire safe shutdown. For the purposes of fire induced safe shutdown the EDG's and the SBO DG's are functionally equivalent.

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 incorporating the use of the SBO DG's for certain fires into the Safe Shutdown Report (SSR) does not increase the quantity or physical arrangement of combustible material, nor does it create additional ignition sources. The specific purpose of the SSR is to describe the methods available to mitigate an accident (in this case, fire as postulated in the fire hazards analysis) that has already occurred. This change does not increase the probability that the accident (Appendix R fire) may occur.

   Incorporating the use of the SBO DG's into the SSR does not impact the systems responsible for the control and mitigation of off-site dose releases. The systems responsible for control of off-site dose will function in the same manner regardless of the source of 4 kV electrical power to the ESS busses during post-fire safe shutdown activities. No new pathways for release are created. The SBO DG's are able to perform the intended function because their location and associated post-fire circuit analyses assures they will be free of fire damage. The manual start and load of the SBO DG's can be completed within the time necessary for initiation of AC powered safe shutdown equipment. All other safe shutdown equipment will operate the same, regardless of the specific generator set which is providing power. The Safe Shutdown Analysis demonstrates that the SBO DG's can successfully provide electrical power to operate safe shutdown equipment. Satisfying

the SSR performance goals established in Generic Letter 81-12 (reactivity control, reactor coolant makeup, reactor heat removal, etc.) assures that no adverse off-site dose consequences will occur. Therefore, the activity does not increase the consequences of any accident or transient 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 SSR and supporting analysis has demonstrated that the SBO DG's can be used to support safe shutdown equipment. The detailed circuit analysis shows that the SBO is free of fire damage for certain fire areas. Using the SBO DG's to provide power is functionally equivalent to using the EDG's and would not introduce a new failure mode.

   The EDG's and the SBO DG's provide power to the same electrical busses. Failure of a generator set has no different impact on the bus or its connected loads when power is provided by the SBO DG's than would be the case when the EDG's are providing power. The connections between the SBO DG's and the 4 kV busses is through a safety-related breaker, which provides the same protection for the bus as when the EDG's are providing power. Therefore, changing the power source does not introduce any new failure modes to the AC power distribution system. Note, in accordance with GL 86-10, additional accidents or system failures need not be considered simultaneous with non-fire related failures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the SBO DG's are not addressed in plant TS. The bases for TS 3/4.9.A "Electrical Power Systems," specifies that the EDG's have sufficient fuel for 2 days of operation under design basis accident conditions and that additional diesel fuel can normally be obtained and delivered to the site within an eight-hour period. This change does not affect the fuel availability for the EDG's. Nor does the activity affect the LCOs for the AC sources and therefore, the margin of safety as described in the basis is not reduced by this activity. To extend diesel generator run-time, both the original SSR and the TS Bases credit fuel replenishment from an off-site source.

Tracking No. SE-00-095
Activity No. FPR 00-09

DESCRIPTION:

This activity updates the existing Fire Protection Report - Safe Shutdown Analysis (FPR - SSA) to include the HPCI System as a system evaluated to satisfy the 10 CFR 50 Appendix R criteria for reactor coolant makeup. The Unit 1 HPCI system will be credited to provide reactor water makeup to the Unit 1 reactor vessel for fire in Fire Area TB-II. Currently, the Unit 1 Reactor Core Isolation Cooling (RCIC) system is credited.

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 physical changes being made to the plant by the activity. The probability of a design basis fire occurring is based on the amount and type of combustibles in an area, the number of ignition sources in the area, the type of work being

performed in the area, whether fire suppression and detection are installed in the area, and the response of the fire brigade. The change has no impact on these fire protection attributes.

No new pathways for radiological release are created by the activity. The safe shutdown analysis (SSA) demonstrates that the HPCI system will provide reactor water makeup for areas where the HPCI system is free of fire damage. Procedures have been established to ensure HPCI will be operated within the system design limits. The improved SSA ensures equipment is available and operated in the time frame to ensure the performance goals established in Generic Letter 81-12 (reactivity control, reactor coolant makeup, reactor heat removal, etc.) are met. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated is 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 there are no SSCs affected by this change. The use of the HPCI system equipment in the safe shutdown analysis does not require any SSCs to be modified or operated in an abnormal manner. Procedures have been established to ensure HPCI will be operated within the system design limits. Note, in accordance with GL 86-10, additional accidents or system failures need not be considered simultaneous with non-fire related failures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because utilizing the HPCI system as a reactor coolant makeup source in order to achieve hot shutdown in the event of fire in Fire Area TB-II does not impact the basis for the Emergency Core Cooling System (ECCS). The change has no impact on the design basis function of the HPCI ECCS subsystem.

Tracking No. SE-00-096
Activity No. FPR 00-08 & FPR 00-09

DESCRIPTION:

This activity updates the existing Fire Protection Reports, Volume 1 - Fire Hazards Analysis (FHA) and Volume 2 - Safe Shutdown Report (SSR) to reflect the creation of five (5) new fire areas in the plant. Note that the changes to the FHA are administrative in nature reflecting the designation of the new fire areas. The following fire zones were used to create the new fire areas:

Fire Zones	New Fire Area
1.Fire Zones 6.1.A, 6.1.B and 7.1 (currently part of Fire Area TB-III)	Fire Area BC-1
2.Fire Zone 8.2.4 (currently part of Fire Area TB-III)	Fire Area CT-1
3.Fire Zones 6.2.A, 6.2.B and 7.2 (currently part of Fire Area TB-I)	Fire Area BC-2
4.Fire Zone 8.2.5 (currently part of Fire Area TB-I)	Fire Area CT-2
5.Fire Zone 9.2 (currently part of Fire Area TB-I)	Fire Area EDG-2*

* Note: The fire area designation of Fire Area EDG-2 and Fire Area DG-2 is used interchangeably in the fire protection reports, documentation and analyses.

A plant Fire Area is that portion of a building or plant that is separated from other areas by 3-hour rated fire barriers (walls, floors, or roofs). Any openings or penetrations are protected with seals or closures having a fire resistive rating equal to that of the barrier. Exceptions are justified with engineering evaluations prepared in accordance with NRC Generic Letter (GL) 86-10.

Fire zones are subdivisions of fire areas. Fire zones are established using boundaries within a given fire area.

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 a design basis fire occurring is largely based on the amount and type of combustibles in an area, the number of ignition sources in the area, the type of work being performed in the area (which may add ignition sources and/or combustibles), whether fire suppression and detection are installed in the area, and the response of the fire brigade (the last two can affect the spread and severity of the fire). The change has no impact on these fire protection attributes. Only the designation for the fire areas were changed. The fire zone designations were not changed and the fire hazards information is maintained by fire zone, not fire area. No new pathways for radiological release are created by the activity. The safe shutdown analysis demonstrates that for a fire in new fire areas safe shutdown will be achieved. Thus, satisfying the SSR performance goals established in Generic Letter 81-12 (reactivity control, reactor coolant makeup, reactor heat removal, etc.) assures that no adverse off-site dose consequences will occur. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated is 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 there are no SSCs affected by this change. The use of the five (5) new fire areas in the safe shutdown analysis does not require any SSCs to be modified or operated in an abnormal manner. The new fire areas are comprised of fire zone barriers which have a 3-hour fire rating thus providing additional protection from the spread of fire between fire areas.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the only equipment for which Technical Specifications are established for post-fire Safe Shutdown is the Safe Shutdown Makeup Pump (SSMP). These changes do not impact the basis for the SSMP. The changes enhance the operation of the SSMP by improving the system availability from the control room for certain fire areas.

Tracking No. SE-00-097
Activity No. FPR 00-09

DESCRIPTION:

The activity credits an automatic transfer device that provides an alternate source of 125 Vdc control power to the emergency 4 kV buses 13-1, 14-1 (Unit 1) and 23-1, 24-1 (Unit 2) from the Unit 1 and Unit 2 SBO 125 Vdc system to support post-fire safe shutdown for certain fire areas.

The physical changes to install the auto transfer feature were made and evaluated under the Design Change Process (DCP). Therefore, during post-fire safe shutdown activities, the automatic repowering of the 4 kV switchgear control circuits can be credited when the normal 125 Vdc power source (125-Vdc station battery system) is lost due to fire.

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 crediting the SBO 125 Vdc system in the SSA does not increase the quantity or physical arrangement of combustible material, nor does it create additional ignition sources. The specific purpose of the SSA is to describe the methods available to mitigate the consequences of a design basis fire that has already occurred. This change does not increase the probability that the accident (fire) may occur.

