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SVP-98-358

December 9, 1998

U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Document Control Desk

Subject: Quad Cities Nuclear Power Station Units 1 and 2
Summary Report of Changes, Tests, and Experiments Completed
Facility Operating License Numbers DPR-29 and DPR-30
NRC Docket Numbers 50-254 and 50-265

Enclosed please find those 50.59 Safety Evaluations associated with Quad Cities Station, Units 1 and 2, DPR-29 and DPR-30. Summaries of the safety evaluations are being reported in compliance with 10 CFR 50.59 and 10 CFR 50.71(e). These safety evaluations cover the period of July 16, 1998 through October 31, 1998.

If you have any questions or comments concerning this letter, please refer them to Mr. Charles Peterson, Regulatory Assurance Manager, at (309) 654-2241, extension 3609.

Sincerely,

L.W. Pearce for JPD

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

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

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SUMMARY REPORT OF CHANGES, TESTS AND EXPERIMENTS COMPLETED

ATTACHMENT A

SVP-98-358

SAFETY EVALUATION
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Safety Evaluation Number: None (Unique Numbers Were Not Assigned Prior to 1997)

Type of Safety Evaluation: Modification

Evaluation Reference Number: DCP 8900029; M04-1-89-074-B

Title: Install Automatic Sensor on the West Turbine Building Rollomatic Filter

DESCRIPTION:

This DCP replaced the timer controls, which advanced the filter medium based only on time, with a light sensor which advanced the filter medium based on the amount of light that passed through the filter medium. This DCP was specific to the Unit 1 West Turbine Building Rollomatic Filter.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function of the Turbine Building Ventilation System is unchanged by this DCP. The Turbine Building Ventilation System cannot cause an accident and is not used to mitigate the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function of the Turbine Building Ventilation System remains unchanged, the failure modes are unchanged.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Tech Specs associated with the Turbine Building Ventilation System.

Safety Evaluation Number: None (Unique Numbers Were Not Assigned Prior to 1997)

Type of Safety Evaluation: Exempt Change

Evaluation Reference Number: DCP 9300033; E04-0-93-284

Title: Replacement of Radwaste Building Exhaust Fans From Belt Driven To Direct Drive

DESCRIPTION:

The design change replaced the Radwaste Building Exhaust fans from belt driven to direct drive. This design change was to reduce unavailability time of the fans and reduce the number of high vibration problems.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this design change is to enhance system reliability and operating characteristics. The Radwaste Building Ventilation System does not affect off-site releases and does not interact with any other system, structure, or component that does.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Radwaste Ventilation System provides personnel protection from radioactive contaminants. A loss of the exhaust fans has been analyzed; thus, this design change does not create the possibility of an unanalyzed event.

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

Safety Evaluation Number: SE-97-120

Type of Safety Evaluation: Design Change

Evaluation Reference Number: DCP 9700275

Title: Install New Overload Relay Heater Elements for the Unit 1 And Unit 2 Residual Heat Removal Service Water (RHRSW) Cubicle Cooling Fan Motors.

DESCRIPTION:

On July 18, 1997 Problem Identification Form (PIF) Q1997-02921 identified that the presently installed overload relay heater elements that protect the 2B RHRSW Cubicle Cooling Fan C Motor do not provide sufficient margin during a reduced or degraded voltage condition. Further investigation has determined that all the RHRSW Vault Cooling Fan Motors on Unit 1 and Unit 2 are equipped with same size overload heater element. The above listed DCPs have been initiated to install properly sized elements that will adequately protect the associated motors and prevent erroneous trips. The elements were sized by Calculation QDC-1000-E-0444.

The Safety Evaluation was also used for DCP 9700276 which was not Op authorized during this report period. The summary will be included when the DCP becomes Op authorized.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the RHRSW cooling fans provide air flow across the coils for the RHRSW vault coolers. The RHRSW pumps are used to mitigate the consequences of accidents. The RHRSW pumps and cooling fans as well as the thermal overloads for the cooling fans do not cause or contribute to the cause of any accident or transient. Therefore, the probability of any of these accidents or transients is not increased. Installing the new heater elements decreases the chance of a nuisance trip of the RHRSW cooling fan motors during a reduced or degraded voltage condition. Therefore, the ability of the RHR system to perform its designed function and mitigate the consequences during and/or after any accident or transient is improved.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new overload heater elements are the same type of element as the old element. The likelihood of a failure and failure modes of the new element are the same as the old heater element. A failure of the element may cause the associated fan motor to malfunction so that the affected room cooler would not operate properly and provide adequate cooling for the RHRSW pump motor. This could ultimately cause a malfunction of the RHRSW pump motor and affect the ability of the RHR system to remove decay heat in a post accident condition. After the new elements are installed, a malfunction or failure would be identical to what is currently evaluated. Therefore, the installation of the new elements will not create the possibility of an accident or malfunction of a different type.

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

   There were no Unreviewed Safety Questions identified as a result of this evaluation. There are no changes to the UFSAR required.

Safety Evaluation Number: SE-98-004

Type of Safety Evaluation: Technical Specification Bases Change

Evaluation Reference Number: Technical Specification Bases 3/4.2.D and 3/4.5.D

Title: Reactor Core Isolation Cooling System

DESCRIPTION:

This Safety Evaluation changes the current wording in the previously mentioned Technical Specification Bases. Specifically, the changes are as follows:

> Technical Specification Bases 3/4.2.D will be revised to read: "The reactor core isolation cooling system provides makeup water to the core in the event of a postulated isolation of the reactor from the main condenser with a loss of feedwater. The system automatically initiates upon receipt of a reactor vessel low-low water level signal utilizing level indicating switches in a one-out-of-two taken twice logic scheme. The system may also be manually started."
>
> Technical Specification Bases 3/4.5.D will be revised, in part, to read: "The Reactor Core Isolation Cooling (RCIC) system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the main condenser with a loss of reactor feedwater."

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the action(s) to be taken will only result in a change in the wording of Technical Specifications Bases 3/4.2.D and 3/4.5.D which will elucidate their meanings. Clarification of the wording in the Bases will not change any plant operational condition which could initiate or increase the probability of the affected accidents or transients.

   Neither of the proposed changes to Technical Specification Bases 3/4.2.D and 3/4.5.D impact the operation of the RCIC system or the availability of the other systems required to mitigate the consequences of the identified accidents/transients. Therefore, the consequences are not increased.

   The proposed changes merely clarify statements in Technical Specification bases 3/4.2.D and 3/4.5.D and provides consistency with the description provided in the UFSAR. They have no bearing on the current operation, maintenance, or surveillance activities of the RCIC system or any other systems and/or equipment important to safety, therefore the probability of a malfunction of equipment important to safety is not increased.

   The proposed changes do not affect RCIC operation or equipment, nor do they affect any interactions with other SSCs. Therefore, the consequences of a malfunction of equipment important to safety do not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the proposed changes do not cause the RCIC system to be operated in an abnormal lineup, outside of its design bases, nor in any manner that impacts its design function. Because of this fact, there is no possibility for an accident or malfunction to occur of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the proposed changes do not alter the method in which the system is operated, or any interactions with other SSCs.

Safety Evaluation Number: SE-98-040

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-069

Title: UFSAR Change to Correct Radwaste Items

DESCRIPTION:

This change is to update the UFSAR on the radwaste system, to correct missing or incorrect data on radwaste equipment and systems, and to provide details on High Integrity Containers (HICs) and processing changes when biological activity is present.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the proposed changes to the UFSAR section will not increase the probability of equipment failures or increase the probability of an accident. The equipment involved is not safety-related and is not used during any accident conditions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no new equipment has been added to the radwaste system. These changes correct minor errors in previous UFSAR revisions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no plant equipment is being physically changed. This proposed change is to update the UFSAR to show current conditions, missing information, changes in tank names, and other pertinent data. The revisions to the UFSAR do not require a change to any administrative control procedure, design drawing, or other design basis documents. The proposed changes to the UFSAR will not affect radioactive liquid effluents. The only change related to the UFSAR accident section (Chapter 15 – River Discharge Tank failure) is to show that the Waste Surge Pump can be used in place of the River Discharge Pump to discharge liquid from the River Discharge Tank. Both pumps discharge to the same monitored locations as described in the UFSAR. These revisions to the UFSAR do not increase the volume or concentration of liquid or solid radioactive material

that can be stored at Quad Cities Station. Administrative procedures are in effect to control the storage of this radioactive material.

Safety Evaluation Number: SE-98-047

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-091

Title: Revision to UFSAR Table 6.2-7, Penetrations Of Primary Containment And Associated Isolation Valves

DESCRIPTION:

Quad Cities UFSAR Table 6.2-7 will be revised to delete the Group 6 Containment Isolation Signal for containment isolation valves AO 1(2)-2599-4A/B and FCV 1(2)-2599-5A/B. Table 6.2-7 did not correctly depict that these valves do not receive a Group 6 Primary Containment Isolation System signal.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no interface or connection between the Atmospheric Containment Atmosphere Dilution (ACAD) pressure bleed subsystem containment isolation valves and the primary pressure boundary piping, whose failure could lead to a LOCA. Failure of the ACAD pressure bleed subsystem valves cannot initiate a LOCA. These valves do form part of the primary containment boundary used to mitigate off site releases. These valves are not used during normal operation and are therefore, maintained in the closed position. These valves would only be opened after a LOCA had already occurred to control the pressure build up after a LOCA. The Group 6 isolation signal only occurs after a LOCA has already occurred to isolate the ACAD air injection subsystem to prevent further pressure increase in a post accident LOCA scenario. The containment isolation valves on the ACAD pressure bleed would not need to be isolated since this system cannot cause the pressure to increase inside primary containment. The ACAD pressure bleed subsystem has opposite function of venting the containment to prevent overpressurization of the containment. Therefore, in accordance with the original design requirements of the ACAD pressure bleed subsystem, the Group 6 isolation signal is not applicable to the pressure bleed subsystem containment isolation valves. Therefore, the consequences of failure of the ACAD pressure bleed subsystem containment isolation valves, which could compromise the integrity of the primary containment, is not increased any greater than original design.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because UFSAR Table 6.2-7 did not correctly state the original design function of the ACAD pressure bleed subsystem containment isolation valves. The original design function was established in accordance with 10CFR50.44 criteria and applicable Draft General Design Criteria as stated in Chapter 3 of the UFSAR. A review of the interactions of these valves with other SSCs has concluded that there are no accidents or malfunctions of a different type created. The correction to the error in UFSAR Table 6.2-7 will not invalidate assumptions and inputs to the design of the interacting SSCs and accident analyses and transients.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because correcting the error to UFSAR Table 6.2-7 will not cause the ACAD pressure bleed subsystem containment isolation valves to be inoperable. These valves will maintain their containment integrity in accordance with design requirements. The valves will function in accordance with Tech Spec requirements for Containment Integrity, Tech Spec 3 / 4.7A, Primary Containment Isolation Valves, Tech Spec 3 /4.7D, and Containment Design Temperature Pressure, Tech Spec 5.2.B. The margin of safety in the bases for these Tech Specs is not reduced.

Safety Evaluation Number: SE-98-060

Type of Safety Evaluation: Drawing Change Request; UFSAR Revision

Evaluation Reference Number: DCR 980056; UFSAR-97-R5-047

Title: EQ Zone Map Drawing Revision and UFSAR Change

DESCRIPTION:

Revise UFSAR Section 3.11 and issue DCR 980056. DCR 980056 is issued to create a new Drawing, M-4A, Sheets 1 through 5. This drawing depicts the boundaries and location of the Environmental Qualification (EQ) Zones within the plant. It will identify the temperature, pressure, humidity, and radiation parameters for each EQ Zone under normal, High Energy Line Breaks (HELB), and a Loss of Coolant Accident (LOCA) conditions.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there are no functional or physical changes being made to any SSC due to this UFSAR change or the DCR drawing change. The change made to the UFSAR Section 3.11 will incorporate the revised environmental parameters calculated in Bechtel Specification 13524-069-N202.

