Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon Nuclear

10 CFR 50.90

RS-04-007

January 29, 2004

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Request for Additional Information Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

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

In the referenced letter, Exelon Generation Company, LLC (EGC) requested NRC approval of a proposed amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed amendment would revise TS 3.4.10, "Pressurizer Safety Valves," by changing the existing pressurizer safety valves (PSV) lift setting from "≥ 2460 psig and ≤ 2510 psig" to "≥ 2411 psig and ≤ 2509 psig" to better reflect the design capabilities of the safety valves while maintaining the appropriate overpressure protection for the reactor coolant system.

During the NRC's review of the proposed change, a number of issues were raised regarding the analyses supporting the revision of the PSV lift setting. Subsequently, the NRC requested that EGC provide additional information to clarify these issues. This additional information is provided in Attachment 1.

Should you have any questions related to this matter, please contact J. A. Bauer at (630) 657-2801.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 1-29-04

Kenneth A. Ainger

Kenneth A. Ainger
Manager, Licensing

Attachment 1: Response to a Request for Additional Information (RAI) Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

Attachment 2: Westinghouse Systems & Equipment Engineering Group, "Evaluation of the Operability of Crosby PSV During Loss of Offsite Power with RCP Seal Injection," March 7, 2003

ATTACHMENT 1

Response to a Request for Additional Information (RAI) Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

NRC RAI No. 1

In support of the technical specification (TS) changes, reanalyses were performed for several cases to determine the DNBRs [departure from nucleate boiling ratios] and pressurizer water levels. Discuss the methods and computer codes used in the reanalyses for each case and address acceptability of the use of the methods and computers for licensing applications. Also, identify for the reanalyzed events the values of input parameters that are different from that assumed in the analysis of record.

<u>Response to RAI No. 1</u>

As discussed in Reference 1, the following events have been reanalyzed to assess the impact of the proposed change in pressurizer safety valve (PSV) setpoint and tolerance.

1. loss of load / turbine trip, departure from nucleate boiling (DNB) limiting cases.
2. rod withdrawal at power, DNB limiting cases.
3. loss of alternating current (LOAC) with reactor coolant pump (RCP) seal injection.
4. loss of normal feedwater limiting cases.
5. LOAC limiting cases.

The computer code (i.e., LOFTRAN) and methods used in the reanalysis of these five events are the same as the current analysis of record and are described in Byron/Braidwood Updated Final Safety Analysis Report (UFSAR) Chapter 15.

For events 1, 2 and 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 associated with a decrease in initial feedwater temperature; and changes in input parameters associated with the proposed change in PSV setpoint and tolerance are incorporated into the reanalysis. The reduction in initial feedwater temperature provides more conservative results for events 4 and 5. Therefore, it is conservative to include this change in the reanalysis supporting the PSV setpoint and associated tolerance change.

NRC RAI No. 2

The use of PSV setpoints with inclusion of negative tolerances lowers effective PSV opening pressures, which would cause an earlier opening of the PSVs and a lower increase in the RCS [reactor coolant system] pressure during overpressurization events. Lower RCS pressures result in lower departure from nucleate boiling ratios (DNBRs). As documented in the [U]FSAR [updated final safety analysis report], the loss of reactor coolant and reactor coolant pump locked rotor are limiting DNBR cases. Discuss for both events whether the calculated DNBRs in the [U]FSAR would be affected by a lower PSV opening pressure or not.

Response to a Request for Additional Information (RAI) Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

Response to RAI No. 2

As noted in RAI No. 2 above, the loss of forced reactor coolant flow and reactor coolant pump shaft seizure (i.e., locked rotor) events are the limiting events for DNBR. The DNBR calculations for these events conservatively do not credit the increase in RCS pressure consistent with the discussion provided in UFSAR Section 15.3, "Decrease in Reactor Coolant System Flow Rate." The initial RCS pressure at the start of these events is used in these calculations. Therefore, the proposed change in PSV setpoint and tolerance has no impact on the DNBR cases for these events.

NRC RAI No. 3

The information discussed on pages 11 and 12 of Reference 1 for a qualitative evaluation indicated that the spurious SI event would have similar results from the LOAC with the RCP seal injection event in terms of the change in the number of PSV water cycles and PSV discharge water temperature. The information is not sufficient for the staff to determine the accuracy of the results of the qualitative evaluation. Perform a quantitative analysis using the approved methods and provide the results to show the accuracy of the qualitative evaluation results.