   Incorporating the use of the SBO 125 Vdc system into the SSA does not impact the systems responsible for the control and mitigation of off-site dose releases. The systems responsible for control of off-site dose will function in the same manner regardless of the source of dc control power for the 4 kV ESS busses during post-fire safe shutdown activities. No new pathways for release are created. The SBO 125 Vdc system is able to perform the intended function as long as post-fire circuit analyses assures that the system is free of fire damage (a component is free of fire damage as defined in GL 86-10 if it is capable of performing its intended function during and after the postulated fire, as needed). The Safe Shutdown Analysis demonstrates that the SBO 125 Vdc system can successfully provide dc control power to operate safe shutdown equipment. Satisfying the SSA performance goals established in Generic Letter 81-12 (reactivity control, reactor coolant makeup, reactor heat removal, etc.) assures that no adverse off-site dose consequences will occur. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated is 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 change involves no physical changes to SSC's important to safety. The change modifies the SSA to credit two independent sources of dc control power to essential 4kV switchgear. The SSA and supporting analysis has demonstrated that the SBO 125 Vdc system can be used to support safe shutdown equipment. Use of the SBO 125 Vdc system is only credited when the circuit analysis shows that the SBO 125 Vdc system is free of fire damage. The loading on the SBO 125 Vdc battery system during post-fire safe shutdown has been evaluated and found acceptable. The SBO 125 Vdc system provides an equivalent source of dc control power as the station batteries.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the SSA credits an automatically connecting alternate source of 125 Vdc control power to the essential service 4 kV buses (13-1, 14-1 and 23-1, 24-1) from the Unit 1 and Unit 2 SBO 125 Vdc systems due to loss of the station 125 Vdc system during certain fires. The change to the SSA does not reduce the margin of safety of the station 125 Vdc system since it does not adversely affect the station 125 Vdc system. The installation of the switch that provides the auto-transfer capabilities was installed and evaluated under the DCP process.

DESCRIPTION:

The Safe Shutdown Analysis (SSA) has been revised and re-written in response to the issues raised in the NRC's Confirmatory Action Letter (CAL) [CAL RIII-98-001]. Through a combination of plant design changes, detailed circuit analysis, new safe shutdown strategies, and revised procedures, the post-fire safe shutdown capabilities have been strengthened. These actions have reduced the number of time-critical manual actions conducted outside of the Main Control Room for certain fire areas. Through detailed circuit analyses, opposite unit effects (i.e., the impact on the non-fire unit) have been analyzed. In general, the opposite unit will be shutdown using normal and abnormal station procedures, without the need to resort to post-fire safe shutdown procedures. A number of plant modifications were completed that directly improve the availability of 125 Vdc control power, which coupled with system modifications have improved the ability to remain in the control room during fire events.

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 does not increase the quantity or change the physical arrangement of combustible material, nor does it create additional ignition sources. The specific purpose of the SSA is to describe the methods available to mitigate an accident (in this case, an Appendix R Fire) that has already occurred.

   No new pathways for radiological release are created by the activity. The safe shutdown analysis demonstrates that systems will be free of fire damage, satisfying the SSR performance goals established in Generic Letter 81-12 (reactivity control, reactor coolant makeup, reactor heat removal, etc.) assures that no adverse offsite dose consequences will occur. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated is 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 revision to the SSA will not modify any plant systems. The SSA determines which SSCs are free of fire damage and can be used to effect post-fire safe shutdown. The SSA and implementing procedures (QCARPs) ensure no new operating failure modes for equipment have been introduced. In some instances, equipment is controlled from local stations; however, procedures have been developed to ensure equipment is operated within design limits. The procedures identify when (which fires and for which system function) and where (local control station or at the component) these local operations are required. Local control of equipment was required by the original SSA; however, the improved SSA has reduced the number of local manual actions for certain fire areas (in particular, the time critical actions associated with core injection).

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because through a combination of plant design changes, detailed circuit analysis, new safe shutdown strategies, and revised procedures, the post-fire safe shutdown capabilities have

been strengthened. The Technical Specification (TS) do not provide a direct margin of safety for the fire protection program. License condition h.3.F allows changes to the fire protection program provided the changes do not adversely impact the ability to achieve and maintain safe shutdown in the event of a fire. The revised SSA has been shown to not adversely impact the ability to achieve and maintain safe shutdown.

SE-00-099
Activity No. UFSAR-99-R6-146

DESCRIPTION:

Revise UFSAR paragraphs 6.3.3.1.3.1 and 6.3.2.3.8 to clarify the HPCI safety function and design features during "continuous operation". The HPCI safety function is to automatically start once. The HPCI design features are to trip the HPCI turbine at a Reactor Vessel level of +48", and to restart the HPCI turbine when –59" Reactor Vessel level is reached.

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 UFSAR clarification adds no new equipment or procedure revisions. The HPCI sub-system will continue to operated in the same manner. This clarification does not impact the Station's analyzed design basis in response to a LOCA. The need to restart HPCI in response to a design basis accident is not required since no single failure can prevent the ADS from successfully depressurizing the Reactor Vessel below LPCI and CS discharge pressures. The HPCI's ability to mitigate the consequences of an accident are unchanged by this clarification. This UFSAR clarification does not change the consequences of any equipment malfunction.

   The probability of an accident is derived from the probabilities of the precursors to the accident. The proposed UFSAR change includes no physical changes to any plant system. Thus the probability of an accident is 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 this proposed activity is to clarify the HPCI safety function to initally automatically start once and inject into the Reactor Vessel in response to a low-low Reactor Vessel level, or high Drywell pressure initiation signal. The HPCI design features are to trip the HPCI turbine when Reactor Vessel level reaches +48", and to restart the HPCI turbine when -59" Reactor Vessel level is reached. This is the current design of the HPCI sub-system. No procedure revisions or new equipment will be added by this UFSAR revision. No new accident or transient is created by the clarification.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the proposed activity will have no adverse affects on the operation of the HPCI system. The HPCI system will still meet its safety function of delivering 5000 gpm within 45 seconds to the Reactor Vessel.

The high Drywell pressure and low-low Reactor Vessel level HPCI initiation signals are not affected.

Tracking No. SE-00-100
Activity No. OOS To Support WR 980102773

DESCRIPTION:

The Unit 2 fuel pool level switch was found to be in disrepair and subsequently taken Out-Of-Service (OOS). A Design Change Package (DCP) has been prepared to install a replacement switch under Work Request (WR) 980102773. This supporting 50.59 evaluation is for the extended duration that the OOS will remain in place to support the preparation and installation of the new spent pool level switch on Unit 2. With this switch disabled, the high and low level alarms are not available from the Unit 2 spent fuel pool. However, the Unit 1 and Unit 2 fuel pools are typically tied together, and any change in the Unit 2 fuel pool is reflected in the Unit 1 fuel pool which will continue to provide level alarm functions. The switch is passive in nature and provides no trip or interlock functions. Furthermore, this condition does not increase the possibility of a leak from the spent fuel pool, and has no affect on the ability to add or reject water from the pool or to remove heat from the pool.

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 fuel pool level switch is passive in nature in that it provides only an alarm function (i.e., no trip or actuation functions are effected by this OOS). The switch is not an initiator of any transient or accident and is not used for mitigation of any transient or accident. For this reason, the consequences of an accident or a malfunction of equipment important to safety previously evaluated is 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 absence of the fuel pool level switch does not create the possibility of a new accident or transient. The fuel pool system is designed to minimize the possibility of a leak that will result in uncovering the fuel stored in the pool. The design function of the spent fuel pool is not adversely affected by this condition. For this reason, the possibility of an accident or malfunction of a different type is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because as stated in the UFSAR, the fuel pool system is designed to minimize the loss of water from the pool and to prevent the water level from falling below a safe level. For example all penetrations into the pool, except for valved drains, are located at a height such that there will always be a safe level of water above the fuel." The fuel pool level switch simply provides a method to monitor the level by use of an annunciator. This activity does not adversely affect the fuel pool design function. For this reason, the margin of safety as defined in the basis for any Technical Specification is not reduced.

