Exelon Generation.

10 CFR 50.59
10 CFR 72.48

SVP-21-001

January 4, 2021

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 Kenneth S. Ohr (Exelon Generation Company, LLC) to U. S. NRC, "10 CFR 50.59 / 10 CFR 72.48 Summary Report," dated January 4, 2019

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, 2019 and December 31, 2020. The referenced letter provided the previous summary report. There were no 10 CFR 72.48 evaluations completed for QCNPS during the specified time period.

Should you have any questions concerning this letter, please contact Sherrie Grant at (309) 227-2800.

Respectfully,

Kenneth S. Ohr
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

ATTACHMENT

Summary Report of Completed Changes, Tests, and Experiments

Tracking Number: QC-E-2018-001 Rev 2

Unit: Unit 1 and 2

Activity Description:

This activity performs a full 50.59 evaluation for site modifications under EC 404570 and 404571. These modifications eliminate the identified single point vulnerability by changing the failure position of the 1(2)-5406 air-operated valve (AOV) from closed to open. This is the Off-Gas Filter Outlet valve and is in the vault north of the main plant stack and is not accessible during normal operations. In the event of a loss of power or instrument air pressure concurrent with valid valve closure signal (when high dose conditions are present), operators will manually isolate the offgas discharge to the stack prior to the release reaching maximum allowable dose criteria at the Exclusion Area Boundary (EAB) or Low Population Zone (LPZ).

Impact of Activity:

The offgas isolation valve 1(2)-5406 was originally designed to close upon detection of high dose conditions. However, in the event of a single failure of local power or instrument air, the isolation valve would close (when high dose conditions may or may not be present) resulting in a loss of condenser vacuum and unit scram. As such, a loss of control power or instrument air to the 1(2)-5406 AOV is considered a single point vulnerability for the unit. These modifications eliminate the single point vulnerability. The overall function of the offgas isolation valve is not changed and the offgas system discharge will continue to be monitored for radiation prior to being released to the plant chimney stack.

Basis for Not Requiring NRC Prior Approval:

The 50.59 evaluation concludes that this activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR because the modified valve cannot initiate an accident and therefore does not impact the frequency of occurrence of any accident that might result in high radiation in the offgas. Analysis performed in support of this activity determined that Operations will have a total of 109 hours following the high radiation signal to manually isolate the Off-Gas system from the plant chimney before 100 mrem at the EAB or LPZ is reached. This duration is considered sufficient with margin. This activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR because the modified valve cannot initiate a malfunction of any SSC important to safety and is not considered substitution of manual action for automatic action for performing a UFSAR-described design function. The manual action is only required under certain failure scenarios and is consistent with UFSAR Chapter 11, which identifies a performance objective of the system as being "to provide sufficient time for operator decision and action in the event of off-standard conditions." The proposed activity does not result in more than a minimal increase in the consequences of an accident or a malfunction of an SSC important to safety previously evaluated in the UFSAR, and the consequences are enveloped by the consequences of the hypothetical failure of the holdup piping considered in the UFSAR Section 15.7.4. The change in valve failure position as a result of this activity cannot cause an accident or create a possibility for an accident of a different type than any previously evaluated in the UFSAR. This change cannot create the possibility for a malfunction of equipment important to safety with a different result than a malfunction of the previous configuration. This activity does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered. This activity issues an analysis to determine the time limit for a manual action to close the offgas discharge isolation valve such that the dose limit is not exceeded at the EAB or the LPZ if the valve fails open concurrent with high radiation in the offgas line. This analysis does not change or replace any method of analysis used in to establishing the design bases or in the safety analyses in the UFSAR. This analysis is used only to establish the time available for manual action and is performed in accordance with the methodology of the Standard Review Plan (SRP) 11.3 Revision 2, Branch Technical Position ETSB 11.5, Postulated Radioactive Release Due to a Waste Gas System Leak or Failure.

Therefore, this activity may be implemented in accordance with the governing procedure without prior NRC approval.