Response to RAI No. 3

Spurious Safety Injection (SI) Evaluation History

The currently approved evaluation addressing a spurious SI at power event was performed as part of the Byron and Braidwood Stations power uprate initiative. This evaluation is documented in References 2 and 3. In Reference 4, the NRC approved the subject evaluation. Based on the documented evaluation, the NRC stated that "...the staff finds that the PORVs, block valves, and associated discharge piping and supports are qualified for the spurious SI event fluid conditions." The NRC further stated that, "...the staff finds the licensee's crediting of the PSVs to discharge liquid water during the spurious SI event to be acceptable."

As noted in Reference 3, the lowest water relief temperature addressed in the Electric Power Research Institute (EPRI) test of Crosby safety valves was 530 °F. This temperature bounded the lowest PSV water relief temperature expected for the assumed duration (i.e., 20 minutes) of the spurious SI at power event at Byron Station and Braidwood Station (i.e., 590 °F).

Comparative Evaluation

The spurious SI at power event was evaluated using engineering judgment and performing a comparative evaluation based on the LOAC with RCP seal injection event analysis performed in support of this license amendment request. Both the spurious SI at power event and the LOAC with RCP seal injection event involve RCS inventory addition. Based on the results obtained for the LOAC with RCP seal injection event analysis, it is concluded that the key results for the existing spurious SI at power evaluation, relative to PSV functionality (i.e., pressurizer water temperature, number of steam and water relief cycles and pressurizer fill time), would also not be significantly affected by the PSV lift setting change.

In the LOAC with RCP seal injection scenario, continued injection of water into the RCS through the RCP seals during the LOAC results in a filled pressurizer and water relief through the PSVs. A similar phenomenon occurs in the spurious SI at power event, where continued injection of SI water results in a filled pressurizer and water relief through the PSVs. The proposed reduction in PSV setpoint and increased tolerance will result in an earlier opening of the PSV in both the LOAC with RCP seal injection analysis and the spurious SI at power evaluation. Given that the PSVs open earlier in each transient due to the lower setpoint, a greater number of PSV water cycles could result for each transient.

The results of the LOAC with RCP seal injection event analysis at the lower PSV lift setting showed an increase of no more than one PSV water cycle, with each cycle being approximately 150 seconds in duration. The RCS water discharge temperature at the end of the transient showed a decrease of 0.5 °F. Since the proposed PSV setpoint is lower than the existing setpoint, the PSV would open 150 seconds earlier in the LOAC with RCP seal injection event as the termination time of the transient is fixed at one hour by operator action.

The water injection rate to the RCS assumed in the spurious SI at power event is approximately eight times greater than the water injection rate assumed in the LOAC with RCP seal injection event.

As described in Reference 3, the existing evaluation assumes that the spurious SI at power event is 20 minutes in duration, terminated by operator action. EGC performed a confirmatory calculation using the LOFTRAN computer code. The only difference between the confirmatory calculation and the Reference 3 evaluation is the change in PSV setpoint and tolerance. The LOFTRAN calculation for this case showed a PSV cycle time of approximately 30 seconds. Note that this 30 second cycle time includes the time to complete a full valve cycle from start of the PSV opening to the start of the next PSV opening, (e.g., the PSV starts to open at time=0, remains open for approximately five seconds, closes and remains closed for 25 seconds, then starts to open again at the beginning of the next 30 second cycle). This behavior as predicted by LOFTRAN is consistent with the evaluation described in Reference 3 that stated, "...a PSV will cycle a number of times (i.e., approximately 20) with a duration of 5-8 seconds per cycle." In both evaluations, the PSV is physically passing water for approximately 5-8 seconds.

The time it takes to reach the PSV lift setpoint is inversely proportional to the water injection flow rate; therefore, the time to reach the proposed lower PSV setpoint during the spurious SI at power event would conservatively be approximately 20 seconds earlier in the event (i.e., one-eighth the time of the LOAC with RCP seal injection event PSV cycle time of 150 seconds; specifically 150 seconds/8 = 18.75 seconds). Since the LOFTRAN calculation indicates that a PSV cycles approximately once every 30 seconds during the spurious SI at power event, lifting the PSV 20 seconds earlier in the event would suggest no more than one additional PSV cycle as the event is terminated by operator action in 20 minutes. The actual results of the LOFTRAN calculation indicated an increase of two PSV cycles.