Tracking No. SE-00-101
Activity No. QCTP 0130-14 Rev. 1, QCTS 0820-01 Rev. 7, QCTS 0820-02 Rev. 6,
QCTS 0820-03 Rev. 6

DESCRIPTION:

These procedure changes reflect a change in the methodology used to evaluate the acceptability of the Residual Heat Removal Service Water (RHRSW) Vault flood protection barriers. This Safety Evaluation supercedes SE-99-096 and SE-98-163. The changes are summarized as follows:

* The acceptability of the RHRSW vault flood protection barriers will be evaluated based on the total leakage measured for unisolable barriers associated with a specific internal flooding scenario.
* The acceptance criteria has been revised to permit minor leakage into the vault. Previously, the acceptance criteria was no visible leakage using a soap bubble solution to check for leaks.
* At the maximum allowable leakage rate, the operability of the safety-related equipment inside the RHRSW vaults would not be affected for a minimum of 48 hours following a design basis internal flood of the condensate pump room area because the water level in the vaults would remain lower than the equipment.
* During a internal flood of a RHRSW vault, the non-flooded RHRSW vaults will remain accessible. At the maximum allowable leakage rate, the operability of the safety-related equipment inside the other RHRSW vaults would not be affected for a minimum of 8 hours following a design basis internal flood of a RHRSW vault area because the water level in the vaults would remain lower than the equipment.
* Appropriate procedure changes were made to remove provisions that allowed the performance of a visual inspection of the components every other operating cycle instead of a quantitative leakage test. A quantified leakage test provides better assessment of the condition of the flood barriers and will be performed once per operating cycle. The implementation of a more restrictive testing frequency provides increased assurance that the flood barriers will function as designed.

Previous experience has shown that the majority of the test "failures" have been attributed to minor air leakage and not gross failure or excessive leakage. The current acceptance criteria has caused numerous repairs, delays, and increased exposure that were not required to ensure adequate flood protection of the safety-related equipment in the RHRSW Vaults.

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 procedural changes to the flood barrier test methodology, the acceptance criteria, and the test frequency do not affect piping systems or other equipment in any way that could create a failure and initiate an internal flooding event. Therefore, the probability of the occurrence an internal flooding event described in UFSAR Section 3.4.2.1.2 is not increased.

As described in UFSAR Section 3.4.1.2.1, an internal flood of the Condensate Pump area or a RHRSW vault will not result in an event with radiological consequences. The basis for that conclusion was that following such an event, sufficient equipment (1 RHRSW pump and 1 DGCW pump) would remain available to safely shutdown the unit. The changes continue to ensure that the sufficient equipment will be available for use to safely shutdown the unit; therefore, the consequences an internal flood of these areas is unchanged.

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 flood protection seals are passive components. The basic test methodology (i.e. hook up air source, pressurize against barrier and check for leaks) has not changed. The test pressure has not changed. Recording the leakage rate does not physically affect the flood protection barrier. The flood barrier test frequency is unrelated to the types of malfunction, accidents or transients that have been evaluated. These changes clearly do not have the potential for creating a new or different type of accident or transients.

   Changing the test procedure to accept a small amount of cumulative leakage into a RHRSW vault will require that mitigating actions be taken to ensure the long term availability of the RHRSW and DGCW pumps in the vaults. At the maximum allowable leakage rate, the operability of the safety-related equipment inside the RHRSW vaults would not be affected for a minimum of 48 hours following a design basis internal flood of the condensate pump room area because the water level in the vaults would remain lower than the safety-related equipment needed to support unit shutdown. In 1972, a large break in the circulating water system flooded the condensate pump room. Following this event, the flood water was removed in approximately 24 hours. During a internal flood of a RHRSW vault, the non-flooded RHRSW vaults will remain accessible. At the maximum allowable leakage rate, the operability of the safety-related equipment inside the other RHRSW vaults would not be affected for a minimum of 8 hours following a design basis internal flood of a RHRSW vault area because the water level in the adjacent vaults would remain lower than the equipment. Eight hours provides ample time to mitigate the long term effect of this leakage based on high water level alarms in the vaults (i.e. early notification of problem) and operator actions can easily divert the leakage outside of the vault. Based on this discussion, changing the acceptance criteria will not create the possibility of an accident or malfunction 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 changes addressed by this Safety Evaluation are not associated with any Technical Specification requirements. The ability of the RHRSW pumps and the Diesel Generator Cooling Water pumps to perform their design functions is not affected by the changes.

Tracking No. SE-00-105
Activity No. UFSAR Change 99-R6-138

DESCRIPTION:

UFSAR Change # 99-R6-138, Implements changes to the UFSAR to address issues identified during DBI review.

a. Revise Section 6.2.1.3.5.2 , by deleting the words "...on a reload/cycle specific basis"

b. Revise Section 9.3.5.3 by replacing the words "enables faster shutdown for ATWS events" with "is required".

c. Revise Section 15.3.6, by deleting the words "..., and the idle loop is sufficiently well isolated".

d. Revise Section 15.5, by updating the text and figures with the available information from Quad Cities 2 Cycle 10 licensing analysis 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 activity only enhances information presented in the UFSAR by clarifying existing information and does not have a direct or indirect effect on any plant SSCs identified in any accident/transient analysis; therefore, the activity does not alter any accident or anticipated transient initial conditions. The changes have been previously analyzed and approved. These changes update the UFSAR to reflect the results of those analyses. Additionally, since the activity will not affect SSCs, it will have no effect on the operation, or the failure modes of the SSCs that could lead to an increase in the probability of occurrence of any accident/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 activity is to enhance the description of, or clarify information presented in Sections 6.2.1.3.5.2, 9.3.5.3, 15.3.6, and 15.5. of the UFSAR. Since SSCs are neither directly nor indirectly impacted by the change, the functions of plant systems will not be impacted by this change. The changes identified will not have any effect on specific SSCs or on the operation or failure of the SSCs, which could lead to any accident/transient. Therefore, there is no possibility that the activity could create an accident or transient of a different type than previously evaluated in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the UFSAR change does not affect any parameters upon which Technical Specifications are based. The sections that discuss the SBLC System and the Reactor Recirc System support these changes. Therefore, there is no reduction in the margin of safety.

Tracking No. SE-00-106
Activity No. ODCM Revision

DESCRIPTION:

The generic portions of this revision to the ODCM adopt the methodology listed in NUREG 0133, Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, and maintain portions of Regulatory Guide 1.109 to calculate the maximum offsite dose recipient. The reason for the generic changes is to change the dose calculation methodology to be consistent with the industry standard. Additionally, there are site specific changes that:

(1) Make the SJAE alarm setpoint consistent with the requirements of the MSL Radiation Monitor Technical Specification Amendment, Amendment No. 196 to DPR-29 & Amendment No. 192 to DPR-30.
(2) Improve accuracy of the hydrogen addition multiplication factor used to determine skyshine dose rates based upon H2 addition flow rate reduction resulting from Noble Metal Injection.
(3) Make the definition of the Process Control Program (PCP) exactly match the definition of PCP in the Technical Specifications.

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluents monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. Other than as described below, the changes being made to the ODCM are considered editorial or administrative in nature.

The ODCM is being revised to utilize the calculation methodology per NUREG 0133, which is the industry standard. Use of the new calculation methodology will not significantly change the resultant doses and dose rates determined. There are no instrument trip setpoints impacted, and the only alarm setpoint impacted is that of the SJAE Radiation Monitors, which is revised conservatively to '1.5 times normal full power background with hydrogen addition'. The hydrogen addition multiplication factor change results in more accurate determination of skyshine dose rates.

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 SAR accidents or transients that the affected activities and instruments associated with the ODCM could impact. No new equipment failure modes will be created by this ODCM revision. This change has no interface with plant equipment, other than the conservative alarm setpoint change for the SJAE Radiation Monitors. The new alarm setpoint is within the instrument's capabilities, and administrative controls have been put in place to minimize the probability of unwarranted spurious alarms due to changing plant conditions.

   This ODCM revision is in accordance with NUREG 0133 and Regulatory Guide 1.109. The new calculation methodology will not result in significant change to the resultant doses and dose rates determined. The hydrogen addition multiplication factor change results in more accurate determination of skyshine dose rates.

