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AMS-94-017

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

SUBJECT: Quad Cities Nuclear Station Units 1 and 2
Changes, Tests, and Experiments Completed
NRC Docket Nos. 50-254 and 50-265

Enclosed please find a listing of those facility and procedure changes, tests, and experiments requiring safety evaluations completed during the month of May, 1994, for Quad-Cities Station Units 1 and 2, DPR-29 and DPR-30. A summary of the safety evaluations are being reported in compliance with 10CFR50.59 and 10CFR50.71(e).

Respectfully,

ComEd
Quad-Cities Nuclear Power Station

AMScott
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System Engineering Supervisor

AMS/dak

Enclosure

cc: J. Martin, Regional Administrator
C. Miller, Senior Resident Inspector

SAFETY\NRC.LTR

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SE-94-37
Software Activity Request #991

DESCRIPTION:

The Rod Worth Minimizer (RWM) software was upgraded to include two new features. First, a select block was added to the previous rod blocks. This causes a rod block upon selection of an out-of sequence control rod. This select block function included an on/off 'toggle' on the RWM touch screen. The other change is the provision of an insert block signal in Rod Exercise mode as soon as a control rod is moved in one notch. In addition, a rod block was applied to all other rods at this time. The blocks are removed when the current rod is withdrawn to its original position. Previously the RWM applied a rod block only after a rod had traveled more than one notch past its target position.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Rod Drop Accident UFSAR SECTION 15.4.10

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the changes that will be made are to the RWM software only. No modifications will be made to the RWM computers, nor will any interactions with other systems be changed. The blocks that are being added will serve as an additional barrier to control rod mispositionings and all of the current rod blocks will be retained. Because no other systems will be affected, there will be no adverse system interactions or accidents created in other systems.

In addition, because the RWM itself will not be altered, the failure modes will remain the same as before the software upgrade. As a result, there will be no new type of RWM malfunction not evaluated in the UFSAR.

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

SE-94-38
QCOS 2300-1 Rev. 6

DESCRIPTION:

This procedure was revised to have the operator verify lube oil temperature indicating switch setpoints are set to their proper value as listed in the procedure. Also, the HPCI pump is verified to be filled and vented locally prior rolling the HPCI turbine.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Small Break LOCA   UFSAR SECTION 15.6.4, 15.6.5

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because verifying lube oil temperature indicating switch setpoints prior to rolling the HPCI turbine doesn't adversely impact systems or functions so as to create an accident of a type different from those previously evaluated in the UFSAR. This procedure change will only aid in preventing possible damage to the HPCI turbine and thus decreasing the probability of any accident.

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

SE-94-039

One time extension to 67 day Safe Shut Down Administrative

DESCRIPTION:

This change to the fire protection program has two parts:

- allow the extension of the 67 day safe shutdown ATR for safe shutdown path B by 24 days.
- establish additional compensatory measures.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.
   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.
   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Appendix R Fire as described in the Fire Hazards Analysis

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because this is an administrative change to the procedure and therefore does not effect equipment operation. The type of accident that could occur (Appendix R Fire) has already been evaluated. The stations approved program addresses the consequences of this fire. No new accident types will be created by this change.

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

SE-94-042
Temp Alt 94-2-32 Disabling Personnel Interlock on U2

DESCRIPTION:

This Safety Evaluation made changes to the Unit 2 Primary Containment Air Lock Doors and Air Lock Mechanism. The following is a description of the changes:

- The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.c.
- A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the interlock doors has been gagged in the CLOSED Position.
- The "Stongbacks" have been left installed to ensure the Drywell side door is aligned and seated properly.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.
   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.
   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Decrease in Reactor Coolant Inventory UFSAR SECTION 15.6

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the following changes have been made to the Unit 2 Personnel Interlock Doors for Primary Containment.

   - The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.b and c.

     One interlock door closed will ensure the integrity of Primary Containment. Administrative controls as required by TS, are being implemented that will ensure only one door is open at a time.

   - A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the air lock doors has been gagged in the CLOSED Position.

     The air lock doors are being leak rate tested to verify that there is no leakage or acceptable leakage out of the air locks. The valve that communicates the Drywell volume and air lock volume has been gagged in the CLOSED position to ensure that during a seismic event and/or a decrease in reactor coolant that the valve will remain in the CLOSED (as tested) condition.

   - The "Strongbacks" have been left installed.

   The strongbacks have been installed and will be left on the Drywell air lock door. The strong back is a series of structural steel that bolts onto the Drywell door and secures the door in the closed position for leak rate testing. The strongbacks have been previously evaluated and found acceptable for operation.

   Based on the above information and the fact that the Technical Specification LCO is being implemented the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the LCO requirements for an air lock door inoperable and the air lock interlock mechanism inoperable. Therefore, the Technical Specifications will be meet. This will ensure the margin to safety will be maintained.

SE-94-043
SESR #4-2156

DESCRIPTION:

The installation of a larger U-bolt (5/8" versus 1/2") on pipe support M-987D-75 because of increased weight of a parts upgrade of the Standby Liquid Control (SBLC) Accumulators for Unit 1. The replacement accumulators evaluation is ME-93-0541-00, Revision 2.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Anticipated Transient Without SCRAM UFSAR SECTION 9.3.5

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the larger U-bolt does not create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR. The larger U-bolt functions in the same manner as the smaller U-bolt it is replacing. The higher weight of replacement SLC accumulators requires the larger U-bolt to maintain the system seismically. Sizing of the U-bolt has been evaluated by seismic calculation (SESR 4-2156).

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because SESR 4-2156 evaluated the increase in weight of the replacement accumulator and the replacement of a larger U-bolt on pipe support M-987D-75. The evaluation determined that these changes were within the design loadings of the SBLC system.

M04-0-90-003
CRD Repair Room A/C Installation

DESCRIPTION:

Provided cooling for the CRD Repair Room. This modification installed an air cooled condenser located outside the room and an air handling unit located inside the room. Electrical power is supplied from a GE MCC which replaced the existing Westinghouse MCC 42R-2-1.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because this modification will install an air conditioning system for the CRD Repair Room. The air handling unit, plenum, return grill and thermostat will be installed inside the CRD repair ante room. The air cooled condenser will be installed outside the CRD Repair Room. This equipment will be powered from a 480V GE MCC which will replace the existing Westinghouse MCC 42R-2-1. Operation of the CRD Repair Room A/C System will offset the constant addition of heat incurred during maintenance on control rod drives. The design includes locally mounted disconnect switches for periods when this equipment will not be required.

Possible new failure modes or unacceptable conditions include:

1. Electrical failures in the new equipment.
2. Leaks in the new refrigerant lines.
3. Failures in the modified block walls.
4. Spread of contamination.

Possible impact of the above failures during all operating modes are:

1. The electrical requirements for this modification include the installation of properly sized breakers in non-safety related MCC 42R-2-1 to protect existing plant electrical equipment from any faults which may occur in the new HVAC equipment. MCC 42R-2-1 receives electrical power from non-safety related transformer T42R-2. The only loads on MCC 42R-2-1 will be the CRD Repair Room HVAC System. Therefore, a fault in the new electrical equipment will result in the tripping of breakers in MCC 42R-2-1 which will have no impact on any other plant equipment.

2. A leak in the refrigerant lines installed by this modification would result in the release of refrigerant-22 into the Unit 1 Reactor Building. The Reactor Building Ventilation System, designed to produce a negative differential pressure, evacuates the Reactor Building at a rate of approximately 1 free volume/hour. Therefore, leakage of refrigerant into the Reactor Building free volume would have no credible impact from a human safety standpoint and have no impact on equipment operation.

3. The structural requirements for this modification include design changes to the west (blocking-in an existing louver opening) and north (installation of electrical supply and refrigerant supply and return lines) block walls. As part of the designer's walkdown, it was identified that no safety related equipment was attached to these two block walls. The actual design will require structural changes meet the seismic 2-over-1 criteria but, if a failure of the wall were to occur, no safety related equipment would be affected.

4. Increased local air flow from the air handling unit could result in unacceptable spread of contamination. The location of the air handling unit inside the ante room instead of the CRD Repair Room provides the highest air flow in the area of least contamination to prevent an unacceptable airborne contamination problem. Blocking-in the louver opening seals the ante room to prevent the spread of contamination to an uncontrolled area.

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

DCR 4-93-205

DESCRIPTION:

The torus level indication was found to be in error and the source was traced to the faulty Narrow Range Level Transmitter (LT-001-1602-9). NWR Q08293 replaced the Rosemount Model 1151DP3B12 (obsolete designation) with a Rosemount Model 1151DP3G12M1B1. This model includes the mounting bracket which was previously ordered separately and an optional integral meter.

The Reactor Building Exhaust Fan 2C was auto-tripping and the source was traced to the setpoint of differential pressure switch (DPS-002-5741-261C) being at the low end of the switch's range. NWR Q08212 replaced the Dwyer Model 1821-2 with Dwyer Model 1823-1. This model has a range of 0.3" to 1.0" water column (WC) whereas the previous model's range was 0.5" to 2.0" WC. The setpoint of 0.5" WC remains unchanged.

DCR 4-93-205 updated the appropriate data sheets to reflect these changes.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

Internal Flood Measures	UFSAR SECTION 3.4.1.2
LOCA	UFSAR SECTION 7.5

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the instrument model changes will not affect the function or operation of the systems since the replacement instruments function the same as the original instruments.

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

DCR 4-94-046

DESCRIPTION:

This DCR revised the Master Equipment List (MEL) and selected drawings to incorporate the results of Component Classification (CC) of the Standby Gas Treatment (SBGT) System. As part of this DCR, 1) no physical change was made to any plant structure, system equipment or component and 2) some components were upgraded from NSR to SR because they are required for the SBGT system to perform its SR function (Secondary Containment Radioactive Effluent Control). Documentation specifically addressing these changes is included in Component Classification Binder #CC-QC009. The CC program is an ongoing controlled program that is supervised by Station Engineering.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment	UFSAR SECTION 15.6.2
Loss of Coolant Accidents Resulting from Piping Breaks Inside Containment	UFSAR SECTION 15.6.5
Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings	UFSAR SECTION 15.7.2

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because this DCR does not involve any physical changes to plant systems, structures, equipment or components. The Component Classification (CC) process for the SBGT system identified the operating mode for each component in the system and also identified that component's role in accomplishing the SBGT system safety function. The CC process also considered all applicable accidents analyzed in the SAR and all potential equipment or component malfunctions. The CC process provides assurance that the changes made by this DCR do not affect any existing accidents analyzed in the SAR and do not create any new accidents. The SBGT system CC process is documented in the SBGT system CC binder.

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

DCR 4-94-055

DESCRIPTION:

The implemented change incorporates the actual location of pressure test point connections for the condensate booster pump discharge piping on Unit 1 Piping and Instrumentation Diagram (P&ID); and incorporate the addition of pressure test point connection for the condensate booster pump discharge piping on Unit 2 P&ID. These Unit 1 and Unit 2 P&ID as-built changes reflect the original designed and installed conditions.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

Loss of Normal AC Power	UFSAR SECTION 15.8.2
Loss of Normal Feedwater Flow	UFSAR SECTION 15.8.3

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the change as described does not cause a functional change in the system or its interaction with other plant systems. It does not alter any physical parameters or process variables of the plant. Due to the nature of the change, there are no new inherent failure modes introduced to the system and the change does not add any new components or process routes.

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

DCR 4-94-064

DESCRIPTION:

Schematic Diagrams 4E-1351B, Sheet 2; 4E-2345, Sheet 1; 4E-2345, Sheet 2; 4E-2430, Sheet 2; and 4E-2430, Sheet 4: These drawings update cross references and descriptions on relays and control contacts to more accurately reflect the installed conditions.

Piping Diagram M-84, Sheet 1: This drawing revised the Equipment Piece Number (EPN) for the Unit 2A Off-Gas Filter Outlet Valve from 2-5499-55 to 2-5499-51. This change was made to match the configuration and numbering of the Unit 1 valve.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

Loss of auxiliary power	UFSAR SECTION	8.3.1
Power bus loss of voltage	UFSAR SECTION	8.3.1
Failure of one diesel generator to start	UFSAR SECTION	8.3.1.6.4
Load rejection without bypass	UFSAR SECTION	15.2.2.1
Load rejection with bypass (Loss of electrical load)	UFSAR SECTION	15.2.2.2
Loss of Coolant	UFSAR SECTION	15.6.2, 15.6.5

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because no new accident scenarios are created by this DCR. The function of the Core Spray, Diesel generator and Off-Gas Systems and their ability to operate are unchanged. This DCR will not adversely impact systems or functions nor will the possibility of an accident malfunction be created that is different from those previously evaluated in the SAR.

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

SE-93-109
M04-1(2)-89-115 Work Packages Q02824, Q02825

DESCRIPTION:

The work performed under this package calibrated two 0-100 psi pressure indicators (SI#208039) and a flow switch (SI#699224) prior to installation of modification M04-1(2)-89-115 (Modification of the service water radiation monitoring system sample delivery piping). The pressure indicators (PIs) will be used to ensure the sample stream eductor is operating properly. Under this package, the PIs will be used to gather system sample pressures while throttling the two glove valves on either side of the eductor. Also, during this test, a flow indicator was installed to give flow indications. This information was used to determine proper system operating pressures. The flow indicator was then removed and the flow switch was installed. The low flow setpoint was then verified. If erratic indication occured during performance of the traveler, individual instrument calibrations can be performed.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because installation and testing of this equipment cannot cause any plant accident or transient not described within the UFSAR. The installation does not alter the interconnecting systems so as to create abnormal lineups or operating modes. The installation will be passive with respect to the potential to initiate a different type of accident.

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

SE-93-112
M04-1(2)-89-115
Service Water Radiation Monitor

DESCRIPTION:

The work to be performed under these packages demolished the existing Service Water Radiation Monitor (SWRM) sample delivery system (receiver tank, pump, all associated piping and valves) and installed a new, eductor driven system powered by domestic water. The sample system inlet isolation valve was replaced and the service water return header was open through a 1-1/2" pipe to the turbine building 595' level during the replacement. This valve acts as the isolation point for further installation work. Domestic water was isolated for installation of the back flow preventer. All other items (skid, detector, eductor) were then installed. A flow indicator was installed to facilitate Instrument Maintenance work and testing on the flow switch and pressure gauges. The indicator was removed and replaced with the switch. A leak test was then performed.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because this work only interfaces with the domestic water system and the service water system. Both interfaces are mechanical only. No other SSC will be impacted by the scope of this work. The worst case scenario would involve a failure of the installed isolation valve on the service water return header. This would lead to leakage onto the turbine building first floor. But, this leakage will not be of greater magnitude than the capability to remove water by the floor drain system. Therefore, this event will not result in flooding. No other SSC will be adversely impacted so as to create a new UFSR accident or transient.

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

E04-1-93-094

Replace Turbine Rotor Unstacking Transformer with Dry Type

DESCRIPTION:

The subject exempt change replaced an oil-filled 1 MVA transformer with a dry-type 500 KVA transformer on elevation 639' of the Unit 1 Turbine Building. The existing wet pipe system was demolished.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because T42R-5A receives power from the 13.8 KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8 kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. The replacement of the oil-type transformer with a dry-type one results in no new accident type.

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

E04-1-93-245
Install Welding Receptacles on Turbine Shield Wall

DESCRIPTION:

The subject exempt change replaced the existing panel with a 10 circuit distribution panel and installed six 60 amp welding receptacles powered from this new panel. Five receptacles were mounted on the outside of the turbine shield wall west of the new circuit panel. A sixth was installed on the inside of the turbine shield wall. This new configuration provides a safer and more efficient means for providing power on the turbine deck.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the subject design change will not result in changed operation of the existing panels. Therefore, no new accident that has not been previously analyzed will be created.

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

E04-1-93-325
UAT Change-out Concrete Work

DESCRIPTION:

The subject exempt plant change installed new concrete piers in support of the replacement of the Unit 1 Unit Auxiliary Transformer (UAT). New concrete piers were required for the fire suppression deluge system which was redesigned due to physical differences between the existing GE and the new SMIT transformer.

Two other exempt changes were required to complete the replacement of the Unit 1 UAT:

E04-1-93-326 replaced the existing fire protection system piping and fire detection method. The deluge piping was replaced due to the physical differences between the existing GE UAT and the new SMIT UAT. The detection method was changed in order to make it more reliable. The overall operation of the system did not change.

E04-1-93-327 reinstalled the transformer control circuitry. These changes were necessary due to slight differences between the GE and SMIT transformers. The control circuitry changes do not affect the operation of the plant.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   The accidents which meet these criteria are listed below:

   Loss of Auxiliary Power UFSAR SECTION 8.3.1

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because the UAT is being replaced by a newer transformer. The failure mode of this new transformer, fire protection system, and control circuitry is the same as for the existing transformer. The failure rate due to these changes is reduced due to the more reliable transformer and enhancements to the fire protection system. Therefore, an accident different from those previously evaluated in the SAR is not created.

