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10 CFR 50.59
10 CFR 72.48

SVP-11-001

January 5, 2011

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

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254, 50-265, and 72-53

Reference: Letter from T. J. Tulon (Exelon Generation Company, LLC) to U. S. NRC, "10 CFR 50.59 / 10 CFR 72.48 Summary Report," dated January 5, 2009

Subject: 10 CFR 50.59 / 10 CFR 72.48 Summary Report

In accordance with 10 CFR 50.59, subpart (d)(2), and 10 CFR 72.48 subpart (d)(2), "Changes, tests, and experiments," Exelon Generation Company, LLC is submitting a summary of completed changes, tests, and experiments for Quad Cities Nuclear Power Station (QCNPS). This summary is provided as an attachment to this letter, which describes the 10 CFR 50.59 evaluations that were completed for QCNPS between January 1, 2009 and December 31, 2010. The referenced letter provided the previous summary report. Note that there were no 10 CFR 72.48 evaluations completed for QCNPS between January 1, 2009 and December 31, 2010.

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

Respectfully,

William R. Gideon
Site Vice President
Quad Cities Nuclear Power Station

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

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

IE47
NMSS01

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ATTACHMENT
Summary Report of Completed Changes, Tests, and Experiments

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1 **Tracking Number:** **QC-E-2008-001**

Unit: **Unit 1 / Unit 2**

Activity Description

The activity replaces the existing Reactor Recirculation (RR) Motor-Generator (MG) Sets with Adjustable Speed Drive (ASD) units (two per unit). The ASDs provide the motive power to the RR Pumps. The main components being replaced are the MG Set drive motor, fluid drive coupler, scoop tube positioner, variable frequency generator and associated cooling system. The ASD components include a multiple winding step-down transformer, power converters, cooling system, and a dual control microprocessor system.

Impact of Activity

The existing RR MG Sets are original equipment relying heavily on large rotating electrical and hydraulic energy conversion components. These components require significant maintenance to achieve reliable performance. Additionally, some of the major MG Set components are either obsolete or no longer supported by the original supplier. The primary function of the RR system is to control reactor power during power operation, which is not affected by the ASD upgrade.

Bases for Not Requiring NRC Prior Approval

The conversion from RR MG Sets to ASDs will not increase transient, accident, or malfunction consequences. Further, the probability or likelihood of an accident or malfunction are not more than minimally affected by this change. A change to the Technical Specifications is not required to implement this activity. For these reasons the proposed activity may be implemented without obtaining a License Amendment.

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2 **Tracking Number:** **QC-E-2008-002**

Unit: **Unit 1 / Unit 2**

Activity Description

The existing Reactor Recirculation (RR) Motor-Generator (MG) Sets are being replaced with Adjustable Speed Drive (ASD) units (two per unit). This change requires a modification to the Anticipated Transient Without Scram (ATWS) function, which trips the RR pumps under pre-determined conditions. When the MG Sets are decommissioned, the original generator field breakers will be removed (the current ATWS tripping device). With the MG Sets, tripping the field breaker removed the excitation from the MG Set generator causing the associated RR pump to coast down. The ASD design retains the ATWS instrumentation logic; however, the tripping devices are different due to the ASD design. One division of ATWS trip logic will trip the ASD feed breaker. The other ATWS division will trip the ASD emergency stop circuitry (E-STOP). The E-STOP function is also being added as a manual trip function to the main control room to provide an independent stop function for the ASDs.

Impact of Activity

The proposed change retains the current ATWS tripping logic scheme. Using the ASD feed breaker and E-STOP function as the tripping devices functionally preserves the ATWS trip function. The E-Stop function is also being added to the main control room as an independent manual trip function. Analysis has demonstrated that tripping the RR pumps with ASDs produces a response similar to tripping the existing RR MG-Sets.

Bases for Not Requiring NRC Prior Approval

Only the ATWS tripping devices are affected by this activity. The 10 CFR 50.59 review has determined the change does not involve a new methodology described in the UFSAR, does not introduce any test or experiment not described in the UFSAR, and does not require a change to the Technical Specifications. Further, the probability or likelihood of an accident or malfunction are not more than minimally affected by this change. For these reasons the proposed activity may be implemented without obtaining a License Amendment.

