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

Subject: 10 CFR 50.59 Summary Report

In accordance with 10 CFR 50.59, "Changes, tests, and experiments," subpart (d)(2), we are forwarding a summary report of changes, tests, and experiments completed. This submittal contains a summary of 10 CFR 50.59 evaluations completed between March 13, 2001 and December 31, 2002.

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

Respectfully,

Wally John Beck for.

Timothy J. Tulon
Site Vice President
Quad Cities Nuclear Power Station

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

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

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ATTACHMENT

SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

MARCH 13, 2001 TO DECEMBER 31, 2002

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SUMMARY REPORT OF CHANGES, TESTS, AND
EXPERIMENTS COMPLETED
MARCH 13, 2001 TO DECEMBER 31, 2002

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10 CFR 50.59 Evaluation Summaries

[1] **Tracking Number:** SE-99-105 | **Change Document(s)**
Unit: Unit 2 | DCP 9900184; DCN 0018941

Activity Description

The activity involves a modification of the Unit 2 Reactor Protection System (RPS) logic, Primary Containment Isolation System (PCIS) logic, and Off-Gas System logic. This change will remove and reconfigure the functions performed by the Main Steam Line Radiation Monitor (MSLRMs). The MSLRMs will no longer provide scram, MSIV closure, or Offgas related System Trip Functions.

Impact of Activity

The effect of this activity is to reduce forced shutdowns due to inadvertent scrams and provide an economic benefit by increasing system availability without compromising a significant increase in radiological consequences. Additionally, removal of these scram and trip functions will reduce the possibility of inadvertent reactor trips.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59. The MSLRMs are not an initiator of any accident or event. The change has been determined not to increase the consequences of an accident or a malfunction of equipment important to safety. No physical changes are involved that would increases the possibility for an accident or malfunction of a different type. The margin of safety was also found acceptable. Note that this change has been generically approved by the NRC in Topical Report NEDO-31400A. A Technical Specification change request has been submitted to support this change.

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[2] **Tracking Number:** SE-00-044 | **Change Document(s)**
Unit: Common | DCP 9900344; DCP 9900345; UFSAR-99-R6-118

Activity Description

The activity involves a modification that removes the refueling interlocks associated with the Unit 1 and Unit 2 Service Platforms. These interlocks are for personnel protection and provide a rod block whenever the service platform is in use during refueling activities. The Service Platforms are no longer used and have been removed. They were originally used for installing, removing and shuffling reactor vessel components.

Impact of Activity

This activity will have no effect on the remaining refueling interlocks. The modified rod block logic will continue to function in the same manner as the existing circuit.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The removal of the Service Platform interlocks will not adversely affect the rod block logic. The rod block logic circuits will be rewired such that they will provide the proper rod block function when required. As such, the probability of occurrence or the consequences of an accident or transient is not increased. In addition, these changes do not create the possibility of an accident of a different type nor reduce the margin of safety.

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3 **Tracking Number:** SE-00-050 **Change Document(s)**

Unit: Unit 2 DCP 9700262; DCN 0017201; DCN 0019801

Activity Description

The activity involves the installation of the Oscillation Power Range Monitor (OPRM), Phase II. When completed, the OPRM modification will initiate a reactor scram via the existing Reactor Protection System (RPS) trip logic when core power oscillations (from potential thermal hydraulic instabilities) are detected. The OPRM system, when fully operational, will be automatically enabled at high power and low recirculation flow conditions. Phase 1 provided alarm and display functions only. Phase II adds the necessary hardware to the allow the OPRM system trip functions to be made fully operational. Note that the OPRM modification will not be made fully operational until resolution of the related 10 CFR Part 21 issue (reference 10 CFR Part 21 2001-23-5, "Stability Reload Licensing Calculations Using Generic DIVOM Curve," issued July 2001).

Impact of Activity

The effect of this activity is to allow the OPRM system to be made fully operational. Quad Cities station committed to install a "detect and suppression" system in response to NRC Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors."

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59. This modification enhances the ability to detect and suppress potential core instabilities. The modification does not adversely affect the existing APRM and RPS design or trip philosophy. Furthermore, the OPRM systems have been designed consistent with NRC approved guidance (e.g., GE Topical Report NEDO-32465-A).

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4 **Tracking Number:** SE-00-103 **Change Document(s)**

Unit: Common DCP 9900404; DCP 9900405

Activity Description

The activity involves a plant modification to support the Extended Power Uprate (EPU) project. As a result of the EPU project, the main steam line high flow differential pressure switches require new setpoints. To accommodate the new EPU setpoint, the existing switches are being replaced with Barton model 288A differential pressure switches. These new switches have a range of 0-400 psig, which can accommodate the calculated setpoints required for EPU.

Impact of Activity

The effect of this activity is to provided new main steam line high flow differential pressure switches in support of the EPU project. The new switches are procured as safety related. The Primary Containment Isolation System (PCIS) interface, function, and method of operation is not effected by this modification.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The replacement main steam line high flow differential pressure switches will continue to operate as required to provide input into the PCIS system. The new differential pressure switches are safety related and will not create the possibility of a new or different type of malfunction. In addition, the setpoint methodology used to generate instrument setpoints will ensure an adequate margin of safety.

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SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

5 **Tracking Number:** **SE-01-003** **Change Document(s)**

Unit: **Unit 2** **EC 24142**

Activity Description

The activity involves a modification to upgrade the existing reactor vessel water level switches manufactured by Yarway. The initiation trip functions/trip functions of the Yarway level switches will be replaced by existing reactor vessel water level instrumentation signals.

Impact of Activity

The functions now provided by the Yarway reactor vessel water level switches will be provided by more reliable instrumentation channels. There will be no degradation of system function during normal operation or in response to off-normal accident events.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. A review of existing accident scenarios has shown that the consequences of any accident is not increased. The reactor water level instrumentation loops are not an initiator of any accident or transient. Design measures and considerations will ensure that the modification will not adversely affect equipment failures or malfunctions. Replacement of the obsolete Yarway instrumentation will improve overall reliability. In addition, the setpoint methodology used to generate instrument setpoints will ensure an adequate margin of safety.

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6 **Tracking Number:** **SE-01-008** **Change Document(s)**

Unit: **Common** **DCP 9900667; DCP 9900668**

Activity Description

The activity involves a plant modification to support the Extended Power Uprate (EPU) project. The modification will upgrade certain main steam piping supports and supporting structures to accommodate the increased loads caused by transients initiated from EPU conditions.

Impact of Activity

The effect of this activity is to modify certain main steam piping supports, drywell steel as well as the addition of new supports to accommodate the increased loads at EPU conditions. The new or modified supports/supporting structures for the main steam piping will be designed, fabricated and installed to acceptable codes, standards and acceptance criteria and will ensure the main steam system is capable of withstanding design basis loads under EPU conditions.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The modification is designed to ensure an adequate margin of safety such that the reactor coolant pressure boundary design limitations are not exceeded during normal and anticipated operational occurrences. The proposed modifications increase the load capacity such that the probability of a main steam line break is not increased. The failure modes of the main steam piping are not adversely effected by this change because the proposed modification will be installed using proper codes and standards.