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

P04-1-91-127
Replacement of existing T-Quencher Bolts

DESCRIPTION:

Installed new replacement welded and/or bolted threaded rod to replace rods anchoring the T Quencher supports located in the torus.

The sample of rods was removed for examination to confirm the absence of stress corrosion cracking.

SAFETY EVALUATION SUMMARY:

1. The change described above has been analyzed to determine each accident or anticipated transient described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis.

   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident.

   - Operation or failure of the changed structure, system, or component could lead to the accident.

   For each of these accidents, it has been determined that the change described above will not increase the probability of an occurrence or the consequence of the accident, or malfunction of equipment important to safety as previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR is not created because change does not affect equipment operations or functions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no change to Technical Specifications.

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE-94-37	Date: 5-9-94
Subject: RWM Software Upgrade	
Submitted by: Frank Amato	

FOR REVIEW:

1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
   - [ ] a. Changes to procedures as described in the Safety Analysis Report.
   - [ ] b. Changes to equipment or systems as described in the Safety Analysis Report.
   - [ ] c. Tests or experiments NOT described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - [ ] a. Procedure changes.
   - [ ] b. Equipment or system changes.
   - [ ] c. Tests or experiments.
- [ ] 3. Proposed changes to the Technical Specifications or Operating License.
- [ ] 4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- [ ] 5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
- [ ] 6. All REPORTABLE EVENTS (LERs only).
- [ ] 7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
- [ ] 8. All changes to the Station Emergency Plan prior to implementation.
- [ ] 9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

- [x] 10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

ATTACHMENT G (Page 1 of 8)
10CFR50.59 SAFETY EVALUATION

GENERAL INFORMATION:	
Safety Evaluation Number: SE - 94 - 37	
Document Identifier: SAR (Software Activity Request) # 991 (Modification, Temp. Alt., Work Request Number, etc.)	
Unit(s): 1 & 2	System(s): 207 (Rod Worth Minimizer)
Applicable Plant Mode(s): ALL (Run, Startup/Hot Standby, Refuel, Shutdown)	
Plant Mode Restriction(s): NONE	

List Multiple Procedures Affected Below:

Procedure Number	Procedure Number	Procedure Number	Procedure Number
QCOP 207-1			

CHANGE DESCRIPTION:

1. Describe the proposed change:

The Rod Worth Minimizer (RWM) software will be upgraded to include two new features. First, a select block will be added to the current rod blocks. This will cause a rod block upon selection of an out-of-sequence control rod. This select block function will include an on/off 'toggle' on the RWM touch screen. The other change is the provision of an insert block signal in Rod Exercise mode as soon as a control rod is moved in one notch. In addition, a rod block will be applied to all other rods at this time. The blocks are removed when the current rod is withdrawn to its original position. Currently the RWM applies a rod block only after a rod has traveled more than one notch past its target position.

2. Reason for the change:

These changes are being made to address several control rod movement errors. The software changes are the corrective actions of NTS item #2650193007305.

3. Is the change:

X	Permanent	
	Temporary - Expected Duration:	

ATTACHMENT G (Page 2 of 8)
10CFR50.59 SAFETY EVALUATION

REFERENCE DOCUMENTS:

4. List reference documents used which describe the structure, system, or component. Identify documents referenced even if no information was found in that section.

a. UFSAR Section(s): 7.7.2 (RWM), 15.4.10 (Rod Drop Accident)

b. SER Section(s):

c. Tech. Spec. Section(s): 3.3/4.3B (Control Rods)

d. Fire Protection Program Document Pkg Section(s):

e. Code of Federal Regulations Section(s):

f. Regulatory Guides/NUREGs:

g. Other: RWM User's Documentation (Rev 3.1, October, 1990)

EVALUATION:

5. Describe how plant operation is affected when the structure, system, or component function is changed as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems, or components.

Using the new software, plant operation will be affected by the addition of the select block and the blocks applied in Rod Exercise mode. The select block is in addition to the current blocks and will be applied when an out-of-sequence rod is selected. This block will improve plant operation in that it adds another barrier to the NSO selecting and moving an incorrect control rod. The Rod Exercise blocks currently take effect only after a rod has traveled more than one notch PAST its target position. The upgraded Rod Exercise blocks will take effect immediately at the target in position, in addition to blocking movement of all other rods while a rod is inserted. The Rod Exercise blocks will improve operation by adding a barrier to mispositionings during the rod exercise procedures.

No physical changes will be made to the RWM, so there will be no changed interactions with other structures, systems, or components after the software upgrade. All interlocks with the RWM will remain the same, as will all other existing rod blocks provided by the RWM.

6. Describe how the change will affect equipment failures. Describe any new failure modes and their impact during all applicable operating modes.

The only changes that will be made are to the software of the RWM - no physical changes will be made. All interactions with other systems and components will be unchanged. As a result, the failure modes of the RWM will remain the same as before the software upgrade.

ATTACHMENT G (Page 3 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis.
- The changed structure, system, or component is explicitly or implicitly assumed to function during or after the accident.
- Operation or failure of the changed structure, system, or component could lead to the accident.

15.4.10 Rod Drop Accident	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine factors affecting the specification, it is necessary to review the UFSAR and SER where the Technical Specification Bases section does not explicitly state the basis.

3.3.B.3	

9. Will the change involve a Technical Specification revision?

	YES
X	NO

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indiocate that a Technical Specification revision is required.

10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of these pages to answer the following questions for each accident listed in Step 7. Provide rationale for all NO answers.

Affected accident: Rod Drop Accident	UFSAR Section: 15.4.10

a. May the probability of the accident be increased?

	YES
X	NO

The operation of the RWM is independent of the Control Rod Drive (CRD) System and has no impact on the withdrawal of a rod. It provides rod blocks to the Reactor Manual Control System (RMCS), but does not affect the interaction of the control rods with the reactor internals. Because this interaction is the means by which a control rod becomes stuck and later drops to the full out position, the RWM cannot affect this event. As a result, the probability of the Rod Drop Accident will not be increased by this software upgrade.

b. May the consequences of the accident (off-site dose) be increased?

	YES
X	NO

The RWM impacts the Rod Drop Accident only by limiting the worth (reactivity) of control rods during a reactor startup by enforcing a withdrawal sequence. This sequence must follow Banked Position Withdrawal Sequence (BPWS) rules to 20% power, which limits the energy deposited in the fuel to 280 cal/gm. These rules are not affected by the software upgrade, nor is the method by which the RWM enforces them. As a result, the rod worths after the software upgrade will not be increased. In addition, the RWM does not provide any mitigating effects after the accident. Because the same BPWS rules will be enforced (no increase in deposited enthalpy), and the RWM cannot mitigate the Rod Drop Accident after it occurs, the consequences of the accident will not be increased by the software upgrade.

ATTACHMENT G (Page 5 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

c. May the probability of a malfunction of equipment important to safety increase?

	YES
X	NO

No changes will be made to the RWM computers to add the rod blocks in the Rod Exercise Mode and selection blocks for out-of-sequence rods. These are software modifications only. No physical modifications will be made to the RWM computers, and no interactions with other systems will be altered. The probability of malfunction will remain the same for the RWM.

d. May the consequences of a malfunction of equipment important to safety increase?

	YES
X	NO

The RWM is comprised of two independent computers designed to enforce the control rod withdrawal sequence. For the withdrawal of the first 12 control rods, one RWM must be operable. If one RWM computer failed, the other would be available. If neither were available for the first 12 rods, startup would not be permitted. This consequence will remain the same after the new software is installed because startup will still be prohibited. After the first 12 rods are fully withdrawn, the RWM is required to be operable up to 20% power. However, if both RWM computers have failed, an additional verifier may be used as a substitute. Again, the consequences of this malfunction will not change with the new software, as a second verifier will still be required.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 6 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

11. Based on the answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

	YES
X	NO

Describe the rationale for your answer.

The changes that will be made are to the RWM software only. No modifications will be made to the RWM computers, nor will any interactions with other systems be changed. The blocks that are being added will serve as an additional barrier to control rod mispositionings and all of the current rod blocks will be retained. Because no other systems will be affected, there will be no adverse system interactions or accidents created in other systems.

In addition, because the RWM itself will not be altered, the failure modes will remain the same as before the software upgrade. As a result, there will be no new type of RWM malfunction not evaluated in the UFSAR.

If any answer to Question 11 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 7 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists. Proceed to Step 14.

Technical Specification: 3.3.B.3

Determine which of the following is true for the above specification:

	All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. The actual acceptance limit need not be identified to determine that no reduction in margin of safety exists. Proceed to Step 13.
	The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s) or margin(s) below.
	The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit or margin for the Margin of Safety determination. List the limit(s) or margin(s) below.
X	The change does not affect any parameters upon which the Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to Step 14.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits identified in Step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

ATTACHMENT G (Page 8 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

14. Check one of the following:

	An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The Proposed change **MUST NOT** be implemented without NRC approval.
X	No unreviewed Safety Question will result (Steps 10, 11, and 13) **AND** no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.
	A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Indicate applicable type(s) below:
	The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.
	The change is a plant modification or minor plant change. Indicate applicable type(s) below:
	A revision to an existing Technical Specification is required. The change **MUST NOT** be installed until receipt of an approved Technical Specification revision.
	The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize the installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.
	Nuclear Licensing has authorized installation, but no operation, prior to receipt of the NRC approval of License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result, and provides authority for installation only.

Preparer/Date: Frank M. Scott 4-27-94

15. Documentation is adequate to support the above conclusion and the conclusion is valid.

Reviewer/Date: BRStuck 5-7-94

16. Obtain a Safety Evaluation number from the Systems Engineering Clerk. Record on Page 1.

17. Leave 1 Safety Evaluation copy with Systems Engineering Clerk. File original with package.

18. Forward Safety Evaluation copy to FSAR Coordinator (ANI Audit Recommendation 88-1).
Completed: Systems Engineering Clerk Initials: DK Date: 5-9-94

(final)

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE-94-038	Date: 5-12-94
Subject: QCOS 2300-1 Rev 6	
QCOS 2300-5 Rev 6	
Submitted by: Bill Finke	

FOR REVIEW:

	1. Safety Evaluations **NOT** involving an unreviewed safety question as defined in 10CFR50.59 for:
X	a. Changes to procedures as described in the Safety Analysis Report.
	b. Changes to equipment or systems as described in the Safety Analysis Report.
	c. Tests or experiments **NOT** described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
	a. Procedure changes.
	b. Equipment or system changes.
	c. Tests or experiments.
	3. Proposed changes to the Technical Specifications or Operating License.
	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
	6. All REPORTABLE EVENTS (LERs only).
	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
	8. All changes to the Station Emergency Plan prior to implementation.
	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

	10. Other OSR Items/Documents **NOT** addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) **for information only**. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

ATTACHMENT G (Page 1 of 8)
10CFR50.59 SAFETY EVALUATION

GENERAL INFORMATION:	
Safety Evaluation Number: SE - 94 -038	
Document Identifier: (Modification, Temp. Alt., Work Request Number, etc.) QCOS 2300-1 Rev. 6	
Unit(s): 1 and 2	System(s): HPCI (2300)
Applicable Plant Mode(s): All Modes (Run, Startup/Hot Standby, Refuel, Shutdown)	
Plant Mode Restriction(s): None	

List Multiple Procedures Affected Below:

Procedure Number	Procedure Number	Procedure Number	Procedure Number
QCOS 2300-5 Rev. 6			

CHANGE DESCRIPTION:

1. Describe the proposed change:

This procedure is being revised to have the operator verify lube oil temperature indicating switch setpoints are set to their proper value as listed in the procedure. Also, the HPCI pump is verified to be filled and vented locally prior to rolling the HPCI turbine.

2. Reason for the change:

This procedure change is being performed to ensure proper setpoints prior to manual startup for routine surveillances to reduce the probability of spurious high temperature alarms which would require system shutdown.

3. Is the change:

X	Permanent	
	Temporary - Expected Duration:	

ATTACHMENT G (Page 2 of 8)
10CFR50.59 SAFETY EVALUATION

REFERENCE DOCUMENTS:

4. List reference documents used which describe the structure, system, or component. Identify documents referenced even if no information was found in that section.

a. UFSAR Section(s): 6.3, 7.3, 9.3.3.9, 15.5.1, 15.6.2, 15.6.5

b. SER Section(s): 3.5.2.1

c. Tech. Spec. Section(s): 3.5.C/4.5.C, 3.5.G/4.5.G, 3.7.A/4.7.A

d. Fire Protection Program Document Pkg Section(s):

e. Code of Federal Regulations Section(s):

f. Regulatory Guides/NUREGs:

g. Other:

EVALUATION:

5. Describe how plant operation is affected when the structure, system, or component function is changed as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems, or components.

These temperature indicating switches give alarms only to the Control Room Operator. They do not provide any trip functions. Verifying these setpoints will eliminate unnecessary alarms but yet will still provide alarm protection as designed.

6. Describe how the change will affect equipment failures. Describe any new failure modes and their impact during all applicable operating modes.

This change does not affect the operation of any equipment and therefore will not affect any failure modes. By ensuring proper alarm setpoints, equipment damage can be averted due to operator response to high temperature conditions.

ATTACHMENT G (Page 3 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis.
- The changed structure, system, or component is explicitly or implicitly assumed to function during or after the accident.
- Operation or failure of the changed structure, system, or component could lead to the accident.

Accident	UFSAR Section
Small break LOCA	15.6.4, 15.6.5

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine factors affecting the specification, it is necessary to review the UFSAR and SER where the Technical Specification Bases section does not explicitly state the basis.

None	N/A

9. Will the change involve a Technical Specification revision?

	YES
X	NO

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indiocate that a Technical Specification revision is required.

ATTACHMENT G (Page 4 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of these pages to answer the following questions for each accident listed in Step 7. Provide rationale for all **NO** answers.

Affected accident: **None**	UFSAR Section: **N/A**

a. May the probability of the accident be increased?

	YES
X	NO

The probability of an accident will not be increased because this procedure change does not affect any equipment which is currently considered an initiator of any analyzed accident.

b. May the consequences of the accident (off-site dose) be increased?

	YES
X	NO

The consequences of an accident (off-site dose) will not be increased because automatic operation of HPCI is not affected by the temperature switches involved in this procedure change, thus, plant response to accidents is unaffected from previous analyses.

ATTACHMENT G (Page 5 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

c. May the probability of a malfunction of equipment important to safety increase?

	YES
X	NO

The probability of a malfunction of equipment important to safety will not increase because by verifying the switch setpoints prior to rolling the HPCI turbine will only ensure the HPCI lube oil system will operate as designed.

d. May the consequences of a malfunction of equipment important to safety increase?

	YES
X	NO

The consequences of a malfunction important to safety will not increase because automatic operation of HPCI is not affected by the temperature switches involved in this procedure change, thus, plant response to accidents is unaffected from previous analyses.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 6 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

11. Based on the answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

	YES
X	NO

Describe the rationale for your answer.

Verifying lube oil temperature indicating switch setpoints prior to rolling the HPCI turbine doesn't adversely impact systems or functions so as to create an accident of a type different from those previously evaluated in the UFSAR. This procedure change will only aid in preventing possible damage to the HPCI turbine and thus decreasing the probability of any accident.

If any answer to Question 11 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 7 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists. Proceed to Step 14.

Technical Specification:

Determine which of the following is true for the above specification:

	All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. The actual acceptance limit need not be identified to determine that no reduction in margin of safety exists. Proceed to Step 13.
	The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s) or margin(s) below.
	The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit or margin for the Margin of Safety determination. List the limit(s) or margin(s) below.
	The change does not affect any parameters upon which the Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to Step 14.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits identified in Step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

ATTACHMENT G (Page 8 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

14. Check one of the following:

	An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The Proposed change **MUST NOT** be implemented without NRC approval.
X	No unreviewed Safety Question will result (Steps 10, 11, and 13) **AND** no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.
	A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Indicate applicable type(s) below:
	The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.
	The change is a plant modification or minor plant change. Indicate applicable type(s) below:
	A revision to an existing Technical Specification is required. The change **MUST NOT** be installed until receipt of an approved Technical Specification revision.
	The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize the installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.
	Nuclear Licensing has authorized installation, but no operation, prior to receipt of the NRC approval of License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result, and provides authority for installation only.

