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U. S. Nuclear Regulatory Commission
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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

References:

(1) 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

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

In Reference 1, 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 questions were raised regarding the analyses supporting the revision of the PSV lift setting and the NRC requested that EGC provide additional information to clarify these issues. This information was provided in Reference 2.

One of the responses (i.e., the response to Question No. 4) provided in Reference 2 addressed the following NRC request:

> "*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.*"

Our evaluation of this issue identified that the probability of having a PSV lift (i.e., with the new setpoint of 2460 psig) before achieving a pressurizer pressure – high reactor trip signal (i.e., with a setpoint of 2385 psig) is less than 1% for any given pressure. Based on this information, the NRC requested that EGC evaluate this potential event to ensure that all accident analyses criteria remain satisfied. Our deterministic evaluation of this issue, documented in Reference 3, confirmed that all applicable accident analysis acceptance criteria remain satisfied.

During the NRC's review of Reference 3, teleconference calls were held between members of the NRC and EGC on March 9, 2004 and March 18, 2004, to discuss certain analysis assumptions documented in Reference 3. At the conclusion of the teleconference calls, the NRC requested that EGC formally document resolution of the issues discussed. This information is provided in Attachment 1 to this letter.

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 _____ _____

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 1

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

Introduction

In Reference 1, 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 (RCS).

During the NRC's review of the proposed change, a number of questions were raised regarding the analyses supporting the revision of the PSV lift setting and the NRC requested that EGC provide additional information to clarify these issues. This information was provided in Reference 2.

One of the responses (i.e., the response to Question No. 4) provided in Reference 2 addressed the following NRC request:

> "*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.*"

Our evaluation of this issue identified that the probability of having a PSV lift (i.e., with the new setpoint of 2460 psig) before achieving a pressurizer pressure – high reactor trip signal (i.e., with a setpoint of 2385 psig) is less than 1% for any given pressure. Based on this information, the NRC requested that EGC evaluate this potential event to ensure that all accident analyses criteria remain satisfied. Our deterministic evaluation of this issue, documented in Reference 3, confirmed that all applicable accident analysis acceptance criteria remain satisfied.

During the NRC's review of Reference 3, teleconference calls were held between members of the NRC and EGC on March 9, 2004 and March 18, 2004, to discuss certain analysis assumptions documented in Reference 3. At the conclusion of the teleconference calls, the NRC requested that EGC formally document resolution of the issues discussed. Specifically, the following issues/assumptions were discussed and are addressed in detail below:

- computer code used in the analysis of record (AOR) for the loss of load/turbine trip (LOL/TT) and rod withdrawal at power (RWAP) overpressure events;
- PSV loop seal purge delay assumptions;
- alternate reactor trip signal assumptions and associated environmental qualifications;
- length of time PSVs remain open following a LOL/TT or RWAP event assuming a zero second PSV loop seal purge delay; and

- confirm that the PSVs do not relieve water and reseat after a LOL/TT or RWAP event.

Evaluation

Computer Codes Used in the AOR LOL/TT and RWAP Events

The computer code used in the AOR for the LOL/TT and RWAP overpressure events is LOFTRAN, as described in Byron/Braidwood Updated Final Safety Analysis Report (UFSAR) Sections 15.2.3, “Turbine Trip,” and 15.4.2, “Uncontrolled Rod Cluster Assembly Bank Withdrawal at Power.”

PSV Loop Seal Purge Delay Assumptions

As discussed in Reference 3, the AOR for the LOL/TT and RWAP overpressure events assumes a one second delay in steam relief through the PSVs to account for PSV loop seal purge. This is a conservative assumption as it maximizes the pressure transient. The loop seal purge delay of one second was confirmed to be a reasonable assumption based on the physical design of the system piping upstream of the PSVs. The PSV loop seals are depicted on piping and instrument diagrams, M-60 Sheet 5 (Unit 1) and M-135 Sheet 5 (Unit 2) and are documented on the corresponding piping isometric drawings. In addition, there are a number of references to the PSV loop seals in the UFSAR. Specifically, UFSAR Section 5.4.13, “Safety and Relief Valves,” states,

> "The pressurizer safety valves are of the pop type. The valves are spring loaded, open by the direct fluid pressure action, and are designed with backpressure compensation features. The 6-inch pipe connecting the pressurizer nozzles to their respective code safety valves, are shaped in the form of a loop seal. Condensate resulting from normal heat losses accumulates in the loop. The water prevents any leakage of hydrogen gas or steam through the safety valve seats. If the pressurizer pressure exceeds the set pressure of the safety valves, they start lifting, and the water from the seal discharges during the accumulation period."

UFSAR Section 15.3.3, “Reactor Coolant Pump Seizure (Locked Rotor),” also states,

> "Upon actuation of the pressurizer safety valves at an opening pressure of 2549.9 psia (including 1% allowance for drift and 1% for pressure shift), purge of the water in the safety valve loop seal occurs and full valve relief capacity is achieved within 1 second."