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7 **Tracking Number:** QC-E-2001-001 **Change Document(s)**

Unit: Common TS Bases B3.3.6.2 and B3.3.7.1

Activity Description

The activity involves a change to the Technical Specifications Bases sections B3.3.6.2 and B3.3.7.1. These Bases sections provide combined requirements for the Reactor Building Exhaust Radiation - High and Refueling Floor Radiation - High isolation functions. The requirements for the Reactor Building Exhaust Radiation - High function are being modified to state that each trip system is required to have one channel inoperable or in trip unless for one unit only the associated reactor building ventilation system is isolated.

Impact of Activity

The effect of this activity is to allow plant operation for 24 hours with the reactor building ventilation isolated on one unit to facilitate maintenance activities, without being required to challenge plant equipment by having all reactor building ventilation systems isolated, control room ventilation isolated, and the standby gas treatment system running.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. This change is consistent with UFSAR accident safety analysis. Because the reactor building and control room ventilation isolation functions are preserved, the safety analysis of the plant is not adversely affected by this change.

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8 **Tracking Number:** QC-E-2001-008 **Change Document(s)**

Unit: Common DCP 9900400; DCP 9900401

Activity Description

The activity involves a modification to support the Extended Power Uprate (EPU) project. The modification will add a logic circuit to automatically trip the "D" condensate/booster pump in the event of a Loss of Coolant Accident (LOCA) while all four condensate/booster pumps are running. As part of EPU, all three feedwater pumps and all four condensate/booster pumps are required at full power. In the event of a LOCA (without a concurrent Loss of Offsite Power), all loads will be transferred to the Reserve Auxiliary Transformer (RAT), including the ECCS loads. By tripping the "D" condensate/booster pump, the RAT loads will remain within currently analyzed conditions.

Impact of Activity

The effect of this activity is to provide trip logic to the "D" condensate/booster pump upon detection of a LOCA with all four condensate/booster pumps operating. By tripping the "D" condensate/booster pump, the RAT loads will remain within currently analyzed conditions.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The new trip logic is similar in manufacture to the existing relays and performs to the same standards and is no more likely to fail than existing components. The components are not initiators of any accident condition. The modification has no adverse impact on existing LOCA detection/actuation logic systems.

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9	**Tracking Number:**	QC-E-2001-010	**Change Document(s)**
	Unit: Common		FPR 01-01

Activity Description

The activity involves a change to the Quad Cities Fire Hazards Analysis (FHA). The change removes certain commitments made prior to the implementation of 10CFR50, Appendix R and are identified in the July 1979 NRC Fire Protection Safety Evaluation Report (SER). Specifically, the commitment to: (a) coat with fire retardant material cables from different divisions that are not separated by greater than three feet horizontal or five feet vertical, (b) seal certain electrical panels and MCC's, and (c) replace the non-rated fire door between the turbine building and service building (elevation 595').

Impact of Activity

The effect of this activity is to eliminate commitments that are no longer required to meet 10CFR50, Appendix R requirements. A different methodology for assuring safe shutdown has been developed and implemented as a result of the Appendix R regulation. The proposed change does not adversely impact post-fire safe shutdown capabilities.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The methodology used to ensure post-fire safe shutdown was changed following the implementation of the Appendix R Rule. These commitment changes do not adversely effect the current post-fire safe shutdown methodology; hence, do not adversely affect the ability of the plant to safely shutdown in the event of a fire.

10	**Tracking Number:**	QC-E-2001-011	**Change Document(s)**
	Unit: Common		UFSAR-99-R6-189

Activity Description

The activity involves a change to the UFSAR. The change modifies the discussion on the Reactor Protection System (RPS) shorting link requirements in Section 7.2.2.5.A. The change will allow control rod exercising and stroking prior to tensioning the reactor head without removing the RPS shorting links provided the core verification is complete and the one-rod-out refueling interlocks have been demonstrated operable.

Impact of Activity

The effect of this activity is to reduce the time when the RPS shorting links are required to be removed. The change will allow for single control rod withdrawal when the vessel head is not tensioned. The requirement to remove the shorting links will remain for activities that require multiple control rod withdrawals with the head not tensioned.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. This change maintains the necessary controls to ensure that the reactor will not inadvertently go critical with the reactor head less than fully tensioned by relying on the analyzed shutdown margin, the core loading verification, and the one-rod-out refueling interlocks. The intermediate range monitors (IRMs) and average range power monitors (APRMs) continue to provide scram protection in the unlikely event a reactivity excursion occurs.

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11 **Tracking Number:** QC-E-2001-014 **Change Document(s)**

Unit: Unit 1 DCP 9900183

Activity Description

The activity involves a plant modification. The design change modifies the 1B service water pump control circuit by adding an auto start of the pump when service water pressure decreases to 75 psig. The service water system demand during cold weather periods (winter months) is normally less than a single pump; however, operating only a single service water pump increase the plant's vulnerability from a single service water pump trip. The proposed change adds an auto-start feature to the 1B service water pump control logic which will restore system pressure should an operating service water pump trip causing reduced system pressure.

Impact of Activity

The effect of this activity is to enhance the operation of the service water system such that the 1B service water pump will auto-start on low service water header pressure signifying a trip of a running service water pump.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. Service water is a non-safety related system. The addition of the auto-start feature improves system reliability by reducing reliance on manual operator actions following a service water pump trip. Providing an auto-start capability on the 1B service water pump has no affect on the performance of the remaining pumps and provides additional assurance system pressure will be restored should a pump trip occur during single pump operation.

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12 **Tracking Number:** QC-E-2001-018 **Change Document(s)**

Unit: Unit 1 DCP 9900376; DCP 9900652

Activity Description

The activity involves a modification to upgrade the existing reactor vessel water level switches manufactured by Yarway. The initiation trip functions/trip functions of the Yarway level switches will be replaced by existing reactor vessel water level instrumentation signals.

Impact of Activity

The effect of this activity is to upgrade the Yarway level instruments. The proposed activity will provided sensing devices and indications for reactor vessel water level that are more reliable with less drift and less frequent surveillance requirements. The transmitters supplying these signals will be updated to current models. There will be no degradation of system function during normal operation or in response to off-normal plant conditions.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. A review of existing accident scenarios has shown that the consequences of any accident is not increased. The reactor water level instrumentation loops are not an initiator of any accident or transient. Design measures and considerations will ensure that the modification will not adversely affect equipment failures or malfunctions. Replacement of the obsolete Yarway instrumentation will improve overall reliability. In addition, the setpoint methodology used to generate instrument setpoints will ensure an adequate margin of safety.

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13 **Tracking Number:** QC-E-2001-019 **Change Document(s)**

Unit: Common EC 24578; EC 332259; EC 332260

Activity Description

The activity involves a plant modification. The change removes a solenoid valve on the Emergency Diesel Generator (EDG) fuel oil system. Specifically, solenoid (SO) valves SO 0(1)(2)-5201 will be removed and, on Units 2 and 3, a normally open manual isolation valve will be installed in the fuel oil supply line to the EDG day tanks. The existing SO valves on Units 1 and 2 prevent flow to the EDG day tanks during transfer of fuel oil to the diesel driven fire pump day tanks (the Unit 1/2 EDG fuel oil transfer pump does not have a connection to the fire pump day tank - hence a new manual isolation valve is not required in place of SO 0-5201). The new manual isolation valves will be closed during a transfer of fuel oil to the diesel driven fire pump day tanks. There have been previous failures of these solenoid valves, which has decreased the reliability of the EDG fuel oil transfer system. The modification will remove this potential failure mode by removing the solenoid valves and installing normally open manual valves.