Preparer/Date: Bill Finke 5-12-94

15. Documentation is adequate to support the above conclusion and the conclusion is valid.

Reviewer/Date: Alex L. Misak 5/12/94

16. Obtain a Safety Evaluation number from the Systems Engineering Clerk. Record on Page 1.

17. Leave 1 Safety Evaluation copy with Systems Engineering Clerk. File original with package.

18. Forward Safety Evaluation copy to FSAR Coordinator (ANI Audit Recommendation 88-1).
Completed: Systems Engineering Clerk Initials: DK Date: 5-12-94

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: 94-039	Date: 5-17-94
Subject: One time extension to 67 day Safe Shut Down Administrative Technical Requirement (ATR) for Unit 2 during Q1R13.	
Submitted by: Jim Masterlark	

FOR REVIEW:

	1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments NOT described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

X	10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

ATTACHMENT G (Page 1 of 8)
10CFR50.59 SAFETY EVALUATION

GENERAL INFORMATION:	
Safety Evaluation Number: SE - 94 - 039	
Document Identifier: **67 Day SSD ATR Extension for SSD Path B** (Modification, Temp. Alt., Work Request Number, etc.)	
Unit(s): **1 and 2**	System(s): **4100, 287, 1000, 1300, 2900, 6600, 6700, 7300, 8300, 8350**
Applicable Plant Mode(s): **All modes** (Run, Startup/Hot Standby, Refuel, Shutdown)	
Plant Mode Restriction(s): **No restrictions**	

List Multiple Procedures Affected Below:

Procedure Number	Procedure Number	Procedure Number	Procedure Number

CHANGE DESCRIPTION:

1. Describe the proposed change:

This change to the fire protection program will have two parts:

-allow the extension of the 67 day safe shutdown ATR for safe shutdown path B by 24 days

-establish additional compensatory measures .

(See Attached document)

2. Reason for the change:

This change will allow the work to continue without requiring the operating unit to shutdown when the 67 day ATR expires. Additional compensatory measures will be established to ensure safe operation during the extension of the ATR.

(See Attached document)

3. Is the change:

	Permanent	
X	Temporary - Expected Duration **May 21 thru June 15**	

ATTACHMENT G (Page 2 of 8)
10CFR50.59 SAFETY EVALUATION

REFERENCE DOCUMENTS:
4. List reference documents used which describe the structure, system, or component. Identify documents referenced even if no information was found in that section.
a. UFSAR Section(s): **9.5.1**
b. SER Section(s): **Fire Protection Reports, Volume 3--Fire Protection SERs**
c. Tech. Spec. Section(s): **6.0 Administrative Requirements**
d. Fire Protection Program Document Pkg Section(s): **Fire Protection Reports, Volume 2, Safe Shutdown Reports, Section Administrative Technical Requirements**
e. Code of Federal Regulations Section(s): **10CFR50 Appendix R, 10CFR50.48**
f. Regulatory Guides/NUREGs: **None**
g. Other: **GL 86-10, GL 88-12, Attached Documentation (Justification)**

EVALUATION:
5. Describe how plant operation is affected when the structure, system, or component function is changed as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems, or components. **This is an administrative change to the Fire Protection Program. This change impacts the compensatory measures required when Safe Shutdown Equipment is Inoperable. This change will allow equipment to be inoperable for more than 67 days if compensatory measures are estblished which would compensate the for the weekness in the 3rd echelon for fire protection (See Step 10.a).**
6. Describe how the change will affect equipment failures. Describe any new failure modes and their impact during all applicable operating modes. **There are no new failure modes. The plant would continue to operate within the bounds of the ATR. The only change is that the ATR would be extended and additional compensatory measures will be established.**

ATTACHMENT G (Page 3 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis.
- The changed structure, system, or component is explicitly or implicitly assumed to function during or after the accident.
- Operation or failure of the changed structure, system, or component could lead to the accident.

Appendix R Fire as described in the Fire Hazards Analysis	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine factors affecting the specification, it is necessary to review the UFSAR and SER where the Technical Specification Bases section does not explicitly state the basis.

None	

9. Will the change involve a Technical Specification revision?

	YES
X	NO

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of these pages to answer the following questions for each accident listed in Step 7. Provide rationale for all **NO** answers.

Affected accident: **Appendix R Fire**

a. May the probability of the accident be increased?

	YES
X	NO

Basis of Fire Protection Program:

The fire protection program is based upon the concept of defense in depth. This defense consists of three echelons of protection between fire initiators and a possible uncontrolled release. No one of these echelons is perfect or complete by itself. These echelons are as follows:

1. **FIRE PROTECTION PROGRAMS TO PREVENT FIRE INITIATION:** This echelon helps to ensure that a fire does not initiate, or if it does initiate, that it does not expand beyond the incipient stages. These programs include the transient combustible programs, welding and grinding permits, and good housekeeping practices.

2. **RAPID SUPPRESSION AND DETECTION:** This echelons will help to ensure that a fire that is in its incipient stage does not propagate into an "Appendix R" fire where safe shutdown of a unit will be required. This barrier consists of numerous automatic detection and suppression systems located throughout the power block.

3. **FIRE BARRIERS/SAFE SHUTDOWN:** This barrier consists of two parts, the first part is physical fire barriers that separate safe shutdown equipment from a Design Basis Fire as described in the Fire Hazards Analysis. The second part is the safe shutdown equipment/procedures that would safely shut down a unit while the fire is contained.

If an Appendix R fire (as described in the Fire Hazards Analysis) were to occur when a safe shutdown path is inoperable, safe shutdown could not be achieved within the bounds of the safe shutdown procedures or within the required time limits meet the requirements by Appendix R. Therefore, compensatory measures and 67 day ATR are established to help ensure that a fire will not reach the Appendix R fire stage.

Basis for ATR extension:

When the 67 day ATR is exceeded on 5/21/94, additional compensatory measures will be initiated to further reduce the probability that a fire could reach the Appendix R Fire stage where safe shutdown would be required These compensatory measures described within this document will strengthen the first two barriers to make-up for a reduction in the third. These compensatory measures will be limited to 24 days. This will allow work to continue on Unit 1 systems during it's refueling outage. After expiration of the 24 days, If all safe shutdown path B is not returned to operability, Unit 2 will be required to shutdown.

The extension of the ATR in itself will increase the probability of an Appendix R fire. However, the increase in compensatory measures will decrease the probability of a fire occurring and will increase the probability of mitigating a small fire before it becomes an Appendix R fire. Therefore, the overall probability for the accident to occur is equal or less than without this extension.

b. May the consequences of the accident (off-site dose) be increased?

	YES
X	NO

The change to the program allows 24 additional days to the safe shutdown ATR while safe shutdown path B will not be available. There will be no change to the consequences of the Appendix R fire when compared to the original ATR criteria. In addition, compensatory measure will be established to mitigate the consequenses of an incipiant fire.

ATTACHMENT G (Page 5 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

c. May the probability of a malfunction of equipment important to safety increase?

	YES
X	NO

This change is administrative in nature and does not affect plant equipment. Therefore, this change does not affect the probability of malfunction. Compensatory measures will help to ensure that a fire does not occur to challenge plant equipment.

d. May the consequences of a malfunction of equipment important to safety increase?

	YES
X	NO

This is an administrative change and does not affect the operation of equipment. This change allows safe shutdown equipment to be inoperable while compensatory measures are established to reduct the probability of a design basis fire to occur.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 6 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

11 Based on the answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

	YES
X	NO

Describe the rationale for your answer.

This is an administrative change to the procedure and therefore does not effect equipment operation. The type of accident that could occur (Appendix R Fire) has already been evaluated. The stations approved program addresses the consequences of this fire. No new accident types will be created by this change.

If any answer to Question 11 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 7 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists. Proceed to Step 14.

Technical Specification:

Determine which of the following is true for the above specification:

	All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. The actual acceptance limit need not be identified to determine that no reduction in margin of safety exists. Proceed to Step 13.
	The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s) or margin(s) below.
	The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit or margin for the Margin of Safety determination. List the limit(s) or margin(s) below.
X	The change does not affect any parameters upon which the Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to Step 14.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits identified in Step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

ATTACHMENT G (Page 8 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

14. Check one of the following:

	An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The Proposed change **MUST NOT** be implemented without NRC approval.
X	No unreviewed Safety Question will result (Steps 10, 11, and 13) **AND** no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.
	A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Indicate applicable type(s) below:
	The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.
	The change is a plant modification or minor plant change. Indicate applicable type(s) below:
	A revision to an existing Technical Specification is required. The change **MUST NOT** be installed until receipt of an approved Technical Specification revision.
	The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize the installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.
	Nuclear Licensing has authorized installation, but no operation, prior to receipt of the NRC approval of License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result, and provides authority for installation only.

Preparer/Date: James Masterlark [illegible] 5/13/94

15. Documentation is adequate to support the above conclusion and the conclusion is valid.

Reviewer/Date: DC Bucknell 5-18-94

16. Obtain a Safety Evaluation number from the Systems Engineering Clerk. Record on Page 1.

17. Leave 1 Safety Evaluation copy with Systems Engineering Clerk. File original with package.

18. Forward Safety Evaluation copy to FSAR Coordinator (ANI Audit Recommendation 88-1).
Completed: Systems Engineering Clerk Initials: DK Date: 5-18-94

(final)

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE 94-042	Date: 5/23/94
Subject: Temp. Alt. 94-2-32 Dissabling Personnel Interlock on Unit 2.	
Submitted by: Alan Blamey	

FOR REVIEW:

1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
   - [ ] a. Changes to procedures as described in the Safety Analysis Report.
   - [X] b. Changes to equipment or systems as described in the Safety Analysis Report.
   - [ ] c. Tests or experiments NOT described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - [ ] a. Procedure changes.
   - [ ] b. Equipment or system changes.
   - [ ] c. Tests or experiments.
- [ ] 3. Proposed changes to the Technical Specifications or Operating License.
- [ ] 4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- [ ] 5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
- [ ] 6. All REPORTABLE EVENTS (LERs only).
- [ ] 7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
- [ ] 8. All changes to the Station Emergency Plan prior to implementation.
- [ ] 9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

- [ ] 10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

ATTACHMENT G (Page 1 of 8)
10CFR50.59 SAFETY EVALUATION

GENERAL INFORMATION:	
Safety Evaluation Number: SE - 94 - 042	
Document Identifier: Temp. Alt. 94-2-32 (Modification, Temp. Alt., Work Request Number, etc.)	
Unit(s): **Two**	System(s): **010**
Applicable Plant Mode(s): **All modes** (Run, Startup/Hot Standby, Refuel, Shutdown)	
Plant Mode Restriction(s): **No restrictions**	

List Multiple Procedures Affected Below:

Procedure Number	Procedure Number	Procedure Number	Procedure Number
NONE			

CHANGE DESCRIPTION:

1. Describe the proposed change:

This SE is for changes made to the Unit 2 Primary Containment Air Lock Doors and Air Lock Mechanism. The following is a description of the changes:

- **The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.c.**
- **A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the interlock doors has been gagged in the CLOSED Position.**
- **The "Strongbacks" have been left installed to ensure the Drywell side door is aligned and seated properly.**

2. Reason for the change:

During a Drywell entry the interlock mechanism failed. These actions listed above will ensure that the Air Lock will be maintained operable during operation.

3. Is the change:

	Permanent	
X	Temporary - Expected Duration **Next Unit 2 Shutdown of sufficient duration - no longer than 8 Months**	

ATTACHMENT G (Page 2 of 8)
10CFR50.59 SAFETY EVALUATION

REFERENCE DOCUMENTS:	
4.	List reference documents used which describe the structure, system, or component. Identify documents referenced even if no information was found in that section.
a.	UFSAR Section(s): **1.0, 6.2, 3.4, 3.7, 3.8, 15.6**
b.	SER Section(s): **NA**
c.	Tech. Spec. Section(s): **3.7.A.7**
d.	Fire Protection Program Document Pkg Section(s): **NA**
e.	Code of Federal Regulations Section(s): **NA**
f.	Regulatory Guides/NUREGs: **None**
g.	Other: **NA**

EVALUATION:

5. Describe how plant operation is affected when the structure, system, or component function is changed as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems, or components.

The following changes have been made to the Unit 2 Primary Containment Air Lock doors

- **The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.c.**

  One interlock door closed will ensure the integrity of Primary Containment. Administrative controls, as required by TS, are being implemented that will ensure only one door is open at a time.

- **A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the air lock doors has been gagged in the CLOSED Position.**

  The air lock doors are leak rate tested to verify that there is no leakage or acceptable leakage out of the air locks. The valve that communicates the Drywell volume and air lock volume has been gagged in the CLOSED position to ensure that during a seismic event and / or a decrease in reactor coolant event the valve will remain in the CLOSED (as tested) condition.

- **The "Strongbacks" have been left installed.**

The strongbacks have been installed and will be left on the Drywell air lock door. The strong back is a series of structural steel beams that bolts onto the Drywell door and secures the door in the closed position for leak rate testing. Leaving the strongbacks in place has been previously evaluated and found acceptable for operation.

This change will not alter the plant response to any accidents or transients. This change will required additional administrative controls as per Technical Specification 3.7.A.7.b and c, which will require the operable interlock door locked closed and verified once every 31 days.

6. Describe how the change will affect equipment failures. Describe any new failure modes and their impact during all applicable operating modes.

There are no new equipment failures, other than defeating the interlocks, that may be caused by this alteration. The current configuration described above will ensure the leak tightness and structural integrity of Primary Containment and the Primary Containment Air Lock.

ATTACHMENT G (Page 3 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis.
- The changed structure, system, or component is explicitly or implicitly assumed to function during or after the accident.
- Operation or failure of the changed structure, system, or component could lead to the accident.

UFSAR Section 15.6 - Decrease in Reactor Coolant Inventory	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine factors affecting the specification, it is necessary to review the UFSAR and SER where the Technical Specification Bases section does not explicitly state the basis.

3.7.A.7 - Primary Containment Air Locks	

9. Will the change involve a Technical Specification revision?

	YES
X	NO

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

ATTACHMENT G (Page 4 of 8)

10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of these pages to answer the following questions for each accident listed in Step 7. Provide rationale for all **NO** answers.

Affected accident: **Decrease in Reactor Coolant Inventory**	

a. May the probability of the accident be increased?

	YES
X	NO

The following changes have been made to the Unit 2 Primary Containment Air Lock doors.

- **The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.b and c.**
- **A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the interlock doors has been gagged in the CLOSED Position.**
- **The "Strongbacks" have been left installed to ensure the Drywell side door is aligned and seated properly.**

These changes do not interface with primary system boundaries and therefore, will not increase the probability of a Decrease in Reactor Coolant Inventory event.

b. May the consequences of the accident (off-site dose) be increased?

	YES
X	NO

Based on the answer to question 5 the air lock doors will continue to maintain the integrity of Primary Containment and off-site dose will not be increased.

ATTACHMENT G (Page 5 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

c. May the probability of a malfunction of equipment important to safety increase?

	YES
X	NO

The probability of a malfunction of equipment important to safety has decreased. The outer air lock door has been locked in the closed position. The Drywell air lock door has the strongback installed which will prevent inadvertent opening of that air lock door. The valve that equalizes pressure between the air locks and Drywell has been locked CLOSED which prevents that valve from inadvertently opening and creating an additional leakage path. Because of the final configuration of the interlock doors malfunction of two of the interlocks (door and valve) has been eliminated.

d. May the consequences of a malfunction of equipment important to safety increase?

	YES
X	NO

As stated above the current plant configuration will not result in off-site dose rates increasing (b) and the probability of malfunction of equipment important to safety decreases (c). This coupled with the required administrative controls listed in the Technical Specification ensure that the consequences of this malfunctioning equipment are no greater than if the air locks were functioning properly.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 6 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

11. Based on the answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

	YES
X	NO

Describe the rationale for your answer.

This following changes have been made to the Unit 2 Personnel Interlock Doors for Primary Containment.

- **The interlocks that prevent more than one door open at a time have been defeated and are being controlled as per Technical Specifications 3.7.A.7.b and c.**

  One interlock door closed will ensure the integrity of Primary Containment. Administrative controls, as required by TS, are being implemented that will ensure only one door is open at a time.

- **A valve that is connected to the interlock mechanism that equalizes pressure between the Drywell and volume between the air lock doors has been gagged in the CLOSED Position.**

  The air lock doors are being leak rate tested to verify that there is no leakage or acceptable leakage out of the air locks. The valve that communicates the Drywell volume and air lock volume has been gagged in the CLOSED position to ensure that during a seismic event and / or a decrease in reactor coolant that the valve will remain in the CLOSED (as tested) condition.

- **The "Strongbacks" have been left installed.**

The strongbacks have been installed and will be left on the Drywell air lock door. The strong back is a series of structural steel that bolts onto the Drywell door and secures the door in the closed position for leak rate testing. The strongbacks have been previously evaluated and found acceptable for operation.

Based on the above information and the fact that the Technical Specification LCO is being implemented the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR is not created.