Tracking Number: QC-E-2020-001 Rev 1

Unit: Unit 1 and Unit 2

Activity Description:

This activity performs a 50.59 evaluation for the compensating actions outlined in Op Eval EC 632028 Rev 3, MO 1(2)-0220-2 Setup with Lower Closing Thrust at Torque Switch Trip than that Required by the Sizing Calculation. The Op Eval is supporting that the Main Steam Line (MSL) Outboard Drain Valve (MOV 1-(2)-0220-2) is still operable in the current, closed position. The 50.59 evaluation will address the compensating actions which will require an operator to be staged near the valve, if opened for any reason, to take manual action of hand wheeling closed the partially open valve and applying the appropriate torque to the handwheel in the event of a small break Loss of Coolant Accident (LOCA) and/or Group 1 isolation with the MSL inboard Drain Valve 1(2)-0220-1 failing to close.

NOTE: This 50.59 evaluation and compensating action was used for a short time until the MSL Outboard Drain Valves (MOV 1(2)-0220-2) were declared inoperable and the appropriate TS actions were taken. See LER 254/2020-002-00, "Main Steam Line Drain Valve Declared Inoperable Due to Improper Torque Switch Setting," for more information.

Impact of Activity:

Per Operability Evaluation EC 632028 Rev 3, the operability of MOV 1(2)-0220-2 in the current configuration and position (closed) is supported. However, to support operability during the time the valve is reopened, the compensating actions will be required until the torque switch can be readjusted during the next outage of sufficient duration or refueling outage. Numerous operating procedures were updated as part of this compensating action to ensure that the operator is staged whenever MOV 1(2)-0220-2 is opened and the reactor is pressurized. Design limits, accident frequencies, malfunctions and consequences are unaffected by these changes.

Basis for Not Requiring NRC Prior Approval:

The 50.59 Evaluation concludes that this activity does not result in a more than minimal increase in the frequency of occurrence of an accident previously evaluated in the UFSAR because the MOV cannot initiate an accident; therefore, does not impact the frequency of occurrence of any accident. This activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR because no new failure modes or causes are introduced as a result of this change. The proposed activity 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. The compensating actions cannot cause an accident or create a possibility for an accident of a different type than any previously evaluated in the UFSAR. This activity cannot create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR. This activity does not result in a design basis limit for fission product barrier as described in the UFSAR being exceeded or altered. This activity does not change or replace any method of analysis described in the UFSAR used in establishing the design bases or in the safety analyses as the compensatory actions for MOV 1(2)-0220-2 do not involve a method of evaluation as defined in the UFSAR. No changes to the Technical Specifications or the Facility Operating License is required.

Tracking Number: QC-E-2020-002 Rev 0

Unit: Unit 1 and Unit 2

Activity Description:

This activity performs a 50.59 evaluation to support a Document Change Request (DCR) that provides the basis for accepting the use of the BELSIM software and the Data Validation and Reconciliation (DVR) thermal model of the plant provided by True North. This DCR allows the addition of any new procedures, as applicable and revision of existing procedures that determine and apply a correction factor to the feedwater flow measurement used to calculate core thermal power. This will change the calculated thermal power that is used to calibrate the APRMs and ultimately will allow the calculated thermal power to be closer to the actual core thermal power. No physical changes are made to the plant by this DCR.

Impact of Activity:

This activity provides a process which significantly improves the uncertainty of the feedwater flow measurement resulting in a more accurate reactor heat balance and consequently more accurate APRM indication.

Basis for Not Requiring NRC Prior Approval:

This activity affects calibration of the average power range monitors (APRMs). The APRMs have a design function to trip the reactor on high neutron flux. UFSAR description of design function and procedures remain accurate as written, however, this activity will result in a change the reactor power and neutron flux. As a result, the activity was screened in and a 10CFR50.59 evaluation was performed.