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3 **Tracking Number:** **QC-E-2009-001**

Unit: **Unit 2**

Activity Description

The existing main turbine low pressure turbines will be replaced with new Alstom turbine rotors and new inner casings. The new turbines are an improved design and will improve plant thermal efficiency. The new rotors will be resistant to Stress Corrosion Cracking (SCC); the new casing material will be significantly less susceptible to erosion.

Impact of Activity

The activity will not impact the design function of the main turbine, condenser or connected systems. The replacement rotors have been designed to permit existing extraction steam, condenser, and turbine auxiliary systems to operate within their respective design limits. The new turbine will be more efficient, which will result in increased electric output from the main generator. The main generator will continue to be operated within design capability.

Bases for Not Requiring NRC Prior Approval

The 10 CFR 50.59 review determined that the change does not increase the frequency of occurrence or consequences of an accident evaluated in the UFSAR. Further, the turbine upgrade does not increase the likelihood of occurrence or consequences of a malfunction of an SSC important to safety. The change does not create the possibility of an accident of a different type or malfunction of an SSC important to safety different from those previously evaluated in the UFSAR. This change does not result in a design basis limit for fission product barrier being exceeded or altered. For these reasons the proposed activity may be implemented without obtaining a License Amendment.

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4 **Tracking Number:** **QC-E-2010-001**

Unit: **Unit 2**

Activity Description

The proposed activity replaces the foam spacers used on the Unit 2 Safety Related 250 VDC Battery. To facilitate installation, the foam replacement will occur one battery section at a time, with the battery logically segregated into three sections. As each section is disconnected a portion of the Unit 2 125 VDC Alternate Battery will be connected to the 250 VDC Battery string to form a complete (operable) 250 VDC battery. The current foam spacers (0.25 inch) will be replaced with 0.50 inch foam spacers. Temporary jumper cables will be used when aligning the Unit 2 125 VDC Alternate Battery to the Unit 2 250 VDC Battery. This connection will be enclosed in a non-conductive rubber sleeve for personnel protection and to prevent the cable from creating an inadvertent path to ground.

Impact of Activity

The 0.25 inch foam spacers currently in use do not provide adequate seismic support. An engineering evaluation determined the spacers should be 0.50 inches thick. Due to the complexity of the replacement task, taking the 250 VDC battery out-of-service and replacing all spacers within the Technical Specification allowed timeframe is not feasible. Accordingly, the spacers will be replaced in three sections utilizing the Unit 2 125 VDC Alternate Battery. Since the Alternate Battery is safety related and composed of the same type of battery cells, it can be aligned to power sections of the 250 VDC battery that are temporarily disconnected during the maintenance. The 125 VDC Alternate Battery is normally disconnected from the plant 125 VDC system. The 125 VDC Alternate Battery will be tested to ensure that the 250 VDC battery will remain fully operable during the replacement activities.

Bases for Not Requiring NRC Prior Approval

Both the 125 VDC alternate and 250 VDC batteries are safety related, seismically mounted, and are comprised of the same type of battery cells. The Alternate 125 VDC battery is more susceptible to tornado missiles which could adversely affect operation of the 250 VDC battery when in the modified condition. The use of the Unit 2 125 VDC Alternate Battery to facilitate replacement of the Unit 2 250 VDC safety related battery is acceptable provided that its use is limited to 52 days per year as specified in the UFSAR.

The 10 CFR 50.59 review determined that the change does not increase the frequency of occurrence or consequences of an accident evaluated in the UFSAR. Further, the change does not increase the likelihood of occurrence or consequences of a malfunction of an SSC important to safety. The change does not create the possibility of an accident of a different type or malfunction of an SSC important to safety from those previously evaluated in the UFSAR. This change does not result in a design basis limit for fission product barrier being exceeded or altered. For these reasons the proposed activity may be implemented without obtaining a License Amendment.

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5 **Tracking Number:** **QC-E-2010-002**

Unit: **Unit 1 / Unit 2**

Activity Description

The proposed activity implements fleet-wide procedure OP-AB-117-101, "Operations With the Potential to Drain the Reactor Vessel (OPDRV)," and associated station procedure OP-QC-117-101, which standardizes the definition of OPDRV across the Exelon BWRs. The changes associated with the implementation of a standard definition include: acceptance of 5 gpm as incidental leakage; discussion of credit for automatic or manual isolation capability in the definition of an OPDRV; and new guidance that treats Control Rod Drive Mechanism (CRDM) removal as a non-OPDRV condition provided certain conditions are met.