Revision 7, NDIT QDC-98-0124, and QDC-98-0161 and NDIT QDC-98-0169. The revised parameters do not affect normal plant operation or operation of the SSCs required to operate under these conditions. All EQ Equipment required to operate in these conditions has been reviewed and found acceptable.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes in parameters calculated are the direct results of the postulated DBAs and do not affect any other accidents. This change does not affect normal plant operation or operation during any other DBA or transient. No functional changes have been made to any SSC due to this proposed change.

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

Safety Evaluation Number: SE-98-070

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-054

Title: NPSH Requirements for HPCI

DESCRIPTION:

UFSAR changes to address NPSH requirements for the High Pressure Coolant Injection Subsystem of ECCS and to increase the maximum suppression pool temperature expected during an Anticipated Transient Without Scram (ATWS), for which HPCI would have to operate, from 146°F to 156°F.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because these changes to the UFSAR involve the required and available net positive suction head to the HPCI pump and the expected pressure suppression pool temperature during an ATWS event. Since the NPSH of the HPCI pump has nothing to do with initiating any accident, these changes cannot increase the probability of occurrence of an accident.

   The consequences of an accident would not increase because the HPCI subsystem, if needed, would respond to any event in the same manner as before the change. The required NPSH for the HPCI pump is not being changed by this UFSAR change. This change includes a reanalysis of what the maximum temperature in the torus will be during an ATWS event. The HPCI pump will supply the required flow to cool the core during any event and the consequences of the event will not change. The new pressure suppression pool temperature expected during an ATWS event is bounded by the analyzed temperature for the torus, torus attached piping, other ECCS pumps and the RCIC system. Therefore, this change will not affect how the containment or other systems respond to any accident and therefore, will not affect the consequences of any event.

The reanalysis of the available NPSH for the HPCI pump during a LOCA and an ATWS prove that the required NPSH is available and that the HPCI pump will perform as required. The HPCI turbine auxiliaries which will be affected by this higher maximum suppression pool temperature are designed for temperatures in excess of the newly calculated maximum and will perform their required design functions at the higher temperature. The torus, torus attached piping, other ECCS pumps and the RCIC system are designed for temperatures in excess of the newly calculated maximum torus temperature. Therefore, the probability of a malfunction of equipment important to safety is not increased.

The changes being made to the UFSAR do not affect the operation of the HPCI subsystem or any other equipment important to safety. If there were a failure of equipment important to safety, the consequences would remain the same. The HPCI pump, other ECCS pumps, the RCIC system and the torus will still operate as required by their design bases. No release paths or release rates are being affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes being made to the UFSAR do not affect the operation of the HPCI subsystem or any other equipment important to safety. No initiators of any accident are affected by these changes. A further description of the required and available NPSH for the HPCI pump cannot affect the operation of the plant such that the possibility of an accident or malfunction is created.

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

Safety Evaluation Number: SE-98-074

Type of Safety Evaluation: Modification; UFSAR Revision

Evaluation Reference Number: DCP 9800181; DCP 9800182; UFSAR-97-R5-056

Title: Install Check Valve and Strainer in HPCI Room Cooler Service Water Line

DESCRIPTION:

This modification installs a redundant safety-related check valve in series with the existing check valve to protect against single active component failure impacting the capability of the Safety Related Diesel Generator Cooling Water (DGCW) system from performing its safety function. A duplex strainer and associated instrumentation and valving is also being installed in a section of non-safety related Service Water system piping. The purpose of this strainer is to remove debris from the cooling water supply from the alternate Service Water supply to the HPCI Room Cooler. The strainer is isolated from the cooling water flow-path when the HPCI Room Cooler receives cooling water supply from the safety-related DGCW system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Service Water system is non-safety related, is not assumed to function in any accident, and is not credited for mitigating the consequences of an accident. The applicable piping and components installed by these modifications are seismically supported. A failure of the piping system for these modifications would not significantly affect the Service Water system. Floor drains are also available in the event of piping failure. The new check valves provide double isolation between the Service Water system and the Diesel Generator Cooling Water system and are included in the IST Program.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new duplex strainers provide better filtration of the water being supplied by Service Water to the HPCI Room Cooler than is normally supplied by the safety-related DGCW system. These modifications do not impact the ability of the DGCW system to provide cooling water flow to the HPCI Room Cooler when necessary. The second check valve ensures adequate separation between the safety-related DGCW system and the non-safety related Service Water system in the event Service Water system pressure is lost.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because DGCW supply to the HPCI Room cooler is still available when required and is not affected. Adequate separation exists between safety-related and non-safety related systems, and applicable equipment installed via these modifications has been analyzed seismically.

Safety Evaluation Number: SE-98-096

Type of Safety Evaluation: Technical Specification Bases Change

Evaluation Reference Number: Technical Specification Sections 3/4.9.E.

Title: Bases Change for Tech Spec Sections 3/4.9.E. (also supports Tech Spec 3/4.9.F)

DESCRIPTION:

The bases for Tech Spec Surveillance requirement 4.9.E contains the sentence "The surveillance requirements verify that the A.C. and D.C. electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power." This change deletes "and the buses energized from normal power."

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. The elimination of the words "and the buses energized from normal power" from bases section 4.9.E does not allow for any new alternate electrical lineups. Alternate electrical system lineups that are allowed (both normal and abnormal), are controlled by other Technical Specifications and by administrative controls. One of the methods of limiting the consequences of an accident is the flexibility built into the design of the electrical distribution system. There is no impact to the ability of the electrical distribution system to support systems that mitigate consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change clarifies the purpose of the electrical distribution weekly verification surveillance requirement as verifying that the electrical power distribution systems have the correct circuit breaker alignment. UFSAR Section 8.3 discusses the various cross-ties and alternate arrangements within the capability of the distribution system. Alignments that are correct, but other than normal, are controlled by other Technical Specifications and by administrative controls.

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

Safety Evaluation Number: SE-98-098

Type of Safety Evaluation: Modification; UFSAR Revision

Evaluation Reference Number: DCP 9800235; UFSAR-97-R5-073

Title: Appendix R EDG Fuel Oil Transfer Pump and Fire Pump Day Tank Modification

DESCRIPTION:

Perform logic changes to the control circuits of the EDG 1 and 2 Fuel Oil Transfer Pumps to address Appendix R concerns. The Fire pump day tank fuel oil piping will also be modified by adding bypass valves around solenoid valves SO ½-5202 and –5203. These changes are necessary due to the possibility of a single fire damaging both the Unit 1 and 2 EDG transfer pumps. Fires could also cause fire pump fuel oil inlet solenoid valves to mal-operate preventing fuel transfer.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the failure of any new component produces the same result as a failure in the existing configuration (i.e. loss of the transfer pump, loss of the fire pump). These failures are already addressed in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new system configurations are enveloped by the existing configurations as described in the UFSAR. All post-modification failures produce the same results as failures of the pre-modification configurations.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification requirements for the fire pumps or its piping. The changes made to the EDG 1 and 2 fuel oil transfer pump only affect the fuel transfer operation in regards to the fire pumps. The method of transferring fuel to the EDG system remains unaffected.

Safety Evaluation Number: SE-98-103

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-080

Title: Update UFSAR Section 9.1.3 To Allow A More Rigorous Analysis Of Fuel Stored In The Spent Fuel Pool

DESCRIPTION:

The heat load of spent fuel(s) will be calculated based upon the number of fuel assemblies to be discharged, the time between reactor shutdown and the start of fuel offload, the rate of fuel offload, and the number of fuel assemblies to be transferred into the other unit's spent fuel pool.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there are no physical changes made as a result of this issue. The procedure changes implement administrative controls that do not affect how the system is operated. This change does not interact with any system or process that may cause a loss of fuel pool cooling event.

   The current analysis of rod assembly drop accident assumes that fuel moves commence within 24 hours after the reactor scram. The results of this analysis are not changed by this safety evaluation.

   A transfer canal allows the opposite unit's fuel pool cooling system and RHR-fuel pool cooling assist mode to cool both fuel pools at Quad Cities station. The opposite units FPC system and RHR-FPC systems provide similar heat load removal capability.

   The RBCCW, Service Water (SW), RHRSW system will be administratively controlled to a value less than the design temperature for the system. Prior to each shutdown, outage specific calculations will be performed to demonstrate that the time to boil, boiloff rate, and the local boiling temperatures are not exceeded.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no physical changes being made as a result of this safety evaluation. The equipment and systems will operate at the higher heat loads predicted by this change. There are no new equipment failures created as a result of this change.

   Analysis for Q1R15 has demonstrated that the maximum pool temperature (normal discharge), peak pool temperature (following a full core offload), maximum boil-off rate, minimum time to boil, and temperature at the fuel exit can be maintained at the higher heat loads.

   This change will not impact fuel/ fuel pool equipment failures associated with the loss of fuel pool cooling. Three constraints defined in the HDFR-SER associated with the station's response to a loss of fuel pool cooling event are boil off rate (51 gpm), time to boil (7.4 hours), and local boiling (167 ° f). This change will not introduce the possibility of a dual unit loss of fuel pool cooling, because the heat load in the other unit is maintained within the heat removal capability of one fuel pool cooling train. Similar to previous outages, these limits can be administratively controlled to ensure the assumptions made in the calculation remain valid. Required action for exceeding these limits or changes, to the pool configuration, would be to stop fuel moves.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced. Incorporation of this change into the plant design increases the calculated worst case k_{eff} by 0.000069. The worst case k_{eff}, incorporating this change will then become 0.932007, which is less than the limit of $k_{eff} < 0.95$. Therefore, the administrative limit is increased but the margin of safety has not been reduced by this change. Because the bulk pool temperature is maintained below 145.6°F, the structural integrity of the fuel pool racks is also maintained.

Safety Evaluation Number: SE-98-106

Type of Safety Evaluation: UFSAR (SSR) UPDATE

Evaluation Reference Number: N/A

Title: Revise The Existing SSR To Include Actions To Mitigate The Consequences Of Normal Initiation Of The RHR System Pumps On High Drywell Pressure And/Or Low-Low Water Level Initiation Signals Concurrent With A Loss Of Off-Site Power

DESCRIPTION:

The current SSR does not consider an auto-start of the RHR system based upon a valid initiation signal(s). This auto start, in conjunction with one (1) spurious operation (auto closure) of the 1001-18A or 18B valve(s) could result in damage to the affected RHR pump(s) due to the lack of a discharge flow path.

Based upon the results of these evaluations, additional SSR actions are required to mitigate this potential for pump damage because the RHR system is utilized to achieve and maintain cold shutdown at a later point in the Appendix R fire response timeline. The subject procedures are being revised to direct implementation of the SSR revision.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the RHR pumps will be isolated to prevent potential damage due to a spurious operation of the 1001-18A or 18B valve concurrent with a valid RHR initiation. The proposed changes serve to isolate this SSD equipment which could start without being properly aligned and may have an adverse impact on the safe shutdown process. Thus, the revision to the SSR and subject procedures serve an additional protective function for the

equipment and for the overall safe shutdown process. These changes, then, will not increase the potential for the malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because these revisions are necessary to add a requirement and provide implementing instructions to electrically isolate the RHR system pumps within the first 9 minutes of accident initiation to prevent them from being damaged due to a potentially impaired discharge flow path. Electrically isolating the RHR pumps within the specified timeframe serves to ensure that these pieces of SSD equipment will be free of damage when they are required to perform their SSD design functions (decay heat removal & suppression pool cooling) at a later point/mode in the procedural response timeline for the identified accidents. Therefore, no new accident scenario or malfunction of equipment required for SSD is created by these revisions to the SSR or subject procedures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameter on which the Technical Specification is based.

Safety Evaluation Number: SE-98-107

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-075

Title: Revision to UFSAR Section 6.2.6.3.1 (6.2-53)

DESCRIPTION:

Revise UFSAR section 6.2.6.3.1 to reflect the station requirement to reduce load to less than 75% power prior to stroke timing the MSIVs, and to delete a statement that the MSIVs are seat leak tested by monitoring drainage from test taps during hydrostatic tests which is a practice no longer used at the station.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because full-closure exercise testing at a reduced power level (<75% Power) is a more conservative approach to stroke timing the MSIVs. Testing the MSIVs at a lower power level (< 75%) reduces the risk to the operating unit by avoiding reactor power fluctuations and pressure spikes that can lead to a reactor trip.