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 except for the SJAE Radiation Monitors and the RB Vent Radiation Monitors, the ODCM instruments provide monitoring functions only and can therefore, have no direct impact on the reactor coolant pressure boundary nor primary & secondary containment. This change does not affect any equipment functions, nor introduce any new failure modes. The RB Vent Radiation Monitors are unaffected by this ODCM revision.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Requirement of an a priori LLD for Ce-144 of $5x10^{-6}$ μ (Ci/ml does not reduce the margin of safety for offsite dose. This value applies to only this radionuclide. The liquid effluent concentration limit of 10 times the 10 CFR 20 (App B, Table 2, Column 2) concentration of $3x10^{-6}$ μ (Ci/ml, which is $3x10^{-5}$ μ, (Ci/ml is 6 times higher in concentration than the new LLD. In addition, NUREG 1302 (Table 4.11-1; footnote 3) provides the basis to raise this LLD value.

Tracking No. SE-01-001
Activity No. CRN 00-10

DESCRIPTION:

Revise Fire Protection Report Vol. I Chapter 5 (Guidelines of Appendix A to APCSB 9.5-1) to clarify the actual hydraulic flow requirements for fire suppression systems. At Quad Cities, 500 gpm allowance for inside and outside hose streams has been previously justified and used for sprinkler hydraulic design calculations. This change will revise the assumed allowances for flow (hydraulic) calculation for sprinkler systems. Additionally, the calculation will be referenced for the allowances on a case by case basis.

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 in the hydraulic calculation requirements does not alter the combustible loading or ignition sources in the plant; therefore, the probability of a fire has not been changed. The change does not affect any plant equipment so the probability and consequences of a malfunction of equipment has not been increased. The change does not impact the availability of any Safe Shutdown Equipment; therefore, it has no effect on the consequences of the accident.

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 other accidents or transients are required to be analyzed in combination with a fire. The change in the methods of fire fighting within the plant does not affect any other accident or transient. This change will not effect the systems function. Therefore, there are no new equipment failures being introduced and the impact of the equipment failing is not changed in any mode of operation.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change affects the method (allowance used) of calculating the supply of water available to fight a fire. Since, the pant's design does not credit fire suppression capabilities in the event of a fire, this change will not affect the plants ability to achieve and maintain safe shutdown. Therefore, the margin of safety is not affected.

Tracking No. SE-01-007
Activity No. ER-QC-0930-01

DESCRIPTION:

Procedure ER-QC-0930-01 controls the temporary installation, operation, and removal of a chemical injection rig developed for injecting activated aqueous sodium nitrate directly into the CRD Cooling Water header. This injection supports efforts to determine the source of Unit 2 unidentified drywell leakage. The activated sodium serves as a chemical tracer. Presence of the sodium in the Drywell Floor Drain Sump (DWFDS) following the injection is indicative of under vessel leakage from the CRD system.

The injection is accomplished using two pumps, a hydrostatic test pump to generate adequate flow and a chemical sample injection pump to precisely meter the sodium injection. CRD Cooling Water flow is lowered prior to the injection to ensure the total flow remains within the system's normal range during the injection.

The disassociated ions of the injected sodium nitrate are already present in the reactor vessel. The quantity of sodium nitrate to be injected is quite small (~ 20 ml or 200 uCi) and has an insignificant impact on reactor vessel chemistry control.

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 operation of the CRD system for manual rod control and automatic scram is unaffected. The injection rig is installed on a non-safety-related portion of the CRD system; furthermore, the procedure maintains all system parameters at their normal values. Since the sodium injection has no impact on the components required for the CRD system to provide its safety function, the probability or consequences of an accident or malfunction of equipment important to safety is 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 sodium injection constitutes an additional source of cooling water for the CRDM's. As such, the hydraulic-mechanical operation of the control rod drives is not altered in any manner while performing the injection procedure and therefore will not create the possibility for an accident or malfunction 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 no Technical Specifications are affected. The means by which the CRD system provides its safety function is not changed. The ability to move CRDMs, scram the reactor,

and operate the plant will not have changed. The procedure will not create any variations in system performance from normal operation.

Tracking No. SE-01-009 (Supercedes SE-00-064)
Activity No. UFSAR-99-R6-130

DESCRIPTION:

The following UFSAR sections were revised by removing stated requirements for locks and other devices which more closely reflects the requirements of IEEE-279 paragraph 4.18, titled "Access to Set Point Adjustments, Calibration, and Test Points."

Revises section 7.2.3.8 from "To gain access to the calibration and trip setting controls located outside the control room, a cover plate, access plug, or sealing device must be removed, " to say; "Administrative controls are used as the basis for assuring that access to Setpoint Adjustments, Calibrations, and Test Points are limited to qualified, plant personnel and that permission of Operations is obtained to gain access."

Revises sections 7.3.1.1.1.18, 7.3.1.2.1.18, 7.3.1.3.1.18, 7.3.1.4.1.18 and 7.3.2.6.18 from "Setpoint adjustments ... are integral with the sensors on the local instrument racks and cannot be changed without the use of tools to remove covers over these adjustments." to say; "Administrative controls are used as the basis for assuring that access to core spray Setpoint Adjustments, Calibrations, and Test Points are limited to qualified, plant personnel and that permission of Operations is obtained to gain access."

Finally, deletes from 7.3.1.3.1.18: "The only adjustable setpoints provided in the HPCI system are those provided on the flow controller on the main control room panel and are administratively controlled."

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 access controls to setpoint adjustments, calibrations, and test points provide no function during an accident nor are they accounted for in any SAR analysis. Physical barriers which require tools for removal, such as cover plates, access plugs, and sealing devices, are just one method that plant operational configuration is controlled. These barriers do not guarantee that setpoints have not been mis-adjusted, that calibrations have been performed correctly, or test points do not have jumpers installed around safety related contacts. Instead, a combination of administrative controls and physical access restrictions are in place that integrate to provide both direction to those personnel granted unescorted access to key areas of the facility and limit physical access to those areas to just the personnel accountable to meet that direction. These controls constitute key elements in the station's configuration management processes. Company employees and contractor personnel are required to meet configuration management requirements including basic elements such as procedure/work instruction adherence, operations authority to perform work activities in the plant, equipment/component operation restrictions, and verification processes and are so instructed during on-going training and/or initial indoctrination as applicable. Physical restrictions include security-controlled

doors throughout the facility and special limited-access locks on other key component locations such as the switchyard and relay house. Physical barriers, which require tools for removal, such as cover plates, access plugs, and sealing devices, themselves do not effect the consequences of any accident or transient. Only a setpoint adjusted in a non-conservative manner, or jumper installed around a safety related contact when it is required to be operable have any impact on the consequences of an accident or transient. Operational configuration is still maintained per plant procedures. Predefines still verify instrument channel function, calibration and logic function on intervals required by station technical specifications.

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 instrument and logic system failures have been evaluated by bounding accident analyses. No new accident or transient is created by changing the technique of configuration management.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because normal setpoint drift, reliability, accuracy, response time and single failure criteria are addressed in our existing margin of safety. These parameters are not impacted adversely by the existing plant configuration (instrument designs that do not have integral covers, etc.) and the proposed UFSAR changes described herein. Specific operational control configuration methodologies for access to setpoint adjustments, calibration, and test points are not considered in the existing design basis calculations. The proposed UFSAR changes does not adversely impact the plant's current margin of safety. Operational configuration of plant equipment is maintained by existing plant procedures and the verification of Technical Specification-required system surveillance requirements (e.g., instrument channel function, calibration and logic functions) are controlled by the predefine process at prescribed intervals.

Tracking No. QC-E-2001-005
Activity No. UFSAR-99-R6-173

DESCRIPTION:

Note that this 50.59 Review was completed after the 50.59 Rule change effective in March 2001. As such, the format of this summary reflects the format of the new Rule and Exelon implementing procedures.

This activity adds discussion to the UFSAR regarding an analysis performed in 1981 to verify the adequacy of the suppression chamber to drywell vacuum breakers. This activity does not involve any physical change to the plant and does not impact the Technical Specification requirements for the system. This activity also includes the correction of two typographical errors on affected UFSAR page 6.2-8 that omitted the word "on" from a sentence and another that listed an area in terms of ft^3 instead of ft^2.

The Quad Cities Improved Technical Specifications submittal included a discussion of the results of this analysis in the bases section for Specification 3.6.1.8 "Suppression Chamber to Drywell Vacuum Breakers" and identified this analysis as an applicable safety analysis that supports the requirements of the Technical Specification. Therefore, discussion of the results of this analysis in

the Containment Section of the UFSAR is warranted. This UFSAR revision is required in support of the ITS Submittal.

This activity does not involve any physical change to the plant and does not impact the Technical Specification requirements for the system. The analysis verifies that seven or more operable suppression chamber to drywell vacuum breakers are required to ensure that the design differential pressure across the containment vent header is not exceeded for the limiting scenario. Technical Specifications currently require a minimum of nine operable vacuum breakers. The correction of the typographical errors included in this activity has no effect on the technical content of the UFSAR or on the facility itself.