If any answer to Question 11 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 7 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):	
12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists. Proceed to Step 14.	
Technical Specification:	
Determine which of the following is true for the above specification:	
	All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. The actual acceptance limit need not be identified to determine that no reduction in margin of safety exists. Proceed to Step 13.
X	The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s) or margin(s) below.
	The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit or margin for the Margin of Safety determination. List the limit(s) or margin(s) below.
	The change does not affect any parameters upon which the Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to Step 14.
List Acceptance Limit(s)/Margin(s) of Safety	
3.7.A.7.b / 3.7.A.7.c	
13. Use the above limits identified in Step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.	
The LCO requirements for an air lock door inoperable and the air lock interlock mechanism inoperable. Therefore, the Technical Specifications will be meet. This will ensure the margin to safety will be maintained.	

ATTACHMENT G (Page 8 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):	
14. Check one of the following:	
	An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The Proposed change **MUST NOT** be implemented without NRC approval.
X	No unreviewed Safety Question will result (Steps 10, 11, and 13) **AND** no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.
	A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Indicate applicable type(s) below:
	The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.
	The change is a plant modification or minor plant change. Indicate applicable type(s) below:
	A revision to an existing Technical Specification is required. The change **MUST NOT** be installed until receipt of an approved Technical Specification revision.
	The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize the installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.
	Nuclear Licensing has authorized installation, but no operation, prior to receipt of the NRC approval of License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result, and provides authority for installation only.
Preparer/Date: Al Rs 5/22/94	
15. Documentation is adequate to support the above conclusion and the conclusion is valid.	
Reviewer/Date: DC Bucknell 5-22-94	
16. Obtain a Safety Evaluation number from the Systems Engineering Clerk. Record on Page 1.	
17. Leave 1 Safety Evaluation copy with Systems Engineering Clerk. File original with package.	
18. Forward Safety Evaluation copy to FSAR Coordinator (ANI Audit Recommendation 88-1). Completed: Systems Engineering Clerk Initials: Tx Date: 5-25-94	

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE-94-043	Date: 5/25/94
Subject: SESR #4-2156	
Submitted by: WILLIAM LAMB	

FOR REVIEW:

1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
   - [ ] a. Changes to procedures as described in the Safety Analysis Report.
   - [x] b. Changes to equipment or systems as described in the Safety Analysis Report.
   - [ ] c. Tests or experiments NOT described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - [ ] a. Procedure changes.
   - [ ] b. Equipment or system changes.
   - [ ] c. Tests or experiments.
- [ ] 3. Proposed changes to the Technical Specifications or Operating License.
- [ ] 4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- [ ] 5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
- [ ] 6. All REPORTABLE EVENTS (LERs only).
- [ ] 7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
- [ ] 8. All changes to the Station Emergency Plan prior to implementation.
- [ ] 9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

- [ ] 10. Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

ATTACHMENT G (Page 1 of 8)
10CFR50.59 SAFETY EVALUATION

GENERAL INFORMATION:

Safety Evaluation Number: SE - 94 - 043

Document Identifier: Work Request Q11619 & Q11620, SESR #4-2156
(Modification, Temp. Alt., Work Request Number, etc.)

Unit(s): 1 | System(s): 1100

Applicable Plant Mode(s): All modes
(Run, Startup/Hot Standby, Refuel, Shutdown)

Plant Mode Restriction(s): No restrictions

List Multiple Procedures Affected Below:

Procedure Number	Procedure Number	Procedure Number	Procedure Number
None			

CHANGE DESCRIPTION:

1. Describe the proposed change:

The installation of a larger U-bolt (5/8" versus 1/2") on pipe support M-987D-75 because of increased weight of a parts upgrade of the Standby Liquid Control (SBLC) Accumulators for Unit 1. The replacement accumulators evaluation is ME-93-0541-00, Revision 2.

2. Reason for the change:

The original accumulators are no longer available from the manufacturer and the replacement accumulator weighs 95 lbs. which is more than 67 lbs., the weight of the original accumulator. This increase in weight has been evaluated (SESR 4-2156) and requires a larger U-bolt (5/8" versus 1/2") be installed on pipe support M-987D-75 to support design loads.

3. Is the change:

X	Permanent	
	Temporary - Expected Duration:	

ATTACHMENT G (Page 2 of 8)
10CFR50.59 SAFETY EVALUATION

REFERENCE DOCUMENTS:
4. List reference documents used which describe the structure, system, or component. Identify documents referenced even if no information was found in that section.
a. UFSAR Section(s): 3.0, 3.2, 3.7, 3.9, 4.6, 9.3.5
b. SER Section(s): None
c. Tech. Spec. Section(s): 3.4/4.4
d. Fire Protection Program Document Pkg Section(s): None
e. Code of Federal Regulations Section(s): None
f. Regulatory Guides/NUREGs: None
g. Other: DBD-QC-139, Rev A; Vetip manual C0116; Drawing C68514-200, Drawing M-40, Rev AG

EVALUATION:

5. Describe how plant operation is affected when the structure, system, or component function is changed as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems, or components.

The SBLC operates in the same manner as prior to installation of the heavier accumulator and larger U-bolt on support M-987D-75. The larger U-bolt performs the same functions as the smaller U-bolt. The function is to provide support in a seismic event and restrain pipe movement. The interactions with other structures, systems and components does not SIGNIFICANTLY change. AWK 5/29/99

6. Describe how the change will affect equipment failures. Describe any new failure modes and their impact during all applicable operating modes.

The larger U-bolt will not affect any equipment failures that had not been considered in earlier evaluations. There are no new failure modes which would occur from the installation of a larger U-bolt. The larger U-bolt performs the same design function as the U-bolt being replaced. A larger U-bolt is required due to the increased weight of the new accumulator.

ATTACHMENT G (Page 3 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis.
- The changed structure, system, or component is explicitly or implicitly assumed to function during or after the accident.
- Operation or failure of the changed structure, system, or component could lead to the accident.

Anticipated Transient Without SCRAM	UFSAR Section 9.3.5
(includes seismic)	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting, or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine factors affecting the specification, it is necessary to review the UFSAR and SER where the Technical Specification Bases section does not explicitly state the basis.

Standby Liquid Control System	Technical Specification 3.4/4.4

9. Will the change involve a Technical Specification revision?

	YES
X	NO

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indiocate that a Technical Specification revision is required.

ATTACHMENT G (Page 4 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of these pages to answer the following questions for each accident listed in Step 7. Provide rationale for all NO answers.

Affected accident: Anticipated Transient Without Scram (ATWS) (includes seismic)	UFSAR Section: 3.9

a. May the probability of the accident be increased?

	YES
X	NO

The larger U-bolt and increased weight from the replacement accumulators do not increase the probability of the ATWS accident. The SBLC system is required in the event that an accident would occur.

b. May the consequences of the accident (off-site dose) be increased?

	YES
X	NO

The consequences of the accident are not increased. The SBLC system will perform its design function with the increased weight and installation of the larger U-bolt. This is based upon SESR 4-2156, which evaluated the increased weight of the replacement accumulator.

ATTACHMENT G (Page 5 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

c. May the probability of a malfunction of equipment important to safety increase?

	YES
X	NO

The probability of a malfunction of equipment important to safety does not change. The U-bolt function does not change and therefore does not change the design basis function of the SBLC system. SESR 4-2156 evaluated the design loading for SBLC piping and determined that with the larger U-bolt, the SBLC will meet loading requirements. Therefore, the probability of a malfunction of equipment does not increase.

d. May the consequences of a malfunction of equipment important to safety increase?

	YES
X	NO

The consequences of a malfunction of the U-bolt remain the same. The function of the larger U-bolt is the same function as the smaller U-bolt. The larger U-bolt operates in the same manner as the smaller U-bolt.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 6 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

11. Based on the answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

	YES
x	NO

Describe the rationale for your answer.

The larger U-bolt does not create the possibility of an accident or malfunction of a type different from those evaluated in the USFAR. The larger U-bolt functions in the same manner as the smaller U-bolt it is replacing. The higher weight of replacement SLC accumulators requires the larger U-bolt to maintain the system seismically. Sizing of the U-bolt has been evaluated by seismic calculation (SESR 4-2156).

If any answer to Question 11 is YES, then an Unreviewed Safety Question exists.

ATTACHMENT G (Page 7 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists. Proceed to Step 14.

Technical Specification: Standby Liquid Control System, 3.4/4.4

Determine which of the following is true for the above specification:

X	All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. The actual acceptance limit need not be identified to determine that no reduction in margin of safety exists. Proceed to Step 13.
	The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s) or margin(s) below.
	The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit or margin for the Margin of Safety determination. List the limit(s) or margin(s) below.
	The change does not affect any parameters upon which the Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to Step 14.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits identified in Step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

SESR 4-2156 evaluated the increase in weight of the replacement accumulator and the replacement of a larger U-bolt on pipe support M-987D-75. The evaluation determined that these changes were within the design loadings of the SBLC system.

ATTACHMENT G (Page 8 of 8)
10CFR50.59 SAFETY EVALUATION

EVALUATION (cont'd):	
14. Check one of the following:	
	An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The Proposed change **MUST NOT** be implemented without NRC approval.
X	No unreviewed Safety Question will result (Steps 10, 11, and 13) **AND** no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.
	A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Indicate applicable type(s) below:
	The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.
	The change is a plant modification or minor plant change. Indicate applicable type(s) below:
	A revision to an existing Technical Specification is required. The change **MUST NOT** be installed until receipt of an approved Technical Specification revision.
	The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize the installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.
	Nuclear Licensing has authorized installation, but no operation, prior to receipt of the NRC approval of License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result, and provides authority for installation only.
Preparer/Date: [signature] 5/24/94	
15. Documentation is adequate to support the above conclusion and the conclusion is valid.	
Reviewer/Date: James W. Wething 5/25/94	
16. Obtain a Safety Evaluation number from the Systems Engineering Clerk. Record on Page 1.	
17. Leave 1 Safety Evaluation copy with Systems Engineering Clerk. File original with package.	
18. Forward Safety Evaluation copy to FSAR Coordinator (ANI Audit Recommendation 88-1). Completed: Systems Engineering Clerk Initials: DK Date: 5-25-94	

(final)

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: M04-0-90-003	Date: 5/11/94
Subject: CRD REPAIR ROOM A/C INSTALLATION	
Submitted by: RICK GEIGER	

FOR REVIEW:

	1.	Safety Evaluations **NOT** involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a.	Changes to procedures as described in the Safety Analysis Report.
☐	b.	Changes to equipment or systems as described in the Safety Analysis Report.
☐	c.	Tests or experiments **NOT** described in the Safety Analysis Report.
	2.	Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a.	Procedure changes.
☐	b.	Equipment or system changes.
☐	c.	Tests or experiments.
☐	3.	Proposed changes to the Technical Specifications or Operating License.
☐	4.	Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5.	Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6.	All REPORTABLE EVENTS (LERs only).
☐	7.	All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8.	All changes to the Station Emergency Plan prior to implementation.
☐	9.	All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

☒	10.	Other OSR Items/Documents **NOT** addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) **for information only**. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

10CFR50.59 SAFETY EVALUATIONS

Safety Evaluation Number: SE-91-561

Document Identifier: M04-0-90-003
(Modification, Temp Alt, Work Request Number, etc.)

Unit(s): 1

System(s): 5700

Applicable Plant Mode(s): ALL Modes Apply
(RUN, STARTUP/HOT STNBY, REFUEL or SHUTDOWN)

1. Describe the proposed change:

   Provide cooling for the CRD Repair Room. Modification will install an air cooled condensor located outside the room and an air handling unit located inside the room. Electrical power will be supplied from a GE MCC which will replace the existing Westinghouse MCC 42R-2-1.

2. Reason for the change:

   Provide cooling for maintenance personnel when working on control rod drives. This is a Performance Enhancement Program item.

3. Is the change:

   (x) Permanent

   (   ) Temporary - Expected Duration: ______
   Plant Mode(s) Restrictions: ______

4. List the reference documents reviewed which describes the structure, system or component. (Identify documents referenced even if no information was found in that section.)

   a. UFSAR Section(s): 1.3.5, 10.10.1, 12.2.2.8

   b. SER Section(s): None.

   c. Tech Spec Section(s): 3.10/4.10.D

   d. Fire Protection Program Document Pkg Section(s): FHA ~~4.1.4~~ 4.1.2 / Dec. 1990

   e. Code of Federal Regulations Section(s): None.

   f. Regulatory Guides (NUREGs): IE Bulletin 80-11.

APPROVED
DEC 31 1990
Q.C.O.S.R.

5. Describe how the change will affect plant operation when the changed structure, system or component function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems or components.

   See Attachment.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   See Attachment.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

   - The change alters the initial conditions used in the UFSAR analysis
   - The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident
   - Operation or structure, system or component failure of the changed structure, system or component could lead to the accident

ACCIDENT	UFSAR SECTION
None.	

8. List each Technical Specification (Safety LImit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected.

   None.

APPROVED

DEC 3 1 1990

Q.C.O.S.R.

9. Will the change involve a Technical Specification revision?

( ) Yes (X) No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing step 14, indicate that a Technical Specification revision is required.

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of this page to answer the following questions for each accident listed in step 7. Provide the rationale for all NO answers.

Affected accident N/A UFSAR Section: ______

May the probability of the accident be increased? ( ) Yes ( ) No

May the consequences of the accident (off-site dose) be increased? ( ) Yes ( ) No

May the probability of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

May the consequences of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

<u>If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.</u>

APPROVED
DEC 3 1 1990
Q.C.O.S.R.

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

( ) Yes (X) No

Describe the rationale for your answer.

Discussed in response to Questions 5 and 6.

<u>If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.</u>

APPROVED
DEC 3 1 1990
Q.C.O.S.R.

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in step 8. If no technical Specifications are impacted, then no reduction in margin of safety exists, proceed to step 14.

Technical Specification N/A

Determine which of the following is true for the above specification:

( ) All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists, proceed to question 13.

( ) The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) below.

( ) The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination. List the limit(s)/margin(s) below.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

<u>If a Margin of Safety is reduced, an Unreviewed Safety Question exists.</u>

APPROVED
DEC 3 1 1990
Q.C.O.S.R.

14. Check one of the following:

( ) An Unreviewed Safety Question was identified in step 10, step 11, or step 13. The proposed change MUST NOT be implemented without NRC approval.

(X) No Unreviewed Safety Question will result (steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

( ) A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

( ) The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

( ) The change is a plant modification or minor plant change. Mark below as applicable.

( ) A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

( ) The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

( ) Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Preparer H. M. Stark 11/20/91
Signature Date

APPROVED
DEC 3 1 1990
Q.C.O.S.R.

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer DCBucknell 11-22-91
Signature Date

16. Obtain a safety evaluation number and note at top of page 1.

17. Forward a copy of this Safety Evaluation to the FSAR Coordinator, Monthly Report Coordinator and Tech Spec Coordinator. (ANI Audit, August 1990).

Completed: Initial ZMJ Date 11/22/91

APPROVED
DEC 3 1 1990
Q.C.O.S.R.

5. This modification will install an air conditioning system for the CRD Repair Room. The air handling unit, plenum, return grill and thermostat will be installed inside the CRD repair ante room. The air cooled condensor will be installed outside the CRD Repair Room. This equipment will be powered from a 480 V GE MCC which will replace the existing Westinghouse MCC 42R-2-1. Operation of the CRD Repair Room A/C System will offset the constant addition of heat incurred during maintenance on control rod drives. The design includes locally mounted disconnect switches for periods when this equipment will not be required.

6. Possible new failure modes or unacceptable conditions include:

   (a) Electrical failures in the new equipment.
   (b) Leaks in the new refrigerant lines.
   (c) Failures in the modified block walls.
   (d) Spread of contamination.

   Possible impact of the above failures during all operating modes are:

   (a) The electrical requirements for this modification include the installation of properly sized breakers in non-safety related MCC 42R-2-1 to protect existing plant electrical equipment from any faults which may occur in the new hvac equipment. MCC 42R-2-1 receives electrical power from non-safety related transformer T42R-2. The only loads on MCC 42R-2-1 will be the CRD Repair Room HVAC System. Therefore, a fault in the new electrical equipment will result in the tripping of breakers in MCC 42R-2-1 which will have no impact on any other plant equipment.

   (b) A leak in the refrigerant lines installed by this modification would result in the release of refrigerant-22 into the Unit 1 Reactor Building. The Reactor Building Ventilation System, designed to produce a negative differential pressure, evacuates the Reactor Building at a rate of approximately 1 free volume/hour. Therefore, leakage of refrigerant into the Reactor Building free volume would have no credible impact from a human safety standpoint and have no impact on equipment operation.

   (c) The structural requirements for this modification include design changes to the west (blocking-in an existing louver opening) and north (installation of electrical supply and refrigerant supply and return lines) block walls. As part of the designer's walkdown, it was identified that no safety related equipment was attached to these two block walls. The actual design will require structural changes meet the seismic 2-over-1 criteria but, if a failure of the wall were to occur, no safety related equipment would be affected.

