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10 CFR 50.90

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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 a License Amendment to Revise the Pressurizer Safety Valves Lift Settings

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an 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 valve (PSV) lift settings from "≥ 2460 psig and ≤ 2510 psig," to "≥ 2411 psig and ≤ 2509 psig." The existing TS represents a ±1% tolerance band around a lift setting of 2485 psig. The proposed lift setting range of "≥ 2411 psig and ≤ 2509 psig" represents a ±2% tolerance band around a lift setting of 2460 psig.

The current TS Limiting Condition for Operation (LCO) 3.4.10 requires that three PSVs shall be operable with lift settings ≥ 2460 psig and ≤ 2510 psig. As noted above, these values represent a ±1% tolerance around a lift setpoint of 2485 psig. The ±1% setpoint tolerance for "as-found" and "as-left" values, as currently required by TS Surveillance Requirement (SR) 3.4.10.1, is the original nuclear industry TS standard for pressurizer safety valve lift settings. This standard was prescribed by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components." Industry experience has shown that normally functioning PSVs periodically exceed the ±1% tolerance after a cycle of operation; therefore, other licensees have revised their TS PSV as-found tolerance limits to ±3%, consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2, dated April 2001, and ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The specified as-left tolerance is ±1% to minimize the probability of setpoint drift going beyond ±2% during the fuel cycle.

Braidwood and Byron Stations have PSVs manufactured by Crosby. There have been numerous instances where the PSV lift settings have fallen outside the ±1% tolerance limits during end of cycle inservice testing. These as-found lift settings have prompted the generation of licensee event reports (LERs) in accordance with 10 CFR 50.73(a)(2)(i)(b), "Any operation or condition prohibited by the plant's Technical Specifications..." Most of the as-found lift settings have not exceeded ±2% of the pressure setpoint. Subsequent evaluations have shown that there are no material condition concerns with these valves and they are performing within their design capabilities. Therefore, generating a LER for a PSV that is performing satisfactorily within its design capability becomes an unnecessary burden for both the licensee and the NRC.

The current associated TS SR 3.4.10.1 states, "Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%." This SR will remain unchanged and the Inservice Testing Program supporting documentation will be revised to reflect the new PSV lift settings.

A reanalysis/evaluation was performed to assess all transients that are potentially impacted by the PSV lift setting and tolerance band change. These reanalyses and evaluations concluded that all of the subject transients continue to have acceptable results.

The attached amendment request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes and contains the following sections:

1.0 Description

2.0 Proposed Change

3.0 Background

4.0 Technical Analysis

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

This section describes our evaluation performed using the criteria in 10 CFR 50.91(a), "Notice for public comment," paragraph (1), which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92, "Issuance of amendment," paragraph (c).

5.2 Applicable Regulatory Requirements/Criteria

6.0 Environmental Consideration

This section provides information supporting an environmental assessment. We have determined that the proposed changes meet the criteria for a categorical exclusion set forth in paragraph (c)(10) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or

otherwise not requiring environmental review."

7.0 References

Attachments 2-A and 2-B include the marked-up TS pages with the proposed changes indicated for Braidwood Station and Byron Station, respectively.

Attachments 3-A and 3-B include the associated typed TS pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively. Also, included in Attachments 3-A and 3-B are the associated typed TS Bases pages for Braidwood Station and Byron Station with the proposed changes incorporated. The Bases pages are included for information only.

We request approval of the proposed amendment by March 1, 2004, in advance of the Spring 2004 Byron Station Unit 2 refueling outage. Once approved, we request that the amendment be implemented upon startup of Byron Station Unit 2 Cycle 12 in Spring 2004, Braidwood Station Unit 1 Cycle 12 in Fall 2004, Byron Station Unit 1 Cycle 14 in Spring 2005, and Braidwood Station Unit 2 Cycle 12 in Spring 2005.

The NRC has previously approved a similar change for the Wolf Creek Generating Station in Amendment No. 133, issued March 23, 2000.

The proposed amendment has been reviewed by the Braidwood Station and the Byron Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

EGC is notifying the State of Illinois of this application for a change to the TS by sending a copy of this letter and its attachments to the designated State Official.

