July 15, 2004

MEMORANDUM TO: G. Dick, Project Manager
Project Directorate III
Division of Licensing Project Management

FROM: J. Uhle, Chief ***/RA/***
PWR Systems Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: BRAIDWOOD AND BYRON TECHNICAL SPECIFICATION CHANGES OF PRESSURIZER SAFETY VALVE SETPOINT LIFT SETTINGS (TAC NOS. MB9760, MB9761, MB9762 AND MB9763)

Plant Name:	Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2
Utility:	Exelon Generation Company, LLC
Docket Nos.:	50-456, 50-457, 50-454 and 50-455
TAC Nos.:	MB9760, MB9761, MB9762, and MB9763
Project Directorate:	PD-III
Project Manage:	G. Dick
Review Branches:	SRXB/DSSA and EMEB/DE
Review Status:	Complete

Exelon Generation Company, LLC proposed to change the pressurizer safety valves lift settings specified in Technical Specification (TS) 3.4.10, “Pressurizer Safety Valves,” for Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2.

We have completed the review of the proposed TS 3.4.10 and concluded that it is acceptable. The attached evaluation documents the basis of acceptance.

This completes our review efforts for TAC Nos. MB9760, MB9761, MB9762, and MB9763.

Attachment: As stated

Contact: S. Sun (SRXB/DSSA), 415-2868
W.K. Poertner (EMEB/DE), 415-3409

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
REGARDING TECHNICAL SPECIFICATION CHANGES OF
PRESSURIZER SAFETY VALVE LIFT SETTINGS
EXELON GENERATION COMPANY, LLC
BRAIDWOOD UNITS 1 AND 2, AND BYRON UNITS 1 AND 2
DOCKET NOS. 50-456, 50-457, 50-454 AND 50-455

1.0 BACKGROUND

By letter dated June 27, 2003 (Ref. 1), Exelon Generation Company, LLC (EGC), the licensee for Braidwood Station Units 1 and 2 and Byron Station Units 1 and 2, proposed changes to the Braidwood and Byron Technical Specification (TS) 3.4.10, "Pressurizer Safety Valves." The current Braidwood and Byron TS 3.4.10 requires that three pressurizer safety valves (PSVs) shall be operable with "as-found" lift settings ≥ 2460 psig and ≤ 2510 psig. The current TS values represent a ±1 percent setpoint tolerance around a nominal lift setting of 2485 psig for the PSVs. The licensee indicated that there had been many instances where one or more of the Braidwood and Byron PSVs were found to have setpoints outside the ±1 percent setpoint tolerance, which resulted in PSVs being declared inoperable. It also indicated that most of the "as-found" lift settings had not exceeded ±2 percent of the nominal pressure setting. Therefore, the licensee proposed TS changes to reduce the nominal setpoint and increase the setpoint tolerance for the PSVs to minimize TS violations caused by setpoint drift.

The licensee proposed to change the existing "as found" PSV lift settings to a range of ≥ 2411 psig and ≤ 2509 psig. The proposed TS reflects changes in the allowed PSV tolerance from ±1 percent to ± 2 percent and a reduction in the nominal lift setting from 2485 psig to 2460 psig. The TS change allows a decrease in the valve actuation pressure and therefore provides the potential for earlier pressurizer relief at a reduced reactor coolant system (RCS) pressure. The licensee would not change the ±1 percent "as-left" setpoint tolerance for the PSVs specified in the current TS Surveillance Requirement (SR) 3.4.10.1. The proposed change would revise the associated Bases for SR 3.4.10.1 to reflect the proposed PSV settings.

In support of the proposed TS changes, the licensee provided the results of its technical evaluation (Ref. 1) and responses (Refs. 2 through 4) to the staff requests for additional information for the staff to review.

2.0 REGULATORY EVALUATION

General Design Criteria (GDCs) 10, "Reactor Design," and 15, "Reactor Coolant System Design," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR 50 require that the specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary (RCPB) must not be exceeded during normal operation, including anticipated operational occurrences.

10 CFR 50.36 specifies the Commission's regulatory requirements related to the content of TSs. Specifically, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining

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whether a limiting condition for operation (LCO) is required to be included in TSs. These criteria are: (1) Criterion 1 - installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB; (2) Criterion 2 - a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) Criterion 3 - a structure, system or component (SSC) that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and (4) Criterion 4 - an SSC which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. An SSC which satisfies any of the above criteria must include an LCO and associated SRs in the TSs.

PSVs are part of the primary success path to mitigate consequences of design-basis events (DBEs), and are credited in the Braidwood and Byron Final Safety Analysis Reports (UFSARs) analyses. In accordance with Criterion 3 of 10 CFR 50.36(c)(2)(ii) discussed above, a TS LCO is required for the PSVs.