(d) Increased local air flow from the air handling unit could result in unacceptable spread of contamination. The location of the air handling unit inside the ante room instead of the CRD Repair Room provides the highest air flow in the area of least contamination to prevent an unacceptable airborne contamination problem. Blocking-in the louver opening seals the ante room to prevent the spread of contamination to an uncontrolled area.

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number:	Date: 5-19-94
Subject: DCR 4-93-205	
Submitted by: Alice Beverth	

FOR REVIEW:

1. Safety Evaluations <u>NOT</u> involving an unreviewed safety question as defined in 10CFR50.59 for:
   - [ ] a. Changes to procedures as described in the Safety Analysis Report.
   - [ ] b. Changes to equipment or systems as described in the Safety Analysis Report.
   - [ ] c. Tests or experiments <u>NOT</u> described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - [ ] a. Procedure changes.
   - [ ] b. Equipment or system changes.
   - [ ] c. Tests or experiments.
- [ ] 3. Proposed changes to the Technical Specifications or Operating License.
- [ ] 4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
- [ ] 5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
- [ ] 6. All REPORTABLE EVENTS (LERs only).
- [ ] 7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
- [ ] 8. All changes to the Station Emergency Plan prior to implementation.
- [ ] 9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

- [X] 10. Other OSR Items/Documents <u>NOT</u> addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Technical Specification Revisions for Modification

Station Quad Cities

Unit(s) 1&2

Modification # DCR 4-93-205

To: ______________ (Systems Design Superintendent)

J. Shrage (NLA)

N. Chrissotimos (Station Regulatory Assurance Supervisor)

List required Technical Specification revisions:

No Technical Specification revision is required as a result of this DCR.

Recommend effective date for revision (i.e., calendar date, beginning of outage #, or end of outage #)

Prepared by: KHA Scott Darm Date: 5/18/94

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   DCR 4-93-205, NWR Q08212, NWR Q08293

2. Describe the proposed change and the reason for the change.

   The torus level indication was found to be in error and the source was traced to the faulty Narrow Range Level Transmitter (LT-001-1602-9). NWR Q08293 replaced the Rosemount Model 1151DP3B12 (obsolete designation) with a Rosemount Model 1151DP3G12M1B1. This model includes the mounting bracket which was previously ordered separately and an optional integral meter.

   The Reactor Building Exhaust Fan 2C was auto-tripping and the source was traced to the setpoint of differential pressure switch (DPS-002-5741-261C) being at the low end of the switch's range. NWR Q08212 replaced the Dwyer Model 1821-2 with Dwyer Model 1823-1. This model has a range of 0.3" to 1.0" water column (WC) whereas the previous model's range was 0.5" to 2.0" WC. The setpoint of 0.5" WC remains unchanged.

   DCR 4-93-205 updates the appropriate data sheets to reflect these changes.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ______

   AND
   Plant Mode(s) restrictions while installed ______
   (NONE if no plant mode restrictions apply) ______

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   3.4.1.2, "Internal Flood Measures"
   7.5.3, "Safety Parameter Display System"
   9.4.7, "Reactor Building Ventilation System"
   15.6.5.4.4, "Fission Product Release from Reactor Building to Atmosphere"

Exhibit E
10CFR50.59 SAFETY EVALUATION

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

Torus Narrow Range Level Transmitter:

The "B" designator in the old transmitter model number indicates 10-50 mA DC output. The "G" designator of new model number indicates 10-50 mA DC output which is the same as the old value and is the same as the value given in the instrument data sheet.

The new transmitter (with bracket and integral meter) weighs 13.34 lb. while the old transmitter and bracket weighs 12 lb and 1.12 lb. = 13.12 lb. This weight difference is negligible.

The mounting of the transmitter is identical for both transmitter models.

Per the Master Equipment List, Rev. 30, the transmitter is non-EQ.

The Level Transmitter change will not affect plant operation. the new LT has the same accuracy and function as the original. The new LT provides local level indication and a level signal to an indicator and recorder in the control room. The local display meter is an enhancement. The LT has no control function.

Reactor Building Exhaust Fan Pressure Switch:

The existing pressure switch is Dwyer Model 1821-2 while the new switch is Dwyer Model 1823-1. The differences between the new model and the old model are the approving agencies and the instrument range.

Model 1821-2 is UL Safety Control listed only and Model 1823-1 is UL, CSA and FM approved. Since there are no commitments for approving agencies, this difference is acceptable.

The old switch, Model 1821-2, has a range of 0.5" to 2.0"

Exhibit E
10CFR50.59 SAFETY EVALUATION

Water Column (WC) with a setpoint of 0.5" WC. This model will reset at 0.6" WC. The new switch, Model 1823-1, has a range of 0.3" to 1.0" WC. The setpoint remains at 0.5" WC and this model will reset at 0.58" WC. The deadband is slightly narrower for the new switch. This will have an insignificant affect on system operation. Having the setpoint in midrange instead of at the low end will enhance system operation by reducing or eliminating spurious trips.

The Differential Pressure Switch change will not affect plant operation. All dimensions, materials and method of operation are the same. The switch has the same set point and function as the original and better accuracy than the original.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

This change will not affect equipment failures nor will it introduce any new failure modes. The replacement LT has the same method of operation, accuracy and performance as the original.

The replacement DPS with a setpoint of 0.5" WC and a range of 0.3" to 1.0" WC will enhance system operation by reducing or eliminating spurious trips.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
-Internal Flood Measures	3.4.1.2
-LOCA	7.5

Exhibit E
10CFR50.59 SAFETY EVALUATION

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

   The torus level measurement requirements are referenced in Sections 3.2/4.2E and 3.7/4.7.A.1 and Tables 3.2-4 and 4.2-2.

9. Will the change involve a Technical Specification revision?

   [ ] Yes [X] No

   If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident -Internal Flood
-LOCA

SAR Section: 3.4.1.2
7.5

May the probability of the accident be increased? [ ] Yes [X] No

The replacement LT has the same accuracy and function as the original. The function of the LT is to provide local level indication and a level signal to an indicator and recorder in the Control Room. The LT has no control function.

The new DPS provides a fan trip function under no flow conditions. The old switch has a range of 0.5" to 2.0" Water Column (WC). Since the setpoint for the switch is 0.5" WC, a Model 1823-1 with a range of 0.3 to 1.0 in. WC will enhance system operation by reducing or eliminating spurious trips. Therefore, the pressure switch replacement will not increase the probability of an accident.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The new LT has the same accuracy and function as the original. The replacement LT will function as originally designed under all operating and accident conditions. Thus, the consequences of the accident are not affected by this change.

The new DPS has the same setpoint, accuracy and function as the original. The replacement DPS will function as originally designed under all operating and accident conditions. Thus, the consequences of the accident are not affected by this change.

Exhibit E
10CFR50.59 SAFETY EVALUATION

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The LT functions the same as the original transmitter. Thus, this change does not increase the consequences of a malfunction of equipment important to safety.

The DPS with a setpoint of 0.5" WC and with a range of 0.3 to 1.0" WC will enhance system operation by reducing or eliminating spurious trips. Therefore, the pressure switch replacement will decrease the probability of a malfunction of equipment important to safety.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The LT and DPS function the same as the original instruments. Thus, this change does not increase the consequences of a malfunction of equipment important to safety.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

The instrument model changes will not affect the function or operation of the systems since the replacement instruments function the same as the original instruments.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.
3.2/4.2

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

The design of the post-accident instrumentation system and components as described in Section 3.2/4.2 has not been changed or modified by this DCR. The parameters used to establish the Technical Specifications have not changed.

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ____________________

SAR Section ____________________

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

SER Section ______

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

N/A

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Mod # DCR 4-93-205

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer KJA Scott Dam 5/18/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer ClAR 5/19/94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number:	Date: 5-6-94
Subject: DCR 4-94-046	
Submitted by: Alice Beverath	

	FOR REVIEW:	
	1.	Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
[ ]	a.	Changes to procedures as described in the Safety Analysis Report.
[ ]	b.	Changes to equipment or systems as described in the Safety Analysis Report.
[ ]	c.	Tests or experiments NOT described in the Safety Analysis Report.
	2.	Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
[ ]	a.	Procedure changes.
[ ]	b.	Equipment or system changes.
[ ]	c.	Tests or experiments.
[ ]	3.	Proposed changes to the Technical Specifications or Operating License.
[ ]	4.	Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
[ ]	5.	Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
[ ]	6.	All REPORTABLE EVENTS (LERs only).
[ ]	7.	All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
[ ]	8.	All changes to the Station Emergency Plan prior to implementation.
[ ]	9.	All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.
	FOR INFORMATION:	
[X]	10.	Other OSR Items/Documents NOT addressed above.
		This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Technical Specification Revisions for Modification

Station Quad Cities

Unit(s) 1 & 2

Modification # DCR 4-94-046
Standby Gas Treatment

To: ______________________ (Systems Design Superintendent)

J. Shrage ______________________ (NLA)

N. Chrissotimos ______________________ (Station Regulatory Assurance Supervisor)

List required Technical Specification revisions:

None

Recommend effective date for revision (i.e., calendar date, beginning of outage #, or end of outage #)

Prepared by: Scott Davis Date: 5/3/94

U 5/4/94

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   DCR 4-94-046 Standby Gas Treatment (SBGT) System

2. Describe the proposed change and the reason for the change.

   This DCR revises the Master Equipment List (MEL) and selected drawings to incorporate the results of Component Classification (CC) of the Standby Gas Treatment (SBGT) System. As part of this DCR, 1) no physical change was made to any plant structure, system equipment or component and 2) some components were upgraded from NSR to SR because they are required for the SBGT system to perform its SR function (Secondary Containment Radioactive Effluent Control). Documentation specifically addressing these changes is included in Component Classification Binder # CC-QC009. The CC program is an ongoing controlled program that is supervised by Station Engineering.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ____________

   AND
   Plant Mode(s) restrictions while installed ____________
   (NONE if no plant mode restrictions apply) ____________

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   6.0.1.4 "Engineered Safeguard Features - Standby Gas Treatment System"
   6.2 "Engineered Safeguard Features - Containment Systems"
   6.5 "Fission Product Removal and Control Systems"
   15.6.2 "Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment"
   15.6.5 "Loss of Coolant Accidents Resulting from Piping Breaks Inside Containment"
   15.7.2 "Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings"

Exhibit E
10CFR50.59 SAFETY EVALUATION

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   No physical change was made to any plant system, structure, equipment or component. The component classification program specifically addressed the effect of the drawing and classification changes on the SBGT system safety function and operating modes. Documentation of this is included in the SBGT system component classification binder. Plant operation is not affected by this DCR.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   There were no physical changes made to any plant system, structure, equipment or component by this DCR. The changes documented in this DCR do not create any new operating or failure modes and have no impact on any existing operating or failure modes. Equipment failures are not affected by this DCR.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment	15.6.2
Loss of Coolant Accidents Resulting from Piping Breaks Inside Containment	15.6.5

Exhibit E
10CFR50.59 SAFETY EVALUATION

Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings	15.7.2

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

   The SBGT system and components are described in Technical Specifications Section 3.7/4.7. As part of this DCR, no physical change was made to any plant system, structure, equipment or component. The SBGT component classification process determined that the drawing and classification changes made by this DCR did not alter the safety limits or other parameters used to establish the Technical Specifications. No Technical Specifications are affected by this DCR.

9. Will the change involve a Technical Specification revision?

   [ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident Break in Reactor Coolant Pressure Boundary Instrument Line Outside Containment

Loss of Coolant Accident (LOCA) Resulting from Piping Breaks Inside Containment

Design Basis Fuel Handling Accidents Inside Containment and Spent Fuel Storage Buildings

SAR Section: 15.6.2

15.6.5

15.7.2

May the probability of the accident be increased? [ ] Yes [X] No

This DCR does not involve any physical changes to plant systems, structures, equipment, or components. The Component Classification (CC) process evaluated all SBGT system components and identified the operating mode required for each component to accomplish the SBGT system safety function. As a result of the SBGT system CC process, several components were reclassified from NSR to SR. The effect of these classification changes was evaluated through the CC process and was found to have no impact on either the SBGT system safety function or on the accident scenarios analyzed in the UFSAR. The CC process provides assurance that the probability of an accident is not increased

Exhibit E
10CFR50.59 SAFETY EVALUATION

due to the component classification changes. Furthermore, the CC process provides assurance that these changes do no alter the initial conditions used in any FSAR accident analysis. This CC process is documented in the SBGT system CC binder.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The Component Classification (CC) process evaluated all SBGT system components and identified the operating mode required for each component to accomplish the SBGT system safety function and to mitigate the accidents analyzed in the UFSAR. As part of the CC process, several components were reclassified from NSR to SR. These classification changes were evaluated through the CC process and were found to have no impact on the SBGT system's ability to mitigate the affects of an accident. The CC process provides assurance that the consequences of an accident are not increased due to the changes in component classification. This CC process is documented in the SBGT system CC binder.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The SBGT system Component Classification (CC) process considered all possible equipment and component malfunctions in determining the classification of each SBGTS system component. As part of the CC process, several components were reclassified from NSR to SR. The classification changes were evaluated through the SBGT system CC process and were found not to have any impact on the SBGT system. The CC process provides assurance that the probability of a malfunction in equipment important to safety is not increased as a consequence of the component classification changes. The function of the SBGT system and its ability to operate are unchanged.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The Component Classification (CC) process identified the operating and failure modes of all SBGT system components and their role in accomplishing the SBGT system safety function. As part of CC process, several components were reclassified from NSR to SR. These classification changes were evaluated through the CC process and were found to have no impact on the SBGT system. The CC process provides assurance that the consequences of a malfunction in equipment important to safety are not increased

Exhibit E
10CFR50.59 SAFETY EVALUATION

due to the changes made by this DCR. Results of the CC process for the SBGT system are recorded in the CC binder for the SBGT system.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

This DCR does not involve any physical changes to plant systems, structures, equipment or components. The Component Classification (CC) process for the SBGT system identified the operating mode for each component in the system and also identified that component's role in accomplishing the SBGT system safety function. The CC process also considered all applicable accidents analyzed in the SAR and all potential equipment or component malfunctions. The CC process provides assurance that the changes made by this DCR do not affect any existing accidents analyzed in the SAR and do not create any new accidents. The SBGT system CC process is documented in the SBGT system CC binder.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.
Technical Specification Section 3.7/4.7

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

The design basis for the SBGT System and components, as described in Technical Specification section 3.7/4.7, has not been changed or modified by this DCR. The parameters and limits used in the Technical Specifications are not changed.

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ______

SAR Section ______

Exhibit E
10CFR50.59 SAFETY EVALUATION

SER Section ______________________________

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

N/A

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer [signature]
(Cognizant Engineer)

5/3/94
Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer [signature]
(Design Superintendent/Supervisor)

5/4/94
Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number:	Date: 5-6-94
Subject: DCR 4-94-055	
Submitted by: Alice Beveroth	

FOR REVIEW:

	1. Safety Evaluations <u>NOT</u> involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments <u>NOT</u> described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

☒	10. Other OSR Items/Documents <u>NOT</u> addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Technical Specification Revisions for Modification

Station Quad Cities

Unit(s) 1&2

Modification # DCR 4-94-055

To: ______________ (Systems Design Superintendent)

J. Shrage (NLA)

N. Chrissotimos (Station Regulatory Assurance Supervisor)

List required Technical Specification revisions:

There are no required revisions to the Technical Specifications as a result of this change.