This analysis involves no physical change to the plant or to the method of operation of any system. No failure modes or malfunctions are introduced, impacted, or identified by this activity. Technical Specifications require nine of the vacuum breakers to be operable, while the analysis verifies that a minimum of seven vacuum breakers are required to perform the required design function. Therefore, the analysis supports the conclusion that the failure of one of the operable vacuum breakers will not cause the containment pressure to exceed the design limit of 2 psid during even the most limiting scenario. The basis of the original sizing of the vacuum breakers is described in the UFSAR as the Bodega pressure suppression system tests. However, this activity does not affect this statement, but documents a more conservative supplementary analysis performed to independently verify the adequacy of the vacuum relief system. Since this activity falls within the provisions of 10 CFR 50.59, a license amendment is not required to implement this UFSAR change.

Tracking No. QC-E-2001-006
Activity No. UFSAR-99-R6-181

DESCRIPTION:

Note that this 50.59 Review was completed after the 50.59 Rule change effective in March 2001. As such, the format of this summary reflects the format of the new Rule and Exelon implementing procedures.

This activity adds discussion to the UFSAR regarding safety analyses performed to determine the radiological effects of a postulated failure of a component in the Off-Gas system. This potential failure had been previously described in the Technical Specification bases for the maximum allowed condenser off-gas activity. This change does not affect maximum Off-Gas activity allowed, which is the same per Technical Specifications 3/4.8.I and ITS Section 3.7.6. However, during the preparation of the ITS submittal, it was determined that the previous analysis (Special Report No. 1), which was submitted to and accepted by the AEC, was based on an assumed Off-Gas activity of 100,000 μCi/s after 30 minute decay, which was described in the report as the standard conditions while the maximum Off-Gas activity allowed by plant Technical Specifications is 251,100 μCi/s (100 μCi/s/MWt) after 30 minute decay. Therefore, an assessment was performed to determine the effects of a bounding activity of 350,000 μCi/s after 30-minute decay on the calculated dose rates from Special Report No. 1. No other assumptions in the analysis are changed, only the assumed Off-Gas activity.

This activity does not involve any physical change to the plant or the manner in which any system or component is operated or controlled. The radiological effects of this postulated accident had been previously evaluated in Special Report No. 1, which was submitted to and accepted by the AEC. As part of ITS, an assessment was performed against this report to determine the effects of greater Off-Gas activity allowed by Technical Specifications than what was previously assumed in the special report. As a result, there was a minimal increase in the calculated dose rates. The end result of this activity is that the UFSAR will be updated to describe the applicable safety analyses for a postulated failure of a component in the Off-Gas system.

As a result of the increased Off-Gas activity in this assessment, for even the most bounding postulated malfunction, the resulting increase in calculated exposure is minimal: less than 0.6% of the 10 CFR 100 limit and less than the SRP (BTP ETSB 11-5) acceptance criteria of 500 mR. This activity does not change the Technical Specifications limit for the system activity, and the Operating License is not impacted. Therefore, a license amendment is not required to implement this UFSAR change. Since this activity falls within the provisions of 10 CFR 50.59, a license amendment is not required to implement this UFSAR change.

Tracking No. QC-E-2001-007
Activity No. UFSAR-99-R6-183

DESCRIPTION:

Note that this 50.59 Review was completed after the 50.59 Rule change effective in March 2001. As such, the format of this summary reflects the format of the new Rule and Exelon implementing procedures.

This activity adds discussion to the UFSAR regarding a radiological reassessment performed to evaluate the effects of a fuel handling accident over a spent fuel storage pool. The current assessment described in UFSAR Section 15.7.2 is based on an assumed fuel assembly drop over the reactor core during refueling operations, which would result in the fuel assembly dropping further and more extensive fuel assembly damage. However, if a refueling accident were to occur in the spent fuel pool, any damaged fuel assemblies would have less water level above the damaged fuel rods than an assembly dropped on the core. Therefore, a new radiological analysis was performed, which conservatively assumed the same amount of fuel damage postulated for a fuel assembly drop on top of the core and used the minimum water level in the spent fuel pool required by plant Technical Specifications.

The accident analysis for a fuel assembly drop as described in UFSAR Section 15.7.2 is based on a fuel assembly drop over the reactor core during refueling. However, in support of the Quad Cities ITS submittal, a radiological reassessment was performed to support the minimum water level allowed in the spent fuel pool. This assessment modeled a fuel assembly drop in the spent fuel pool and conservatively assumed that the fuel damage would be the same as for a drop on top of the core, previously evaluated. The NRC approval of the minimum spent fuel pool level in the ITS submittal was based on the fact that the spent fuel pool level required by ITS was 9 inches greater than the level required by previous Technical Specifications. However, the radiological assessment was included in the ITS bases as an applicable safety analysis. Therefore, the UFSAR will be updated to include a discussion of the results of the radiological assessment with a comparison to the applicable regulatory limits.

This activity does not involve any physical change to the plant or the manner in which any system or component is operated or controlled. Even with conservative assumptions regarding the amount of fuel damaged during a bundle drop in the spent fuel pool, this activity results in only a minimal increase in the calculated off-site dose rates, which remain well within the limits of 10 CFR 100 and SRP 15.7.4. Although the control room dose for a refueling accident had not been previously evaluated, this analysis included the evaluation of control room dose, which remains well within the limits of SRP 6.4.

No failure modes or malfunctions are introduced, impacted, or identified by this activity. Because of the lower water level above the fuel in the spent fuel pool as compared to the reactor core, the assessment resulted in a minimal increase in the calculated off-site exposure: less than 3.5% of the 10 CFR 100 limit and less than the SRP 15.7.4 acceptance criteria. The effects of this postulated event on Control Room dose were also evaluated and found to meet the applicable acceptance criteria of SRP 6.4 and is bounded by previous control room dose analyses. This activity does not change the Technical Specifications limit for the water level, which was already increased by 9 inches as part of the ITS submittal, and the Operating License is not impacted. Therefore, a license amendment is not required to implement this UFSAR change. Since this activity falls within the provisions of 10 CFR 50.59, a license amendment is not required to implement this UFSAR change.

Tracking No. SS-H-99-0212
Activity No. DCP 9700011; SE-99-076

DESCRIPTION:

The design is to upgrade the Feedwater Regulating Valve (FRV) Controllers LC 1-0640-18 (Reactor Level Master Controller) and FC 1-0640-20 (Low Flow FRV Controller). The existing GE 50,000 Ω current limiting variable resistance potentiometers in the automatic output circuits of both controllers will be replaced with Bourns Inc. 50,000 Ω potentiometers. As for the master controller LC 1-0640-18, two new 200 Ω variable resistor potentiometers, provided by Bourns Inc., will be added to the manual output circuit as high and low current limiters for the manual mode of operation. High and low limit pots should be operating in mid-range after final adjustments have been made.

SAFETY EVALUATOIN 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 1-0640-18. There are two possible failure states for the new current limiting potentiometers added to the manual output of controller LC 1-0640-18; either fail open or short. If they fail open, this is the equivalent of a loss of output of the feedwater controller, and the FRVs fail "as is". A failure of the FRVs "as is" is bounded by the increase in feedwater flow (Section 15.1.2) and loss of normal feedwater flow (Section 15.2.7) transient analyses. A short across the potentiometers is the equivalent of the manual 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, or the consequence 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 1-0640-18. The high and low current limiting potentiometers added to the manual output circuit of controller LC 1-0640-18 could either fail open or short. If they fail open, this is the equivalent of a loss of output of the feedwater controller, and the FRVs fail "as is". A failure of the FRVs "as is" is bounded by the increase in feedwater flow (Section 15.1.2) and loss of normal feedwater flow (Section 15.2.7) transient analyses. A short across the potentiometers is the equivalent of the manual controller circuit as it is ow, 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 feedwater control system is not required to maintain a margin of safety described in the Technical Specification. Therefore, there an be no effect on the margin of safety.