The staff reviewed the proposed TS changes in accordance with: (1) the requirements of GDCs 10 and 15 related to fuel integrity and integrity of the RCPB, respectively; and (2) the guidance regarding the PSV lift setting tolerance limits provided in NUREG-1431 (Ref. 5), “Standard Technical Specifications - Westinghouse Plants.” Since the NUREG was developed for Westinghouse plants based on the requirements of 10 CFR 50.36, it is applicable to the Braidwood and Byron plants which are Westinghouse four-loop plants.

3.0 TECHNICAL EVALUATION

The staff has reviewed the licensee’s proposed TS changes and the associated supporting analysis (Refs. 1 and 4) for the Braidwood and Byron plants, and prepared the following evaluation.

3.1 PSV Setpoint with Positive Tolerance Analysis

Each unit of the Braidwood and Byron plants has three spring-loaded PSVs with a relief capacity of 420,000 lb/hr for each valve. As described in Section 5.4.13 of the Braidwood and Byron UFSARs, the PSVs provide overpressure protection for the RCS. Together with the reactor protection system, the PSVs ensure that the RCS pressure meets the GDC 15 requirement in terms of the RCS design pressure safety limit. Compliance with the GDC 15 requirement is demonstrated in the analysis of the DBEs. In assessing the effects of the TS changes on the DBE analysis, the licensee evaluated the analysis of record (AOR) and identified that six DBEs relied on PSV actuation to limit the pressure increase to below the pressure safety limit of 110 percent of the design pressure. The events are: (1) uncontrolled rod withdrawal from full power; (2) loss of reactor coolant flow; (3) loss of external electrical load; (4) loss of normal feedwater; (5) loss of all AC power to station auxiliaries; and (6) reactor coolant pump locked rotor.

The licensee indicated that the AOR for the events listed above assumed that the PSVs would fully open when the calculated pressurizer pressure reaches 2534.7 psig, which corresponds to the PSV setpoint of 2485 psig with the associated tolerance of +1 percent and an additional +1 percent allowance to account for a pressure shift due to operation with water-filled pressurizer loop seals. Modeling the proposed PSV setpoint changes would result in a PSV full open pressure of 2533.8 psig, which is based on the proposed PSV lift setting of 2460 psig with the associated +2 percent tolerance and an additional +1 percent to account for pressure shift. The setpoint of 2534.7 psig assumed in the AOR generates a higher effective PSV opening pressure and results in a higher peak pressurizer pressure during an event. Therefore, the staff determined that the AOR remains bounding for overpressure protection and is valid for supporting the proposed PSV lift setting of 2460 psig with an upper tolerance of +2 percent.

3.2 PSV Setpoint with Negative Tolerance Analysis

Use of PSV setpoints with negative tolerances lowers effective PSV opening pressures, which would cause an earlier opening of the PSVs and a slower increase in RCS pressure during overpressurization events. PSV actuation at a lower RCS pressure could result in a lower departure from nucleate boiling ratio (DNBR) that reduces the margin to the safety DNBR limit, and it could also result in a higher pressurizer water level that increases the potential to overfill the pressurizer with water. The licensee evaluated the AOR and identified that the following events assumed PSV opening with negative tolerance modeled:

(1) loss of load (LOL) / turbine trip (TT),
(2) rod withdrawal at power (RWAP),
(3) loss of alternating current (LOAC) with reactor coolant pump (RCP) seal injection,
(4) loss of normal feedwater (LONF), and
(5) LOAC.

The licensee reanalyzed the five events for both Unit 1 (with Babcock and Wilcox International steam generators) and Unit 2 (with Westinghouse Model D5 steam generators). During the course of the review, the staff requested the licensee to discuss the methods used for the reanalysis and identify input parameters that were different from those assumed in the AOR. In response (Ref. 2), the licensee indicated that the computer code (LOFTRAN) and methods used in the reanalysis are the same as those used in the AOR. For events 1 through 3, changes in input parameters are associated with the proposed change in PSV setpoint and tolerance. For events 4 and 5, changes in input parameters are associated with a decrease in initial feedwater temperature and the proposed change in PSV setpoint and tolerance. The reduction in initial feedwater temperature causes a slower increase in RCS pressure and results in a higher pressurizer water level. Therefore, the assumption of a lower initial feedwater temperature is conservative with respect to the calculated pressurizer water level.