As noted above, the RCS water temperature reduction upon termination of the LOAC with RCP seal injection event was approximately 0.5 °F. This temperature reduction is attributed to the additional heat removed by the one additional PSV cycle relieving RCS water. Since the water injection rate to the RCS assumed in the spurious SI at power event is approximately eight times greater than the water injection rate assumed in the LOAC with RCP seal injection event, the one or two additional PSV cycles during the spurious SI at power event would remove eight times the water per valve cycle and proportionately eight times the heat per valve cycle as the LOAC with RCP seal injection event. Therefore, the temperature reduction of the RCS water upon termination of the SI at power event would conservatively be approximately 8.0 °F [i.e., 8(0.5 °F/valve cycle)(2 valve cycles) = 8.0 °F]. The actual results of the LOFTRAN evaluation indicated a water temperature reduction of approximately 3.0 °F. As noted in Reference 3, the water relief temperature assumed in the approved spurious SI at power event evaluation is 590 °F; therefore, the conservative 8.0 °F reduction in water temperature remains bounded by the EPRI analysis that assumed a PSV water relief temperature of 530 °F.

Based on this confirmatory calculation, it is concluded that the results of the spurious SI at power event, considering the proposed PSV lift setting and increased tolerance, are similar to the results of the existing spurious SI evaluation relative to pressurizer water temperature, number of PSV steam and water relief cycles, and pressurizer fill time. Therefore, the spurious SI transient does not progress into a higher condition transient (i.e., a Condition III loss of coolant accident) consistent with the conclusion of the existing evaluation.

NRC RAI No. 4

Specify the pressure measurement uncertainties associated with the high pressure reactor trip and the PSV, and confirm that they are appropriately considered in the error analysis such that a reactor trip will occur prior to PSV actuation.

<u>Response to RAI No. 4</u>

The Pressurizer Pressure – High reactor trip function ensures that protection is provided against overpressurizing the RCS. This trip function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions. The Pressurizer Pressure - High limiting safety system setting is selected to be below the PSV actuation pressure and above the pressurizer power operated relief valve (PORV) setting. This minimizes challenges to the PSVs while avoiding unnecessary reactor trips for those pressure increases that can be controlled by the PORVs.

Figure 1 illustrates the probabilities associated with the actuation of the setpoints for the pressurizer PORVs, Pressurizer Pressure – High reactor trip, and PSVs. The probability of having a PSV (i.e., with the new setpoint of 2460 psig) lift before achieving a Pressurizer Pressure – High reactor trip signal (i.e., with a setpoint of 2385 psig, 2 out of 4 logic) is less than 1% for any given pressure. This determination is based on the PSV having an acceptable as found tolerance of ±2% of nominal setpoint value which equates to ±49.2 psig, and the Pressurizer Pressure – High reactor trip uncertainty of ±5.25% of the 800 psig instrument span (i.e., a span of 1700 psig – 2500 psig) for two channels which equates to ±42.0 psig. The uncertainty associated with the Pressurizer Pressure – High reactor trip was calculated using the methodology described in Westinghouse WCAP-12523, "Bases Document for Westinghouse Setpoint Methodology for Protection Systems - Commonwealth Edison Company Zion/Byron/Braidwood Units," dated October 1990. The probability of having a PSV lift before achieving a Pressurizer Pressure – High reactor trip was determined based on the assumption that each function (i.e., PSV actuation and Pressurizer Pressure – High reactor trip) has uncertainties (i.e., errors) that follow a normal distribution and behave according to the fundamental rules of probability. Based on the above, challenges to the safety valves are minimized and the reactor trip is expected to occur prior to the PSV actuation.

Note that in support of this license amendment request, the Overpressure Protection Report will be revised, as appropriate, to document the change in PSV setpoint and adequacy of the existing analyses demonstrating overpressure protection. The results of the existing analyses were determined to remain applicable since the proposed change in maximum PSV opening pressure is insignificant and in the conservative direction (i.e., 2509 psig compared to 2510 psig). Note that the UFSAR Chapter 15 Loss of Load/Turbine Trip peak pressure case is the limiting transient for overpressure protection and used to support the Overpressure Protection Report. The results of this existing analysis continue to demonstrate that the overpressure protection provided by the PSVs, with the proposed setpoint and tolerance changes, is sufficient to maintain both the peak RCS pressure and main steam system pressure below the American Society of Mechanical Engineers (ASME) Code limit of 110% of the respective system design pressures. Note that the analyses performed in support of the Overpressure Protection Report conservatively do not credit pressurizer PORVs or a reactor trip as a result of a turbine trip. The reactor trip occurs on a Pressurizer Pressure – High signal and the analyses appropriately account for instrument uncertainty.