Impact of Activity

The statement "potential for draining the reactor vessel" appears in various Technical Specification conditions and is intended to ensure reactor water level is maintained greater than the top of active fuel (TAF) in all operating modes. An engineering evaluation has confirmed the guidelines in OP-AB-117-101 and OP-QC-117-101 are adequate to protect TAF. The procedures specify the appropriate precautions and limitations to ensure that removal of a single CRDM will not result in an OPDRV.

Bases for Not Requiring NRC Prior Approval

The 10 CFR 50.59 review concludes that the proposed changes do not result in more than a minimal increase to the frequency or consequences of a vessel drain down event. The activity includes sufficient limitations and controls to ensure that fuel is adequately protected for any activity that is not considered an OPDRV. Additionally, this activity does not create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR. This activity does not impact any fission product barriers and does not involve a method of evaluation as described in the UFSAR. For these reasons the proposed activity may be implemented without obtaining a License Amendment.

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6 **Tracking Number:** **QC-E-2010-003**

Unit: **Unit 0**

Activity Description

The proposed activity updates the offsite dose consequences of a postulated fire in the Interim Radwaste Storage Facility (IRSF). During inspection activities (Material Processing and Transportation Inspection) the NRC identified that the original 10 CFR 50.59 review for the Quad Cities IRSF was out of date. The 10 CFR 50.59 addressed the potential for a release due to a fire event, assuming that the resin was stored in a solidified concrete form. However, as described in UFSAR Section 11.4.2, dewatering is the current process used for processing radwaste resin. Calculation QDC-0000-M-1787 was performed to determine the offsite dose consequences associated with a postulated fire event and demonstrated dose consequences remain significantly less than the regulatory limits.

Impact of Activity

The original IRSF dose assessment determined an offsite dose of 0.3 Rem assuming all HICs were consumed in the fire and that the fire fractured the concrete inside the HICs. Considering resin is currently stored in dewatered form, there would be a measurable increase in the consequences if the contents were released during a fire. Therefore, a very conservative analysis was completed, consistent with NRC Generic Letter 81-38, (Storage of Low-Level Radioactive Wastes at Power Reactor Sites), Regulatory Guide 1.183 (Alternate Source Term) and Regulatory Guide 1.145 (Atmospheric Dispersion Models). The analysis of site area exclusion boundary dose (bounding for this scenario) was modeled and shown to be less than 2.5 Rem, or an increase of less than 2.2 Rem, which is less than a small fraction (10%) of the corresponding 10 CFR 100 limit of 25 Rem. Further, the results remain with the specific radiological dose limit of 2.5 Rem for this accident scenario as defined in NUREG0800 (Standard Review Plan) and NRC Generic Letter 81-38.

Bases for Not Requiring NRC Prior Approval

The storage of dewatered resin in the IRSF instead of resin solidified in concrete does not increase the frequency of occurrence of any accident evaluated in the UFSAR or the likelihood of a malfunction of an SSC important to safety. The consequences of a fire event in the IRSF were determined to be less than 2.5 Rem whole body, which meets the specific radiological dose limit of 2.5 Rem for this accident and is a small fraction of the 10 CFR 100 limit. Therefore, the activity does not result in a more than minimal increase to the consequences of an IRSF fire event. No credit is taken for isolation or mitigation provided by any SSC during the event. Therefore, the malfunction of any equipment associated with the IRSF will have no impact on the maximum calculated dose consequences. This activity does not create the possibility for an accident of a different type or for a malfunction with a different result than any previously evaluated in the UFSAR, and this activity does not impact any fission product barriers. The method of evaluation for an IRSF fire event is not specified in the UFSAR. However, the method utilized for other offsite dose analyses is explicitly described in the UFSAR. Since the methodology for an offsite release for an IRSF fire is not explicitly discussed and since the methodology used is consistent with that discussed for scenarios involving dose consequences, this activity does not result in a departure from a method of evaluation described in the UFSAR. For these reasons the proposed activity may be implemented without obtaining a License Amendment.