The elimination of the statement that an MSIV leak test can be performed by monitoring drainage from the test taps during hydrostatic tests will not affect the accident probability because Appendix J Local Leak Rate testing will still be performed. Appendix J testing provides quantifiable leakage data that is more accurate and can be trended rather than the qualitative approach of monitoring drainage during a hydrostatic test.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this testing is designed to detect and monitor any degradation of the MSIVs to ensure the valves are adequately maintained during shutdown periods to prevent any unexpected failures. The proposed changes do not create any new modes of operation for the MSIVs and do not introduce any new operating conditions that deviate from the MSIVs intended design functions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the acceptance limits for the MSIVs have not changed. The UFSAR changes only the operating conditions at which the MSIVs are tested. The testing procedure QCOS 0250-04 already provides adjustments to accommodate the affects steam flow has on the MSIV closure time. These adjustments reduce the acceptable stroke time range at low power levels (< 500 MWe) to ensure the closure time of the MSIVs remain between 3 and 5 seconds when at normal operating conditions (> 500 MWe).

Safety Evaluation Number: SE-98-108

Type of Safety Evaluation: Procedure Change

Evaluation Reference Number: QOP 6900-11, QCOS 6900-02 and QCOS 6900-15

Title: Place Battery Chargers to Equalize Mode

DESCRIPTION:

Place the 4 battery chargers that feed the A and B banks of the 24/48V batteries to their equalize mode, and leave them there until February of 1999.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the increase in voltage will decrease the probability of malfunction of the Scram Discharge Volume Level Switches by allowing them to operate at their designed voltage. All of the equipment is already designed to operate at the highest equalize voltage. This does not change the values of the

voltage setpoint. The only component that was designed to normally be operated at a lower voltage was the battery. The battery chargers and the AC system that supplies the chargers are designed to allow the chargers to operate continuously at equalize voltage.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the current procedures establish the equalize voltage range. The only new aspect of this change is the time period of the equalize charge. The accelerated aging of the battery will not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR. Maintaining the system in the equalize mode has no adverse effects on the remaining components. The increased voltage will improve the operation of the loads over the performance of that equipment at the present lower voltage.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the plant conditions created by this change do not conflict with the requirements of Tech Specs.

Safety Evaluation Number: SE-98-116

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-083

Title: UFSAR Changes for PIFs Q1998-00556, 00655, 00785, 00789, 00725, 02243, 03256

DESCRIPTION:

Revises Figure 8.3-1 to correct breaker numbers and designation of transformer 18 and 19. Revises Table 8.3-1 to correct loading for 4160 volt and 480 volt loads given in sheets 4 & 5. Changes page 8.3-6 to specify well water pump #5 as a load from bus 24 versus the pump house transformer. Changes page 8.3-32 to clarify that the 125 VDC control power cross-tie between bus 13-1 and 23-1 breakers have no trip elements and have a locking clip versus a lead seal.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the affected changes do not affect the initiating events of any accident. The equipment does not affect the loading on the auxiliary power system or change the function of the equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because equipment failure modes and system interactions have not changed. The loading on the system as evaluated by the ELMS database has not changed.

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

Safety Evaluation Number: SE-98-117

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-085

Title: Revision To UFSAR Tables 6.3-13, 6.2-7, 6.3-5, and 6.3-2; Figures 6.3-3, 5.4-11, 6.3-5 Sheet 1 and 2, 6.3-12, 6.3-2, and 9.2-1; Pages 6.3-12, 6.3-5, and 7.3-4

DESCRIPTION:

Change Table 6.3-13 ECCS Single Valve Failure Analysis, to correct confusing information. The table lists valves as example (only one valve on one loop on Unit 1). The column labeled "Total Number of Valves at Station" makes it clear that valves on both loops on both units are included in the analysis. This change provides the EPN numbers so that all valves are clearly identified on the table and eliminates the Unit 1 P&ID numbers since this is unnecessary.

Change Figure 5.4-11 (Simplified one-line diagram of the RHR System) to correct the position of valve 1(2)-1001-41 (CCST suction valve to the RHR pumps) to color the valve as closed. The figure already shows this valve as "LC" (Locked Closed), however the figure inadvertently showed the valve as open by failing to shade the valve as black. Table 6.3-5, RHR Pump Design Parameters, needs to show pump original head at 4500 gal/min for each pump as 400 feet (from the manufacturer's pump curves). Adds the words "approximately 230 feet is required for pump operability."

Figure 6.3-12 needs to be corrected to show Recirculation System equalizing valve as normally closed. The purpose of this figure is to show the arrangement of the LPCI Logic Control System for the delta pressure instruments. The figure needs to have the correct position shown for the equalizing valves to be totally correct. Section 5.4.1.2.2, Recirculation Pumps, Valves and Piping, states these valves as closed. This change makes the figure agree with this section.

Figure 9.2-1 (Simplified one-line diagram of the RHR Service Water System) incorrectly shows the location of the pressure breakdown orifices for the RHR Service Water Cubicle Coolers. The orifices are actually located on the inlet to the coolers not on the outlet. This lowers the pressure to the coolers to within design limits. This corrects an error that was made when the one-line diagram was drawn from the P&ID.

UFSAR Section 6.3.2.2.3.2 states that the RHR system contains check valves in the containment that are equipped with pneumatic operators to permit remote exercising and testing during normal plant operation. This is clarified that the normal plant operation specified is when the Reactor is depressurized.

UFSAR Table 6.2-7 is clarified with the following corrections:

a. Valves MO 1001-23A/B and MO 1001-34A/B contain a "C" in the Test Class column. These valves are in the "Type C Local Leak Rate Test program" but are not actually tested due to having a qualified water seal on the back side of the valve. A note to this effect is being added to the table.

b. To penetration Number X-210A,B add valves 1001-18A/B, RHR min flow bypass to make the RHR system consistent with Core Spray, HPCI, and RCIC which all contain min flow valves in the line in this penetration.

c. Remove/move Note (4) for several valves which are not actually Pressure Isolation Valves (PIV).

Change UFSAR Section 6.3.2.1.2 to clarify that the horsepower rating of the core spray pump is within the nameplate plus the 1.15 service factor.

Change UFSAR Table 6.3-4 to indicate that the pump impeller is stainless steel not bronze and add the horsepower clarification as above.

Change UFSAR Figure 6.3-2 to correct the Core Spray one-line diagram to show loop tie-ins as they are on Unit 1.

Change UFSAR Figure 6.3-3 to correct the pump curve so that it more closely resembles the CS pumps.

Change UFSAR Figure 6.3-5 sheet 1 and sheet 2 to correct discrepancies in the Functional Control Diagram to show the actual "as built" condition of the system. The only technical change being implemented on the FCD's is the insertion of the 9 minute timer that starts Core Spray.

Change UFSAR Section 7.3.1.1.1.8 to correct the description of the pressure switches that provide the Low Reactor Pressure Permissive for the opening of the Core Spray Injection Valves. The Current description states that there are four switches that provide the function when in fact there are only two switches. The installation of only two switches is substantiated in UFSAR section 7.3.1.1 and the GE Process diagram 729F230.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because these portions of the ECCS systems are not connected to the reactor coolant system, the aux power system, or the reactor protection system. In addition, these components are not connected with any of these accidents' initiators. These systems are used in accident mitigation and are isolated from accident initiators. The systems' responses to an accident are not changed and therefore, there is no change to the consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because these changes revise the UFSAR to the actual plant configuration required to ensure that the design basis requirements are met. The actual plant configuration is correct. In some cases the changes are purely editorial or add additional clarifying information. In all cases, these changes will not result in the creation of any accidents or malfunctions different than those already evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification 4.5.A.2.b provides an acceptance limit of two RHR pumps providing a flow of 9000 gpm against a system head corresponding to a reactor pressure of 20 psig. Calculation QDC-1000-M-0587, which is an input to QCOS 1000-06 and is performed quarterly, assures that this limit is met. The changed pump design parameters table shows pump design flow above the levels in calculation QDC-1000-M-0587. Therefore, acceptance limit is satisfied. For the change to the Core Spray Pump Curve figure in the UFSAR, Technical Specification 3.5.A.2.a provides an acceptance limit of each CS pump develop a flow of at least 4500 gpm against a system head corresponding to a reactor pressure of 90 psig. This acceptance limit is well within the new pump curve figure.

Safety Evaluation Number: SE-98-121

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-084

Title: Updated Analysis of RHR and CS Pump Short Term NPSH Design Basis

DESCRIPTION:

The purpose of this Safety Evaluation is to support the change to the existing Net Positive Suction Head (NPSH) analysis for the RHR and CS pumps as described in UFSAR Section 6.3.3.2.9, "Net Positive Suction Head Availability." UFSAR Section 6.3.3.2.9 describes the results of the original NPSH analyse. for the RHR and CS pumps following a Design Basis Loss of Coolant Accident (DBA-LOCA), and UFSAR Figures 6.3-41 and 42 provide time-dependent curves of the Containment Overpressure (COP) available and the COP required to satisfy the NPSH requirements of the RHR and CS pumps. This change will revise Section 6.3.3.2.9 to describe the updated short-term (first 600 seconds) analysis and add Figure 6.3-57 to incorporate the results of the short-term NPSH calculation QDC-1000-M-0454, Rev. 1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change involves the available COP following a DBA-LOCA and the resulting NPSH available to the RHR and CS pumps. While this change affects the ability of these pumps to perform their required functions following a DBA-LOCA, it does not increase the probability of the accident because there is no impact on reactor recirculation piping or the reactor pressure boundary.

   The COP and NPSH evaluations support the ability of the RHR and CS pumps to provide the flow rates assumed in the Appendix K fuels analysis and have no effect on containment integrity. Therefore, the consequences of an accident are not increased by this change.

   The probability of the malfunction of the RHR and CS pumps is not increased by this change. The evaluations determine that pump tests have proven the ability of the RHR and CS pumps to withstand the effects of short-term cavitation. Therefore, the probability of a malfunction of equipment important to safety has not increased.

   The NPSH evaluations demonstrate the ability of the RHR and CS pumps to provide the flow rates assumed in the Appendix K fuels analyses for even the most limiting single failure. Therefore, the consequences of a malfunction of equipment important to safety are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the probability of the malfunction or accident is not increased by this change. This change deals exclusively with the COP credited following a DBA-LOCA in the short-term (first 600 seconds). The evaluations demonstrate that pump tests have proven the ability of the pumps to withstand the effects of short-term cavitation and the RHR pump tests are directly applicable to the CS pumps. Therefore, a different malfunction or accident has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the CS pump requires about 3.0 psig COP for approximately 3 minutes following a DBA-LOCA. The amount of COP is consistent with the "few psi" reviewed and approved by the SER, and the duration is within the approved "about 8 hours."

Safety Evaluation Number: SE-98-122

Type of Safety Evaluation: Procedure Change

Evaluation Reference Number: QCOS 1300-23, Revision 0 (formerly QCTS 0300-03)

Title: RCIC Logic Functional Test

DESCRIPTION:

Procedure revisions to allow the RCIC logic functional tests to be performed while in reactor Mode 1 or 2. Other changes are to enhance contact testing adequacy of the test.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change to the RCIC Logic Functional Test does not affect any of the initiators of these events. The changes to the procedure temporarily isolate the RCIC system turbine and pump to test the logic while the reactor is either in Mode 1 or Mode 2. Testing is only performed on RCIC system while the system is isolated. The system will be placed back to its normal configuration at the completion of the logic functional test.