Impact of Activity

The new manual isolation valves will be normally open to support automatic fuel oil transfer to the EDG day tanks and will be closed when a diesel fuel oil transfer pump is used to fill the fire pump day tank. The current function to provide makeup fuel to the fire pump day tanks is currently a manual action required to support post-fire safe shutdown (within 8 hours of the fire). The additional action to reposition the new manual valves has been evaluated and determined to have an insignificant impact on the ability to complete this manual function.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. These changes are designed to improve the reliability of the EDGs and do not produce any adverse system effects. These changes do not result in more than a minimal increase in the frequency of an accident or the likelihood of a malfunction of equipment important to safety. Because there are no adverse system effects, there is no increase in the consequences from a postulated accident or equipment malfunction. No fission product barriers or methods of evaluation are impacted by this change.

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14 **Tracking Number:** QC-E-2001-020 **Change Document(s)**

Unit: Unit 2 DCP 9900696

Activity Description

The activity involves a plant modification. The change will provide a new and upgraded digital feedwater level control (FWLC) system. The new digital FWLC system is modeled based on the existing control system; however, the new system will include a "state-of-the-art" digital control system including an improved level control strategy. The reason for the change is that the existing FWCL system has become obsolete. The existing technology is not fault tolerant and has no redundancy and no self-check functions. Because the existing system has become obsolete, it presents a significant challenge in terms of reliability and maintainability.

Impact of Activity

The effect of the activity is to replace the exiting FWLC system with a modern, fault tolerant, state-of-the-art digital system. The new system will also include an improved control strategy. The new digital system and associated control algorithm will provide an improved level of control during normal and abnormal plant conditions.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The FWCL system has been thoroughly evaluated. The new digital FWLC system will continue to control reactor vessel level for all analyzed conditions and transients. The affect of upgrading the FWLC to a digital control system with new control features does not increase the frequency of occurrence of the consequences of any evaluated accidents and transients. The effects of upgrading the FWLC to a Digital Control System with new control features does not introduce the possibility of an accident of a different type or a malfunction with a different result. The failures previously postulated in the safety analysis bound any failure modes postulated with the new control system.

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15 **Tracking Number:** QC-E-2001-021 **Change Document(s)**

Unit: Common EC 331303; EC 331304; UFSAR-R7-01-18/19

Activity Description

The activity involves a plant design change. The modification enhances the existing reactor recirculation (RR) pump shaft seal system by installing a seal purge to the first stage seal. The seal purge will use Control Rod Drive (CRD) water. This existing seal system design does not minimize the potential for crud build-up that can reduce seal life. The proposed seal purge system will inject filtered water from the CRD system into the first seal cavity through existing seal cavity vent and pressure connections. The feed from the CRD system will be further filtered to ensure water quality. A relief valve, flow restricting orifice and flow regulators will be added to limit pressure and flow to the seals to minimize thermal effects on the pump shaft. Double check valves will also be installed in the CRD feed line to mitigate the consequences of a potential line break.

Impact of Activity

The effect of this activity will be to improve the RR seal purge system. Normally, the RR seals receive water form the recirculation pumps - cooled by a set of seal cooling heat exchangers in the seal assembly with a controlled bleed off of approximately 0.8 gpm. Experience has shown that crud suspended in the reactor water can be deposited on the seals, reducing seal life. The effect of installing the seal purge system will reduce crud buildup and extend seal life. The pre-modification system will continue to function in the event the CRD purge becomes inoperable for any reason.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The modifications will provide a source of clean water form the CRD system for recirculation pump seal purge. Similar modifications have been incorporated at other stations since 1976. The change does not alter or affect the design function of any structures, systems or components. The supplemental seal purge system will reduce crud buildup and increase seal life. There are no changes to any design basis limit for a fission product barrier.

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SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

16 **Tracking Number:** QC-E-2001-022 **Change Document(s)**

Unit: Common UFSAR-01-R7-042

Activity Description

The activity involves a change to the UFSAR. The change will remove selected lead seal and/or lockwire references in sections 7.2.2.8, 7.3.1.4.1.13, 7.3.1.4.1.14 of the UFSAR. Per the UFSAR, the facility design was required to meet the requirements set forth in IEEE-279, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems" [1968]. IEEE-279 requires the capability to administratively control selected instrument isolation valves to prevent inadvertent bypassing of safety functions. IEEE-279 paragraph 4.14 titled "Access to Means for Bypassing" states "The design shall permit the administrative control of the means for manually bypassing channels or protective functions." This UFSAR description revision does not alter any physical valve positions nor will the subject valves be operated in any manner differently from currently required configuration management control processes. Reliance on existing administrative controls (policies, procedures and procedure adherence, formal training and routinely reinforced management expectations) versus the use of selected physical barriers (e.g., lead seals and/or lockwire) meets the requirements of IEEE 279-1968. The physical locking devices are not credited in existing accident or transient analyses and the lack of such a physical barrier introduces no new equipment malfunction. Additionally, the removal of references to instrument isolation valve (lead) seals and/or lock wire for selected Reactor Protection and Engineered Safety Features Systems does not result in a change that would cause any system parameters to change.

Impact of Activity

The proposed change has no impact upon the operation of the facility as alternative means are in place to administratively manage the configuration control of the affected instrument isolation valves. This change is being made to allow existing Instrument Maintenance and Operating work procedures to be revised to expedite the completion of field work activities for ALARA and other efficiencies and to focus the workforce on the critical steps of the work vice the removal and subsequent reinstallation of (lead) seals and/or lockwire deemed unnecessary under the current station configuration control practices.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The physical locking devices are not credited in existing accident or transient analyses and the lack of such a physical barrier introduces no new equipment malfunctions. Additionally, the removal of references to instrument isolation valve (lead) seals and/or lock wire for selected Reactor Protection and Engineered Safety Features Systems does not result in a change that would cause any system parameter to change. Therefore, the removal of selected references to (lead) seals and/or lockwire does not result in a design basis limit for a fission product barrier (DBLPPB) as described in the UFSAR being exceeded or altered. Finally, the removal of references to instrument isolation valve (lead) seals and/or lock wire for selected RPS and ESF Systems does not involve any method of evaluation (e.g., data reduction methodology, data correlation, physical constants/coefficients, mathematical models, computer program limitations, etc.) in any way and therefore has no impact on existing methods of evaluation described in the UFSAR.

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17 **Tracking Number:** **QC-E-2001-023** **Change Document(s)**

Unit: **Common** **T&S DIT SPD-04-01-011**

Activity Description

The activity is a change to the facility. As part of the utility deregulation, state regulations require metering of incoming and outgoing power from generating stations. The metering is to be installed at the point of connection (typically the high side transformer disconnects). As such, revenue metering equipment is being installed in the 345KV switchyard for the Unit 1 and 2 Main Transformers (Tl and T2), the Unit 1 and Unit 2 Reserve Auxiliary Transformers (T12 and T22), and the 345KV Switchyard Transformers (T81 and T82). Testing of the units will be performed prior to and after transformer re-energization. These facility changes are all within the 345KV switchyard.