Recommend effective date for revision (i.e., calendar date, beginning of outage #, or end of outage #)

Prepared by: Scott Davis Date: 5/4/94

5/4/94

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   DCR 4-94-055

2. Describe the proposed change and the reason for the change.

   The implemented change will incorporate the actual location of pressure test point connections for the condensate booster pump discharge piping on Unit 1 Piping and Instrumentation Diagram (P&ID); and incorporate the addition of pressure test point connection for the condensate booster pump discharge piping on Unit 2 P&ID. These Unit 1 and Unit 2 P&ID as-built changes reflect the original designed and installed conditions.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ______

   AND
   Plant Mode(s) restrictions while installed ______
   (NONE if no plant mode restrictions apply) ______

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   Functional aspects or activities related with the Condensate or Condensate Pump Room are described in the following UFSAR Sections:

   1.2.2.2, "Station Arrangements"
   3.4.1.2.1, "Protection of the Condensate Pump Room and Residual Heat Removal Service Water Pump Rooms"
   3.6.1.1.2, "High Energy Systems"
   5.1.3, "Reactor Coolant System Subsystems"
   5.4.7.2.3, "Other Functions of the Residual Heat Removal System"
   6.3.3.2.6, "Summary - Integrated Emergency Core Cooling System Performance Evaluation"
   6.3.3.2.8.1, "Small Line Break"
   7.7.6, "Main Condenser, Condensate and Condensate Demineralizer"
   9.2.8.2, "System Description - Standby Coolant Supply System"

Exhibit E
10CFR50.59 SAFETY EVALUATION

10.1, "Summary Description - Steam and Power Conversion System"
10.4.7, "Condensate and Feedwater System"
Table 10.4-3, "Condensate Booster Pump Characteristics"
11.1.3.7, "Tritium"
Table 11.1-7, "Turbine Building Equipment Drain Sump Sources For Radioactive Material"
Table 12.3-3, "Quad Cities Unit 1 Area Radiation Monitoring System Sensor Location and Range"
Table 12.3-4, "Quad Cities Unit 2 Area Radiation Monitoring System Sensor Location and Range"
14.2.12.1.32.2, "Condensate and Feedwater Systems"

UFSAR Accident Analysis Sections Pertaining to Condensate:

15.8, "Anticipated Transients Without SCRAM"

Other Documents:

Grinnell Erection drawing number 1-3401-ED-1 Revision 01/30/70
Grinnell Erection drawing number 2-3401-ED-1 Revision 08/31/71
Quad Cities Special Report 3A - Condensate Pump Room Modifications (Permananet Flood Protection of the RHR Service Water Pumps and Diesel Generator Cooling Water Pumps)

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

The relocation or addition of the Unit 1 and Unit 2 condensate booster pump discharge pressure test point connection does not produce any functional change in the system. It only revises the Unit 1 and Unit 2 P&IDs to reflect the original designed and installed conditions for pressure testing tap points. Implementation of these changes will not alter any operational parameters of the system or the plant, and therefore will not affect current plant operation.

Exhibit E
10CFR50.59 SAFETY EVALUATION

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

This change does not add any new components to the system, but reflects actual pressure tap locations which were installed in 1970 and 1971 for Units 1 and 2, respectively, as shown on the original Grinnell erection drawings. The operational characteristics of the system will not be affected by this P&ID drafting change, so there is no potential for introduction of any circumstances or conditions that could produce a failure mechanism that did not previously exist.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
-Loss of Normal AC Power	15.8.2
-Loss of Normal Feedwater Flow	15.8.3

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

The applicable Safety Limits, Limiting Safety System Settings and Limiting Conditions for Operation are not directly related to, nor do they mention the Condensate System piping and valves. Therefore, no Technical Specifications require revision as a result of this change. The effects from possible failure of condensate piping are described in Technical Specifications Sections 3.5/4.5 and 3.9/4.9. However the limiting conditions stated for condensate pump room flood protection and liquid radioactive effluents are not affected.

Exhibit E
10CFR50.59 SAFETY EVALUATION

9. Will the change involve a Technical Specification revision?

[ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident Loss of Normal AC Power

SAR Section: 15.8.2

May the probability of the accident be increased? [ ] Yes [X] No

The probability of a Loss of Normal AC Power event is independent of the function or operation of the Condensate System. This change can not increase the probability of the initiating event for the Loss of Normal AC Power.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The loss of Normal AC Power would deenergize the condensate system pumps. Therefore, the possiblity of this change affecting condensate pump operation has previously been analyzed. The potential consequences of this accident which affects system operation and off-site dose are not increased.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The incorporation of the originally designed and installed pressure test connections will not increase the probability of a malfunction of equipment important to safety due to Loss of Normal AC Power. The affects of Loss of Normal AC Power has previously been analyzed which would deenergize the normally operating equipment including the condensate pump system.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The change as described will not affect any operational parameters of the Condensate System. The consequences of a malfunction of equipment important to safety affected by the Loss of Normal AC Power will not increase.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

Affected accident Loss of Normal Feedwater Flow

SAR Section: 15.8.3

May the probability of the accident be increased? [ ] Yes [X] No

The probability of a Loss of Normal Feedwater Flow will not increased due to the incorporation of the originally designed and installed pressure test connections.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The loss of condensate resulting from system failure would result in Loss of Normal Feedwater Flow. This loss of feedwater flow has previously been analyzed. The potential consequences of this accident which affects system operation and off-site dose are not increased.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

This change will not increase the probability of a malfunction of equipment important to safety due to Loss of Normal Feedwater Flow. Failure of this change could result in condensate pump room flooding, however this change does not involve any new components therefore the probability is not increased.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The change as described will not affect any operational parameters of the Condensate System. The consequences of a malfunction of equipment important to safety affected by the Loss of Normal Feedwater Flow or condensate pump room flood will not increase. The effects and preventative measures of a condensate pump room flood has previously been analyzed for its effects on equipment important to safety.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

The change as described does not cause a functional change in the system or its interaction with other plant systems. It does not alter any physical parameters or process variables of the plant. Due to the nature of the change, there are no new inherent failure modes introduced to the system and the change does not add any new components or process routes.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.
Technical Specification Sections 3.5/4.5 and 3.9/4.9.

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

The effects from possible failure of the Condensate System and components as described in Technical Specification Section 3.5/4.5 and 3.9/4.9 have not been changed or modified by this DCR. The parameters and limits used in the Technical Specifications are not changed.

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ______________________________

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

SAR Section

SER Section

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

N/A

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a p[illegible] modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Mod # DCR 4-94-055

Station/Unit Quad Cities / 1&2

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer Scott Davis 5/4/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer ______ 5/4/94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number:	Date: 5-6-94
Subject: DCR 4-94-064	
Submitted by: Alice Beverett	

FOR REVIEW:

	1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments NOT described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

X	10. Other OSR Items/Documents NOT addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Technical Specification Revisions for Modification

Station Quad Cities

Unit(s) 1 & 2

Modification # DCR 4-94-064

To: ______________ (Systems Design Superintendent)

J. Shrage (NLA)

N. Chrissotimos (Station Regulatory Assurance Supervisor)

List required Technical Specification revisions:

A revision to the Technical Specifications is not required.

Recommend effective date for revision (i.e., calendar date, beginning of outage #, or end of outage #)

Prepared by: SRB Scott Darin Date: 5/4/94

C 5/4/94

Mod # DCR 4-94-064

Station/Unit Quad Cities /1 & 2

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   DCR 4-94-064

2. Describe the proposed change and the reason for the change.

   This DCR was submitted to document the following as-builts:

   Schematic Diagrams 4E-1351B, Sheet 2; 4E-2345, Sheet 1; 4E-2345, Sheet 2; 4E-2430, Sheet 2; and 4E-2430, Sheet 4: These drawings update cross references and descriptions on relays and control contacts to more accurately reflect the installed conditions.

   Piping Diagram M-84, Sheet 1: This drawing revises the Equipment Piece Number (EPN) for the Unit 2A Off-Gas Filter Outlet Valve from 2-5499-55 to 2-5499-51. This change is being made to match the configuration and numbering of the Unit 1 valve.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ____________

   AND
   Plant Mode(s) restrictions while installed ____________
   (NONE if no plant mode restrictions apply) ____________

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   5.2, "Integrity of Reactor Coolant System"
   6.0.1.5, "Emergency Core Cooling System"
   6.2, "Containment Systems"
   6.3, "Emergency Core Cooling Systems"
   7.3, "Engineered Safety Features"
   8.0, "Electric Power"
   8.2, "Offsite Power Systems"
   8.3, "Onsite Power Systems"
   9.5, "Other Auxiliary Systems"
   10.4.2, "Main Condenser Evacuation System"
   11.3, "Gaseous Waste Management System"

Exhibit E
10CFR50.59 SAFETY EVALUATION

11.5, "Process & Effluent Radiological Monitoring & Sampling Systems"
15.2.2.1 "Load Rejection (Generator Trip) Without Bypass"
15.2.2.2 "Load Rejection With Bypass"
15.6 "Decrease in Reactor Coolant Inventory"

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   Updating the cross references on the Schematic Diagrams and the EPN on the Piping Diagrams to reflect the actual plant conditions will not impact any plant system. Updating the drawings will simplify operations and maintenance on the systems.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   Updating the Schematic Diagrams to correct cross references and relay designations on the Diesel Generator and Core Spray systems and revising the EPN on the Unit 2A Off-Gas Filter Inlet Valve to match the Unit 1 configuration does not introduce any new failure modes in these systems.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
-Loss of auxiliary power	8.3.1
-Power bus loss of voltage	8.3.1
-Failure of one diesel generator to start	8.3.1.6.4
-Load rejection without bypass	15.2.2.1

Exhibit E
10CFR50.59 SAFETY EVALUATION

-Load rejection with bypass (Loss of electrical load)	15.2.2.2
-Loss of Coolant	15.6.2, 15.6.5

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

The Core Spray system is addressed in the Technical Specifications (Tech Spec) Section 3.5/4.5. The Diesel Generator system is referenced in Tech Spec Section 3.9/4.9. Updating the drawing cross references and descriptions on relays and control contacts does not impact the Tech Specs.

The Off-Gas system is referenced in Tech Spec Section 3.8/4.8. Revising the EPN number on the drawing for the Unit 2A Off-Gas Filter Outlet Valve does not impact the Tech Specs.

9. Will the change involve a Technical Specification revision?

[ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident -Loss of auxiliary power
-Power bus loss of voltage
-Failure of one diesel generator to start
-Load rejection without bypass
-Load rejection with bypass (Loss of electrical load)
-Loss of coolant

SAR Section: 8.3.1
8.3.1
8.3.1.6.4
15.2.2.1
15.2.2.2
15.6.2, 15.6.5

May the probability of the accident be increased? [ ] Yes [X] No

The function of the Core Spray and the Diesel Generator Systems are unchanged by updating the drawing cross references and descriptions on relays and control contacts.

Likewise, the function of the Off-Gas System and its ability to operate are unchanged by the revision to the EPN on the Unit 2A

Exhibit E
10CFR50.59 SAFETY EVALUATION

Off-Gas Filter Outlet Valve. Information on this label required for updating station procedures is being coordinated by the station system engineers. The revised EPN on the drawing will provide consistency with the Unit 1 Off-Gas system.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The function of the Core Spray, Diesel Generator and Off-Gas Systems and their ability to operate are unchanged by the revision to the cross references and descriptions on relays and control contacts on the Schematic Diagrams and the change in EPN for the Unit 2A Off-Gas Filter Outlet Valve (2-5499-51). There is no change in any accident scenarios and no new failure modes are introduced by these changes.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The probability of equipment malfunction is unchanged because there is no physical change to the equipment or operating modes by revising the cross references and descriptions on relays and control contacts on the schematic diagrams or revising the EPN on the P&ID. Operations and maintenance will be enhanced by these revisions.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The probability of malfunction of any equipment or system due to the updating of the cross references and descriptions on relays and control contacts and by revising the EPN for valve 2-5499-51 is not increased and therefore the consequences of a malfunction of equipment important to safety are not increased. All systems will function as originally designed.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

No new accident scenarios are created by this DCR. The function of the Core Spray, Diesel Generator and Off-Gas Systems and their ability to operate are unchanged. This DCR will not adversely impact systems or functions nor will the possibility of an accident malfunction be created that is different from those previously evaluated in the SAR.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Mod # DCR 4-94-064

Station/Unit Quad Cities /1 & 2

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.
Technical Specification Sections 3.5/4.5, 3.8/4.8 and 3.9/4.9.

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

The parameters used to establish the Technical Specifications for the Core Spray, Diesel Generator and Off-Gas systems are not changed by this DCR. This DCR updates cross references and descriptions on relays and control contacts on the Core Spray and Diesel Generator systems and relabels equipment associated with the Off-Gas Filter Outlet Valve.

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ______________________

Exhibit E
10CFR50.59 SAFETY EVALUATION

SAR Section ______

SER Section ______

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

N/A

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Mod # DCR 4-94-064

Station/Unit Quad Cities /1 & 2

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer JRS Scott Darin 5/4/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer [signature] 5/4/94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE-93-109	Date: 5-17-94

Subject: Modification M04-1(2)-89-115 Service Water Radiation Monitor. A temporary flow indicator and pressure indicators are to be used to verify proper system operating pressures and to ensure proper eductor operation.

Submitted by: David Harmon

FOR REVIEW:

1. Safety Evaluations NOT involving an unreviewed safety question as defined in 10CFR50.59 for:
   - ☐ a. Changes to procedures as described in the Safety Analysis Report.
   - ☒ b. Changes to equipment or systems as described in the Safety Analysis Report.
   - ☐ c. Tests or experiments NOT described in the Safety Analysis Report.
2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
   - ☐ a. Procedure changes.
   - ☐ b. Equipment or system changes.
   - ☐ c. Tests or experiments.
3. ☐ Proposed changes to the Technical Specifications or Operating License.
4. ☐ Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
5. ☐ Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
6. ☐ All REPORTABLE EVENTS (LERs only).
7. ☐ All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
8. ☐ All changes to the Station Emergency Plan prior to implementation.
9. ☐ All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

10. ☐ Other OSR Items/Documents NOT addressed above.

This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

10CFR50.59 SAFETY EVALUATIONS

Safety Evaluation Number: SE- 93 - 109

Document Identifier: M04-1(2)-89-115 work packages Q02824, Q02825 Elect, Mech, Inst
(Modification, Temp Alt, Work Request Number, etc.)

Unit(s): 1(2)

System(s): 1700, 3900, 4200

Applicable Plant Mode(s): This evaluation is applicable for all plant modes.
(RUN, STARTUP/HOT STNBY, REFUEL or SHUTDOWN)

Plant Mode Restriction(s): This evaluation contains no mode restrictions.

1. Describe the proposed change:

The work to be performed under this package will calibrate two 0-100 psi pressure indicators (SI# 208039) and a flow switch (SI# 699224) prior to installation of modification M04-1(2)-89-115 (Modification of the service water radiation monitoring system sample delivery piping). The pressure indicators (PIs) will be used to ensure the sample stream eductor is operating properly. Under this package, the PIs will be used to gather system sample pressures while throttling the two globe valves on either side of the eductor. Also, during this test, a flow indicator will be installed to give flow indications. This information will be used to determine proper system operating pressures. The flow indicator will then be removed and the flow switch will be installed. The low flow setpoint will then be verified. If erratic indication occurs during performance of the traveler, individual instrument calibrations can be performed.

2. Reason for the change:

This work package was written to install the instrument portion of modification M04-1(2)-89-115.

3. Is the change:

(X) Permanent

(   ) Temporary - Expected Duration: ____________________

4. List the reference documents reviewed which describes the structure, system or component. (Identify documents referenced even if no information was found in that section.)

   a. UFSAR Section(s): Table 1.8-1, 3.0, 7.1, 7.5, 11.5.2.7, 15

   b. SER Section(s): None

   c. Tech Spec Section(s): 3.2/4.2, 3.8/4.8

   d. Fire Protection Program Document Pkg Section(s): None

   e. Code of Federal Regulations Section(s): None

   f. Regulatory Guides/NUREGs: 1.97

   g. Other: MasterEquipment List

5. Describe how the change will affect plant operation when the changed structure, system or component function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems or components.

   During the completion of this portion of the installation work, the service water radiation monitor (SWRM) will be out of service and grab samples will be drawn and analyzed every twelve hours in accordance with technical specification table 3.2-5. Potential effected systems include process radiation monitoring, service water, and domestic water. The process radiation monitoring system will only be affected at the SWRM. These effects are described above. The service water system will not be affected by this installation due to the SWRM system being OOS during the described installation. During SWRM system testing, the eductor will draw service water sample flow through the SWRM system and discharge back to the service water return header. This will result in no change from the present analyzed condition of the system. Domestic water will be used to drive the eductor. The flow will be stopped, started, and throttled, but no adverse effect will be made to the system.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   A possible failure mode of the PIs would be erratic indication. If this were to occur, system flow set-up could be affected (flow too high or low). Piping for domestic water and service water can withstand the peak system pressure which is domestic water. Possible failure modes of the flow switch is constant no flow and constant normal flow. Each of these conditions is easily detectable and will be detected by performing flow switch setpoint checks.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis
- The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident
- Operation or structure, system or component failure of the changed structure, system or component could lead to the accident

ACCIDENT	UFSAR SECTION
None	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. (To determine the factors affecting the specification, it is necessary to review the UFSAR and SER where the Bases Section of the Technical Specifications does not explicity state the basis).

3.2.G	

9. Will the change involve a Technical Specification revision?

( ) Yes (X) No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing step 14, indicate that a Technical Specification revision is required.

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of this page to answer the following questions for each accident listed in step 7. Provide the rationale for all NO answers.

Affected accident Not Applicable ______________ UFSAR Section: ____________________

May the probability of the accident be increased? ( ) Yes ( ) No

May the consequences of the accident (off-site dose) be increased? ( ) Yes ( ) No

May the probability of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

May the consequences of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

( ) Yes (X) No

Describe the rationale for your answer.