   The function of the RCIC system in all of these events is as a mitigator to the accident by providing cooling water to the reactor and to assist with pressure control during the event. Provided that HPCI system is verified operable, Technical Specification 3.5.D Action Statement indicates that the unit may stay on-line for up to 14 days with RCIC not operable. The HPCI system is designed similar to the RCIC system in that it can provide cooling water to the reactor core whenever the feedwater system is lost. In addition, however, HPCI is designed to provide coolant inventory to compensate for a LOCA. Therefore, HPCI serves a complementary function to RCIC. This revision to the RCIC Logic Functional Test has Operations verify that HPCI is operable. Therefore, the consequences of these events are not increased because core cooling will be maintained by the HPCI subsystem.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because based on the RCIC system being isolated as previously described for testing purposes and the fact that the RCIC turbine or pump are not operated during this testing, the RCIC system will not be operated in an abnormal line-up, outside of its design basis, nor in any manner that impacts its design functions or any other plant equipment. The procedure will be performed in the 901(2)-48 panel. Only RCIC equipment is located in this panel, so no other systems will be impacted. Because of this fact, there is no possibility for an accident or malfunction to occur of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the RCIC logic functional test is required to be performed every 18 months. The RCIC logic functional test procedure, as changed, meets this requirement.

Safety Evaluation Number: SE-98-123

Type of Safety Evaluation: Procedure Change

Evaluation Reference Number: QCAP 1500-01, Rev. 10

Title: Administrative Requirements for Fire Protection Procedure

DESCRIPTION:

This procedure change adds portable lighting as an acceptable alternative to compensatory light packs in high radiation areas with ALARA considerations. This is in accordance with guidance provided by Nuclear Operations Division, Nuclear Engineering Standard, NES-MS-05.4, Appendix R Emergency Lighting Program, Section 5.1.4, Supplemental Portable Lighting.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of occurrence, consequence, or a malfunction of equipment important to safety is not affected by QCAP 1500-01, Revision 10. The procedure provides administrative requirements for Fire Protection Equipment. The change to Section D.9.c, Appendix R Emergency Light Packs compensatory requirement, provides portable lighting in the event that Appendix R light packs are inoperable in high radiation areas for more than 14 days. The equipment used for Appendix R Safe Shutdown does not change nor does the method of operation of the equipment. Portable lighting provides an illumination method when the Appendix R emergency light packs are not operational. The procedure change clarifies the compensatory requirements when an Appendix R emergency light pack has a planned out of service less than 14 days.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because QCAP 1500-01, Revision 10, does not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR. Revision 10 provides compensatory requirements when an Appendix R emergency light pack in a high radiation area is not operational for more than 14 days. The portable lighting provides illumination when the Appendix R emergency light pack in a

high radiation area is not operational for more than 14 days. The implementation method of the Appendix R Safe Shutdown procedures is not impacted by the change to the compensatory requirement. The procedure change clarifying the compensatory requirements when an Appendix R emergency light pack has a planned out of service less than 14 days does not affect plant equipment or operation of that equipment.

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

Safety Evaluation Number: SE-98-126

Type of Safety Evaluation: Temporary Alteration

Evaluation Reference Number: DCP 9800279

Title: Quad Cities Unit 2, Replacement of Safety Relief Valve, 2-0220-238, with a Blind Flange on Reactor Recirculation Sample Line 2-0251-3/4"-A

DESCRIPTION:

Temporary Modification, DCP 9800279, replaced safety relief valve, 2-0220-238, with a blind flange. This valve is located inside primary containment on Reactor Recirculation Sample Line (RRSL), 2-0251-3/4"-A, between containment isolation valves AO 2-0220-44 and AO 2-0220-45. The valve has experienced leakage or spurious operation, which has resulted in increased drywell temperature and pressure. It was observed that when the inboard (upstream primary containment isolation) valve was closed, that drywell temperature and pressure returned to normal.

The relief valve was originally installed under DCP 9700130 to prevent possible thermal overpressurization of the RRSL which could result from a LOCA/HELB if the containment isolation valves, AO 0220-45 and 45 were closed. This overpressurization could result in a failure of the containment penetration boundary on the RRSL. This potential failure mechanism was identified in NRC Generic Letter 96-06.

As part of this temporary modification, requirements are included to maintain the subject containment isolation valves in the closed position and to blow out any water initially present in this volume. This ensures that the volume is initially empty or near empty and therefore thermal overpressurization of the piping is not possible. In the event of leakage past the seats of the containment isolation valves, the volume could become filled with water. However, the same leakage that created the condition would also mitigate any potential thermal overpressure condition. Therefore, it is concluded that the requirements of NRC Generic Letter 96-06 are met.

See safety evaluation SE-98-135 which is Revision 1 to Temporary Modification, DCP 9800279.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new blind flange is fabricated and installed consistent with the requirements of General Work Specification, R-4411, and is therefore suitable for this application. Per calculation QDC-0220-M-0368, the flange and associated piping will be in compliance with all UFSAR requirements for piping deadweight, thermal and seismic loading and will be in compliance with the original code of construction, USAS B31.1.0, 1967 edition. Therefore, it is concluded that the probability of failure leading to small break LOCA is actually somewhat reduced as the potential to lose coolant through spurious opening or sticking of the previously installed relief valve is removed.

   Off-site dose will not increase due to a small line break or large line break inside containment. The blind flange will be installed on a ¾" nominal diameter pipe and will not have an adverse affect on the existing containment boundaries. This line size is well within the size limit discussed in the UFSAR for a small line break and is enveloped by the limiting design basis LOCA. There is no physical interaction between the blind flange and the systems used to mitigate a small break LOCA (HPCI and RHR/Core Spray). Closure of the isolation valves and clearing of the water in the line ensures that there is no possibility of thermal overpressurization of the piping, thus ensuring primary containment integrity in the event of a LOCA resulting from failures unrelated to the RRSL. There is also no physical interaction with the primary containment isolation system and no change in primary containment isolation logic introduced as a result of this temporary modification. Installation of the flange will not adversely affect the ability of these systems to detect an accident condition and initiate primary containment isolation, so it is concluded that primary containment isolation integrity will not be adversely affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because removing the relief valve and installing the blind flange will eliminate the reactor coolant leakage that has been experienced through the relief valve. The new piping components have the same failure modes and consequences as the previously existing components.

   Imposing the requirements to maintain the isolation valves in the closed position following the blowing out of water in the line will eliminate the possibility of containment penetration boundary overpressurization should a LOCA/HELB occur inside drywell, thus meeting the requirements of GL 96-06. As an additional compensatory measure, a baseline pressure reading and subsequent daily pressure readings for the isolated RRSL will be taken to determine if in leakage is occurring in this isolated volume. This temporary alteration does not preclude operation of the 2-0220-44 and 2-0220-45 valves for the purpose of post-accident sampling via the HRSS. The use of the RRSL in this condition is acceptable since the event that would cause the thermal type of overpressurizaton as described in GL 96-06 has already occurred.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because for Tech. Specs. 3 / 4.6.I, PRIMARY SYSTEM BOUNDARY -- Chemistry, and 3 /4.6.J, Specific Activity, even though closing of the containment isolation valves will isolate the primary sample point for reactor water sampling and analysis, alternate sample points to the RRSL are available.

   For Tech Specs. 3 / 4C.N: PRIMARY SYSTEM BOUNDARY --Structural integrity and 3 /4 .7.A: PRIMARY CONTAINMENT INTEGRITY, the structural integrity is maintained. The integrity of these components and primary containment is maintained because the volume of water between the closed containment isolation valves will be blown out to prevent the possibility of thermal overpressurization. As an additional compensatory measure, a baseline pressure reading and subsequent daily pressure readings for the isolated RRSL will be taken.

   For Tech Spec 3 /4 .7.D: PRIMARY CONTAINMENT ISOLATION VALVES, primary containment isolation valves AO-0220-44 and 45 will be fully operable even though they will be maintained in the closed position. Since the valves are normally open and automatically close upon the initiation of a LOCA, they will already be in the required position to provide primary boundary and primary containment integrity.

   For Tech Spec 5.2, CONTAINMENT DESIGN FEATURES, closing the valves and replacing the relief valve with a blind flange will not compromise the design requirements stated in the specification. The flange will be fabricated and installed in accordance with the requirements specified in General Work Specification R-4411 and associated Piping Design Table A.

Safety Evaluation Number: SE-98-129

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-087

Title: Review and Audit Functions

<u>DESCRIPTION:</u>

This change will remove the detailed description of the Review and Audit functions from the UFSAR.

The required Review and Audit functions will not change. The proposed change to the UFSAR will be in section 13.4 "Review and Audit".

The fir[illegible]o sentences in paragraph 13.4-1 which state "The Chief Nuclear Officer … respo[illegible]y for … safety." and "Onsite reviews … are briefly described in the following sections" will be deleted.

Sections 13.4.1 "Onsite Review", 13.4.2 "Independent Review", and 13.4.3 "Audit Program" will be deleted in their entirety.

What remains in section 13.4 is the statement "Review and Audit functions … are established in accordance with the ComEd Quality Assurance Topical Report." (QA Manual).

This change is being made to remove the detailed description of the Review and Audit functions from the UFSAR. This is being done so that changes to the UFSAR are not required simply to keep it current with changes in titles, and or minor administrative changes in the mechanisms for Review and Audit. Also, this change will reduce redundancy with other documents that contain the same information such as the Quality Assurance Topical Report and Nuclear Oversight Procedures.

This change will remove only the descriptions of the Audit and Review activities from the UFSAR. There will be no functional change to the Audit and Review activities. The detailed descriptions of Audit and Review activities are preserved in the Quality Assurance Topical Report. The Topical Report is submitted to and reviewed by the USNRC. Therefore, any proposed change to the substance of the Audit and Review activities will be critically examined in a manner similar to the scrutiny it would receive if proposed as a UFSAR change.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this is an administrative change. There is no direct effect on any system, structure, or component and therefore no effect on any equipment. This change will not introduce any new failure modes for equipment. It will not affect any of the known equipment failure modes nor will affect the accident response of any plant system, structure, or component. Therefore, the probability or consequences of an accident or of a malfunction of equipment important to safety is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the proposed change is administrative only. Because the existing administrative requirements for the Review and Audit functions have not been changed, there can be no direct affect on any system, structure, or component. Since the changed description location has no direct affect on any system, structure, or component, it cannot introduce any new failure modes nor can it affect any known failure modes.

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

Safety Evaluation Number: SE-98-130

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-086

Title: Change to UFSAR Description of RBCCW System

DESCRIPTION:

Revise the UFSAR to:

- Correct Load table to indicate 2 RWCU Non-Regen Hx as the normal lineup.
- Add HWC Autoclave system as a Unit 2 heat load.
- Add Primary Containment Particulate Sample Cooler as a heat load.
- Specify that the Unit 2 Primary Containment Particulate Sample Cooler and Unit 2 Reactor Building Sample Panel Coolers are not isolated by closing the 2-3701 valve.
- Remove the specific flows for system loads on the system load table.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes do not affect the ability of the system to perform its design functions. None of the changes affect equipment important to safety directly and do not decrease the reliability of the system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes do not involve components or functions different from those already included in the description of the system. All changes were installed in accordance with the applicable codes and standards. All changes involve simple, passive components expected to be of high reliability.

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

Safety Evaluation Number: SE-98-131

Type of Safety Evaluation: UFSAR Revision

Evaluation Reference Number: UFSAR-97-R5-088

Title: Revision to Quad Cities UFSAR Table 5.1-1, Reactor Coolant System Data, RCIC Pump Type

DESCRIPTION:

Quad Cities UFSAR Table 5.1-1, Reactor Coolant System Data, incorrectly states the number of stages for the RCIC Pumps. Table 5.1-1 incorrectly states that the RCIC pumps are single stage horizontal, centrifugal pumps. The as installed RCIC pumps are 5-stage horizontal centrifugal pumps. UFSAR Table 5.1-1 is being revised to reflect that the Unit 1 and Unit 2 RCIC pump is a 5-stage, horizontal, centrifugal pump.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the RCIC system pumps are used to mitigate the accidents and transients evaluated in the UFSAR that are applicable to the RCIC system. The RCIC system pumps cannot initiate the accidents and transients analyzed in the UFSAR. The change does not affect the RCIC pumps ability to perform its design functions to mitigate accidents or transients. Therefore, the consequences of an accident or malfunction of equipment important to safety will not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because UFSAR Table 5.1-1 incorrectly states that the RCIC pumps are single stage, horizontal, centrifugal pumps. The RCIC pumps are Bingham 5-stage horizontal, centrifugal pumps that are original equipment installation. These pumps were originally selected and installed because of their capability to operate within the RCIC system design parameters and requirements as stated within the RCIC pump design specifications, UFSAR, and Technical Specifications. Surveillance tests performed on these pumps since initial startup have verified their capability to meet the RCIC system pump design basis requirements. Since the changes to the UFSAR will not change the facility, there is no change to the way the RCIC system will operate.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because changing the description of the RCIC pump from a single stage to a 5-stage, horizontal, centrifugal pump will not reduce the margin of safety as defined for Tech Spec 3 /4.5.D. Technical Specification surveillance tests performed since original plant startup have shown that the 5-stage pumps meet the

flow rate of ≥ 400 gpm for the pressures stated for the Tech Spec surveillances for Tech Spec 4.5.D.

