Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon. Nuclear

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

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

Subject: Additional Information Supporting Request for License Amendment to Increase Main Steam Safety Valve Lift Setpoint Tolerance and Standby Liquid Control System Enrichment (TAC MD3689 and MD3690)

Reference: Letter from D. M. Benyak (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Increase Main Steam Safety Valve Lift Setpoint Tolerance and Standby Liquid Control System Enrichment," dated November 7, 2006

In the referenced document, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. The proposed change revises Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve (MSSV) lift setpoint tolerance from ± 1% to ± 3%. In addition, the proposed change revises SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the Standby Liquid Control System from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

The NRC has requested additional information to complete its review. In response, EGC is providing the attached information.

EGC has reviewed the information supporting a finding of no significant hazards consideration that was previously provided to the NRC in Attachment 1 of the referenced document. The information provided in this submittal does not affect the bases for concluding that the proposed license amendment does not involve a significant hazards consideration.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Michelle Yun at (630) 657-2818.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 24th day of January 2007.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachment 1: Response to NRC Request for Additional Information

ATTACHMENT 1

Response to NRC Request for Additional Information

NRC Request 1

The final paragraph of GE-NE-0000-0053-8435-R1P, page 6-5, states, "Exelon will ensure that the 10CFR50.62 [sic] requirement to inject 86 GPM [gallons per minute] of 13% sodium pentaborate solution, or the equivalent, plus the ATWS [anticipated transient without scram] specific injection requirements stated in Section 3.0 of this report are met for injection against the maximum reactor vessel pressure of 1301 psig at the SLCS sparger occurring during an ATWS event when the SLCS is in operation without opening of the SLCS relief valve." However, TS SR 3.1.7.7 requires a discharge pressure of 1275 psig for each pump. Explain the disparity in discharge pressure between the General Electric (GE) maximum and the SR. Also explain why TS SR 3.1.7.7 should not be revised to reflect the higher discharge pressure.

Response

The Quad Cities Nuclear Power Station (QCNPS) Standby Liquid Control (SLC) system consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel. The following table provides SLC System relief valve setpoints and the system design pressure for each QCNPS unit.

	Unit 1	Unit 2
Relief Valve Setpoint	1602 psig	1602 psig
SLC System Design Pressure	1602 psig	1602 psig

Technical Specification (TS) Surveillance Requirement (SR) 3.1.7.7 is based on NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4." SR 3.1.7.7 of NUREG-1433 requires that each SLC system positive displacement pump develop a flow rate and discharge pressure based on plant-specific values. QCNPS SR 3.1.7.7 is performed to verify a flow rate of ≥ 40 gpm at a discharge pressure of ≥ 1275 psig. The intent of this SR is to ensure that pump performance has not degraded during the fuel cycle. As described in the Bases for SR 3.1.7.7, this test confirms one point on the pump design curve and is indicative of overall performance. The test confirms operability of the SLC pumps, detects incipient failures identified by abnormal performance, and provides assurance that the pumps have not degraded. Verifying that the pump is operating on its curve provides confidence that the pump will meet its design requirements.

In addition, SR 3.1.7.7 is performed to meet the requirements of the QCNPS Inservice Testing Program. In accordance with the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants, OM Code (1998 Edition through 2000 Addenda), subsection ISTB-3300, "Reference Values," the reference values shall be established at a point of operation (i.e., reference point) readily duplicated during subsequent tests. The SLC system uses positive displacement pumps which are designed to meet the required flow rate over the entire range of operating pressure. Thus, testing at a reference point is sufficient to establish operability. This philosophy applies regardless of a change in ATWS analytical injection pressure. The QCNPS SR 3.1.7.7 test pressure remains consistent with the ASME OM Code Inservice Testing requirements.

The above information forms our basis for not requesting a revision to SR 3.1.7.7.

NRC Request 2

As discussed during teleconferences [related to Reference 1 for Dresden Nuclear Power Station] between the NRC staff and Exelon, the NRC staff requests that Exelon submit a commitment to perform a one-time pump test, prior to implementing this amendment, with the flow recorded at sufficiently high pressure and flow rate to demonstrate, after density correction, that the design-basis requirements of the SLCS pumps have been met. In addition, provide the pressure and flow rate that this will be used for this one-time pump test.

Response

Prior to Extended Power Uprate (EPU) operation at QCNPS in 2002, the Unit 1 and Unit 2 SLC systems were upgraded to accommodate the higher pressure that could be experienced as a result of EPU conditions. Following the design change, a test was successfully completed to confirm the system's functional capability. The test procedure established acceptance criteria to ensure each SLC pump could achieve a minimum flow rate of 42 gpm at a pressure range of 1470 psig – 1500 psig (as measured at the pump discharge), confirming the design flow requirement for single pump operation could be achieved. For these reasons, a similar regulatory commitment is not required for QCNPS.