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

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

Executed on 6-27-03

Kenneth A. Ainger

Kenneth A. Ainger
Manager - Licensing
Midwest Regional Operating Group

Attachments:
Attachment 1: Evaluation of Proposed Changes
Attachment 2-A: Markup of Proposed Technical Specifications Page Changes for Braidwood Station
Attachment 2-B: Markup of Proposed Technical Specifications Page Changes for Byron Station
Attachment 3-A: Typed Pages for Technical Specification Changes and Bases Changes (for information only) for Braidwood Station
Attachment 3-B: Typed Pages for Technical Specification Changes and Bases Changes (for information only) for Byron Station

ATTACHMENT 1
Evaluation of Proposed Changes

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1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an 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 reducing the existing pressurizer safety valve (PSV) lift settings from "≥ 2460 psig and ≤ 2510 psig," to "≥ 2411 psig and ≤ 2509 psig."

2.0 PROPOSED CHANGE

The current TS Limiting Condition for Operation (LCO) 3.4.10 states the following:

> "Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig."

The proposed amendment would revise this LCO to state:

> "Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2411 psig and ≤ 2509 psig."

We request approval of the proposed amendment by March 1, 2004, in advance of the Spring 2004 Byron Station, Unit 2 refueling outage. Once approved, we request that the amendment be implemented upon startup of Byron Station, Unit 2 Cycle 12 in Spring 2004, Braidwood Station, Unit 1 Cycle 12 in Fall 2004, Byron Station, Unit 1 Cycle 14 in Spring 2005, and Braidwood Station, Unit 2 Cycle 12 in Spring 2005.

The current associated TS Surveillance Requirement (SR) 3.4.10.1 states:

> "Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%."

This SR will remain unchanged and the Inservice Testing Program supporting documentation will be revised to reflect the new PSV lift settings.

The current TS represents a ±1% tolerance band around a lift setting of 2485 psig. The proposed lift setting range of "≥ 2411 psig and ≤ 2509 psig" represents a ±2% tolerance band around a lift setting of 2460 psig. The ±1% setpoint tolerance for "as-found" and "as-left" values, as currently required by TS SR 3.4.10.1, is the original nuclear industry TS standard for pressurizer safety valve lift settings. This standard was prescribed by the American Society of Mechanical Engineers (ASME), Section III, "Rules for Construction of Nuclear Facility Components." Industry experience has shown that normally functioning PSVs periodically exceed the ±1% tolerance after a cycle of operation; therefore, other licensees have revised their TS PSV as-found tolerance limits to ±3%, consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2, dated April 2001, and ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

Braidwood and Byron Stations have PSVs manufactured by Crosby. There have been numerous instances where the PSV lift settings have fallen outside the ±1% tolerance limits during end of cycle inservice testing. These as-found lift settings have prompted the generation of licensee event reports (LERs) in accordance with 10 CFR 50.73(a)(2)(i)(b), "Any operation or condition prohibited by the plant's Technical Specifications..." Most of the as-found lift settings have not exceeded ±2% of the pressure setpoint. Subsequent evaluations have shown that there are no material condition concerns with these valves and they are performing within their design capabilities. Therefore, generating a LER for a PSV that is performing satisfactorily within its design capability becomes an unnecessary burden for both the licensee and the NRC.

For this reason, EGC is proposing to revise the PSV lift settings to 2460 psig ±2% (i.e., ≥ 2411 psig and ≤ 2509 psig). This lift setting reduction is being requested to obtain a ±2% tolerance about the PSV lift setpoint while having a minimum impact on the existing analysis of record and maintaining sufficient margin between the PSV minimum lift setpoint and the lift setpoint of the pressurizer power operated relief valves (PORVs).

The proposed amendment is reflected on marked-up copies of the affected TS pages for Braidwood and Byron Stations in Attachments 2-A and 2-B, respectively. The typed TS pages, with the changes incorporated, are provided in Attachments 3-A and 3-B along with the associated revised TS Bases pages (for information only). Following NRC approval of this request, the Braidwood Station and Byron Station TS Bases will be revised in accordance with the TS Bases Control Program described in TS Section 5.5.14, "Technical Specifications (TS) Bases Control Program."

3.0 BACKGROUND

System Description

The pressurizer safety valves provide, in conjunction with the reactor protection system, overpressure protection for the reactor coolant system (RCS) as described in the Byron/Braidwood Updated Final Safety Analysis Report (UFSAR), Section 5.4.13, "Safety and Relief Valves," (i.e., Reference 1). The three pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the RCS safety limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which defines the minimum relief capacity for the safety valves. The discharge flow from the PSVs is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the PSVs or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in Modes 1, 2, 3, 4, and 5; however, in Modes 4 and 5, and in Mode 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The current upper and lower pressure limits are based on the ±1% tolerance requirement (see Reference 2) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with Modes 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The PSVs are part of the primary success path and mitigate the effects of postulated accidents. Operability of the PSVs ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit (see Reference 2) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