Safety Evaluation Number: SE-98-132

Type of Safety Evaluation: Procedure Change

Evaluation Reference Number: QCOS 2300-29, Revision 0 (formerly QCTS 0300-04)

Title: HPCI Logic Functional Test

DESCRIPTION:

The reason for these changes is due to procedure QCOS 2300-29 being performed during non-outage time periods as opposed to the previous procedure (formerly QCTS 0300-04) that was performed during outage periods. The HPCI logic is in a different configuration when the unit is on line as opposed to off-line. For instance, the HPCI low reactor pressure isolations and high reactor water isolations that are received during unit outages will not be received during on-line operation.

In addition, other changes are due to additional precautions because this test is being performed during on-line operations. Normally locked open manual pump suction valve 2301-22 will be closed. The precautions of the procedure ensure that both the HPCI pump and turbine are isolated during the test.

Other changes are to enhance contact testing adequacy of the HPCI logic and to re-arrange contact testing to increase the efficiency of the testing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change to the HPCI Logic Functional Test does not affect any of the initiators of these events. The changes to the procedure temporarily isolate the HPCI system turbine and pump to test the logic while the reactor is either in Mode 1 or Mode 2. Testing is only performed on HPCI system while the system is isolated. When the system is placed back to its normal configuration at the completion of the logic functional test, the HPCI steamline is slowly pressurized to prevent a piping failure.

   The function of the HPCI system in all of these events is as a mitigator to the accident by providing cooling water to the reactor and to assist with pressure control during the event. Provided that ADS, RCIC, LPCI, and both loops of Core Spray systems are verified operable, Technical Specification 3.5.D Action Statement indicates that the unit may stay on-line for up to 14 days with HPCI system not operable. Therefore, the consequences of these events are not increased because core cooling will be maintained by these alternate systems.

To increase the consequences of this accident, flow would have to be increased above design injection flow of the HPCI system. The revisions to this procedure do not disturb components involved with the pump flow rate of HPCI. Therefore, this revision to the HPCI logic functional test cannot increase the consequences of an inadvertent initiation of HPCI during power operations.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because based on the HPCI system being isolated as previously described for testing purposes and the fact that the HPCI turbine or pump are not operated during this testing, the HPCI system will not be operated in an abnormal line-up, outside of its design basis, nor in any manner that impacts its design functions or any other plant equipment. The revisions to this procedure will impact steps performed on HPCI equipment only. Because of this fact, there is no possibility for an accident or malfunction to occur of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the HPCI logic functional test is required to be performed every 18 months. The HPCI logic functional test procedure meets this requirement. Hence, there is no change in the margin of safety.

Safety Evaluation Number: SE-98-135

Type of Safety Evaluation: Temporary Alteration

Evaluation Reference Number: DCP 9800279 Rev. 1

Title: Quad Cities Unit 2, Replacement of Safety Relief Valve, 2-0220-238, with a Blind Flange on Reactor Recirculation Sample Line 2-0251-3/4"-A

DESCRIPTION:

Revision 0 to Temporary Modification (TMOD), DCP 9800279, replaced safety relief valve 2-0220-238, with a blind flange. This TMOD was required when it was determined that the relief valve had experienced spurious operation and had failed to reset during plant operation. No suitable replacement valve was available. The original issue of this TMOD required that containment isolation valves, AO 2-0220-44 and 45, be placed in a closed position and the volume in between these valves blown clear of water. The pressure in the volume was to be monitored on a periodic basis to ensure that the volume was initially empty or near empty so that thermal overpressurization of the piping would not be possible. A 10CFR50.59 Safety Evaluation (#SE-98-126) was performed for Revision 0 which concluded that the installation of the TMOD did not alter any station Technical Specifications or represent an unreviewed safety question as long as these actions were taken. Based upon the line being blown clear of water between the two containment isolation valves, it was concluded that thermal overpressurization of the trapped volume was not a concern. Revision 1 of this TMOD will provide for the opening of containment isolation valves, AO 2-0220-44 and 45.

Revision 1 to this TMOD is a recommended interim compensatory action resulting from the operability evaluation for PIF Q1998-04363. As discussed above, the valves were originally closed and the volume between the valves was blown empty or nearly empty as a condition of the installation of the TMOD. As part of the Revision 0 TMOD installation requirements, pressure readings were periodically taken via a pressure gauge at the existing LLRT tap. This was done to monitor the volume between AO 2-0220-44 and 45 to ensure the volume remained empty. During plant startup, following installation of the TMOD, a significant increase in line pressure was observed. It was concluded that the pressurization was a result of leakage past the valve seats of AO 2-0220-44, the inboard containment isolation valve. PIF Q1998-04363 was generated to document this condition and the condition of the piping was considered to be operable in a subsequent operability evaluation. The operability determination was based on a high degree of confidence that the piping and fittings can be shown to meet NRC GL 96-06 requirements for thermal overpressurization without an installed relief valve following analysis utilizing ASME Section III, Appendix F criteria, as allowed by NRC GL 96-06 Supplement 1. Completion of the Appendix F analysis is the long term corrective action for this item and will be the basis for making this modification permanent and ultimately removing the TMOD.

Revision 1 to this TMOD is issued as an interim compensatory measure to place the containment isolation valves in an open position. With known seat leakage associated with the inboard valve, it is considered to be preferable to leave the containment isolation valves open. The opening of the subject valves and establishment of flow in the line will ensure that any potential for thermal overpressurization of the piping between the isolation valves is minimized. This piping takes flow from a tap downstream of the "A" reactor recirculation pump discharge, and is at reactor coolant operating temperature. With flow established and the valves open, the temperature of the fluid inside the pipe will be very close to reactor coolant operating temperature. If the containment isolation valves were to close in response to a Group 1 containment isolation signal, due to a LOCA/HELB, the trapped fluid would initially be at or near reactor coolant temperature. This temperature is above the maximum drywell temperature calculated for a LOCA/HELB event (334 deg. F.). Therefore the piping would not be subject to thermal overpressurization as the trapped fluid will not be subject to any expansion. This safety evaluation evaluates the TMOD installation with the containment isolation valves placed in the open position, as an interim compensatory measure for the condition identified under PIF Q1998-04363. This safety evaluation, as allowed under NRC GL 91-18, Revision 1, evaluates whether this compensatory measure has any impacts on other aspects of the facility as described in the SAR. Thermal overpressurization is considered outside of the scope of this 10CFR50.59 as the leakage through the AO 2-0220-44 valve is considered to be a degraded condition and is addressed under the operability evaluation for PIF Q1998-04363.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new blind flange is fabricated and installed consistent with the requirements of General Work Specification, R-4411, and is therefore,suitable for this application. Per calculation, QDC-0220-M-0368, the flange and associated piping will be in compliance with all UFSAR requirements for piping deadweight, thermal and seismic loading and will be in compliance with the original code of construction, USAS B31.1.0, 1967 edition. The conclusions of the referenced calculation are independent of valve position. Therefore, it is concluded that the probability of a failure leading to this accident scenario is not increased as a result of this interim compensatory measure, opening the inboard and outboard isolation valves.

   Opening the inboard and outboard isolation valves will not result in an increase of off-site doses due to a small line break or large line break inside containment. As previously discussed, the installed blind flange is suitable for all operating conditions with the containment isolation valves in the open position. Opening the valves will not create any adverse physical interaction between the blind flange and the systems used to mitigate a small break LOCA (HPCI and RHR/Core Spray). Opening the valves has no effect on Group 1 containment isolation logic or any Group 1 containment isolation instrumentation. The response of the valves to a Group 1 isolation signal is not altered in any way. Thermal overpressurization of the piping due to a failure unrelated to the RRSL is currently outside the scope of this safety evaluation due to the degraded condition described in PIF Q1998-04363. However, the PIF operability evaluation has determined the piping is operable under these conditions. Therefore, it is concluded that the consequences of the accident are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because removing the relief valve and installing the blind flange will eliminate the reactor coolant leakage that has been experienced through the relief valve. It is concluded that the likelihood of a failure of the flange is not more probable than a failure of any of the existing pipe or pipe component and the consequences would be no worse than a failure of the existing piping.

   The opening of the containment isolation valves, as an interim compensatory action does not create any new interfaces with plant systems or with the containment isolation logic or function. Maintaining flow through open valves will ensure that no thermal overpressure condition will occur as a result of a design basis accident. Operability of the subject piping for thermal overpressure is discussed in the operability evaluation for PIF Q1998-04363.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because for Tech. Specs, 3 / 4.6.I, PRIMARY SYSTEM BOUNDARY – Chemistry, and 3 /4.6.J, Specific Activity, opening of the containment isolation valves will allow the primary system coolant to be sampled utilizing the RRSL which is the normal means of sampling.

   For Tech Specs. 3 / 46.N: PRIMARY SYSTEM BOUNDARY –Structural integrity and 3 /4 .7.A: PRIMARY CONTAINMENT INTEGRITY, the structural integrity is maintained. The integrity of these components and primary containment is maintained because the valves will be left open. If the valves isolated due to a LOCA/HELB, the water temperature in the trapped volume would be at or near reactor coolant temperature, which is greater than the maximum drywell temperature of 334 deg. F. Therefore, the trapped water would not be subject to thermal expansion as the water in the pipe is at a higher temperature than the maximum drywell temperature.

   For Tech Spec 3 /4 .7.D: PRIMARY CONTAINMENT ISOLATION VALVES, primary containment isolation valves AO-0220-44 and 45 will be fully operable when the valves are placed in the open position. This is the normal operating condition for the valves. No physical or electrical alterations will be done to these valves to prevent normal operation of the valves. Therefore, surveillances will be able to be performed on these valves as required by this Tech. Spec.

   For Tech Spec 5.2, CONTAINMENT DESIGN FEATURES, opening the valves and replacing the relief valve with a blind flange will not compromise the design requirements stated in the specification.

Safety Evaluation Number: SE-98-136

Type of Safety Evaluation: Design Change, UFSAR Revision

Evaluation Reference Number: DCP 9800078; 9800079; UFSAR-97-R5-097

Title: Determinate Power Cable(s) From Motor-Operated Valves 1(2)-0220-4

DESCRIPTION:

This design change determinates power cable 15031(25031) at the motor terminals of valve 1(2)-0220-4 and at MCC 18-1A(28-1A), cubicle C5(B4). These cables will be left in place. Breakers, control room pushbuttons and the local push-buttons are administratively maintained by the operations department. This configuration will assure that a hot short on cable 15031(25031) will not spuriously operate Hi-Low pressure interface valve 1(2)-0220-4. The subject valves may still be operated manually. A note was added to Figure 10.3-1 in the UFSAR indicating that valve 220-4 is electrically disconnected making it a manual valve.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the motive force used to operate valve 1(2)-220-4 (manual or electrical) has no affect on the initiating event leading to the accidents (break in instrument line outside containment, break in piping inside containment or a severe, un-controlled fire). The loss of electrical capability for these valves will not increase the probability of a malfunction of equipment important to the mitigation of a LOCA or a severe, uncontrolled fire.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because after this modification, the 1(2)-220-4 valves will function as a normally closed, manually operated valve. The valves will not be susceptible to spurious opening due to fire induced hot shorts involving the power cable. Loss of electrical capability for these valves does not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

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

Safety Evaluation Number: SE-98-142

Type of Safety Evaluation: Procedure Change

Evaluation Reference Number: QCOS 1400-11, Rev. 0

Title: Sesquiannual Core Spray Logic Functional Test

DESCRIPTION:

This activity issues a new procedure QCOS 1400-11: Sesquiannual Core Spray Logic Functional Test to perform functional surveillance testing of the Core Spray system initiation logic while the plant is in Mode 1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. This activity does not impact the initiators of the accident or, in the case of inadvertent opening of a relief valve, it does not significantly change the probability of the event occurring. The consequences of the loss of coolant accident are not increased since the minimum systems that are required

for mitigating the consequences are always maintained operable. The equipment important to safety affected in this procedure is declared inoperable and within an AOT.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created by this activity since the accidents and malfunctions possible under this activity have already been evaluated or are bounded by existing evaluations for similar accidents or malfunctions.