Given the proposed lower PSV setpoint and assumed maximum negative uncertainty in the PSV response and the maximum positive uncertainty in the high pressurizer pressure reactor trip signal response, the high pressurizer pressure reactor trip signal will still occur prior to PSV relief if the one second delay for PSV loop seal clearance is assumed as described in Reference 3. If however, no delay in PSV pressure relief is assumed, the PSVs will lift and relieve RCS pressure prior to reaching the high

pressurizer pressure reactor trip. In this case, the overpressure transient is mitigated as described below.

Alternate Reactor Trip Signals

The above overpressure scenario (i.e., with maximum negative tolerance for the PSVs and no PSV loop seal purge delay) is non-limiting with respect to RCS pressure concerns and therefore was not considered in the AOR for the LOL/TT and RWAP overpressure events. When modeling the negative tolerance for the PSVs with no PSV loop seal purge delay, all acceptance criteria remain satisfied; in particular, the peak RCS pressure and steam generator secondary side pressure remain less than the allowable limits and the pressurizer does not reach an overfill condition. If the PSVs relieve prior to receiving the high pressurizer pressure reactor trip signal, the reactor will trip on overtemperature delta-T (i.e., OTDT), in the LOL/TT case, and on high neutron flux, in the RWAP case, only a short time after it would have tripped on high pressurizer pressure (i.e., rod motion will start approximately 7.6 seconds later for the LOL/TT event and approximately 0.3 seconds later for the RWAP event). This response was validated in a confirmatory analysis using the LOFTRAN computer code with the high pressurizer pressure reactor trip disabled for the LOL/TT and RWAP overpressure events.

OTDT and High Neutron Flux Reactor Trip Instrumentation Environmental Qualifications

Given the above scenario, since the PSVs relieve prior to receiving a reactor trip, a concern was expressed regarding the environmental qualification of the OTDT reactor protection instrumentation. The PSVs relieve to the pressurizer relief tank (PRT) which has a rupture disc that may rupture after a significant PSV relief. The PRT rupture disc relieves directly to the containment atmosphere. It was postulated that the adverse environment potentially created by the PRT rupture disc relieving to the containment may have an adverse effect on the OTDT instrumentation located in the containment.

The confirmatory analysis, noted above, indicated that the pressurizer volume increased above the pressurizer volume currently predicted in the AOR overpressure transients that assumes a reactor trip on high pressurizer pressure. In the proposed scenario with no high pressurizer pressure reactor trip, the pressurizer does not reach an overfill condition and no PSV water relief is expected; therefore, reclosure of the PSVs is not a concern. In this scenario, the PSVs are open only a short time (i.e., approximately 15 seconds as noted below); however, if the PRT rupture disc were to relieve, there would be no effect on the operability of the OTDT reactor trip instrumentation as this instrumentation is environmentally qualified. Similarly, in the RWAP event, the PSVs are open approximately seven seconds and the high neutron flux reactor trip instrumentation is also environmentally qualified should the PRT rupture disc relieve. The environmental qualification of this instrumentation is documented in UFSAR Section 3.11, "Environmental Design of Mechanical and Electrical Equipment," which states, "The mechanical, instrumentation, and electrical portions of the engineered safety features and the reactor protection system are designed to ensure acceptable performance in all environments anticipated under normal, test, and design-basis accident conditions."

ATTACHMENT 1

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

Length of Time PSVs Remain Open Following a LOL/TT or RWAP Event

As noted above, the confirmatory analysis, assuming maximum negative PSV tolerance and no PSV loop seal purge delay, indicated that the PSVs would lift followed by an OTDT reactor trip signal within approximately 7.6 seconds for the LOL/TT event (followed by a high neutron flux reactor trip signal within approximately 0.3 seconds for the RWAP event) after the pressurizer high pressure trip would have occurred as modeled in the current AOR. The PSVs only remain open a short time after the reactor trip occurs, (i.e., for approximately 15 seconds total for the LOL/TT event and seven seconds total for the RWAP event).

Confirm the PSVs Do Not Relieve Water and Reseat After LOL/TT or RWAP Event

As noted above, the confirmatory analysis indicated that, although the pressurizer volume increases above the AOR which assumes a reactor trip on high pressurizer pressure, the pressurizer does not reach an overfill condition and no PSV water relief, subsequent to the initial purge of the loop seal, is expected. Therefore, there is no concern regarding the PSVs reseating due to passing water as the PSVs will only experience approximately 15 seconds of steam relief for the LOL/TT event and seven seconds of steam relief for the RWAP event.

Conclusion

The impact of a PSV lifting prior to reaching the high pressurizer pressure reactor trip setpoint has been evaluated for the peak RCS pressure events (i.e., LOL/TT and RWAP). Based on the re-analyses performed to support the proposed PSV setpoint and tolerance change discussed in References 1 and 2, the evaluation of the current AOR for the LOL/TT peak pressure case and the RWAP peak pressure case documented in Reference 3 and the above information, it is concluded that all acceptance criteria for these events continue to be met.

References

1. 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

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