NRC Request 3

On August 31, 2006, during a telephone conference call [related to Reference 1 for Dresden Nuclear Power Station] between Exelon and the NRC staff, Exelon discussed its testing program for the SLCS pumps that follow the requirements of the American Society of Mechanical Engineers Operation and Maintenance Code (ASME OM Code). Regarding the acceptance criteria for inservice testing, confirm that the testing and test acceptance criteria used in the ASME OM Code testing of the SLCS pumps, as required by Title 10 of the Code of Federal Regulations Section 50.55a, demonstrate that the SLCS pumps are operationally ready and capable of performing their intended function(s).

During a conference call on September 25, 2006, the NRC clarified NRC Request 3. Specifically, the NRC is questioning how future degradation of the SLC pumps will be factored into the results of the one-time test described above, since the ASME OM Code, permits the SLC pumps to degrade as much as 10% during the quarterly testing, and still meet the ASME OM Code acceptance criteria.

Response

As described in the ASME OM Code as well as in EGC procedures that implement the Code, two acceptance criteria must be satisfied when testing the SLC pumps. First, the flow must be within 10% of the reference value established during the pre-service or first inservice test. Second, the flow must be greater than or equal to 40 gpm, as required by Technical Specification (TS) Surveillance Requirement (SR) 3.1.7.7.

10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," requires the SLC system to have the capability of injecting a borated water solution, in which the resulting reactivity control is at least equivalent to that resulting from injection of 86 gpm of 13-weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design.

Under the proposed change, EGC is requesting NRC approval to increase the enrichment of sodium pentaborate used in the SLC system to ≥ 45.0 atom percent boron-10. EGC has evaluated the equivalent SLC system flow requirement, as required by 10 CFR 50.62, using the methodology described in NEDE-31096-P-A (i.e., Reference 2), which was approved by the NRC in Reference 3. Using 14-weight percent sodium pentaborate (i.e., the minimum concentration allowed by TS 3.1.7), and a boron-10 enrichment of 45.0 atom percent boron-10, the equivalent SLC system flow was determined to be 35.2 gpm.

Therefore, based on the above, continued use of the 40 gpm acceptance criterion defined by SR 3.1.7.7 for the quarterly ASME OM Code testing will ensure that the SLC pump is operationally ready and capable of performing its intended function, even if the SLC pump is in a 10% degraded condition.

NRC Request 4

Figure 3-8 of same report provides the bounding pressure against which SLCS must inject. Please provide the figure with the reactor vessel lower plenum pressure scaled in psig.

Response

The requested information was provided to the NRC in Reference 4 as Attachments 2 and 4. The figures are based on a report applicable to both QCNPS and Dresden Nuclear Power Station using the same data. Therefore, the aforementioned material is not being provided as an attachment to this letter but instead is being incorporated by reference.

NRC Request 5

Page 3 of 12 in Attachment 1 to the June 2, 2006 [for Dresden Nuclear Power Station], submittal states that all nine MSSVs are required to be operable by TS 3.4.3, "Safety and Relief Valves," and that the function of all nine safety valves is required to be operable to satisfy the assumptions of the safety analysis. However, on Page 2-2 of GE-NE-0000-0053-8435-R1P in Table 2-1, "Overpressure Results with 3% Setpoint Tolerance," it states that the number of dual safety relief valves (DSRVs) credited for Dresden Units 2 and 3 is zero, and the number of safety valves credited is eight. Based on the information in this table it appears that only eight safety valves were credited for the overpressure analyses at Dresden which is inconsistent with the information on Page 3 of 12 in Attachment 1 to the June 2, 2006, submittal. Please confirm whether the [QCNPS Unit 1 Cycle 19] and [QCNPS Unit 2 Cycle 19] reload analyses credited all nine safety valves.

Response

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that the peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. Each unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV). The reload analysis performed for QCNPS Unit 1 Cycle 19 credits all nine safety valves. The reload analysis currently in progress for QCNPS Unit 1 Cycle 20, which is scheduled to begin in May

2007, credits eight of nine safety valves. The reload analysis for QCNPS Unit 2 Cycle 19 credits eight of nine safety valves.

References

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment to Increase Main Steam Safety Valve Lift Setpoint Tolerance and Standby Liquid Control System Enrichment," dated June 2, 2006

2. NEDE-31096-P-A, "Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.62," dated February 1987

3. Letter from G. Lainas (U. S. NRC) to T. A. Pickens (BWR Owners' Group), "Acceptance for Referencing of Licensing Topical Report NEDE-31096-P, 'Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.62'," dated October 21, 1986

4. Letter from K. M. Nicely (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting Request for License Amendment to Increase Main Steam Safety Valve Lift Setpoint Tolerance and Standby Liquid Control System Enrichment," dated August 18, 2006