Tracking No. SS-H-99-0227
Activity No. DCP 9900060; UFSAR-99-R6-0227; SE-99-070

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 1. The SSMP tie-in point is currently between HPCI valves 1-2301-7 and 1-2301-8. Valve 1-2301-7 is a check valve and valve 1-2301-8 is motor operated. The revised location will be down stream of valve 1-2301-7 and before the HPCI injection piping connection to the feedwater 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 relocating the SSMP tie-in point from the upstream side of check valve 1-2301-7 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 1-2301-7 will remove the burden on the operations staff to close (or verify closed) valve 1-2301-8 or 1-2301-9 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 1 SSMP tie-in to the HPCI system, plant personnel will no longer be required to close (or verify closed) valve no. 1-2301-8 or 1-2301-9 prior to using the SSMP system to mitigate consequences of certain Appendix R fire scenarios. Therefore, margin of safety is not reduced.

Tracking No. SS-H-99-0235
Activity No. DCP 9700261; SE-99-074; UFSAR-99-R6-055

DESCRIPTION:

Installation of Phase 1 of the OPRM subsystem into the Power Range Neutron Monitoring System for Unit 1. The OPRM system, when it is fully operational, is designed to provide an alarm and to initiate a scram to prevent the fuel from exceeding the MCPR safety limit upon detection of thermal hydraulic instabilities. 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 ability to initiate a scram will not be enabled until Phase 2 of the installation (DCN 001602I) which is scheduled to be completed during Q1R17.

The Phase 1 installation (DCN 001555I) will include installation of the ABB OPRM modules and supporting equipment such as voltage regulators, bulk power supplies, analog and digital isolators, OPRM relay boards, and the annunciator/SER alarms during Q1R16. The OPRM system performance will be monitored and fine-tuned during the operating cycle following installation of Phase 1. During this time, the OPRM ability to recognize and respond to thermal hydraulic instability conditions will be evaluated. No licensing credit is taken for the OPRM system during this period, and no changes to the plant TS are necessary while the OPRM is in this mode of operation.

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:

Remove two existing voltage regulators in each APRM page (total 12 voltage regulators) of Panel 901-37. Replace six of these voltage regulators, one in each APRM page of panel 901-37, with new dual voltage regulators (1-0756-VR-1A through 6A). Install eight OPRM signal processing modules (1-0756-0PRM 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.

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 901-37.

Replace eight existing power supplies powering RBM, APRM, and LPRM Group Pages with new bulk power supplies (1-0756-PS 11 through 14 and 1-0756-PS 17 through 20) in Panel 901-37.

Install four new analog signal isolators (1-0756-AI-3 & 4 and 1-0756-AI-3T & 4T) in Panel 901-37. (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.)

Install new instrumentation and control cables inside Panel 901-37 and new control cables between 901-34 and 901-37. 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.

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 901-37 with 20A Gould type ATM fuses. Add RBM 7 and RBM 8 isolation fuses at panel 901-37.

Remove two Flow Converter/RMCS interposing relays (1-0756-101A & 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 Flux-High 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 Flux-High 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 pressure 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 Flux-High 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. SS-H-00-0059
Activity No. - QCOS 2900-04, Rev. 11; QCOP 2900-01, Rev. 15; QOM 1-6800-01, Rev. 5, QOA 6800-03, Rev. 20; SE-99-104

DESCRIPTION:

These procedures are being revised because of the installation of DCP #9900067. This design change has installed an alternate power supply for the flow-indicating controller (FIC) for the Safe Shutdown Makeup Pump (SSMP). This installation allows the operator to select either power supply as needed for the motor operated flow indicating controller valve.

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 no new failures have been introduced by the installation of this alternate power supply. The additional components have been evaluated and have been found acceptable. This installation does not change the function or operational characteristics of the affected 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 normal state of the modified circuit is the same as before. This modification adds the ability to change the power supply to the FIC. There is no new possibility created by a failure of both power supplies due to the installation of the power supply transfer switch. There are no new failure modes or changes to existing failures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this system will be more reliable because of the additional power supply for the FIC. The margin of safety as described in the Technical Specifications is not reduced.

Tracking No. SS-H-00-0067
Activity No. DCP 9900213; UFSAR-99-R6-105; SE-99-106

DESCRIPTION:

The modification revises the Reactor Protection System (RPS) Turbine Generator Load Rejection (40% Mismatch) SCRAM signal logic. This logic consists an Electrohydraulic Control (EHC) system fluid reservoir low-pressure switch in series with a Turbine Control Valve Fast Closure pressure switch. This modification will revise the RPS logic to remove the four (4) turbine EHC system fluid reservoir low pressure scram switches, PS 1-5650-1, 2, 3 and 4, including, instrument service lines back to the process header.

This modification is being performed in the Unit 1 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.

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 1-5650-1, 2, 3 and 4 will be initiated by the Turbine Control Valve Fast Closure switches (PS 1-5641-122, 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 1-5641-122, 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 1-5650-1, 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.

   Appropriate Technical Specification changes, as identified in the Safety Evaluation, have been submitted and approved by the Nuclear Regulatory Commission in accordance with 10CFR50.90 as Amendment Nos. 193/188.

Tracking No. SS-H-00-0071
Activity No. DCP 9700104, Addendum 1; SE-99-008; S-H-99-0188

DESCRIPTION:

Install flex hoses on the Unit 1 EHC FASTC and FCD lines; install isolation valves on the FASTC and FCD lines and install connecting line and support for accumulator (Accumulator is installed by separate DCP). Change method of retaining clevis pins on CV linkage. These changes are being made to reduce vibration being transmitted from the Main Turbine Control valve to the EHC tubing,

provide improved isolation capability for the EHC FASTC and FCD lines and eliminate vibration related failures.

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 installation meets or exceeds the design requirements (USAS B31.1-1967). The flexible hoses will reduce vibration being transmitted from the Main Turbine Control Valve (TCV) to the EHC tubing, the isolation valves allow isolation of EHC leaks at the TCV and the retainers ensure retention of the pins under vibrating conditions. The consequences of an EHC leak have not changed and the probability of a leak in the EHC tubing at the TCV have been reduced.

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 is a change to the piping configuration to reduce vibration. The new method of retaining the pins considers vibrations. An EHC leak in the flex hose, welded or flared fittings is bounded by an EHC leak in the original piping. If the manual valves are required to be isolated (via OOS), the valves do not prevent safety features of the reactor protection system from operating and may reduce the probability of a turbine trip while maintaining the reactor in a safe condition within Technical Specification Limits. Operation of the Unit with a half-scram does not create the possibility of a different type of malfunction because the Main Control Valve will also be closed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Pressure Switches on the CV provide the scram signal to RPS system. Installation of this modification does not affect these pressure switches or isolate them from the CV actuator. All other changes are bounded by the design requirements. Therefore, the margin of safety is not reduced by this change.

Tracking No. SS-H-00-0072
Activity No. DCP 9900185; UFSAR-99-R6-108; SE-99-105

DESCRIPTION:

This Modification of the Unit 1 Reactor Protection System (RPS) logic, Primary Containment Isolation System PCIS) logic, and Off-Gas System logic will remove inputs from the MSL Radiation Monitors' Trip logic. The MSL Radiation Monitors will no longer provide scram, MSIV closure, or Off-Gas System trip functions. This change will also eliminate the high-radiation trip, which initiates the closure of the off-gas chimney isolation valves and the off-gas drain valves. This Modification will also re-identify and rewire four computer points associated with the scram trip functions of the Main Steam Line Radiation Monitors and eliminate the first-out feature of the two annunciator points. The Condenser Vacuum pump RPS trip initiated by the MSL Radiation Monitors and the Monitors' alarm functions will be maintained.

When a high radiation setpoint of the MSLRM or Offgas radiation monitors or both is reached, alarms will annunciate in the control room alerting the operators. Operator actions will then be required to monitor radioactivity in the main steam lines and offgas system.

This Modification is a result of a Boiling Water Reactor Owners Group (BWR) topical report, initiated to minimize inadvertent scrams and Main Steam Isolation Valve closures erroneous radiation monitor actuation. Further details addressed in General electric Report NEDO-31400, dated October 1992,

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the consequences of removing the Main Steam Line Radiation Monitor Scram and MSIV Isolation Functions have been evaluated and deemed within 10CFR100 limits. General Electric Topical Report NEDO-31400 evaluated the consequences of eliminating the MSLRM scram and isolation function by performing two radiological assessments: a CRDA with and without automatic main steam isolation. The NEDO-31400 evaluation demonstrated that removing the scram and main steam line isolation functions of the MSLRM in conjunction with proper use of an augmented off gas system results in acceptable dose consequences following a CRDA (within 25% if the 10CFR100 limits as provided in the SRP Section 15.4.9).