3.2.1 DNBR Reanalysis

The licensee performed the DNBR reanalysis for the LOL/TT and RWAP events that were identified in the AOR as the limiting cases that resulted in the lowest DNBR values. The results of the reanalysis (Ref. 1) indicated that for the LOL/TT event, although the calculated minimum DNBR values were slightly lower than those calculated in the AOR, they remained well above the DNBR safety limit. For the RWAP event, the analysis indicated that the results for Unit 2 remained limiting and that the minimum DNBR remained above that calculated in the AOR. Since the licensee’s reanalysis of the limiting events showed that the calculated DNBRs are either bounded by the AOR or meet the DNBR safety limits, the staff determined that the reanalysis satisfies the GDC 10 requirement related to fuel rod integrity criteria, and concluded that the reanalysis is acceptable to support the proposed TS.

3.2.2 Pressurizer Water Inventory Reanalysis

The licensee performed a reanalysis of pressurizer water level for the LOL/TT, LONF, LOAC and LOAC with RCP seal injection events that were identified in the AOR as the limiting cases that resulted in the highest pessurizer water levels. The results of the reanalysis indicated that for the LOL/TT, LONF, or LOAC events, although the calculated peak pressurizer water level was slightly higher than that calculated in the AOR, it did not reach the top of the pressurizer, and thus, the results demonstrated that no water discharge from the pressurizer occurred.

As for the LOAC with RCP seal injection event, the licensee's analysis indicated that continued injection of water into the RCS through the RCP seals would result in a water-solid pressurizer and water discharge through the PSVs. The proposed PSV setpoint tolerance assuming negative tolerance would result in a lower PSV lift setpoint. With the lower setpoint, the PSV would open earlier and a larger number of PSV water cycles and a lower water discharge temperature could result during the transient. The licensee performed an analysis of the LOAC with RCP seal injection event and determined (Ref. 1) the revised PSV setpoint would result in an increase of about one PSV water cycle and a reduction in the liquid discharge temperature of about 0.5 ^{0}F.

The licensee also performed a qualitative evaluation (Ref. 1) of the effect of the proposed TS changes on the spurious safety injection (SI) at power event, the limiting AOR event with respect to water inventory addition to the pressurizer. Based on its qualitative evaluation, the licensee claimed that the event would show results similar to those of the LOAC with RCP seal injection event in terms of the number of PSV water cycles and the PSV discharge water temperature. However, the licensee did not provide the results of a quantitative analysis of the spurious SI event to support its position. The staff requested the licensee to quantify the effect of the lower PSV setpoint on the AOR limiting event. In response, the licensee performed a reanalysis (Ref. 2) and showed that the revised PSV setpoint would result in an increase of two PSV water cycles and a reduction in the liquid discharge temperature of no more than 3.0 ^{0}F.

A comparison of the reanalysis showed that the spurious SI event remained the limiting event since it resulted in a greater increase in the number of PSV water cycles (two cycles vs. one cycle) and a greater decrease in the PSV discharge water temperature (3.0 ^{0}F vs. 0.5 ^{0}F) than that calculated for the LOAC with RCP seal injection event. As indicated in reference 6, the water discharge temperature in the AOR for the spurious SI event was 590 ^{0}F. The lowest discharge water temperature for the spurious SI event with the revised PSV setpoint is 587 ^{0}F (i.e., 590 ^{0}F - 3.0 ^{0}F). The staff found that the calculated water discharge temperature (587 ^{0}F) is significantly higher than the discharge water temperature of 530 ^{0}F that was used to support operability of the PSVs as discussed in the AOR (Ref. 6). Therefore, the staff concludes that the reanalysis is acceptable to assure that the PSVs will remain operable following a spurious SI event.

Since (1) the reanalyses used the same methods as those used in the AOR, (2) the values of input parameters, except for the PSV opening pressure and the conservative assumption of a lower initial feedwater temperature, used in the reanalyses were the same as that assumed in the AOR, (3) the results of the reanalyses showed that the calculated minimum DNBR did not exceed the DNBR safety limits, and that the calculated PSV operating conditions did not exceed the AOR PSV operability range previously approved by the staff, the staff concludes that the reanalyses are acceptable.

3.3 Margin Between High Pressure Reactor Trip and Opening of PSVs

The licensee indicated that the PSV setpoints were established to be above the setpoint of the high pressure reactor trip to minimize challenges to the PSVs. During the review, the staff requested that the licensee specify the pressure measurement uncertainties associated with the high pressure reactor trip and the PSVs, and confirm that they were appropriately considered in the error analysis such that a reactor trip would occur prior to a PSV actuation. In the licensee response (Ref. 2), the error analysis of the pressure measurement uncertainties showed that the lowest lifting pressure for a PSV (with the proposed setpoint of 2460 psig) is 2411 psig and the highest pressure for a reactor trip (with a nominal setpoint of 2385 psig) is 2427 psig. The licensee concluded that the probability of having a PSV lift before achieving a high pressure trip is less than 1% for any given pressure based on instrument uncertainties and that a reactor trip was expected to occur prior to a PSV actuation. Based on this information, the staff requested that the licensee evaluate the impact on the appropriate accident analysis acceptance criteria.