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

Safety Evaluation Number: SS-H-98-0114

Type of Safety Evaluation: Validation; Drawing Change Request

Evaluation Reference Number: DCR 970213; DCP 9400083; E04-2-94-118

Title: Actual Field Installation of Support on M-1804-18

DESCRIPTION:

This DCR was written to incorporate the as built condition of support on M-1804-18. This support was added to the HPCI piping on Unit 2 under DCP 9400083, (E04-2-94-118). The drawing was not properly incorporated in the modification closeout so a separate DCR was issued.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new supports decrease the nozzle loads on the HPCI pump, thus improving its reliability, and decreasing the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the HPCI supports do not affect the way the HPCI system performs. The new supports decrease nozzle stresses which improve the long term reliability of the system.

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

Safety Evaluation Number: SS-H-98-0119

Type of Safety Evaluation: Validation; Modification

Evaluation Reference Number: DCP 9700268; SE-97-045

Title: Installation of Unit 1 Zinc Injection System

DESCRIPTION:

Install Unit 1 Zinc Injection System.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new Zinc Injection System is a passive system that does not have any adverse affects on the Unit.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new Zinc Injection System is a passive system that does not have any adverse affects on the Unit. Therefore, the change does not create the possibility of an accident or malfunction different from those evaluated in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Zinc Injection is designed to maintain the required levels of Zinc and the Chemistry of the reactor vessel will comply with Tech Specs.

Safety Evaluation Number: SS-H-98-0123

Type of Safety Evaluation: Validation; Procedure

Evaluation Reference Number: QCOS 6600-06, QCOS 6600-08; SE-98-074

Title: Quarterly Diesel Generator Cooling Water Pump Flow Rate Test, Quarterly 1/2 Diesel Generator Cooling Water to Unit 1 and Unit 2 ECCS Room Coolers Flow Test

DESCRIPTION:

Ensure that ECCS Room cooler flow is supplied from the Diesel Generator Cooling Water Pump and not Service Water.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this surveillance tests only DG cooling water flow to the HPCI room cooler. SW is not assumed to function during any accident described in Chapter 15 of the UFSAR. Two check valves ensure no adverse impact on Diesel Cooling water results if a SW failure occurs. Isolation of SW supply for this surveillance increases the degree of protection should a service water failure occur. The HPCI room cooler is not electrically interlocked or physically interlocked to the HPCI system, and therefore, failure of the room cooler will not impact the system in any way other than to control the ambient temperature of the room.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the surveillance isolates SW supply to the HPCI room cooler and thereby precludes creation of an unanalyzed flow path of DG Cooling Water flow backward into the SW system. The Diesel Cooling Water system function remains unaffected. Availability and reliability of the DG Cooling water system and ECCS systems is not compromised.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because isolation of the SW supply to the HPCI Room cooler does not affect any parameters upon which Technical Specifications are based.

Safety Evaluation Number: SS-H-98-0125

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCAP 1500-1; DCP 9600177

Title: ACAD Compressor

DESCRIPTION:

Remove Unit 1 ACAD compressor from the procedure, it has been permanently disabled.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the system has been disabled and is no longer used. There is no reason to provide fire protection for it.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because ACAD does not have the capability to cause an accident or malfunction.

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

Safety Evaluation Number: SS-H-98-0127

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QIS 0003-01 Rev. 14; SE-98-087

Title: APRM Rod Block and Scram Calibration

DESCRIPTION:

Include the use of the RPS test box during the functional portion of the procedure. This will allow the functional test to be performed without causing a ½ scram condition on the Unit.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes to the RPS testing surveillance procedure do not constitute operating bypasses thus the probability is not increased. This activity only affects the conduct of a test performed on an inoperable portion of the system which is being administratively controlled. The RPS is still capable of performing the mitigating function during this activity, therefore the consequences has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no power supplies or other energy sources in the test box to create an over current or over voltage condition in the RPS trip logic therefore, the possibility of an accident different than previously evaluated is not created. The possibility of a malfunction is not created because there will be no change in the Allowed Out of Service Time (AOT), the probability of an equipment malfunction remains unchanged from that already analyzed. All other inputs to the RPS subchannel trip logic remain available during the time the test box is installed.

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

Safety Evaluation Number: SS-H-98-0128

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QIS 0003-S01 Rev. 16; SE 98-087

Title: APRM Rod Block and Scram Calibration and Functional Test Data Sheet

DESCRIPTION:

Include the use of the RPS test box during the functional portion of the procedure. This will allow the functional test to be performed without causing a ½ scram condition on the Unit.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes to the RPS testing surveillance procedure do not constitute operating bypasses. This activity only affects the conduct of a test performed on an inoperable portion of the system which is being administratively controlled. The RPS is still capable of performing the mitigating function during this activity, therefore the consequences have not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no power supplies or other energy sources in the test box to create an over current or over voltage condition in the RPS trip logic therefore, the possibility of a accident different than previously evaluated is not created. The possibility of a malfunction is not created because there will be no change in the Allowed Out of Service Time (AOT), the probability of an equipment malfunction remains unchanged from that already analyzed. All other inputs to the RPS subchannel trip logic remain available during the time the test box is installed.

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

Safety Evaluation Number: SS-H-98-0129

Type of Safety Evaluation: Validation; Modification

Evaluation Reference Number: DCPs 9800181; 9800182; SE-98-074

Title: Install Check Valve and Strainer in HPCI Room Cooler Service Water Line

DESCRIPTION:

Modification test for DCPs 9800181 and 9800182.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Service Water system is Non-Safety-Related, is not assumed to function in any accident, and does not mitigate the consequences of an accident. The piping and components are seismically supported. A failure of the piping system for these modifications would not significantly affect the Service Water system. Floor drains are also available in the event of piping failure. The new check valves provide double isolation between the Service Water system and the Diesel Generator Water Cooling system and are included in the IST program. This has been evaluated in Safety Evaluation SE-98-074.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new strainers provide better-quality water to the HPCI room coolers than the Diesel Generator Cooling Water system. These modifications do not prevent water from the Diesel Generator Cooling Water system from being supplied to the HPCI room coolers. The second check valve ensures adequate separation from the Safety-Related supply of water in the event that service water is lost. This has been evaluated in Safety Evaluation SE-98-074.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Diesel Generator Cooling Water, the Safety-Related supply of water to the HPCI room coolers, is still being supplied and is not affected. There is adequate separation, and these modifications have been analyzed seismically. This has been evaluated in Safety Evaluation SE-98-074.

Safety Evaluation Number: SS-H-98-0130

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCTS 820-01,02,03 &10 in addition to superseded copy of QTS 105-1,2,3 and QTS 170-4

Title: Deletion of Signature Sheet

DESCRIPTION:

The proposed activity is to delete the signature sheets for the procedures which were previously superseded by QCTS 0820. The previous activity deleted the implementing procedure. The Flood Protection Surveillance Activities are performed in accordance with the QCTS 820 series. The activity does not require mode restrictions, special conditions, system line ups or system interactions other than those stipulated in the QCTS procedure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because procedure number has been changed. The method of performing the test remains the same. There are no changes in frequency or class of accident. There are no new accident initiators or failure modes as a result of this change. The probability of occurrence of an accident remains the same. There is no increase in Radiological dose. The consequence of the accident has not been changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change in procedure number does not result in any change of frequency or type of accident. The test methods are unchanged from those previously described in the SAR. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the acceptance criteria for flood protection as described in NRC SER & Tech Spec and method of arriving at the acceptance criteria have not been changed.

Safety Evaluation Number: SE-H-98-0131

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCOS 5750-09 Rev. 12; SE-98-074

Title: ECCS Room and DGCWP Cubicle Cooler Monthly Surveillance

DESCRIPTION:

The procedure has been revised in accordance with DCPs 9800181 and 9800182. This modification installs a redundant safety-related check valve in series with the existing check valve to protect against single active component impacting the capability of the Safety Related Diesel Generator Cooling Water (DGCW) system from performing its safety function. A duplex strainer and associated instrumentation and valving is also being installed in a section of non-safety related Service Water system piping. The purpose of this strainer is to remove debris from the cooling water supply from the alternate Service Water supply to the HPCI Room Cooler. The strainer is isolated from the cooling water flow-path when the HPCI Room Cooler receives cooling water supply from the safety-related DGCW system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Service Water system is non-safety related, is not assumed to function in any accident, and is not credited for mitigating the consequences of an accident. The applicable piping and components installed by these modifications are seismically supported. A failure of the piping system for these modifications would not significantly affect the Service Water system. Floor drains are also available in the event of piping failure. The new check valves provide double isolation between the Service Water system and the Diesel Generator Water Cooling system and are included in the IST Program.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new duplex strainers provide better filtration of the water being supplied by Service Water to the HPCI Room Cooler than is normally supplied by the safety-related DGCW system. These modifications do not impact the ability of the DGCW system to provide cooling water flow to the HPCI Room Cooler when necessary. The second check valve ensures adequate separation between the safety-related DGCW system and the non-safety related Service Water system in the event the Service Water system pressure is lost.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because DGCW supply to the HPCI Room cooler is still available when required and is not affected. Adequate separation exists between safety-related and non-safety related systems and applicable equipment installed via these modifications have been analyzed seismically.

Safety Evaluation Number: SS-H-98-0132

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QOP 5750-17, Rev. 8; SE-98-074

Title: ECCS Room Coolers

DESCRIPTION:

The procedure has been revised in accordance with DCPs 9800181 and 9800182. This modification installs a redundant safety-related check valve in series with the existing check valve to protect against single active component impacting the capability of the Safety Related Diesel Generator Cooling Water (DGCW) system from performing its safety function. A duplex strainer and associated instrumentation and valving is also being installed in a section of non-safety related Service Water system piping. The purpose of this strainer is to remove debris from the cooling water supply from the alternate Service Water supply to the HPCI Room Cooler. The strainer is isolated from the cooling water flow-path when the HPCI Room Cooler receives cooling water supply from the safety-related DGCW system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Service Water system is non-safety related, is not assumed to function in any accident, and is not credited for mitigating the consequences of an accident. The applicable piping and components installed by these modifications are seismically supported. A failure of the piping system for these modifications would not significantly affect the Service Water system. Floor drains are also available in the event of piping failure. The new check valves provide double isolation between the Service Water system and the Diesel Generator Water Cooling system and are included in the IST Program.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new duplex strainers provide better filtration of the water being supplied by Service Water to the HPCI Room Cooler than is normally supplied by the safety-related DGCW system. These modifications do not impact the ability of the DGCW system to provide cooling water flow to the HPCI Room Cooler when necessary. The second check valve ensures adequate separation between the safety-related DGCW system and the non-safety related Service Water system in the event Service Water system pressure is lost.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because DGCW supply to the HPCI Room cooler is still available when required and is not affected. Adequate separation exists between safety-related and non-safety related systems and applicable equipment installed via these modifications have been analyzed seismically.