Impact of Activity

The installation of revenue metering equipment in the 345KV switchyard will connect current and potential (voltage) transformers (CT/PTs) in-line with the 345KV lines for the Main Transformers and Reserve Auxiliary Transformers and in-line with the 13.8KV cables of the Switchyard Transformers. Under normal operation of the metering equipment, there will be no effect on plant equipment or operation. However, because the CT/PT units are installed in-line with the 345KV or 13.8KV lines/cables, the failure of these units could disrupt power to or from the Station and thus have adverse effects on plant or station operation. However, the effects and consequences of these failures are no different than those accidents and transients bounded by existing UFSAR analyses. The CT/PT design and construction is in compliance with industry standards, and industry experience has demonstrated their reliability. Although the CT/PT units are more complex than a 345KV line or 13.8KV cable, it has been determined that there is not more than a minimal increase in the frequency of accidents evaluated in the UFSAR or the likelihood of malfunction of equipment important to safety evaluated in the UFSAR as a result of the installation of the CT/PT units in the 345KV Switchyard.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. Despite the new failure mechanism introduced (failure of in-line CT/PT units) by the installation of the revenue metering equipment, the failure modes remain the same (faulted or open circuited 345KV or 13.8KV electrical conductors). The effects and consequences of these failures are bounded by existing UFSAR analyses (in particular, UFSAR 15.2.6 Loss of Offsite AC Power; UFSAR 15.2.2 Load Rejection; and UFSAR 15.2.3 Turbine Trip). In the case of loss of offsite power, the Emergency Diesel Generators are capable of powering the 4.16KV ESF buses for safe shutdown and post accident recovery. It has been further determined that there is not more than a minimal increase in the frequency of accidents evaluated in the UFSAR or the frequency of malfunction of equipment important to safety evaluated in the UFSAR as a result of the installation of the CT/PT units. The proposed change does not result in more than a minimal increase in the frequency of occurrence of an accident or likelihood of a malfunction of an SSC important to safety previously evaluated in the UFSAR. It does not result in more than a minimal increase in the consequences of an accident or malfunction of an SSC important to safety previously evaluated in the UFSAR. It does not create the possibility for an accident of a different type or malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. It does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. Nor does it result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. Therefore prior NRC approval of the change is not required.

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18 **Tracking Number:** QC-E-2001-024 **Change Document(s)**

Unit: Common UFSAR-01-R7-014

Activity Description

The activity involves a change to the UFSAR. The Small Instrument Line Break Analysis (described in Sections 6.2.3.3 and 15.6.2) is being revised to reflect the analysis from the Quad Cities Station original Final Safety Analysis Report Section 5, Amendment 13, Page 5.9-3. Additionally, the current analysis for a Small Instrument Line Break is being identified as historical in nature and not relied upon for plant activities regarding dose consequences. The secondary containment pressure response analysis remains valid as a bounding analysis. The current analysis for a Small Instrument Line Break is based upon the original proposed Technical Specification Reactor Coolant System (RCS) activity limit of 20 micro-Curies/cc of total Iodine. This coolant activity level results in high Reactor Building (RB) Ventilation (Vents) duct radiation and a corresponding trip signal which isolates RB Vents and initiates an auto-start of the Standby Gas Treatment System (SBGTS), assumed to occur coincident with the accident. The current Technical Specification action limit for required sampling for reactor coolant activity is 0.2 micro-Curies per gram I-131 dose equivalent. Actual reactor coolant activity under normal plant operating conditions is < 0.002 micro-Curies/cc I-131 dose equivalent, or less than a factor of 104 smaller than the 20 micro-Curies/cc limit upon which the current Small Instrument Line Break analysis is based. Consequently, radiation activity levels high enough to exceed the RB Vent Radiation Monitor setpoint (and corresponding isolation of the RB Vents & initiation of the SBGTS) would not be reached. Finally, the UFSAR Section 6.2.3.3 states the blow off panel breakaway pressure is 7 in. H2O gauge. This information is being clarified by this UFSAR change to state that the minimum design pressure of the Reactor Building (RB) is 7 in. H20 gauge, and that the blow off panel breakaway loading is 70 lb/ft2.

Impact of Activity

The safety objective of the secondary containment system, in conjunction with other engineered safeguards and nuclear safety systems, is to limit the release of radioactive materials so that offsite doses resulting from a postulated design basis accident (DBA) will remain below 10CFR100 guideline values. The reactor building is designed to contain a positive internal pressure of at least 7 in. H2O gauge without structural failure and without pressure relief. The secondary containment isolation system and its associated controls are designed to isolate the reactor building in the time required to prevent significant release of fission products through the normal discharge path. The Small Instrument Line Break analysis is being revised to reflect the actual accident scenario that would occur, in which all the steam not condensed in the reactor building is transported out the RB Vent Stack. Dose consequences are related linearly to coolant activity. The Original FSAR dose analysis used a value for coolant activity that is smaller than the current Technical Specification shutdown limit (4.0 micro-Curies/gram) by less than a factor of forty. By extrapolating linearly for coolant activity of 4.0 micro-Curies/cc and conservatively using a factor of 40, the calculated lifetime dose for the duration of the accident release is 12.0 Rem thyroid dose and compliance to 10CFR100 is therefore demonstrated. The pressure response for Secondary Containment was also found acceptable based on existing UFSAR analysis. The Small Instrument Line Break analysis in QC Station Original FSAR Section 5, Amendment 13, Page 5.9-3 used iodine activity with the released liquid of 0.04 micro-Curies/cc of I-131 and 0.3 micro-Curies/cc Iodine-133, which equates to 0.121 I-131 dose equivalent. The QC Station original FSAR analysis represents the Station's licensing basis. RCS activity of 0.121 I-131 dose equivalent is conservative to normal RCS activity by greater than a factor of 50, and is similar in magnitude (within a factor of 2) to the current Technical Specification action limit for required sampling. This conservative analysis demonstrates that the lifetime dose for the duration of the release will be well below 10CFR100 limits.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. There are no physical changes being made to plant equipment, and no potential impact on the reactor coolant pressure boundary. The only accident analysis impacted by this change is the Small Instrument Line Break accident analysis. The original FSAR Section 5 Amendment 13 analysis demonstrates that the dose consequences are well within 10CFR100 limits. To be conservative, the resulting dose is linearly extrapolated to a value representative of the current Technical Specification limit. The calculated dose is demonstrated to meet both the "minimal" increase requirement (<10% of the difference between the analyzed dose and the 10CFR100 limit) and the Standard Review Plan (SRP) requirements (<10% of the 10CFR100 limits for whole body and thyroid doses).