<u>Safety Analysis</u>

All accident and safety analyses in the UFSAR Chapter 15, "Accident Analyses," (Reference 3) that require safety valve actuation assume operation of three PSVs to limit increases in RCS pressure. The overpressure protection analysis (see Reference 4) is also based on operation of three PSVs. Accidents that could result in overpressurization if not properly terminated include:

a. uncontrolled rod withdrawal from full power;
b. loss of reactor coolant flow;
c. loss of external electrical load;
d. loss of normal feedwater;
e. loss of all AC power to station auxiliaries;
f. reactor coolant pump locked rotor; and
g. feedwater line break.

Detailed analyses of the above transients are contained in Chapter 15 of the UFSAR. PSV actuation is required in events a-f (above) to limit the pressure increase.

The three PSVs are currently set to open at the RCS design pressure of 2500 psia, within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (see Reference 2) for lifting pressures above 1000 psig. The limit protected by this TS is the reactor coolant pressure boundary SL of 110% of design pressure. Inoperability of one or more PSVs could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

An analysis of a loss of AC power to the plant auxiliaries (LOAC) with reactor coolant pump (RCP) seal injection event has been performed. This event involves RCS water inventory addition resulting in pressurizer overfill. The results of these analyses confirm that the PSVs remain operable following water relief.

The spurious safety injection (SI) at power event was evaluated 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 spurious SI at power event, relative to PSV functionality, would also not be significantly affected by the PSV lift setting change. This conclusion is discussed in detail in the spurious SI event discussion on

page 11 of 19 below. Thus, the conclusion of the current analysis, that the spurious SI at power event does not progress into a higher condition transient (i.e., a Condition III event), remains valid.

The requirement of SR 3.4.10.1 is specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Reference 5), which provides the activities and frequencies necessary to satisfy the SR. The current PSV setpoint is ±1% of a nominal 2485 psig.

Need for Amendment

As previously noted, Braidwood and Byron Stations have PSVs manufactured by Crosby. There have been numerous instances where the PSV lift settings have fallen outside the ±1% tolerance limits during end of cycle inservice testing. These as-found lift settings have prompted the generation of licensee event reports (LERs) in accordance with 10 CFR 50.73(a)(2)(i)(b), "Any operation or condition prohibited by the plant's Technical Specifications..." Most of the as-found lift settings have not exceeded ±2% of the pressure setpoint. Subsequent evaluations have shown that there are no material condition concerns with these valves and they are performing within their design capabilities. Therefore, generating a LER for a PSV that is performing satisfactorily within its design capability becomes an unnecessary burden for both the licensee and the NRC.

For this reason, EGC is proposing to revise the PSV lift settings to 2460 psig ±2% (i.e., ≥ 2411 psig and ≤ 2509 psig). This lift setting reduction is being requested to obtain a ±2% tolerance about the PSV lift setpoint while having a minimum impact on the existing analysis of record and maintaining sufficient margin between the PSV minimum lift setpoint and the lift setpoint of the pressurizer PORVs. The current TS SR 3.4.10.1 specifies that, "Following testing, lift setting shall be within ±1%." This as left requirement will remain unchanged.

The NRC has previously approved a similar change for the Wolf Creek Generating Station in Amendment No. 133, issued March 23, 2000.