This activity provides a new method for determining a feedwater flow correction factor used in the determination of reactor thermal power. The objective is to correct the actual power produced by the reactor while remaining within the current licensing commitments. The methodology used to provide input to the reactor heat balance based on the feedwater flow measurements is not described in the UFSAR. As a result, the introduction of software to correct for bias and report uncertainty in the feedwater flow measurement and determine a feedwater flow correction factor does not involve an adverse change to an element of a UFSAR described evaluation methodology, or use of an alternative evaluation methodology, that is used in establishing the design bases or used in the safety analyses.

The result of this activity will be a small change in reactor power while remaining within the current licensing commitments. The likelihood of occurrence of accidents or malfunctions evaluated in the UFSAR is already based on the appropriate limiting conditions, such as licensed reactor power. Consequently, a change in reactor power that is still within the limit will not result in a more than a minimal increase in the frequency of occurrence of any accident or malfunction evaluated in the UFSAR. Similarly, the dose consequences of accidents and malfunctions evaluated in the UFSAR are already based on the appropriate limiting conditions, such as maximum neutron flux and licensed reactor power. As a result, the consequences of accidents and malfunctions as evaluated in the UFSAR are unaffected by this activity. The small change in reactor power and neutron flux resulting from this activity will remain within the current licensing commitments. Consequently, this change does not create the possibility of an accident of a different type or with a different result than any previously evaluated in the evaluated in the UFSAR. Evaluations in the UFSAR of the design basis limits associated with fission product barriers are also based on the appropriate limiting conditions. This activity does not change those limiting conditions and, consequently, does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

Therefore, this activity does not require prior NRC notification and may be implemented in accordance with the governing procedures.

Tracking Number: QC-E-2020-003 Rev 0

Unit: Unit 0, Unit 1, and Unit 2

Activity Description:

This activity performs a 50.59 evaluation to allow the revision of procedure CC-QC-409. Procedure CC-QC-409 is a maintenance specification that allows maintenance personnel to replace the obsolete General Electric (GE) RMS-9 trip units with GE EntelliGuard trip units which are the recommended replacement. These trip units contain the programming and setpoints which trip the circuit breakers that are used in non-safety related applications. EC 632901 will revise CC-QC-409 to allow the EntelliGuard trip units to be used in safety-related applications as well.

Impact of Activity:

As the EntelliGuard trip units were designed to be a direct replacement of the RMS-9 units (and originally analyzed electro-mechanical trip units), their use at Buses 18, 19, 28, and 29 will have no adverse effects on behavior of the equipment, or how the plant is operated. The trip units will continue to protect the loads from overload and overcurrent conditions by use of the long time delay, short time delay, and instantaneous trips in the same manner as the existing RMS-9 units. Engineering Change (EC) 632904, which supports this 50.59 Evaluation determined that the EntelliGuard units will not introduce any common mode failures which would disable redundant equipment required per the UFSAR. Therefore, the single failure criteria analysis for these buses remains valid.

Basis for Not Requiring NRC Prior Approval:

EC 632904 performed a "qualitative assessment" of the design, development, and qualification of the new EntelliGuard units per NRC Regulatory Issue Summary (RIS) 2002-22, Supplement 1 and NEI 01-01. EC 632904 reviewed GE Hitachi document ECO#: 0002567 Critical Digital Review (CDR) for their new EntelliGuard trip units. ECO#: 0002567 demonstrated the design, manufacture, and qualification of the new EntelliGuard trip units meets RIS 2002-22, Supplement 1 and NEI 01-01 guidance. EC 632904 concluded that the EntelliGuard trip unit's probability of malfunction/failure/common mode failure is similar to the RMS-9 trip unit it replaces. Subsequently, there will not be more than a minimal increase in occurrence of accident or malfunction, increase in the consequences of an accident or malfunction, create the possibility of an accident of a different type, or a malfunction with different results.

This change, therefore, can be implemented under 10 CFR 50.59 without prior NRC approval and a License Amendment is not required. Since the subject trip units are not described in the Technical Specifications or Operating License, a change to these documents is also not required.