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

19 **Tracking Number:** QC-E-2001-025 **Change Document(s)**

Unit: Common UFSAR-01-R7-050

Activity Description

The activity involves a change to the UFSAR. The following changes are proposed:

1) Revise UFSAR Table 12.3-4, "Quad Cities Unit 2 Area Radiation Monitoring System Sensor Location and Range," to reflect that Area Radiation Monitor (ARM) Station 23, "Standby Gas Treatment - Unit-2", does not provide a local alarm horn. Engineering Request (ER) 9903952 has been processed for an abandoned equipment modification to update plant design documents to reflect that the alarm horn is abandoned in place. This UFSAR revision supports the abandoned equipment modification process.

2) Revise UFSAR Table 12.3-5, "Quad Cities Unit 1/2 Area Radiation Monitoring System Sensor Location and Range," to remove ARM Station 4, "Radwaste Bldg. Barreling Line - North Wall," from the Unit-0 ARM Table. Revise the number of ARM Stations in Section 12.3.4 from 71 to 70 stations, correlating to the removal of Unit-0 ARM #4. ER 9910859 has been processed for an abandoned equipment modification to remove the Indicator/Trip-Unit and Detector, and abandon the cables in place.

3) Provide clarification to Section 12.3.4 that Control Room alarms are received for Unit-1 and Unit-2 ARM's, and the Unit 1/2 ARM's provide local indication and alarm conditions only.

4) Add expected background radiation value for Stations 35 & 36 in Table 12.3-3 and all Stations in Table 12.3-5.

These changes are required for the following reasons:

1) The alarm horn for Unit-2 ARM #23 is not functional due to a break in the cable conductor in a penetration between the Unit-2 Cable Tunnel and the 'D' Feedwater (FW) Heater Bay. This ARM is in a low traffic area that is not a personnel thoroughfare. Additionally, an alarming electronic dosimeter to warn personnel of abnormal dose rates is worn by each plant worker. Based upon the cost required and minimal added benefit of the alarm horn, the plant modification is not cost-justifiable.

2) The RW Barreling Line has not been used in at least 10-years, and any future use of the area for storage would be monitored by a Radiation Protection (RP) Technician. This UFSAR revision supports the abandoned equipment modification process, such that the dose received during maintenance and the cost for calibration and repair of the ARM will be eliminated. Unit-0 ARM #4 provides local indication and alarm functions only. It is therefore not cost-effective and does not support station As Low As Reasonably Achievable (ALARA) Program goals to keep this ARM in service, since the RW Barreling Line is no longer used.

3) The Unit 1/2 ARM's do not report alarm conditions to the Control Room. This change provides clarification regarding the as-built equipment function for Unit 1/2 ARM Stations.

4) UFSAR Section 12.3.4 states that areas monitored by the ARM System are listed in Tables 12.3-3, 12.3-4, and 12.3-5, which also identify expected background radiation. The expected background radiation information was not previously provided but is being provided for Stations 35 and 36 in Table 12.3-3 and all Stations in Table 12.3-5 by this change.

Impact of Activity

The ARM system continuously monitors and records the radiation levels in accessible work areas of the plant. In areas where high radiation is most likely to occur, a local alarm horn is provided. As stated in Section 15.6.2, a general increase in ARM readings throughout the Reactor Building is one of several methods used by Operators to detect a small instrument line break. The local alarm horn function for Unit-2 ARM#23 is the only impacted design function. There is no effect on Technical Specifications as a result of this change. The function of Unit-2 ARM#23 is to provide local and Control Room indication and alarm functions for its area, the 'A' Standby Gas Treatment floor on Unit-2. The expected background radiation at the radiation detector is 1 mR/hr under normal plant conditions and 6 mR/hr when Standby Gas Treatment System (SBGTS) is operating. The Control Room alarm function for the ARM, as well as both local and Control Room indication functions, remain intact. The SBGTS floor is a low personnel traffic area. Additionally, an alarming electronic dosimeter to warn personnel of abnormal dose rates is worn by each plant worker. In a General Station Emergency Plan (GSEP) event, where dose levels may be higher at the detector resulting from accident conditions, non-essential personnel would be evacuated from the plant. Therefore, the potential impact on radiation workers as a result of the local alarm horn not functioning is minimal. The design function of Unit-0 ARM#4, which is to provide local indication and alarm functions the RW Barreling Line area in RW building, is being abandoned by this change. The RW Barreling Line is no longer used. The RP department would monitor any future use of the area. The RW Barreling Line is a very low personnel traffic area. Additionally, an alarming electronic dosimeter to warn personnel of abnormal dose rates is worn by each plant worker. Therefore, the potential impact on radiation workers as a result of the ARM being abandoned is minimal. In the case of the 1/2 ARMs, the affected ARM Stations serve local functions associated with operation of the Reactor Building (RB) Crane and monitoring of activities in the Radwaste (RW) building. Since potentially impacted personnel are involved in the specific local activities that these ARM'S monitor, there is no impact that these ARM's do not provide a Control Room alarm. Finally, expected background radiation information is being provided for ARM Stations where the information had previously been omitted, which makes Tables 12.3-3 & 12.3-5 consistent with Section 12.3.4.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. There are no Technical Specification requirements, nor Technical Requirements Manual (TRM) requirements for the affected ARM's. The affected ARM's do not impact any UFSAR accident analysis. ARM's provide monitoring and alarm function only. Unit-0 ARM #4 is not used in Emergency Operating Procedures (EOPs). Although Unit-2 ARM#23 is used in EOPs, the only function impacted is the local alarm horn which is not required by the EOPs. ARM's have no impact on the reactor coolant pressure boundary nor any fission product barrier, and the ARM functions impacted by this change have no impact on the consequences of an accident (radiation release). The local alarm horn function is provided by ARM's where high radiation is most likely to occur. Neither of the areas affected by this change are expected to have high radiation conditions. The function of the Unit 1/2 ARMs to provide personnel hazard protection is unaffected by the absence of Control Room indication/alarm functions. The ARM local indication/alarm functions provide immediate and adequate protection to warn personnel in those areas of high radiation conditions.

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20 **Tracking Number:** QC-E-2002-001 **Change Document(s)**

Unit: Unit 1 Core Operating Limits Report

Activity Description

The activity involves a change to the plant (core reload design). The proposed change is to discharge a failed (i.e., leaker) ATRIUM-9B fuel assembly (Q7D210) originally installed in Quad Cities Unit 1 during Reload 16 and replace it with a previously installed GE10 fuel assembly (YJ8262) and perform approximately 550 fuel assembly shuffles. For the purposes of this evaluation, the activities, including core shuffling, removal of the failed ATR1UM-9B assembly and the insertion of the GE10 bundle will be referred to as the Q1C17A core reload. The Q1C17A core design consists of a total of 724 fuel assemblies, including 289 previously loaded GE10 assemblies and 435 previously loaded SPC ATR1UM-9B assemblies. The failed ATR1UM-9B fuel assembly (Q7D210) is being removed from the core and placed in the spent fuel pool. The reinsert GE10 assembly has been visually inspected by camera and verified to be physically satisfactory for reinsertion into the Unit 1 core for Q1C17A operation. The GE10 fuel assembly will not exceed its end of life exposure prior to end-of-cycle.