NRC RAI No. 5

The information discussed on page 10 of Reference 1 for a loss of power to plant auxiliaries with RCP seal injection references an assessment by the Westinghouse Systems and Equipment Engineering Group that concluded that the PSVs will remain operable following a LOAC event with water relief through the PSVs. Please provide a copy of the Westinghouse assessment for review.

ATTACHMENT 1

Response to a Request for Additional Information (RAI) Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

Response to RAI No. 5

A copy of the assessment performed by the Westinghouse Systems & Equipment Engineering Group, "Evaluation of the Operability of Crosby PSV During Loss of Offsite Power with RCP Seal Injection," dated March 7, 2003, is provided in Attachment 2.

References

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

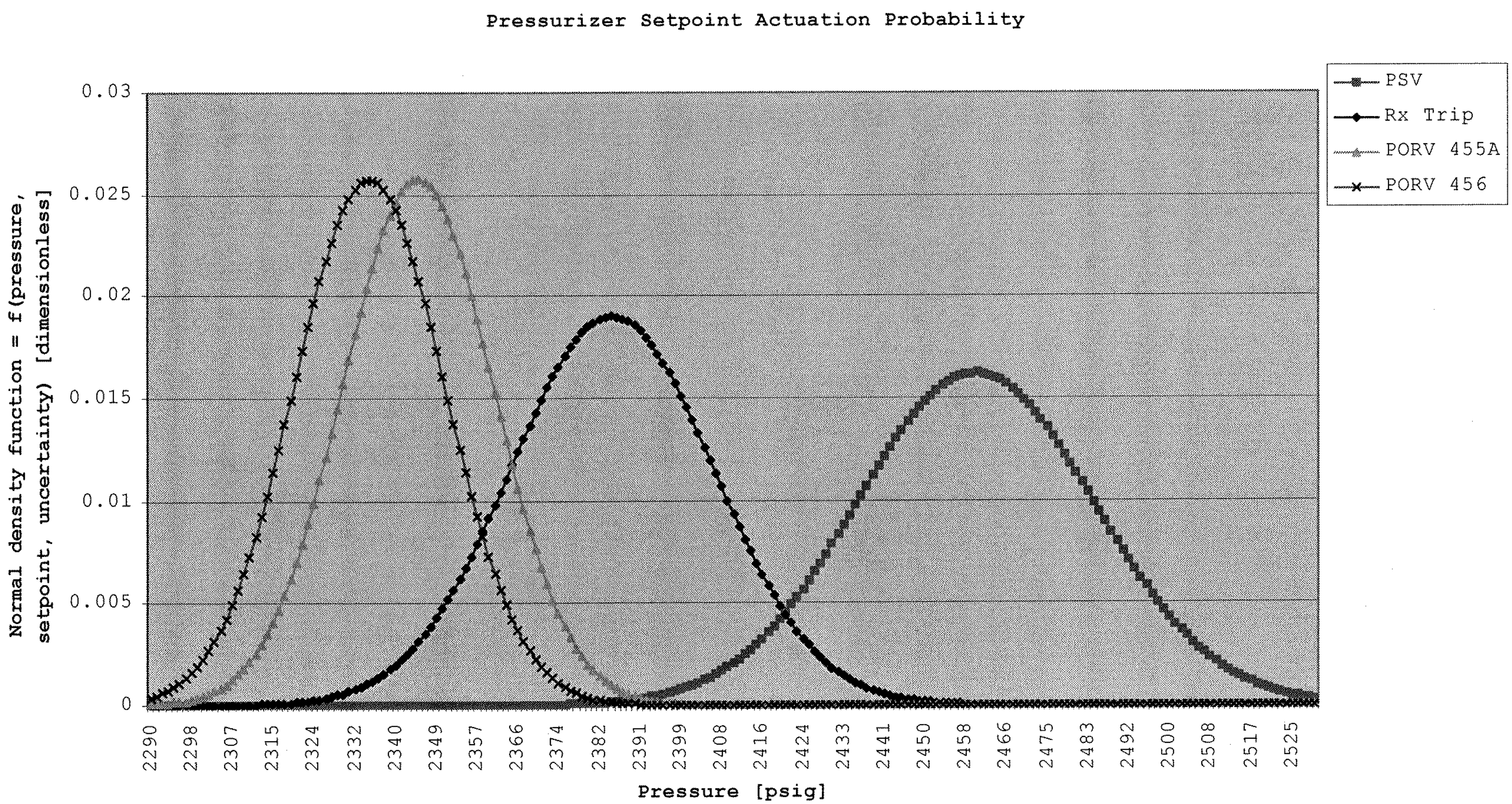
2. Letter from R. M. Krich (Commonwealth Edison Company), to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Station," dated July 5, 2000