The answer to question 5 details the installation will have no effect on the three interconnecting systems assuming no equipment failures. The answer to question six explored the possible equipment failures and found no adverse impact on the three potentially affected systems. Installation and testing of this equipment cannot cause any plant accident or transient not described within the UFSAR. The installation does not alter the interconnecting systems so as to create abnormal lineups or operating modes. The installation will be passive with respect to the potential to initiate a different type of accident.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists, proceed to step 14.

   ~ nical Specification 3.2.G ______________________________

   De...mine which of the following is true for the above specification:

   ( ) All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the ...ual acceptance limit need not be identified to determine that no reduction in ... of safety exists, proceed to question 13.

   ( ) The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) below.

   ( ) The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request ...uclear Licensing assistance to identify the acceptance limit/margin for the ...argin of Safety determination. List the limit(s)/margin(s) below.

   (X) The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety - proceed to Question 14.

   List Acceptance Limit(s)/Margin(s) of Safety

   ______________________________ ______________________________
   ______________________________ ______________________________
   ______________________________ ______________________________
   ______________________________ ______________________________

13. Use the above limits identified in step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced, an Unreviewed Safety Question exists.

14. Check one of the following:

( ) An Unreviewed Safety Question was identified in step 10, step 11, or step 13. The proposed change MUST NOT be implemented without NRC approval.

(X) No Unreviewed Safety Question will result (steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

( ) A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

( ) The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

( ) The change is a plant modification or minor plant change. Mark below as applicable.

( ) A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

( ) The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

( ) Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Preparer Mark L. Budjon ____ June 10, 1993
Signature Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer [signature] 6/10/93
Signature Date

16. Obtain a safety evaluation number from the Tech Staff clerk and record it on page 1.

17. Leave one copy of the safety evaluation with the Tech Staff clerk and file the original with the applicable package(s)

18. The Tech Staff clerk will forward a copy of this safety evaluation to the FSAR Coordinator. (ANI Audit Recommendation 88-1)

Completed: DJC 6-11-93
Initial Date

(final)

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: SE-93-112	Date: 5-17-94
Subject: Modification M04-1(2)-89-115 Service Water Radiation Monitor. Demolish existing sample delivery system and install an eductor driven system using domestic water.	
Submitted by: David Harmon	

FOR REVIEW:

	1. Safety Evaluations <u>NOT</u> involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☒	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments <u>NOT</u> described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

☐	10. Other OSR Items/Documents <u>NOT</u> addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

10CFR50.59 SAFETY EVALUATIONS

Safety Evaluation Number: SE- 93 - 112

Document Identifier: M04-1(2)-89-115 work packages Q02824, Q02825 mechanical scope
(Modification, Temp Alt, Work Request Number, etc.)

Unit(s): 1(2)

System(s): 1700, 3900, 4200

Applicable Plant Mode(s): This evaluation is applicable for all plant operating modes.
(RUN, STARTUP/HOT STNBY, REFUEL or SHUTDOWN)

Plant Mode Restriction(s): This evaluation contains no mode restrictions.

1. Describe the proposed change:

The work to be performed under these packages will demolish the existing service water radiation monitor (SWRM) sample delivery system (receiver tank, pump, all associated piping and valves) and install a new, eductor driven system powered by domestic water. The sample system inlet isolation valve will be replaced and the service water return header will be open through a 1'-1/2" pipe to the turbine building 595' level during the replacement. Once installed, this valve will act as the isolation point for further installation work. Domestic water will be isolated for installation of the back flow preventer. All other items (skid, detector, eductor) will then be installed. A flow indicator will be installed to facilitate Instrument Maintenance work and testing on the flow switch and pressure gauges. The indicator will be removed and replaced with the switch. A leak test will then be performed.

2. Reason for the change:

This work package was written to install the mechanical portion of modification M04-1(2)-89-115.

3. Is the change:

(X) Permanent

(   ) Temporary - Expected Duration: ____________________

4. List the reference documents reviewed which describes the structure, system or component. (Identify documents referenced even if no information was found in that section.)

   a. UFSAR Section(s): Table 1.8-1, 3.0, 7.1, 7.5, 11.5.2.7, 15

   b. SER Section(s): None

   c. Tech Spec Section(s): 3.2/4.2, 3.8/4.8

   d. Fire Protection Program Document Pkg Section(s): FPR Vol. 1 4.7.3, 4.7.4

   e. Code of Federal Regulations Section(s): None

   f. Regulatory Guides/NUREGs: 1.97

   g. Other: SE 93-109

5. Describe how the change will affect plant operation when the changed structure, system or component function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other structures, systems or components.

   During installation of this modification, continuous monitoring of the service water return header will be lost. Instead, the Chemistry Department will take and analyze grab samples in accordance with the requirements of Technical Specification Table 3.2-5. Domestic water will be isolated to the Fish House and Chemistry Labs during installation of the back flow preventer, but this will not affect any system needed to assure continued normal plant operations. During replacement of the inlet isolation valves, the intake sample line will be open to turbine building atmosphere. A funnel and drain hose will be in place to collect any water and route it to a floor drain. Some water may not be caught by the funnel, but the small diameter of the pipe (1'-1/2") means the volume can easily be contained by local floor drains. No other structures, systems, or components (SSC) will be affected by this work.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   The new isolation valve will be a ball valve. The old isolation valve was a gate valve which was susceptible to stem/disk separation. A possible failure mode would be the ball valve sticking partially or fully open after it is installed. This would result in leakage of service water onto the floor. Nearby floor drains, however, will be able to handle this leakage and prevent local flooding. This analysis is also good for any leakage after the sample supply system. The piping supports could fail resulting in an improperly supported line. This may or may not lead to line cracking or rupture, but the resulting leakage is still bounded by the above analysis. This work will not affect any other systems or impact the present failure modes of any SSC.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the UFSAR where any of the following is true:

- The change alters the initial conditions used in the UFSAR analysis
- The changed structure, system or component is explicitly or implicitly assumed to function during or after the accident
- Operation or structure, system or component failure of the changed structure, system or component could lead to the accident

ACCIDENT	UFSAR SECTION
None	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. (To determine the factors affecting the specification, it is necessary to review the UFSAR and SER where the Bases Section of the Technical Specifications does not explicitly state the basis).

3.2.G

9. Will the change involve a Technical Specification revision?

( ) Yes (X) No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing step 14, indicate that a Technical Specification revision is required.

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the UFSAR may be increased, use one copy of this page to answer the following questions for each accident listed in step 7. Provide the rationale for all NO answers.

Affected accident Not Applicable ______________ UFSAR Section: ____________________

May the probability of the accident be increased? ( ) Yes ( ) No

May the consequences of the accident (off-site dose) be increased? ( ) Yes ( ) No

May the probability of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

May the consequences of a malfunction of equipment important to safety increase? ( ) Yes ( ) No

<u>If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.</u>

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the UFSAR?

( ) Yes (X) No

Describe the rationale for your answer.

This work only interfaces with the domestic water system and the service water system. Both interfaces are mechanical only. No other SSC will be impacted by the scope of this work. The worst case scenario would involve a failure of the installed isolation valve on the service water return header. This would lead to leakage onto the turbine building first floor. But, this leakage will not be of greater magnitude than the capability to remove water by the floor drain system. Therefore, this event will not result in flooding. No other SSCs will be adversely impacted so as to create a new UFSAR accident or transient.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists, proceed to step 14.

Technical Specification 3.2.G ______________________________

Determine which of the following is true for the above specification:

( ) All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists, proceed to question 13.

( ) The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) below.

( ) The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the UFSAR. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination. List the limit(s)/margin(s) below.

(X) The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety - proceed to Question 14.

List Acceptance Limit(s)/Margin(s) of Safety

______________________________ ______________________________
______________________________ ______________________________
______________________________ ______________________________
______________________________ ______________________________

13. Use the above limits identified in step 12 to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

<u>If a Margin of Safety is reduced, an Unreviewed Safety Question exists.</u>

14. Check one of the following:

( ) An Unreviewed Safety Question was identified in step 10, step 11, or step 13. The proposed change MUST NOT be implemented without NRC approval.

(X) No Unreviewed Safety Question will result (steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

( ) A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

  ( ) The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

  ( ) The change is a plant modification or minor plant change. Mark below as applicable.

    ( ) A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

    ( ) The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

      ( ) Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Preparer Mark L. Bridges June 15, 1993
Signature Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer ~~Signature~~ 6/22/93

Signature Date

16. Obtain a safety evaluation number from the Tech Staff clerk and record it on page 1.

17. Leave one copy of the safety evaluation with the Tech Staff clerk and file the original with the applicable package(s)

18. The Tech Staff clerk will forward a copy of this safety evaluation to the FSAR Coordinator. (ANI Audit Recommendation 88-1)

Completed: DK 6-22-93

Initial Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: C04-1-93-094	Date: 5/20/94
Subject: REPLACE TURBINE ROTOR UNSTACKING TRANSFORMER WITH DRY TYPE. REMOVE ASSOCIATED DELUGE SYSTEM.	
Submitted by: J.G. WESTON	

FOR REVIEW:	
	1. Safety Evaluations <u>NOT</u> involving an unreviewed safety question as defined in 10CFR50.59 for:
	a. Changes to procedures as described in the Safety Analysis Report.
	b. Changes to equipment or systems as described in the Safety Analysis Report.
	c. Tests or experiments <u>NOT</u> described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
	a. Procedure changes.
	b. Equipment or system changes.
	c. Tests or experiments.
	3. Proposed changes to the Technical Specifications or Operating License.
	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
	6. All REPORTABLE EVENTS (LERs only).
	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
	8. All changes to the Station Emergency Plan prior to implementation.
	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.
FOR INFORMATION:	
X	10. Other OSR Items/Documents <u>NOT</u> addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) **for information only**. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Mod # E04-1-93-094

Station/Unit Quad Cities /1

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   Engineering Change Notice 04-01031E dated 12/6/93.
   Bechtel Calculation QC-429-C-035 dated 12/7/93.

2. Describe the proposed change and the reason for the change.

   The subject exempt change will replace an oil-filled 1 MVA transformer with a dry-type 500 KVA transformer on elevation 639' of the Unit 1 Turbine Building. The existing wet pipe system will be demolished.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ______

   AND
   Plant Mode(s) restrictions while installed ______
   (NONE if no plant mode restrictions apply) ______

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   The following FSAR sections were reviewed for applicability:
   8.0 Electric Power
   8.3 Onsite Power Systems
   The change does not affect these documents.

Exhibit E
10CFR50.59 SAFETY EVALUATION

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   This change does not electically or structurally interact with plant equipment. Therefore, equipment failures are not affected. The failure mode of the transformer is not changed.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
None.	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitely state the basis.

   None.

Exhibit E
10CFR50.59 SAFETY EVALUATION

9. Will the change involve a Technical Specification revision?

[ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Mod # E04-1-93-094

Station/Unit Quad Cities /1

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident None.

SAR Section: N/A.

May the probability of the accident be increased? [ ] Yes [X] No

T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. The probability of an oil fire due to transformer failure is reduced to zero because the new transformer is a dry-type containing no oil.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. Therefore, the transformer has no affect on the consequences of an accident. The consequences of failure of the transformer is reduced significantly since the dry-type transformer will not contribute combustible oil to a fire in the immediate area.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not

Exhibit E
10CFR50.59 SAFETY EVALUATION

electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. The replacement of the oil-type transformer with a dry-type one results in no net change in probability of malfunction of safety related equipment.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. The replacement of the oil-type transformer with a dry-type one results in no net change in consequences of malfunction of safety related equipment. The dry type transformer will reduce the consequences of transformer failure or area fire.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

T42R-5A receives power from the 13.8KV yard. Its 480V secondary will provide power for maintenance activities on the turbine deck. Per the FSAR, the 13.8kv system is not used for plant equipment. Therefore, this transformer will not electrically affect operation of plant equipment. Per the Bechtel calculation listed previously, the supports and attachments have been evaluated for structural acceptability. The replacement of the oil-type transformer with a dry-type one results in no new accident type.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.______

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

N/A.______

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ______

SAR Section ______

SER Section ______

Exhibit E
10CFR50.59 SAFETY EVALUATION

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

Station/Unit Quad Cities /1

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Mod # E04-1-93-094

Station/Unit Quad Cities /1

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer [signature] 1/21/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer RA Dammann 1-21-94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: E04-1-93-245	Date: 5/20/94
Subject: INSTALL WELDING RECEPTACLES ON TURBINE SHIELD WALL	
Submitted by: J. S. WESTON	

FOR REVIEW:

	1. Safety Evaluations **NOT** involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments **NOT** described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

X	10. Other OSR Items/Documents **NOT** addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) for information only. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   Exempt Plant Change E04-1-93-245.
   Engineering Change Notice 04-1053E, dated 1/17/94.
   Bechtel Calculation QC-469-C-001, dated 1/21/94.
   Parameter Assessment and Reconciliation B-130-00360.

2. Describe the proposed change and the reason for the change.

   The subject exempt change will replace the existing panel with a 10 circuit distribution panel and install six 60 amp welding receptacles powered from this new panel. Five receptacles will be mounted on the outside of the turbine shield wall west of the new circuit panel. A sixth will be installed on the inside of the turbine shield wall. This new configuration will provide a safer and more efficient means for providing power on the turbine deck.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ____________

   AND
   Plant Mode(s) restrictions while installed ____________
   (NONE if no plant mode restrictions apply) ____________

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   The following UFSAR Sections were reviewed:
   1.2.2.10 Shielding, Access Control, and Radiation Protection Procedures
   3.5 Missile Protection
   8.0 Electric Power Systems
   8.3 Onsite Power Systems
   10.0 Steam and Power Conversion System
   These sections will not be affected by this design change.

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   The power source for the new 480VAC panel and welding receptacles is from the 13.8KV yard via transformer T42R-5A. Per the UFSAR, the 13.8KV yard is not used for plant equipment. Therefore, the subject design change will not electrically affect plant equipment. The only structural interaction is due to mounting on the turbine shield wall (which is non-safety related / non-seismic / non-II/I). The referenced calculation and associated PAR have determined that the structural loads are acceptable.

Exhibit E
10CFR50.59 SAFETY EVALUATION

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   As stated in Section 5, the design change does not electrically interact with plant equipment. The additional structural loads have been analyzed for acceptability. Therefore, the design change will not affect equipment failures. The addition of the circuit panel and welding receptacles will result in an increase in equipment reliability over the existing configuration. Therefore, the failure mode of the 480VAC panel is lessened in severity.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
None	*N/A*

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitly state the basis.

   The following sections were reviewed:
   3.9/4.9 Auxiliary Electric Systems

9. Will the change involve a Technical Specification revision?

   [ ] Yes [X] No

   If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident: *None*

SAR Section: *N/A*

May the probability of the accident be increased? [ ] Yes [X] No

As stated in section 5, the subject exempt change does not electrically interact with plant equipment. The structural interactions are with plant structures that are not important to safety and have been analyzed for acceptability. The replacement circuit panel and receptacles will be more reliable than the existing configuration. Therefore, the chances of failure of this equipment is reduced. The probability of an design basis accident is unchanged.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

As stated prev usly, the subject design change does not affect equipment important to safety or equipment required for safe shutdown of the plant. Therefore, the consequences of an accident are not changed by this change.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

As previously stated, the subject exempt change does not electrically interact with plant equipment. The structural interactions are with plant structures that are not important to safety and have been analyzed for acceptability. The replacement circuit panel and receptacles will be more reliable than the existing configuration. Therefore, the chances of failure of this equipment is reduced. Therefore, the probability of malfunction of other nearby equipment due to failure of this equipment is reduced.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

As previously stated, the subject exempt change does not electrically interact with plant equipment. The structural interactions are with plant structures that are not important to safety and have been analyzed for acceptability. The replacement circuit panel and receptacles will be more reliable than the existing configuration. The failure mode of this new panel is the same as the existing panel. Such a failure does not change the consequences of a malfunction of other nearby equipment. Therefore, it will not change the consequences of safety related equipment failure.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

    [ ] Yes [X] No

    Describe the rationale for your answer.

    The subject design change will not result in changed operation of the existing panels. Therefore, no new accident that has not been previously analyzed will be created.

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following Questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.

    N/A

    Evaluation of Technical Specification
    (Enter N/A if none are affected and check last option.)

    N/A

    (Check appropriate condition):

    [ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

    [ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

    [ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

    [X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

    List Acceptance Limit(s)/Margin(s) of Safety

    Tech Spec ______________________

    SAR Section ______________________

    SER Section ______________________

Exhibit E
10CFR50.59 SAFETY EVALUATION

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

N/A

<u>If a Margin of Safety is reduced an Unreviewed Safety Question exists.</u>

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

- [ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

- [ ] The change is a plant modification or minor plant change. Mark below as applicable.