Impact of Activity

The function of the reload core is to provide sufficient energy to support Cycle 17A operation while maintaining margin to the prescribed safety criteria. There will be no changed interactions with other plant systems. The discharge of a failed ATR1UM-9B fuel assembly, core shuffling, and insertion of a previously exposed GE10 fuel assembly are the only physical changes to the plant. No other safety related equipment is affected by the reload design change. The core design for Cycle 17A has been analyzed (from a transient and accident perspective) for operation at a rated power level of 2511 MWt. These analyses are used to determine cycle-specific thermal limits for operation at 2511 MWt. The thermal limits are provided in the Q1C17A Core Operating Limits Report (COLR). The analysis concludes that all Cycle 17 limits provided in both the Quad Cities Unit 1 Cycle 17 Reload Analysis and Quad Cities Unit 1 Cycle 17 Plant Transient Analysis continue to remain applicable for Cycle 17A up to a cycle exposure of 4140 MWD/MTU. To support Cycle 17A beyond an exposure of 4140 MWD/MTU (or approximately 200 effective full power days), a new analysis will be provided.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The activities associated with this package were identified as requiring a 50.59 Evaluation. The 50.59 evaluation concluded that the Q1C17A reload core design could be implemented without prior NRC approval. All analyses done to evaluate the Q1C17A reload core design utilized previously NRC reviewed and approved methodologies and were performed within the constraints of Framtome-ANP's (fuel supplier) quality assurance program as reviewed and approved by Exelon Corporation.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

21 **Tracking Number:** QC-E-2002-002 **Change Document(s)**

Unit: **Unit 2** **UFSAR-01-R7-098; Core Operating Limits Report**

Activity Description

The activity involves a change to the facility. Specifically, a core reload design is necessary to allow operation of Quad 2 Cycle 17 (Q2C17) at rated thermal power for two years of operation while maintaining the necessary margin to thermal limits. In order to meet this goal, GE14 fuel is being utilized. The GE14 design is a NRC-approved design. The Q2C17 reload core design contains the first reload of GE14C fuel at power uprate conditions (2957 MWt). The Q2C17 core design consists of a total of 724 fuel assemblies, including 456 previously loaded Framatome-ANP (formerly Siemens Power Corporation) ATRIUM-9B assemblies, and 268 fresh Global Nuclear Fuel (GNF) GE14C assemblies. The Q2C17 cycle-specific reload analyses were performed in accordance with USNRC approved methodologies. The Q2C17 cycle-specific reload analyses evaluated the Maximum Load Line Limit Analysis (MELLLA) and Increased Core Flow extended operating domains and utilized the APRM, RBM Technical Specifications (ARTS) power and flow dependent thermal limits strategy. The GE14C fuel design consists of a 10x10 fuel rod array, with 14 rods being part-length rods. Two large water rods in the center of the array replace eight fuel rods. The ATRIUM-9B assemblies that are being utilized in this reload have been evaluated for extended burnup limits. This exposure extension was accomplished by extending present exposure analysis limits using USNRC approved methodologies. The extended exposure limits are applicable to all FANP ATRIUM-9B fuel at Quad Cities. In addition, GE Marathon control blades will be used in Quad Cities for the first time. The Marathon blades were reviewed and approved by the USNRC (NEDE-31578P-A, October 1991). The Marathon Control Blade design is fully interchangeable with the OEM design and within vendor defined tolerances for weight and dimensions. Therefore, scram and rod drop speeds are not impacted. The Q2C17 COLR has been generated in accordance with the requirements of Generic Letter 88-16 and Technical Specification 5.6.5.

Impact of Activity

The function of the reload core is to provide enough energy to support Cycle 17 operation while maintaining appropriate operating margins. Interactions with other plant systems will not change. The GE14 fuel and use of Marathon control blades are the only physical changes to the plant. No other safety related equipment is affected. Quad 2 Cycle 17 is designed for a 24 month operating cycle at a power level of 2957 MWt. Consequently the Q2C17 core design has a higher enriched reload batch compared with previous Quad Cities Unit 2 cycles. The Cycle 17 core design neutronic and thermal hydraulic analyses incorporate the effects of the higher enrichment. The core design has been analyzed (from a transient and accident perspective) for operation at a rated power level of 2957 MWt. These analyses are summarized in the Supplemental Reload Licensing Report and are used to determine cycle-specific thermal limits. The thermal limits are provided in the Q2C17 COLR. The transient and safety analyses for Cycle 17 were performed assuming a mixed core of GNP GE14 fuel and Framatome-ANP ATRIUM-9B fuel.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The activities associated with this package were identified as requiring a 50.59 Evaluation. The 50.59 evaluation concluded that the Q2C17 reload core design for operation at a rated thermal power level of 2957 MWt utilizing the GE14 fuel design and the Q2C17 COLR could be implemented without obtaining a License Amendment. This conclusion was made with the understanding that the License Amendments needed to utilize GE14 fuel and to operate at EPU conditions (i.e., 2957 MWt) have been submitted to and approved by the NRC. All analyses done to evaluate the Q2C17 reload core design and GE14 fuel design used previously NRC reviewed and approved methodologies and were performed within the constraints of Global Nuclear Fuel/General Electric's quality assurance program.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

[22] **Tracking Number:** **QC-E-2002-003** **Change Document(s)**

Unit: **Common** **EC 336060; EC336165**

Activity Description

The activity involves a change to the facility. The proposed design changes will upgrade the existing POWERPLEX-II (PPLX-II) Core Monitoring Software System (CMSS) to POWERPLEX-III (PPLX-III) CMSS Version (UFEB02). The PPLX CMSS code is used to perform heat balance and fuel thermal limit calculations, collect Traversing In-core Probe (TIP) and Local Power Range Monitor (LPRM) data, and perform LPRM calibrations. The PPLX code may also be used for monitoring shutdown margin (SDM) and determining enthalpy deposited from a Control Rod Drop Accident (CRDA) to ensure it is less than the required limit. The new PPLX-III CMSS will perform the same functions and maintain the same interface with plant equipment as the existing PPLX-II CMSS. The PPLX CMSS is common to both Quad Cities units. The PPLX-III CMSS incorporates a methodology change in the neutronics design code from CASMO-3/MICROBURN-B to CASMO-4/MICROBURN-B2. The new methodology has been accepted by the NRC in their review and acceptance of Licensing Topical Report EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2." The on-line core monitoring software upgrade is being implemented in conjunction with a corporate initiative to install a common, highly accurate core monitoring system at all BWR sites.