   The general conclusions of the Boiling Water Reactor Owners Group (BWROG) study is considered valid for Quad Cities Nuclear Power Station, although the supporting radiological analysis is not directly applicable due to a site-specific release path not accounted for in the study. Therefore, ComEd has completed a site-specific radiological evaluation (Calculation QDC-M-0550) to account for the additional release path from the turbine gland seal exhauster. The analysis was performed using the approach outlined in SRP Section 15.4.9. The total doses calculated are within 25% of the guideline values in the 10CFR part 100.11, or 75 REM for the thyroid and 6 REM for whole-body doses (consistent with the guideline in SRP 15.4.9). Therefore, operation of Quad Cities Unit 2 under the proposed amendment to Facility Operating License No. DPR-29 and DPR-30 does not increase the consequences of a malfunction of equipment important to safety than previously identified.

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 are no new failure modes associated with the elimination of the Main Steam Line High Radiation Monitor input for Scram and MSIV isolation Functions. Main Steam Line Radiation Monitors are still continuously monitoring the radiation from the main steam lines and will annunciate in the control room upon receipt of high radiation signal. When a high radiation setpoint of the MSLRM or Offgas radiation monitors or both is reached, alarms will annunciate in the control room alerting the operators. Procedures will be in place for operations to promptly sample the reactor coolant to determine possible sources of the contamination as well to determine the need for further corrective action that may be required to limit both occupational doses and environmental releases.

   The Removal of the Main Steam Line Radiation Monitor Scram and MSIV Isolation Functions will not increase the consequences of a malfunction of equipment important to safety different than those previously evaluated in the SAR. The proposed change will not result in the failure of any affected (SSC) Safe Shutdown Component. The changes have been evaluated and will not result in the degradation or failure of any SSC. All modified

components have been analyzed and will be tested following installation, as indicated in the testing requirements to insure that they will continue to function exactly as before.

Therefore, the possibility is not created 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 eliminating the Main Steam Line Isolation Function and the Reactor Scram Function of the Main Steam Line Radiation Monitors will not change any plant operation parameters, nor any protective system set points other than removal of these functions. The Main Steam Line Radiation Monitors High alarm setpoint is being revised to 1.5 times normal background with hydrogen addition. Th MSL high high rad alarm and trip setpoint is being maintained at 15 times normal background without hydrogen addition, for alarm and mechanical vacuum pump RPS trip respectively. The offgas radiation monitor alarm set point is also being revised to 1.5 times normal background with hydrogen addition, per licensee agreement.

   In accordance with 10CFFR50.90 ComEd has requested a change to the Technical Specifications (TS) of the Facility Operating Licenses DPR-29, and DPR-30, for the Quad Cities Nuclear Power Station, Units 1 and 2 respectively. This amendment request removes the MSLRM scram and main steam line isolation functions and adds a new specification TS 3/4.2.I for the MSLRM Mechanical Vacuum Pump trip, which is not being eliminated.

   As part of the implementation of the Technical Specification Amendment change, ComEd has committed to review the Quad Cities Nuclear Power Station Operations Annunciator and General Abnormal Conditions Procedures. Review and revise them as required to insure proper operator action to limit occupational doses and environmental releases prior to implementation of this modification.

   The Boiling Water Reactor Owners Group (BWROG) Safety Analysis Report had demonstrated that the consequences of the CRDA, without the MSLRM High Scram, and MSL Valve closure signal, are that the does are well within the guidelines of 10CFR part 100 limits.

   Therefore, this modification does not reduce the margin of safety as described in the basis for any Technical Specification.

Tracking No. SS-H-00-0074
Activity No. DCP 9300156; SE-96-122

DESCRIPTION:

The activity is to install Design Change Package (DCP) 9300156. This modification will install new instrument sensing lines and new gauges PI 1-6641-20, PI 1-5241-25A, PI 1-5241-25B and PI 1-6641-8210 to the Unit 1 Emergency Diesel Generator (EDG). Each gauge will be installed on an instrument line with an isolation valve and test tap. The new gauges will provide more detailed system monitoring and aid in diagnosing Maintenance Functional Failures (MPFFs). The isolation

valve and test tap will facilitate the reading, testing and calibration of the new gauge. This modification has been previously installed on the Unit 2 and the 1/2 EDGs.

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 affect the barriers used to prevent the release of Off Site dose. These instruments do not affect the operation of the EDG during any accident. Addition of the new instruments and sensing lines are designed and will be seismically installed and tested to maintain the Safety-Related pressure boundary. The consequences of a failure of the new instruments or sensing lines to maintain their pressure boundary are the same as the consequences of a failure for the existing instruments.

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 indicating function of the new gauges is a Non-Safety-Related function. The Safety-Related function of the new instruments and sensing lines is to maintain the pressure boundary of the system. The failure modes for the new instruments and sensing lines are no different than the failure modes for the existing instruments and sensing lines on the EDG system.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification requirements affected by the activity.

Tracking No. SS-H-00-0104
Activity No. WR 990158747; WR 990158749; WR 990146943; SE-98-144

DESCRIPTION:

To repair the 101 valves, CRD Hydraulic Control Unit (HCU) insert valves and the 102 valve, CRD Hydraulic Control Unit (HCU) withdrawl valve, a freeze seal must be installed as an out of service boundary. The freeze seal will be installed per approved procedures, QCMM 1500-32 "Establishing Freeze Seals Using Jackets", SMP-M-107" Pipe Freeze Seals" and CC-AA-403 "Maintenance Specification: Selection and Control of Freeze Seal Location". Work Request 990158747 is to repair the 1-0305-101-54-27 valve, work request 990158749 is to repair the 1-0305-101-34-19 valve and work request 990146943 is to repair the 1-0305-102-22-07 valve. Similar work was evaluated by SE-98-144.

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 with the existing procedures the probability of a freeze seal failure is small. The procedures are in compliance with EPRI guidelines for freeze seals and NRC information notice 91-41, "Potential Problems with the Use of Freeze Seals". This work is limited to mode 4, 5 or no mode. There will be a contingency plan in place in the unlikely event that the freeze seal fails. This contingency plan includes methods of stopping and

containing the leakage, limitations on CRD movements, and requirements for monitoring the freeze seal.

If the freeze seal were to fail the event is bounded by the instrument line break outside of containment, therefore, the probability of an accident is not increased. The anticipated leakage rate is small and well within the capability of the systems normally used to control reactor vessel level, so the consequences of the accident are not increased and are bounded by an instrument line break outside of 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 if the freeze seal were to fail the event is bounded by the instrument line break outside of containment. No new failure modes are introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification 3/4.10G, Water Level-Reactor Vessel-Refueling Operations and 3/4.10H, Water Lever-Spent Fuel Storage Pool-Refueling were evaluated and it was determined that they were not affected by this work.

Tracking No. SS-H-00-0124
Activity No. QCTS 0730-01; SE-99-070

DESCRIPTION:

This activity changes procedure QCTS 0730-01: Reactor Feedwater Check Valve 1(2)-220-59A/B Leak Test, Revision 4 to reflect the relocation of the SSMP tie-in point to downstream of check valve 1-2301-7 in the HPCI injection piping (DCP 9900060).

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 1-2310-7 to the downstream side will not adversely alter any starting (precursor) conditions required for an Appendix R fire. This change does not add any combustible materials to the plant. The SSMP system is used in certain Appendix R fire scenarios. Relocating the SSMP tie-in location to the downstream side of check valve 1-2301-7 will remove the burden on the operations staff to close (or verify closed) valve 1-2301-8 or 1-2301-9 prior to using the SSMP system. There are no consequences to an Appendix R fire with the current configuration and the actions of the operations staff will be more effective with the new configuration.

A LOCA inside containment is also not affected. The HPCI system mitigates the accident by injecting water into the vessel via the feedwater piping located outside containment. The HPCI system is used to mitigate the consequences of a LOCA inside containment. Addition of a "tee" has a negligible effect on the hydraulic performance of HPCI and does not affect the system pressure boundary. The HPCI system will continue to perform as designed and this change will not impact the consequences associated with a LOCA inside containment.

This activity does not affect the operation or system description of the Feedwater 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 piping reroute is minor in that the new tie-in location is a few feet from the existing location. The SSMP, HPCI and Feedwater systems will perform the same functions with the same flows, pressures and temperatures as before. The valves used to isolate flows between systems are designed for these conditions and 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 of the HPCI system will not adversely impact the system's hydraulic characteristics. The added head loss is negligible compared to the overall system characteristics and the margin of safety is not reduced.