In response, the licensee evaluated the AOR and identified (Ref. 3) that only the peak pressure cases for the LOL/TT and RWAP analyses resulted in a reactor trip on high pressurizer pressure. The licensee reanalyzed (Ref. 4) these two events by assuming that the PSVs would lift at the low end of the tolerance band with and without including a one second delay in PSV lifting and subsequent steam relief through the PSVs to account for PSV loop seal clearance. The PSVs operate with a water loop seal during normal operation and the loop seal results in a nominal 1 second delay (as documented in WCAP-12910) in PSV lifting while the water is purged from the loop seal. When the PSV loop seal purge delay was included in the analysis, the high pressurizer pressure trip was reached prior to steam relief through the PSVs. When the loop seal purge delay was not included in the analysis, the PSVs relieved prior to receiving the high pressurizer pressure reactor trip signal. The licensee analysis determined that a reactor trip would occur on overtemperature delta-T (OTDT) in the LOL/TT case, and on high neutron flux in the RWAP case if credit was not assumed for the loop seal purge delay. The results of the reanalysis showed that the reactor trip occurred only a short time after it would have occurred on high pressurizer pressure (i.e., rod motion would start approximately 7.6 seconds later for the LOL/TT event and approximately 0.3 seconds later for the RWAP event). The reanalysis further confirmed that all acceptance criteria remained satisfied, in particular, the peak RCS pressure and steam generator secondary side pressure remained less than the allowable limits and the pressurizer did not reach a water-solid condition. Based on the above analysis, the staff determined that a reactor trip would occur during the analyzed events and that all acceptance criteria remained satisfied.

3.5 NUREG-1431 Consistency

The staff found that the proposed tolerance of ±2% for the PSV setpoint is within the allowable range specified in NUREG-1431 (Ref. 4), “Standard Technical Specifications for Westinghouse Plants.” Specifically, the Bases for SR 3.4.10.1 allow a tolerance range of ± 3 percent for the PSV setpoint.

Since (1) the proposed changes to the PSV setpoint tolerances are adequately reflected in the acceptable analyses that satisfy the requirements of GDC 10 for fuel integrity and GDC 15 for integrity of the RCS pressure boundary, and (2) the proposed PSV setpoint tolerances are within the range allowed for Westinghouse-designed plants, such as the Braidwood and Byron plants, as specified in the STS, NUREG-1431 (Revision 2), the staff concludes that the proposed PSV setpoint tolerances are acceptable.

As stated in the Bases for SR 3.4.10.1, the licensee would not change the ±1 percent “as-left” setpoint tolerance for the PSVs. Setting the PSV to this tolerance helps reduce the overall setpoint drift over time, which is acceptable to the staff.

4.0 CONCLUSIONS

The staff evaluated the licensee’s request to amend the Braidwood and Byron TSs to lower the nominal setpoint and increase the tolerance band of the “as-found” lift setting for PSVs specified in TS 3.4.10. The changes are proposed to minimize TS violations caused by setpoint drift. Based on the review discussed in Sections 2.0 and 3.0, the staff determined that the proposed revision to TS 3.4.10, “Pressurizer Safety Valves,” is acceptable. Therefore, the staff concludes that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in this manner; (2) such activities will be conducted in compliance with the Commission’s regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Thus, the licensee amendment request is acceptable.

REFERENCES

1. Letter from K. A. Ainger (Exelon Generation Company, LLC) to NRC, "Request for a License Amendment to Revise the Pressurizer Safety Valves Lift Settings," dated June 27, 2003.

2 Letter from K. A. Ainger (Exelon Generation Company, LLC) to NRC, "Request for Additional Information Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings," dated January 29, 2004.

3. Letter from K. A. Ainger (Exelon Generation Company, LLC) to NRC, "Request for Additional Information Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings," dated March 3, 2004.

4. Letter from K. A. Ainger (Exelon Generation Company, LLC) to NRC, "Request for Additional Information Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings," dated June 4, 2004.

5. NUREG-1431, Revision 2, "Standard Technical Specifications - Westinghouse Plants," April 30, 2001.

6. Letter from G. F. Dick (NRC) to O. D. Kingsley (EGC), "Issuance of Amendments; Increase in Reactor Power, Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated May 4, 2001.