Impact of Activity

Phase 1 of the proposed design change will install the PPLX-III CMSS, but maintain the existing PPLX-II CMSS as the approved on-line core monitoring software. Comparison testing of the PPLX-III CMSS will be performed to validate the acceptability of PPLX-III CMSS prior to the final implementation under Phase 2. Upon completion of Phase 2, PPLX-III will become the approved on-line core monitoring software. PPLX-III will perform the same functions and maintain the same interface with plant equipment as the existing PPLX-II.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. For the proposed design changes, the conclusion of the 50.59 review is that the implementation of PPLX-III CMSS does not change the design function of an SCC or procedure as described in the UFSAR, is not a test or experiment not described in the UFSAR, and does not require a change to the Technical Specifications or Operating License. In addition, the change in methodology is not a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. The PPLX-III upgrade does not impact the safety limits nor do they keep the plant from operating in a manner consistent with the UFSAR and Technical Specifications. Based on the results of this 50.59 review, the proposed activity can be implemented without obtaining a License Amendment.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

23 **Tracking Number:** **QC-E-2002-004** **Change Document(s)**

Unit: **Unit 1** **UFSAR-01-R7-98**

Activity Description

The activity involves a change to the facility. Specifically, the Quad Cities Unit 1 Cycle 18 (Q1C18) core design and the introduction of GE14C fuel and Marathon control blades to Quad Cities Unit 1 to support Q1C18 operation at a rated power level of 2957 MWt. Additional activities within the purview of this evaluation include: new equipment out-of-service (EOOS) flexibility options, implementation of the maximum extended load line limit analysis (MELLLA), implementation (partial) of Average Power Range Monitor (APRM)/Rod Block Monitor (RBM) Technical Specification (ARTS) (i.e. power and flow dependent thermal limits), an exposure extension for ATRIUM-9B LHGR limits, and Final Feedwater Temperature Reduction (FFWTR) of 120°F. The Q1C18 core design consists of a total of 724 fuel assemblies, including 428 previously loaded Framatome-ANP (FANP) ATRIUM-9B (offset) assemblies from Quad Cities Unit 1 Cycles 16 through 17A, and 296 fresh Global Nuclear Fuel (GNF) GE14C (offset) assemblies. The GE14C fuel for the Q1C18 core design is physically identical to the GE14C fuel currently in Quad Cities Unit 2. The Q1C18 reload core safety analysis and cycle-specific reload analyses were performed in accordance with USNRC approved General Electric (GE) analytical methodologies described in NEDE-24011-P-A-14, "General Electric Standard Application for Reactor Fuel (GESTAR)." The Q1C18 core safety analyses and cycle-specific reload analyses included Extended Power Uprate (EPU) conditions, GE14C fuel, the new EOOS options, MELLLA, and partial implementation of ARTS power and flow dependent thermal limits. The Q1C18 Core Operating Limits Report (COLR) includes all limits applicable to the core, including limits for new EOOS flexibility options not previously analyzed for Quad Cities Unit 1. The loss of coolant accident (LOCA) analysis under EPU conditions for all fuel types in the Q1C18 reload is documented in NEDC-32990P Revision 1. The proposed changes are necessary to allow operation of Q1C18 for 26 months at the EPU power level while maintaining the necessary margin to thermal limits.

Impact of Activity

The effect of implementing the proposed changes is to allow Q1C18 to operate at EPU conditions within all core thermal limits. These activities require a revision to the COLR for operation of Q1C18, as well as UFSAR changes.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. All analyses done to evaluate the Q1C18 reload core design used previously NRC reviewed and approved methodologies and were performed within the constraints of Global Nuclear Fuel/General Electric's quality assurance program as reviewed and approved by Exelon Corporation. The activities associated with this package were identified as requiring a 50.59 Evaluation. The 50.59 Evaluation concluded that the Q1C18 reload including new EOOS options could be implemented without prior NRC approval.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

10 CFR 50.59 Validation Summaries

1 **Tracking Number:** QC-V-2001-0008 **Change Document(s)**

Unit: Unit 2 DCP 9900530; DCP 9900531

Activity Description

The activity involves a plant modification to add automatic start logic to the 2-0202-47A(B) and 2-0202-48A(B) Fluid Coupler Lube Oil Pumps which supply coupling oil to 2A(B) Recirculation System Motor Generator Sets (i.e., RR MG Sets). This activity will increase the reliability of the of the Unit 2 RR MG Sets and consequently prevent potential plant transients. Currently, a loss of power to (or failure of) a Fluid Coupler Lube Oil Pump causes a trip of the associated RR MG Set. This activity will modify the start logic such that the standby Fluid Coupler Lube Oil Pump will automatically start and restore system oil pressure before the associated RR MG Set trips.

Impact of Activity

The effect of this activity is to increase the reliability of the Unit 2 RR MG Sets and consequently prevent potential plant transients. The modification will allow the standby Fluid Coupler Lube Oil Pump to automatically start if system oil pressure reaches a specified low pressure OR if logic control power is lost to the operating Fluid Coupler Lube Oil Pump. The associated RR MG Set scoop tube will lock up on the same start signal to minimize scoop tube movement and potential flow transients.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The activity does not have any adverse effects on the plant. There is no design basis limit for a fission product barrier associated with or affected by the proposed activity. In addition, the proposed activity does not represent in a departure form a method of evaluation described in the UFSAR.

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2 **Tracking Number:** QC-V-2001-0016 **Change Document(s)**

Unit: Common QCOP 6900-09

Activity Description

The activity deletes station procedure QCOP 6900-09. QCOP 6900-09 provides guidance for placing the Alternate Shutdown Method (ASM) batteries on equalize charge, adjusting battery voltage, and subsequently returning the batteries to a float (trickle) charge. These batteries were in place for use during an Appendix R fire to provide an alternate power supply to the reactor relief valves when normal 125V DC power is lost. The ASM batteries are no longer in use and have been physically removed from the plant.

Impact of Activity

The effect of this activity is to delete a procedure that is no longer in use. The mobile battery carts (ASM batteries) were in place solely for use during Appendix R fires. Since they no longer provide a function, they have been removed from the plant and a procedure for charging these batteries is no longer required.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. Under the current Appendix R safe shutdown program, these batteries provide no function and have been removed. For this reason, no UFSAR described design function is adversely affected by this change.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

3	Tracking Number:	QC-V-2001-0030	Change Document(s)
	Unit: Unit 1		EC 23254; EC 26695

Activity Description

The activity involves a change to the facility. The proposed design change is the installation of the Unit 1 Oscillation Power Range Monitor (OPRM), Phase II design. The OPRM system utilizes the existing Local Power Range Monitor (LPRM) signals to detect reactor core thermal hydraulic instabilities. The OPRM system monitors the output of all available LPRMs, in parallel with the existing Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) functions. When completed the OPRM system will be capable of initiating a reactor scram via existing Reactor Protection System (RPS) trip logic when detection of core power oscillations from thermal hydraulic instabilities under high power and low core flow conditions occurs. The OPRM system, when fully operational, will be automatically enabled at high power and low recirculation flow conditions. The first phase of OPRM installation provided alarm and display functions only. The second phase provides the capability to make the OPRM system trip operational by employing the automatic suppression function (ASF) trip (SCRAM) which will suppress postulated reactor core oscillations. The OPRM System consists of four redundant OPRM trip channels, two per RPS Trip System. Each channel consists of two OPRM modules, each module pair providing trip outputs to the corresponding RPS channel. Quad Cities station committed to install a detect and suppress system in response to Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors." Note that the OPRM modification will not be made fully operational until resolution of the related 10 CFR Part 21 issue (reference 10 CFR Part 21 2001-23-5, "Stability Reload Licensing Calculations Using Generic DIVOM Curve," issued July 2001).