3. Letter from R. M. Krich (Exelon Generation Company, LLC) to the NRC, "Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations," dated January 31, 2001

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

Response to a Request for Additional Information (RAI) Regarding a License Amendment Request to Revise the Pressurizer Safety Valves Lift Settings

Figure 1



ATTACHMENT 2

Westinghouse Systems & Equipment Engineering Group

"Evaluation of the Operability of Crosby PSV During Loss of Offsite Power with RCP Seal Injection"

March 7, 2003

Westinghouse

To:	S. V. Andre'	Date:	3/7/2003
cc:	E. M. Monahan J. Galando		
From:	Systems & Equipment Engineering	Your ref:	See Below
Ext:	6643	Our ref:	LTR-SEE-03-46
Fax:	3451		

Subject: **Evaluation of The Operability of Crosby PSV During Loss of Offsite Power with RCP Seal Injection**

References:

1. LTR-TA-03-50
2. LTR-SEE-01-287
3. G-678838 Specification Sheet 2

Reference 1 performed a Loss of Offsite Power (LOOP) analysis assuming a net mass addition to the RCS via RCP seal injection for Byron Braidwood Units. The analysis showed that previous evaluation of the PSVs covered in Reference 2 bounds the analyzed conditions except for the revised setpoint and tolerance of Reference 3. Further, Reference 1 requested SEE to assess the operability of the PSVs based on the analyzed LOOP conditions.

SEE review of References 1 and 2 confirmed that the limiting PSV relieving conditions are governed by the results of D5 – Case 3 with Revised PSV Set Point and Tolerance, which are:

- PSV Opening Pressure (psia) = 2413.2
- Minimum Water Relief Temperature (^{o}F) = 633.5
- Total Steam relief Cycles = 14
- Total Water Relief Cycles = 12

These conditions differ slightly from the data used to qualify PSV operability in Reference 2 in the following respects:

1. The PSV Opening Pressure (psia) for the D5-Case 3 (Reference 1) is 36.8 psi lower than the PSV Opening Pressure (psia) of Reference 2. The discharge flow of subcooled water is directly proportional to the inlet pressure. Thus, reducing the inlet pressure by 36.8 psi will conservatively reduce the flowrate by 1.5%. However, in Reference 2 it was shown that the assumed 158 lbm/sec flowrate of Byron / Braidwood PSVs is bounded by the expected minimum flowrate (198 lbm/sec) of the PSVs under similar postulated relieving conditions. Therefore, applying the 1.5% reduction factor on the expected minimum flowrate of 198 lbm/sec results in a flowrate of 195 lbm/sec at the reduced PSV Opening Pressure. This adjusted flowrate still bounds the assumed 158-lbm/sec of Reference 1. Thus, the reduction

in PSV Opening Pressure of 36.8 psi has no impact on the relief capacity of 158 lbm/sec assumed in Reference 1.

2. Also, the Minimum Water Relief Temperature (°F) for the D5-Case 3 (Reference 1) is 0.3 °F lower than the Minimum Water Relief Temperature (°F) of Reference 2. A delta of 0.3 °F in the accepted water temperature of 633.7 °F has no impact on the operability of the PSV.

Based on the foregoing, it is concluded that the Byron / Braidwood PSVs will be operable during the Loss of Offsite Power with RCP Seal Injection event.

If there are any questions, contact the undersigned.

L. I. Ezekoye
Systems & Equipment Engineering

Verified: T. J. Matty
Systems & Equipment Engineering

Approved: J. C. Bass, Manager
Systems & Equipment Engineering