   The minor piping reroute has also been evaluated for its effect on SSMP system hydraulics. The SSMP system can continue to deliver at least 400 gpm against a reactor vessel pressure of 1120 psig and the margin of safety is not reduced.

Tracking No. SS-H-00-0127
Activity No. DCP 9900506; UFSAR-99-R6-104; SE-00-023

DESCRIPTION:

DCP 9900311 issued safety evaluation SE-00-023 to evaluate a modification to MO 2-2301-8. This modification will also be performed for MO 1-2301-8 under DCP 9900506. Since DCP 9900506 performs the same changes that are contained in DCP 9900311, DCP 9900506 is enveloped by safety evaluation SE-00-23. Changes to QCEM 0600-12, Rev 11, are also necessary to implement DCP 9900506. The 9-9C contact of the MO 1-2301-8 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.

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 by SE-00-023. The inputs and assumptions used for SE-00-023 are valid for this modification also, with the only difference being that this modification is for Unit 1. 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 the changes made to the procedure and equipment are enveloped by the previously performed safety evaluation. The revision to the procedure to incorporate changes and the equipment modification made by DCP 9900506 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 equipment and procedure changes are enveloped by SE-00-023 for DCP 9900506. Safety Evaluation SE-00-023 determined that no changes to the Technical Specifications are required.

Tracking No. SS-H-00-0129
Activity No. DCP 9900169 (Rev. 2); SE-99-082

DESCRIPTION:

This modification supports the Quad Cities safe shutdown analysis (SSA) optimization program.

Cables 14216 and 14217 provided the normal feed to Unit 1 125Vdc Bus 1B-1 from Unit 2 Bus 2A. The modification will perform the following: The existing feed will be disconnected at Unit 2, 125 Vdc Bus 2A in the 2A Battery Charger Room, cut back and left abandoned-in-place. New cables will be connected to Bus 2A and routed in tray and a new dedicated conduit. The conduit will be wrapped with Fire Protective Wrap in designated areas. A new splice box will be installed in Fire Area TB-III. Existing cables 14216 and 14217 will be rerouted to the new splice box and the existing cable will be spliced with the newly routed cables. To facilitate the reroute, penetrations will be opened in five existing fire barriers. These penetrations will be fire sealed to the rating of the barrier using station-approved methods and details following installation. As the new cables will be spliced to the reused portion of the existing cables, the new cables will be assigned the same numbers as the existing cables. To allow for tracking of the abandoned cables, the abandoned-in-place cables will be identified as cables 13993 and 13994.

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 supports the SSA optimization program. The effect of this modification will not alter the existing plant 125 Vdc system operating breaker configuration, any protective devices or the SSA. The power cables being rerouted and/or protected are currently not credited for post fire operation in Fire Zones TB-I and TB-II. A portion of the relocated cable will be covered with fire barrier material. Vendor data provided indicates that the Darmatt KM-1 fire barrier material used is designed to meet the requirements of NRC Generic Letter 86-10, Supplement 1. Derating factors for the protected cables has been provided to SLICE to properly evaluate the current load and any future load additions to these cables. The additional weight of the fire wrap has been accounted for in the design of the new conduit and conduit supports. Cables that are abandoned-in-place by this design change are identified with new cable numbers to allow the abandoned cables to be tracked by SLICE. The abandoned-in-place cables cannot increase the probability of occurrence of an accident, as they are de-energized. The impact of this modification on all affected SSCs has been determined and evaluated and is acceptable. Therefore, the change cannot increase the probability of an accident.

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 cable loading with its reduced ampacity is still within the design requirements of the existing loads. The revised ampacity is provided to SLICE to properly evaluate future load additions to these cables.

The additional weight of the fire wrap has been accounted for in the design of the new conduit and conduit supports. All aspects of the installation have been evaluated and the installation will not adversely impact any structures, systems or components (SSCs). Therefore, there can be no 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 bases to Technical Specifications 3/4.9.E & F provide no specific limits but rather discuss the availability of ac and dc busses which provide power to ECCS components. Following installation, the modification will not affect the reliability, availability or operability of any ECCS component, because the function of Cables 14216 and 14217 is unchanged. During installation, loss of the normal power feed to Bus 1B-1 will occur as a result of the cable determination, re-routing, splicing and re-termination of Cables 14216 and 14217. The appropriate actions as described in Technical Specification Sections 3/4.9.E & F will be implemented when the normal feed to Unit 1 Bus 1B-1 is out of service. Following the installation of the change all SSCs will perform their designed safety functions, meet all of their design requirements and no margins of safety will be reduced with respect to the affected systems.

Tracking No. SS-H-00-0130
Activity No. DCP 9900119, Rev. 2; SE-00-019

DESCRIPTION:

This DCP revision moves the scope of work associated with replacing the Moisture Separator (MS) internals per DCN 001909M from DCP 9900119 to DCP 9900590, which will be completed during a future outage. Because of dose and contamination problems discovered when the Low Pressure Heater Bay was entered during Refuel Outage Q1R16, the replacement of the MS internals has been deferred. The replacement of these internals was considered a system improvement, which would improve the moisture removal efficiency of the MS and increase the electrical output of Unit 1. There are no regulatory commitments to replace these internals. Therefore, operation with the current internals is acceptable, and the installation of the MS Internals may be deferred to DCP 9900590. The Heater Drain valve modifications (per DCN 001910M) will continue to be installed.

The Safety Evaluation was also used for DCP 9900590 and UFSAR-99-R6-065, which were 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 functions of the moisture separators and feedwater heater drain system, heaters, and valves are not being changed by this modification. The system and its components will function as required during accident or transient conditions because component failure modes are unchanged. Therefore, there is no increase in the probability or consequences of an accident or 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 the existing plant equipment to function as required during all operational or accident modes. The moisture separators and heater drain system, and valves will continue to perform their intended functions. There will be no effect on equipment failures or malfunctions as a result of this modification. Therefore, the possibility of a different 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 not Technical Specifications relevant to or affected by this modification.

Tracking No. SS-H-00-132
Activity No. NWR 990141931; SE-98-087

DESCRIPTION:

The activity installs RPS Test Boxes to facilitate the removing of temporary jumpers installed in the neutron monitoring trip logic. The jumpers are installed to allow installation of the Unit 1 OPRM modification (DCP 9700261) without a full scram being maintained. The RPS Test Box will be installed on the Neutron Monitoring (NM) System RPS Trip relay 590-107A thru H to prevent receiving a 1/2 scram during jumper removal and subsequent operability test of the affected NM trip channel. Installation and removal of the temporary jumpers in the 901-37 Neutron Monitoring trip logic was previously evaluated by SS-H-00-098.

The tasks will functionally check the restored APRM channel scram function following jumper removal. Each task will employ administrative controls normally used for surveillance testing, as the 590-107A thru H relays is required trip relays in modes 4 and 5 for the IRM function.

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 test boxes are constructed of passive devices as described in question 6 of the safety evaluation. There are no power supplies or other energy sources in the test box to create an over current or over voltage condition in the RPS trip logic. The failure modes of the test boxes have been reviewed and documented in the original safety evaluation. The test boxes are installed during testing activities where the equipment under test is considered inoperable and will be removed within the allowed out of service time (AOT) required by the Technical Specifications. Because there will be no change in the AOT, the probability of an equipment malfunction remains unchanged from that already analyzed. All other inputs to the RPS subchannel trip logic remain available during the time the test box is installed.

   This activity will require the total time that the associated Scram Logic Subchannel is bypassed during performance of the referenced procedures to include time required to remove the jumpers. The added time for removal of jumpers is minutes and will be done during the allowed outage time for this trip function. This is not considered an increase in the probability of a malfunction in the Reactor Protection System, because the total time

that the associated Scram Logic Subchannel may be bypassed is still bounded by Technical Specification Table 3.1.A-1, Note (a).

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 test boxes are constructed of passive devices. There are no power supplies or other energy sources in the test box to create an over current or over voltage condition in the RPS trip logic. The failure modes of the test boxes have been evaluated previously. The test boxes are installed during testing activities where the equipment under test is considered inoperable and will be removed within the allowed out of service time (AOT) required by the Technical Specifications because there will be no change in the AOT, the probability of an equipment malfunction remains unchanged from that already analyzed. All other inputs to the RPS subchannel trip logic remain available during the time the test box is installed.

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