Safety Evaluation Number: SS-H-98-0134

Type of Safety Evaluation: Validation; Design Change

Evaluation Reference Number: DCP 9700302; SE-97-104

Title: Abandon the Control Room Humidifier System

DESCRIPTION:

The Previous Safety Evaluation:

(1) Specified that the +1/8 in. w.g. over-pressure requirement for the Control Room applies only when Train A and Train B of the Control Room Emergency Ventilation System (CREVS) is operated in the Emergency mode. CREVS will still be required to provide a slight positive pressure in the Control Room when operated in the Normal mode.

(2) Specified that the humidification system for Train A of the CREVS is no longer used.

(3) Clarified the Radwaste ventilation supply and exhaust fan configuration.

(4) Editorial changes for clarification without changing intent.

This validation incorporated the abandoned humidifier equipment into plant drawings and design.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because changes associated with this activity are for systems that do not initiate accidents (Accidents with Radiological consequences). The activity does not result in a physical plant change, change in design requirements, or a change to any station operating procedure which would make any accident more likely to occur (Fire and Toxic Gas).

   The consequences of an accident are not increased because changes associated with this activity will not result in a change in the requirements for pressurizing the Control Room to its required level of over-pressure following a DBA (CREVS pressurization change) and are not associated with systems used to mitigate the off-site or Control Room dose (Radwaste and steam humidifier changes).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity will not result in a physical change to any SSC, deviation from any design requirement, or operating procedure (CREVS pressurization and Radwaste ventilation changes). Discontinued use of Train A CREVS steam humidifier will not impact any SSC,

since the use of a steam humidifier is not required to maintain relative humidity levels within design requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the only applicable Technical Specification Safety Margin associated with this activity is the +1/8 inch w.g. over-pressure criteria for the Control Room. This Safety limit only applies when the CREVS is operated in the Emergency mode. Since the activity only applies to CREVS operation in the Normal mode, there is no reduction in the Margin of Safety.

Safety Evaluation Number: SS-H-98-0141

Type of Safety Evaluation: Validation; Design Change

Evaluation Reference Number: SE-98-015

Title: Fabrication, construction and installation of raceway supports associated with the re-route of Unit 2 SBO 4KV feeder cables to switchgear 23-1

DESCRIPTION:

Supports will be installed on the east outer wall of the Unit 2 Reactor Building and the north outer wall of Unit Turbine Building. Work is to be performed "at risk" in accordance with QCAP 2200-02 and PIF # 98-03549 to expedite re-location of referenced cables now temporarily traversing the Unit 1 Turbine floor. This area must be cleared to support refuel outage Q1R15 work activities. Finalized design documentation will not be ready to support normally scheduled installation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the conduit installed by this portion of the design change (ECN 001644E) is passive in nature and does not interact with any other plant systems or components that could affect the probability of the malfunction of any equipment important to safety. The attachment of the conduit and associated supports for this design change will in no way affect the ability of the NCAD system, HRSS, Unit 1/2 EDG or Secondary Containment to perform their previously stated design functions. The size and depth of the holes being drilled to install the conduit supports will not penetrate the wall and will not damage any of the concrete reinforcing framework (re-bar).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new raceway will consist of rigid conduit, field routed along the outside east wall of the Turbine and Reactor building. It is not credible that this new raceway will cause the failure of any existing systems, specifically the Nitrogen Containment

Atmosphere Dilution System, HRSS, the Unit 1/2 EDG or Secondary Containment. This is because an evaluation has qualified the raceway for structural integrity (S&L Calculation 92C0-EO-S).

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification or SAR does not provide a margin of safety or acceptance limit for the applicable parameter or condition.

Safety Evaluation Number: SS-H-98-0143

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-067; NFS:BSS:98-058

Title: UFSAR Change for Velocity Limiter Specifications

DESCRIPTION:

Update the UFSAR Sections 4.6.2.2 and Table 1.2-3 to reflect the current control blade velocity limiter design specifications for rod free fall velocity of ABB and GE control rods from 5 ft/s to 3.11 ft/s and to be consistent with UFSAR section 15.4.10.2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change does not physically affect any plant system, structure, or components, any plant processes or procedures. Updating the UFSAR so that section 4.6.2.2 and Table 1.2-3 agree with the analysis in section 15.4.10.2 with respect to the control rod drop speed will not increase the probability of an accident or the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because updating the UFSAR text to agree with current analytical methods and control rod design criterion will not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR. The plant components, systems and processes remain unchanged. The analysis itself did not change. The UFSAR is being corrected to be consistent with the current blade specifications from ABB and GE, the control rod drop accident methodology and section 15.4.10.2 of the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not physically affect any plant system, structure, or component, any plant processes or procedures. The proposed change

corrects the control rod drop speed in Table 1.2-3 and section 4.6.2.2 to be consistent with section 15.4.10.2 of the UFSAR. The control rods and control rod drive system will continue to function without change. None of the Tech Spec LCO surveillances are changed. The actions that must be taken if the control rods are outside of the Tech Spec LCO's also have not changed. Therefore, there is no change to a Tech Spec margin of safety.

Safety Evaluation Number: SS-H-98-0145

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-074; NFS:BSA:98-090

Title: Update the UFSAR to Reflect Accurate LOCA Analysis Inputs

DESCRIPTION:

Update the UFSAR to reflect the results of formal 10CFR50.46 peak cladding temperature (PCT) analysis which was performed in response to Quad Cities Station PIFs Q1998-00606, Q1998-00608, and Q1998-00695. These three PIFs identified discrepancies in the existing Loss Of Coolant Accident (LOCA) analysis. All Technical Specification requirements continue to be met.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because any cause of a rupture or split of the reactor recirculation suction pipe or other NSSS pipes coincident with Loss of Offsite Power is unaffected by the change to revised LPCI, CS and HPCI system performance description. The proposed UFSAR revision has no physical changes to the LPCI, CS and HPCI associated with it, nor does it affect the containment system, its related systems, or any function of any system or component other than the fuel.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because only the LPCI, CS and HPCI system performance descriptions are being changed. The impact of these changes was analyzed and was shown to not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the new LOCA analysis determines the ATRIUM-9B fuel and GE with its MAPLHGR limits will not violate the 2200 °F Peak Cladding Temperature limit, nor any of the other 10CFR50.46 acceptance criterion.

Safety Evaluation Number: SS-H-98-0150

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCAP 0210-04; SE-98-086

Title: Shift Change For Nuclear Station Operators

DESCRIPTION:

Add verification of valve line up for Unit 2 RCIC that is in an abnormal condition due to back leakage from either the feedwater or Reactor Water Clean up system past the RCIC injection check valve and RCIC system isolation valve. This resulted in increased temperatures in the pump discharge piping and pressurization of the pump suction piping. These valve lineups are part of the NSO's responsibility during a shift change.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change does not affect any of the initiators of the events RCIC is credited to mitigate or the initiators of any other event. The change revised the standby valve line up for the RCIC system that is used to mitigate these events and implements the appropriate operating procedures to support the new line up. The function of the RCIC system is as a mitigator to the subject events by providing cooling water to the reactor and to assist with pressure control during some of the events. Since this change will not affect the ability of RCIC to provide these functions within the design time requirements, this change does not increase the consequences of any event. The change to system operation will not increase the probability of a malfunction of equipment important to safety because:

    a. The pump discharge valve now being used for injection isolation has been evaluated to be able to perform the required functions.
    b. The affected piping and equipment is rated for the expected conditions and will be monitored so that it is maintained within those conditions.
    c. The RCIC discharge piping will be maintained full and verified full as required by the Technical Specifications.

    This change will not affect the consequences of a malfunction of equipment important to safety and does not make a release of radioactive material more likely or affect any of the equipment designed to prevent and mitigate the release of radioactive material.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change to the RCIC system standby line up has no impact on the system design

functions. The pump discharge valve has been verified to be an acceptable injection isolation valve of the system, and system integrity will be maintained. The pressure gauge installed under the interim procedures will be isolated if an RCIC initiation occurs during use. Therefore, this change does not create the possibility of a different type of accident or malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the intent of the Technical Specification surveillance requirement is to ensure that the RCIC discharge piping is full so that the system will inject water in the least amount of time and prevent water hammer events. This is normally accomplished by the routine surveillance through verification of water flow from the high point vent. The water flow verification will still be accomplished; however, due to the new standby system line up, an additional static pressure measurement will be performed to verify that the pump discharge piping remains full between surveillances.

Safety Evaluation Number: SS-H-98-0163

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCIS 1000-16, Revision 4; SE-98-049

Title: LPCI Pump Discharge Flow Loop Transmitter Calibration

DESCRIPTION:

Safety Evaluation SE-98-049 evaluated DCR's 980047 (Unit 1) and 980048 (Unit 2) to change the position of MO-1(2)-1001-18A/B minimum flow valves from "normally closed" to "normally open" to reduce the probability of damaging the residual heat removal (RHR) pumps due to failure of the minimum flow valves to open as a result of an Appendix R event. The current proposed activity merely revises procedure QCIS 1000-16 to account for the fact that the MO-1(2)-1001-18A/B valves may not be closed at the start of the transmitter calibration. The valve position checks were removed from the procedure because the initial position of the valves is not critical to the performance of the calibration. Information was added to the impact statement to ensure that Operations is informed that the calibration procedure will result in applying a "closed" signal to the valve. The NSO is notified upon completion of testing that RHR minimum flow valves(s) MO 1(2)-1001-18A/B may need to be repositioned in accordance with current plant conditions.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no increase in the probability of a

malfunction of equipment important to safety based on changing the normal valve position of MOV 1(2)-1001-18A/B. There is no physical change to the valves or their circuitry so the probability of the valves malfunctioning is not changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change to MOV 1(2)-1001-18A/B normal position has no adverse impact on the ECCS/RHR system, containment system or reactor systems to the extent of creating an accident or malfunction different from those evaluated in the SAR. There are no new interactions or functions created so there is no possibility of creating an accident or malfunction or a different type than already evaluated.

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

Safety Evaluation Number: SS-H-98-0164

Type of Safety Evaluation: Validation; Design Change

Evaluation Reference Number: DCP 9700394; SE-97-160

Title: Install Closing Device On Unit 1 MSIV Vent Door

DESCRIPTION:

Install a closing device on the Unit 1 MSIV Room Reactor Building Vent Door. The closing device will isolate the Reactor Building in the event that smoke is detected in the Reactor Building.

The purpose of this modifcation test was to ensure that operators could respond to an inadvertant closure of the door. The testing was performed within the bounds of the operating temperatures for the room.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because installation of MSIV vent door closing device will not cause MSIV line break. During break, primary/secondary containment is maintained by the main steam isolation valves. The blow out panels relieve pressure. Blow out panels will perform their function regardless of the MSIV vent door position.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because when the MSIV Room is part of the Turbine Building the door will function as previously designed. When MSIV is part of the Reactor Building the closing device provides additional protection for operators who must access the area. Therefore, spurious operation of MO1-2301-8 will not affect the ability of SSMP system to perform its design function during an Appendix R fire.

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

Safety Evaluation Number: SS-H-98-0165

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QCCP 0500-06 Rev. 02; SE-97-045

Title: Zinc Injection System Operation

DESCRIPTION:

This procedure change added the operation of the Unit One Zinc injection skid to the existing procedure for operation of the Unit Two skid.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the operation of the Unit One zinc skid is the same in all aspects as the operation of the Unit Two skid. The analysis completed in SE-97-045 is applicable in all cases to this procedure. The probability or consequences of an accident are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the operation of the Unit One zinc skid is the same in all aspects as the operation of the Unit Two skid. Any possible malfunction or accident is bounded by the accident analysis contained in SE-97-045 and the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the zinc injection system is a passive system that will have no adverse effects on either the feedwater piping, feedwater pumps, the primary coolant system, or the reactor vessel and internals. Injection of depleted zinc oxide into the reactor feedwater will result in a slight increase in reactor water

conductivity. The reactor water will still be a factor of 10 below the Technical Specification limit of 1 0 uS/cm.