Impact of Activity

The effect of this activity is to allow the OPRM system trip to be fully operational by employing an automatic suppression function (ASF) trip (SCRAM signal) to suppress the reactor core power oscillations prior to exceeding the Minimum Critical Power Ratio (MCPR) safety limit. Operating procedures will be revised to replace the interim manual methods as the primary means of protection. The OPRM system, when it is fully operational, will function to automatically prevent the fuel from exceeding the MCPR safety limit during postulated thermal hydraulic instability events. The instrumentation installed by this design change will improve plant safety and protection of the reactor core.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59. The final OPRM installation does not cause a change to the existing APRM and RPS design or trip philosophy but only augments the existing APRM trip outputs such that the OPRM trip will logically function in the same manner as the existing APRM trips. The OPRM system is designed to detect core power oscillations in response to the thermal hydraulic instability that can occur under high power, low core flow during any condition of normal operation and initiate a scram via the existing RPS trip circuit. The installation of the OPRM and ABB flow units does not cause a change to parameters or plant system controls that contribute to these transients or accidents. The installation of the OPRM equipment is intended to improve mitigation to the instability region transients by providing detect and suppress functions. Providing reliable equipment to perform these functions can eliminate the sole reliance on operator manual action. This system has been thoroughly tested by the BWROG, vendors, and reviewed by the NRC. The single failure tolerant design of the APRM system assures that the APRM protective function is not affected by a worst-case OPRM failure. In addition, no new interfaces with operating nuclear safety systems or BOP systems are created. There is no failure of the OPRM system that can prevent the APRM or RPS circuits from responding to the possible accidents evaluated in the UFSAR.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

4 **Tracking Number:** **QC-V-2001-0040** **Change Document(s)**

Unit: **Common** **EC 24422; EC 24423**

Activity Description

The activity involves a change to the facility. The design change will replace the Unit 1 and 2 Emergency Diesel Generator (EDG) speed sensing panels (SSP) with an updated design. These speed sensing panels are obsolete and the components are reaching their end-of-life. The existing SSPs provide control permissives at 200 and 800 rpm to facilitate EDG start up. The SSPs accomplish this via a tuned circuit connected to the EDGs frequency generator. The vendor's recommended replacement uses a magnetic pickup unit (MPU) mounted at the EDG flywheel. With the MPU placed in close proximity to the flywheel teeth (ring gear), a voltage signal will be produced when the flywheel is rotated. This voltage signal is converted to engine speed by the SSP which actuates relay contacts at 200 and 800 rpm. The new SSP will provide the same output as the existing SSP. No logic or parameter changes will be made to the EDG control circuits.

Impact of Activity

The use of the new SSP will not adversely affect EDG operation. These SSPs have been designed specifically to replace the existing SSPs and will provide the same 200 and 800 rpm logic permissives upon EDG start up. The architecture of the speed sensing panel and the MPU will not exhibit a single failure that is more severe than the SSP and frequency generator they replace. The replacement SSP and MPU have the same response time as the existing SSP and frequency generator. Their use will allow the EDGs to function per their UFSAR described design requirements.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The changes made by these ECs are within the bounds of the plant's design basis. They will not require a change or adversely affect any equipment required to be operable per the UFSAR or the Technical Specifications. The changes are intended to increase EDG reliability.

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5 **Tracking Number:** **QC-V-2002-0022** **Change Document(s)**

Unit: **Unit 1** **EC 24462**

Activity Description

The activity involves a plant modification. The change will provide a new and upgraded digital feedwater level control (FWLC) system. The new digital FWLC system is modeled based on the existing control system; however, the new system will include a "state-of-the-art" digital control system including an improved level control strategy. The reason for the change is that the existing FWLC system has become obsolete. The existing technology is not fault tolerant and has no redundancy and no self-check functions. Because the existing system has become obsolete, it presents a significant challenge in terms of reliability and maintainability.

Impact of Activity

The effect of the activity is to replace the exiting FWLC system with a modern, fault tolerant, state-of-the-art digital system. The new system will also include an improved control strategy. The new digital system and associated control algorithm is expected to provide an improved level of control and level fluctuation during normal pant operations.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The FWCL system has been thoroughly evaluated. The new digital FWLC system will continue to control reactor vessel level for all analyzed conditions and transients. The affect of upgrading the FWLC to a digital control system with new control features does not increase the frequency of occurrence or the consequences of any evaluated accidents and transients. The effects of upgrading the FWLC to a Digital Control System with new control features does not introduce the possibility of an accident of a different type or a malfunction with a different result. The failures previously postulated in the safety analysis bound any different failure modes that could be postulated within the new control design.

ATTACHMENT
SUMMARY REPORT OF CHANGES, TESTS, AND EXPERIMENTS COMPLETED

6 **Tracking Number:** QC-V-2002-0026 **Change Document(s)**

Unit: Unit 1 EC 333046

Activity Description

The activity involves a change to the facility. The proposed design change will install an automatic Reactor Recirculation (RR) Runback function in the speed control circuits for the 1A and 1B RR Motor Generators (MG's) under Design Change Package (DCP) Engineering Change (EC) number 333046. This activity is being performed in support of the Extended Power Uprate (EPU) project and to provide the control room operators with more thorough indications associated with the performance of the speed control circuits for the RR MG Sets. Once EPU is implemented at Quad Cities Unit 1, the plant will be required to operate all three Reactor Feed Pumps (RFP's) and all four of the Condensate/Condensate Booster (CD/CDB) pumps in order to achieve full EPU power levels. The automatic Runback circuit is being installed to reduce the potential of a Reactor (RX) trip on Low Water Level if a RFP automatically trips while all three are running or on the loss of a CD/CDB pump. Both RR MG Sets will "Runback" in speed to a predetermined level that corresponds to 70% total core flow when the automatic Runback function is initiated by the automatic trip of a RFP or loss of a CD/CDB pump at or above 85% steam flow. The new dual parameter deviation meters are being installed to provide the control room operators with more thorough information/indication associated with the operation of the RR speed control system.

Impact of Activity

The affect of the proposed activity will be that both the 1A and 1B RR MG Set will automatically runback in speed to a level that will produce a total core flow of approximately 70%. The runback will initiate a decrease in Reactor power corresponding to the negative reactivity added by the flow decrease. This will occur if a RFP automatically trips or on a loss of a CD/CDB pump while the unit is at or above 85% steam flow. A faster rate of speed change will also be experienced during the Runback scenario to reach the 70% core flow more quickly. The initiation of the Runback function will be automatic and must be manually reset to increase speed. New dual parameter deviation meters will allow the control room operators to compare the RR Generator speed with the speed signal (speed controller input) and the RR Generator speed with the speed demand (speed controller output). This dual indication will allow the operator to better identify and evaluate mismatches in the speed control system signals and also provide further assurance that the speed signals and demands are nulled (equalized) before unlocking the scoop tube. This will help prevent unwanted transients in RR Generator/Pump speed when unlocking the scoop tube and enabling the speed control circuits.

Bases for Not Requiring NRC Prior Approval

This activity has been evaluated under the criteria of 10CFR50.59 and determined not to require prior NRC approval. The activity does not alter the design function of any System Structure or Component (SSC) as described in the licensing basis of the plant. Normal plant operating conditions will not be changed. This activity is being accomplished in support of the EPU project that will be implemented at the same time as the activity. The automatic Runback function will reduce the potential of a reactor trip on Low Level during automatic trips of the RFP and/or the loss of CD/CDB pumps. No adverse system interactions are created. All existing design functions described in the UFSAR are maintained.