  - [ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

  - [ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

    - [ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Exhibit E
10CFR50.59 SAFETY EVALUATION

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer [signature] 2/2/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer RADammann 2-3-94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: E04-1-93-325	Date: 5/20/94
Subject: UAT CHANGE-OUT CONCRETE WORK	
Submitted by: J. S. WESTON	

FOR REVIEW:

	1. Safety Evaluations **NOT** involving an unreviewed safety question as defined in 10CFR50.59 for:
☐	a. Changes to procedures as described in the Safety Analysis Report.
☐	b. Changes to equipment or systems as described in the Safety Analysis Report.
☐	c. Tests or experiments **NOT** described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
☐	a. Procedure changes.
☐	b. Equipment or system changes.
☐	c. Tests or experiments.
☐	3. Proposed changes to the Technical Specifications or Operating License.
☐	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
☐	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
☐	6. All REPORTABLE EVENTS (LERs only).
☐	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
☐	8. All changes to the Station Emergency Plan prior to implementation.
☐	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.

FOR INFORMATION:

X	10. Other OSR Items/Documents **NOT** addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) **for information only**. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Station/Unit Quad Cities / 1

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

   Exempt Plant Change E04-1-93-325.
   Engineering Change Notice 04-001149S and associated calculations.

2. Describe the proposed change and the reason for the change.

   The subject exempt plant change will install new concrete piers in support of the replacement of the Unit 1 Unit Auxiliary Transformer (UAT). New concrete piers are required for the fire suppression deluge system which is being redesigned due to physical differences between the existing GE and the new SMIT transformer.

   Two other exempt changes are required to complete the replacement of the Unit 1 UAT:

   E04-1-93-326 will replace the existing fire protection system piping and fire detection method. The deluge piping must be replaced due to the physical differences between the existing GE UAT and the new SMIT UAT. The detection method is being changed in order to make it more reliable. The overall operation of the system will not change.

   E04-1-93-327 will reinstall the transformer control circuitry. These changes are necessary due to slight differences between the GE and SMIT transformers. The control circuitry changes will not affect the operation of the plant.

3. Is the change:

   [X] Permanent

   [ ] Temporary -
   Expected duration ______________________

   AND
   Plant Mode(s) restrictions while installed ______________________
   (NONE if no plant mode restrictions apply) ______________________

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

   The following section of the Quad Cities Station UFSAR has been reviewed:

   Section 8.3 Onsite Power Systems

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

   The replacement of the Unit 1 General Electric UAT with a new SMIT UAT does not affect plant operation in any operating mode. The new UAT has electrical characteristics that are compatible with the existing GE UAT. It has the

Exhibit E
10CFR50.59 SAFETY EVALUATION

capability to supply auxiliary power for the unit while in run mode or in backfeed mode. The control wiring changes are necessary due to the new UAT's slight differences and enhancements. The replacement of the fire protection deluge piping and installation of associated concrete supports are necessary due to the minor differences in the new UAT's physical layout. The replacement of the existing thermal detectors with linear-type detection cable will improve the reliability of the system by reducing the inadvertent actuations. These changes to the Unit 1 UAT do not change operation of the UAT as it relates to the plant or plant systems.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

   The failure mode of the new UAT is the same as the existing UAT. The failure rate of the new UAT should be lower than that of the existing UAT due to the age of the existing UAT.

   The replacement of the deluge system piping and installation of associated concrete pad supports does not affect any other equipment. The failure mode of these components is the same as for the existing system.

   The change in fire detection method will increase the fire detection reliability. The replacement "protecto-wire" detection method will reduce the number of DC ground problems, and therefore increase the reliability of the system. The failure mode of the fire detection system is the same as for the existing detectors.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
   - The change alters the initial conditions used in the SAR analysis
   - The changed SSC is explicitly or implicitly assumed to function during or after the accident
   - Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
Loss of Auxiliary Power	*8.3.1*

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected. To determine the factors affecting the specification, it is necessary to review the FSAR and SER where the bases section of the Technical Specifications does not explicitly state the basis.

   The following section was reviewed:

   Section 3.9.A.3, "One other 345-kv line capable of carrying auxiliary power to an essential electrical bus of the unit through the 4160-volt bus tie shall be available."

Exhibit E
10CFR50.59 SAFETY EVALUATION

9. Will the change involve a Technical Specification revision?

[ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

Exhibit E
10CFR50.59 SAFETY EVALUATION

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident: *Loss of Auxiliary Power*

SAR Section: *Section 8.3.1*

May the probability of the accident be increased? [ ] Yes [X] No

The replacement UAT will provide the same function as the existing UAT. The new UAT should be more reliable than the existing UAT because it is new and has been constructed using latest technology. Therefore, the probability of the accident will not increase by this change.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

The consequences on plant operation of failure of the new UAT are the same as for the old UAT. Therefore, the consequences of a failure of the Unit 1 UAT will not change due to the replacement of the existing GE UAT with the new SMIT UAT.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

This change is compatible with interfacing plant systems. The replacement of the GE UAT with a new SMIT UAT will reduce the probability of a Unit 1 UAT failure by improving the reliability of the transformer and fire detection circuitry. Therefore, the probability of a malfunction of equipment important to safety will be reduced.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

The consequences of failure of the UAT or the associated fire protection system is the same as for the existing UAT. Therefore, the consequences of failure of equipment important to safety is unchanged as a result of this project.

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

The UAT is being replaced by a newer transformer. The failure mode of this new transformer, fire protection system, and control circuitry is the same as for the existing transformer. The failure rate due to these changes is reduced due to the more reliable transformer and enhancements to the fire protection system. Therefore, an accident different from those previously evaluated in the SAR is not created.

<u>If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.</u>

Station/Unit Quad Cities / 1

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following Questions for each Technical Specification listed in Step 8. List the Technical Specification, Technical Specification Bases, SAR and SER Sections reviewed for this evaluation.

Section 3.9.A.3, "One other 345-kv line capable of carrying auxiliary power to an essential electrical bus of the unit through the 4160-volt bus tie shall be available."

Evaluation of Technical Specification
(Enter N/A if none are affected and check last option.)

The replacement of the UAT does not affect the requirements for this Technical Specification. The new UAT will be capable of providing auxiliary power to the unit during backfeeding operations

(Check appropriate condition):

[ ] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification or SAR provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) and applicable reference for the margin of safety below - proceed to question 13.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and neither the Technical Specification, the SAR, or the SER provides a margin of safety or an acceptance limit. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination by consulting the NRC, SAR, SER's or other appropriate references. List the agreed limit(s)/margin(s) below.

[X] The change does not affect any parameters upon which Technical Specifications are based; therefore, there is no reduction in the margin of safety. Proceed to question 14.

List Acceptance Limit(s)/Margin(s) of Safety

Tech Spec ____________________

SAR Section ____________________

SER Section ____________________

Exhibit E
10CFR50.59 SAFETY EVALUATION

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

    N/A

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Note: Partial Modifications and/or separate 10CFR50.59 reviews for portions of the work may be used to facilitate installation.

Preparer [signature] 2/28/94
(Cognizant Engineer) Date

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer RA Dammann 3/1/94
(Design Superintendent/Supervisor) Date

ATTACHMENT A (Page 1 of 1)
OFFSITE REVIEW AND INVESTIGATIVE FUNCTION TRANSMITTAL

Quad Cities Nuclear Power Station

Reference Number: PO4-1-91-127	Date: 6/6/94
Subject: REPLACEMENT OF EXISTING T-QUENCHER BOLTS	
Submitted by: Craig Baldwin	

FOR REVIEW:	
	1. Safety Evaluations <u>NOT</u> involving an unreviewed safety question as defined in 10CFR50.59 for:
[ ]	a. Changes to procedures as described in the Safety Analysis Report.
[ ]	b. Changes to equipment or systems as described in the Safety Analysis Report.
[ ]	c. Tests or experiments <u>NOT</u> described in the Safety Analysis Report.
	2. Proposed changes which involve an unreviewed safety question as defined in 10CFR50.59.
[ ]	a. Procedure changes.
[ ]	b. Equipment or system changes.
[ ]	c. Tests or experiments.
[ ]	3. Proposed changes to the Technical Specifications or Operating License.
[ ]	4. Noncompliance with codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.
[ ]	5. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affects nuclear safety.
[ ]	6. All REPORTABLE EVENTS (LERs only).
[ ]	7. All recognized indications of an unanticipated deficiency in design or operation of safety-related structures, systems, or components.
[ ]	8. All changes to the Station Emergency Plan prior to implementation.
[ ]	9. All items referred by the Systems Engineering Supervisor, Station Manager, Site Vice President, and General Manager of Quality Programs and Assessments.
FOR INFORMATION:	
[X]	10. Other OSR Items/Documents <u>NOT</u> addressed above.
	This Transmittal is being made in accordance with Quad Cities Nuclear Power Station Technical Specifications 6.1.G.2.d(1) **for information only**. No specific action is required unless deemed necessary by Offsite Review and Investigative Function.

Mod # P04-1-91-127

Station/Unit QUAD CITIES/1

Exhibit E
10CFR50.59 SAFETY EVALUATION

1. List the documents implementing the proposed change.

MINOR PLANT CHANGE P04-1-91-127
DWG B-1470, B-1471, B-1468, & B-1467

2. Describe the proposed change and the reason for the change.

INSTALL NEW REPLACEMENT WELDED AND/OR BOLTED THREADED ROD TO REPLACE RODS ANCHORING THE T QUENCHER SUPPORTS LOCATED IN THE TORUS.
A SAMPLE OF RODS IS BEING REMOVED FOR EXAMINATION TO CONFIRM THE ABSENCE OF STRESS CORROSION CRACKING.

3. Is the change:

[X] Permanent

[ ] Temporary — Expected duration ______
AND
Plant Mode(s) restrictions while installed ______
(NONE if no plant mode restrictions apply)

4. List the SAR sections which describe the affected systems, structures, or components (SSCs) or activities. Also list the SAR accident analysis sections which discuss the affected SSCs or their operation. List any other controlling documents such as SERs, previous modifications or Safety Evaluations, etc.

MAIN STEAM LINE RELIEF VALVE DISCHARGE - UFSAR SECT. 4.4.2
PRESSURE SUPRESSION CHAMBER (TORUS) - UFSAR SECT. 5.2.2.

Mod # P04-1-91-127

Station/Unit QUAD CITIES /1

Exhibit E
10CFR50.59 SAFETY EVALUATION

5. Describe how the change will affect plant operation when the changed SSCs function as intended (i.e., focus on system operation/interactions in the absence of equipment failures). Consider all applicable operating modes. Include a discussion of any changed interactions with other SSCs.

NO CHANGE TO OPERATION
THE VERY SMALL AMOUNT OF WATER DISPLACED AS A RESULT OF THIS MPC IS NOT OBSERVABLE WITHIN THE ACCURACY OF EXISTING WATER LEVEL INDICATORS.

6. Describe how the change will affect equipment failures. In particular, describe any new failure modes and their impact during all applicable operating modes.

THERE IS NO OPERATING EQUIPMENT INVOLVED IN THIS WORK.

7. Identify each accident or anticipated transient (i.e., large/small break LOCA, loss of load, turbine missiles, fire, flooding) described in the SAR where any of the following is true:
- The change alters the initial conditions used in the SAR analysis
- The changed SSC is explicitly or implicitly assumed to function during or after the accident
- Operation or failure of the changed SSC could lead to the accident

ACCIDENT	SAR SECTION
NONE	

8. List each Technical Specification (Safety Limit, Limiting Safety System Setting or Limiting Condition for Operation) where the requirement, associated action items, associated surveillances, or bases may be affected.

NONE. THE TECHNICAL SPECIFICATION MAXIMUM AND MINIMUM WATER LEVELS IN THE PRESSURE SUPPRESSION CHAMBER ARE NOT AFFECTED BY THE VERY SMALL AMOUNT OF WATER DISPLACED BY THIS MPC

Station/Unit QUAD CITIES/1

Exhibit E
10CFR50.59 SAFETY EVALUATION

9. Will the change involve a Technical Specification revision? [ ] Yes [X] No

If a Technical Specification revision is involved, the change cannot be implemented until the NRC issues a license amendment. When completing Step 14, indicate that a Technical Specification revision is required.

10. To determine if the probability or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, use one copy of this page to answer the following questions for each accident listed in Step 7. Provide the rationale for all NO answers.

Affected accident NONE SAR Section: ________

May the probability of the accident be increased? [ ] Yes [X] No

THE MINIMUM REQUIRED FACTOR OF SAFETY FOR THE CONNECTION HAS NOT BEEN CHANGED WITH THE USE OF ANY OF THE NEW REPLACEMENT ASSEMBLIES.

May the consequences of the accident (off-site dose) be increased? [ ] Yes [X] No

THIS CHANGE HAS NO EFFECT ON ACCIDENT ANALYSIS.

May the probability of a malfunction of equipment important to safety increase? [ ] Yes [X] No

THE MINIMUM REQUIRED FACTOR OF SAFETY FOR THE CONNECTION HAS NOT BEEN CHANGED WITH THE USE OF ANY OF THE NEW REPLACEMENT ASSEMBLIES.

May the consequences of a malfunction of equipment important to safety increase? [ ] Yes [X] No

THIS CHANGE HAS NO EFFECT ON SAFETY RELATED EQUIPMENT OPERATION

If any answer to Question 10 is YES, then an Unreviewed Safety Question exists.

Mod # P04-1-91-127

Station/Unit QUAD CITIES/1

Exhibit E
10CFR50.59 SAFETY EVALUATION

11. Based on your answers to Questions 5 and 6, does the change adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR?

[ ] Yes [X] No

Describe the rationale for your answer.

CHANGE DOES NOT AFFECT EQUIPMENT OPERATIONS OR FUNCTIONS

If the answer to Question 11 is Yes, then an Unreviewed Safety Question exists.

Mod # P04-1-91-127

Station/Unit QUAD CITIES / 1

Exhibit E
10CFR50.59 SAFETY EVALUATION

12. Determine if parameters used to establish the Technical Specification limits are changed. Use one copy of this page to answer the following questions for each Technical Specification listed in Step 8. If no Technical Specifications are impacted, then no reduction in margin of safety exists - proceed to Step 14.

Technical Specification NONE,

Determine which of the following is true for the above specification:

[X] All changes to the parameters or conditions used to establish the Technical Specification requirements are in a conservative direction. Therefore, the actual acceptance limit need not be identified to determine that no reduction in margin of safety exists - proceed to Question 13.

[ ] The Technical Specification provides a margin of safety or acceptance limit for the applicable parameter or condition. List the limit(s)/margin(s) below.

[ ] The applicable parameter or condition change is in a potentially non-conservative direction and the Technical Specification neither provides an acceptance limit nor explicitly references a limit in the SAR. Request Nuclear Licensing assistance to identify the acceptance limit/margin for the Margin of Safety determination. List the limit(s)/margin(s) below.

List Acceptance Limit(s)/Margin(s) of Safety

13. Use the above limits to determine if the margin of safety is reduced (i.e., the new values exceed the acceptance limits). Describe the rationale for your determination. Include a description of compensating factors used to reach that conclusion.

NO CHANGE TO TECHNICAL SPECIFICATIONS.

If a Margin of Safety is reduced an Unreviewed Safety Question exists.

Mod # P04-1-91-127

Station/Unit QUAD CITIES/1

Exhibit E
10CFR50.59 SAFETY EVALUATION

14. Check one of the following:

[ ] An Unreviewed Safety Question was identified in Step 10, Step 11, or Step 13. The proposed change MUST NOT be implemented without NRC approval.

[X] No Unreviewed Safety Question will result (Steps 10, 11, and 13) AND no Technical Specification revision will be involved. The change may be implemented in accordance with applicable procedures.

[ ] A Technical Specification revision is involved; but no Unreviewed Safety Question will result. The proposed change requires a License Amendment. Notify Station Regulatory Assurance and Nuclear Licensing that a Technical Specification revision is required. Mark below as applicable.

[ ] The change is not a plant modification or minor plant change and will not be implemented under 10CFR50.59. Upon receipt of the approved Technical Specification change from the NRC, the change may be implemented.

[ ] The change is a plant modification or minor plant change. Mark below as applicable.

[ ] A revision to an existing Technical Specification is required. The change MUST NOT be installed until receipt of an approved Technical Specification revision.

[ ] The change will not conflict with any existing Technical Specifications and only new Technical Specifications are required. In these cases, Nuclear Licensing may authorize installation, but not operation, prior to receipt of NRC approval of the License Amendment. If such authorization is granted, the block below should be checked.

[ ] Nuclear Licensing has authorized installation, but not operation, prior to receipt of NRC approval of the License Amendment. The 10CFR50.59 Safety Evaluation indicates that no Unreviewed Safety Question will result and provides authority for installation only.

Preparer ____________ (Signature) 7-1-92 (Date)

15. The reviewer has determined that the documentation is adequate to support the above conclusion and agrees with the conclusion.

Reviewer B MICW____ (Signature) 7-2-92 (Date)

SJ 7/2/92