Safety Evaluation Number: SS-H-98-0169

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-082; NFS:BSA:98-106

Title: Update the UFSAR to Reflect Accurate LOCA Analysis Inputs

DESCRIPTION:

Update the UFSAR to reflect the results of 10CFR50.46 peak cladding temperature (PCT) assessments which were performed in response to Quad Cities Station PIFs Q1998-00688. There were ten additional PIFs that require a UFSAR correction to resolve. All Technical Specification requirements continue to be met.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because any cause of a rupture or split of the reactor recirculation suction pipe or other NSSS pipes coincident with Loss of Offsite Power is unaffected by the change to revised LPCI, CS and HPCI system performance description. The proposed UFSAR revision has no physical changes to the LPCI, CS and HPCI associated with it, nor does it affect the containment system, its related systems, or any function of any system or component other than the fuel.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because only the revised LPCI, CS and HPCI system performance descriptions are being changed. The impact of these changes was analyzed and was shown to not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the new LOCA analysis determines the ATRIUM-9B fuel and GE with its MAPLHGR limits will not violate the 2200 °F Peak Cladding Temperature limit, nor any of the other 10CFR50.46 acceptance criterion.

Safety Evaluation Number: SS-H-98-0182

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-015; SE-97-108

Title: Revision to UFSAR Sections 6.2.5.2, and 6.5.3 Due to the Removal of the Atmospheric Containment Atmosphere Dilution (ACAD) Pressure Bleed Subsystem from Service

DESCRIPTION:

The ACAD pressure bleed subsystem was removed from service by locking closed manual isolation valves 1(2)-2599-10A/B and 1(2)-2599-32. UFSAR Sections 6.2.5.2 and 6.5.3 are being revised to reflect that ACAD pressure bleed was removed from service.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the ACAD valves have no effect on any equipment assumed to cause an accident. The valves are located outside of the drywell and their failure will not effect the plant's ability to remove combustible gases from the drywell. The Hardened Vent System performs this function. The ACAD pressure bleed subsystem no longer performs a function for the Station. So closing the valves to remove the ACAD pressure bleed subsystem from service will not increase the probability or consequences of an accident or malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the ACAD system is not required to function after an accident. Closing of the manual valves will not affect the mounting of the associated piping and primary containment boundary. Therefore, no new accidents or malfunction are created by this change. The system has been removed from plant operating procedures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety for this system is not defined in the basis for any Technical Specification. Therefore, the safety margin is not reduced. The margin of safety as defined in the basis for Technical Specifications for the containment are not reduced because the Hardened Vent system performs the function that ACAD pressure bleed subsystem did. This function is to vent the containment to prevent overpressurization of the containment during LOCA post accident conditions.

Safety Evaluation Number: SS-H-98-0183

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-015; DCP 9700177; DCP 9700179

Title: Revision to UFSAR Sections 7.3.2.2, 9.3-1, and Figure 7.3-1 Due to the Abandonment of the Quad Cities Unit 1 and Unit 2 Atmospheric Containment Atmosphere Dilution (ACAD) Air Dilution Subsystem

DESCRIPTION:

The ACAD air dilution subsystem was removed from service and abandoned in place to comply with NRC Generic Letter 84-09, "Recombiner Capability Requirements of 10CFR50.44 (c) (3) (ii)" and NRC SER "Safety Evaluation by the Office of Nuclear Reactor Regulation Regarding Post-Accident Combustible Gas Control System at Dresden Units 2 and 3, and Quad Cities Units 1 and 2 Commonwealth Edison Company, Docket Nos. 50-237, 50-249, 50-254, and 50-265" dated June 29, 1993. UFSAR Sections 7.3.2.2, 9.3-1 and Figure 7.3-1 are being revised to enhance and clarify statements regarding the abandonment of the ACAD air dilution subsystem.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the air dilution subsystem does not interface or interact with any primary pressure boundary piping that could lead to a small or large break LOCA inside or outside containment. The abandoned ACAD equipment cannot act as a missile that could cause a rupture in any of the piping that could lead to a LOCA. The containment penetration integrity for the ACAD air injection piping is assured by cutting and welding caps to the piping stubs at the containment penetrations in accordance with piping design code requirements.

   The ACAD system was removed from service because the system injected air into the primary containment to dilute the hydrogen concentration that could result from a LOCA. The injection of air also injected oxygen inside containment which, when mixed with the hydrogen, could cause an explosive gas mixture that could compromise the integrity of the primary containment. The result of removing the ACAD air injection subsystem from service is to reduce the probability of a malfunction of the primary containment and therefore reduce the consequences of the off site dose. In addition, the ACAD system does not interact with secondary containment such that the consequences or malfunction of the secondary containment could be increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the removal of the ACAD air dilution subsystem will not create any new failure modes. All equipment and piping that is abandoned in place will meet seismic II over I criteria

to ensure that the abandoned piping and equipment will not become missiles during a seismic event. This will prevent the possibility of damage to other SSCs that could lead to a malfunction of a different type than that evaluated in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specifications that apply to the ACAD air dilution subsystem. The ACAD air injection subsystem which could cause an explosive gas mixture was replaced with the Nitrogen Containment Atmospheric Dilution (NCAD) system which cannot cause an explosive gas mixture when used to dilute the hydrogen concentration inside the primary containment during post LOCA conditions. The ACAD air injection piping was capped and sealed at the containment penetrations in accordance with applicable piping design codes to assure that the margin of safety as defined in the basis for the Technical Specification 3 /4.7A, Primary Containment Integrity, is maintained.

Safety Evaluation Number: SS-H-98-0186

Type of Safety Evaluation: Validation; UFSAR Change

Evaluation Reference Number: UFSAR-97-R5-095; NFS:BSS:96-159

Title: Revision to UFSAR Section 9.1.2.1, Design Bases, Paragraph G

DESCRIPTION:

A revision to UFSAR Section 9.1.2.1, Design Bases is being made to remove the statement that the GE BWR 8 x 8 fuel rod array is the design basis fuel assembly and replace the statement with the sentence, "Design basis fuel assembly parameters are stated in UFSAR Section 9.1.2..3." This change is being made because Siemen's ATRIUM-9B 9x 9 fuel rod array is also a design basis fuel assembly. Section 9.1.2.3 states the design basis input parameters used for both types of fuel assemblies in the safety evaluation to justify storage of both types of fuel assemblies in the spent fuel pool.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the following Transients and Accidents are applicable to the spent fuel pool:

   Inadvertent criticality in the spent fuel pool, Bundle drop onto spent fuel racks, and Safe shutdown and operating basis earthquake. The probability of occurrence or consequences of an accident or malfunction of equipment important to safety for these accidents is not increased. The proposed limitations on the design characteristics of the ATRIUM-9B fuel have been established to meet the UFSAR criteria of K_{eff} less than 0.95 in the spent fuel pool storage racks. The ATRIUM-9B fuel is compatible with existing fuel handling equipment. The fuel bundle meets the same handling specification as the GE fuel and the fuel bundle is approximately the same weight and dimensions as the GE fuel. Also a change in fuel type will not increase the probability of an earthquake. In addition, the change in fuel design will not affect pool water level, shielding, secondary containment so that the consequences of the accidents are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change allows the storage and handling of Siemens Power Corporation (SPC) ATRIUM-9B fuel at Quad Cities Station. The only change is in the manufacturer of the fuel and the allowed reactivity of the fuel assembly. The proposed limitations on the design characteristics of the ATRIUM-9B fuel have been established to meet the UFSAR spent fuel pool k_{eff} criteria. The ATRUIM-9B fuel is compatible with existing fuel handling equipment, meets the same handling specification as the GE fuel, and is approximately the same weight and dimensions as the GE fuel. The ATRUIM-9B bail handle utilized in the lifting the bundle is of the same functional design as the GE bail handle and will interface with existing fuel handling equipment in the same manner as the GE fuel. This is supported by the fact that SPC fuel (with the same bail design) has been utilized at Dresden Station successfully for many years with fuel handling equipment equivalent to Quad Cities' equipment. No physical changes are being made to the plant. The plant equipment and procedures related to the storage and handling of the fuel are not affected by this change as the fuel will be handled and stored in the same way as the GE fuel. No new activities are involved with the handling and storing of the SPC fuel.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the proposed limitations on the design characteristics of the ATRIUM-9Bfuel assembly has been established to meet the UFSAR and Technical Specification criteria of K_{eff} less than 0.95 for the Quad Cities Unit 1 and 2 spent pools. The UFSAR is being updated to include reference to the analysis that will be performed to ensure the spent fuel rack criteria is met for both the GE Fuel and SPC ATRIUM-9B fuel bundles.

Safety Evaluation Number: SS-H-98-0193

Type of Safety Evaluation: Validation; Procedure Change

Evaluation Reference Number: QOM 1-2700-01, Rev. 6; SE-97-098

Title: U1 H2 Water Chemistry Valve Checklist

DESCRIPTION:

The procedure is being revised in accordance with DCPs 9300138 and 9300292 to install a Permanent Feedwater Oxygen Injection System on Unit 1 and 2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Feedwater oxygen injection system is designed to maintain the Feedwater oxygen rates within the EPRI and Technical Specification guidelines to minimize the corrosion rates of the Feedwater piping. There are no negative system interactions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Feedwater oxygen injection system is designed to maintain the Feedwater oxygen rates within the EPRI and Technical Specification guidelines to minimize the corrosion rates of the Feedwater piping. There are no negative system interactions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification 3/4.6 was reviewed and is not affected by this change.

Safety Evaluation Number: SS-H-98-0198

Type of Safety Evaluation: Validation: Procedure Change

Evalration Reference Number: QCOS 1400-01, Rev. 9; SE-98-030

Title: Quarterly Core Spray Pump Flow Rate Test

DESCRIPTION:

The procedure is being revised in accordance with DCP 9800063 to incorporate steps to perform a functional test of the new LOCAL/REMOTE control switch that was installed at MCC 29-1. This will allow control of the Air Handling Unit (AHU) located in the RCIC/ Core Spray Room and ensure that the AHU remains reliable for a postulated Appendix R fire in the Turbine Building.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report

is not increased. The new LOCAL/REMOTE control switch interfaces only with the control circuit for the AHU and does not adversely impact any SSC in such a way as to increase the probability of creating an accident. When the switch is in REMOTE, operation of the AHU is the same as before installation of the new switch. When the switch is placed in LOCAL, that portion of the control circuit which is vulnerable to an Appendix R fire in the Turbine Bldg. is isolated and the AHU is provided with a continuous run signal. Therefore, this change does not create any new modes of operation which could increase the probability of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created. The LOCAL/REMOTE control switch for the AHU allows portions of the AHU control circuit which could be damaged by an Appendix R fire in the Turbine Bldg. to be isolated and allows the AHU to run continuously. This ensures the AHU remains reliable and that adequate cooling is provided to the RCIC/Core Spray Room. The switch and associated terminal block were purchased and installed as safety-related to ensure they remain functional during and after a postulated design event. Therefore, the reliability of the AHU control scheme and MCC 29-1 is not reduced in any mode of operation. The AHU logic does not interface or impact any SSC in a manner which could create the possibility of an accident or malfunction of a different type from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced. Technical Specification 3/4.5.D was reviewed and this change does not affect any parameters upon which Tech Specs are based, therefore, there is no reduction in the margin of safety.

Safety Evaluation Number: SS-H-98-0232

Type of Safety Evaluation: Validation; Temporary Alteration

Evaluation Reference Number: DCP 9800260

Title: Installation Of A Temporary Alteration On The A Fire Pump

DESCRIPTION:

The Temporary Alteration removed the low oil pressure and high temperature alarm feature.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis

Report is not increased because the fire diesels supply water to suppression systems are used to mitigate fires and reduce the probability of a design basis fire. The alarms are installed to provide notification and to allow corrective action during non-emergency operation. There is no change to the fire diesels' function during a design basis fire.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the fire diesels supply water to suppression systems are used to mitigate fires and reduce the probability of a design basis fire. The alarms are installed to provide notification and to allow corrective action during non-emergency operation. There is no change to the fire diesels' function during a design basis fire. Additionally, the operators monitor the diesels during non-emergency runs; therefore, the probability of a failure or malfunction is not increased.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the functions bypassed by the temp mod are not relied upon during a design basis fire; therefore, the elimination of these alarms does not reduce the margin of safety.