4.0 TECHNICAL ANALYSIS

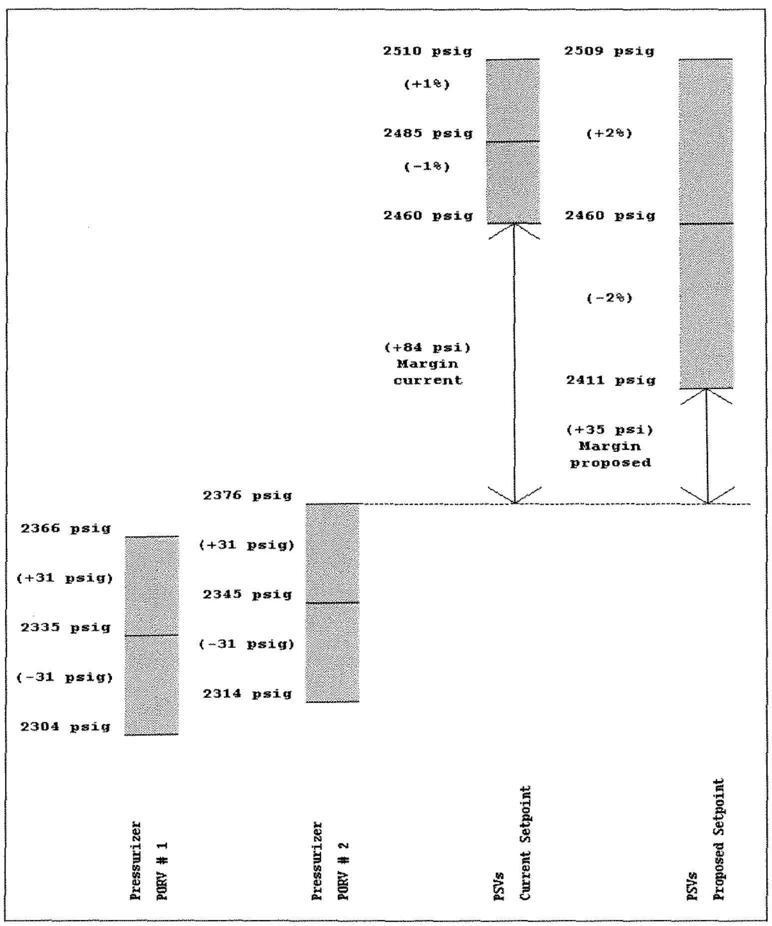
Reanalyses/evaluations were performed to assess all transients that could be potentially impacted by the PSV lift setting and tolerance band change. The proposed change in the PSV tolerance from ±1% to ±2% with a reduction in the lift setting from 2485 psig to 2460 psig allows a decrease in the valve minimum opening pressure and therefore provides earlier pressurizer relief and a reduced RCS pressure. Except for the PSV lower lift setting and increased tolerance, the design and operation of the PSVs remains unchanged. Note that the proposed change did not result in a change in the maximum opening pressure assumed in the accident analyses.

Adequacy of Margin Between Opening of Pressurizer PORVs and PSVs

The design function of the pressurizer PORVs is to limit pressurizer pressure to a value below the "Pressurizer Pressure – High" reactor trip setpoint following a step reduction of 50% of full load with steam dumps available. The pressurizer PORVs also serve to minimize the

challenges to the PSVs. The impact of the proposed change on the adequacy of margin between the opening of the pressurizer PORVs and the opening of the PSVs was evaluated. The pressurizer PORVs are designed to lift prior to the PSVs. At Braidwood, the two pressurizer PORVs have staggered setpoints of 2335 psig and 2345 psig. The calculated uncertainty associated with opening the pressurizer PORVs was determined to be ±31 psig based on a recent design change that removed the compensated pressure signal from one of the pressurizer PORVs. That is, currently at Braidwood, both pressurizer PORVs receive input from uncompensated pressure signals. Conservatively applying the pressurizer PORV instrument uncertainty (i.e., ±31 psig) to the nominal actuation setpoint of the pressurizer PORV with the highest setpoint (i.e., 2345 psig), results in an actuation band between 2314 and 2376 psig. When compared to the current PSV lift setpoint and associated setpoint tolerance (i.e., 2485 psig ±1% resulting in an actuation band of 2460 and 2510 psig), the margin between the highest possible actuation of a pressurizer PORV (i.e., 2376 psig) and the lowest possible actuation of a PSV (i.e. 2460 psig) is 84 psi. When compared to the proposed PSV lift setpoint and associated setpoint tolerance (i.e., 2460 psig ±2% resulting in an actuation band of 2411 and 2509 psig), the margin between the highest possible actuation of a pressurizer PORV (i.e., 2376 psig) and the lowest possible actuation of a PSV (i.e. 2411 psig) is 35 psi. Refer to Figure 1 below.

Figure 1



Although the proposed change results in a reduction in the margin between opening the pressurizer PORVs and opening the PSVs, the resulting margin is still sufficient to ensure that the PORVs will actuate prior to the PSVs.

Currently at Byron, the pressurizer PORVs both have an actuation setpoint of 2335 psig. However, one pressurizer PORV receives an uncompensated pressure signal, while the other pressurizer PORV receives a compensated pressure signal. Design changes to modify the design consistent with the discussion above are scheduled to be implemented prior to implementation of this license amendment request for each unit.

Instrument Measurement Uncertainty

An evaluation of the instrumentation used to conduct PSV testing required by TS SR 3.4.10.1 was performed to identify the associated instrument measurement uncertainty based on the current test method. The acceptance criteria in the associated test procedure will be appropriately revised to account for this instrument measurement uncertainty to assure that the as-found lift setting of the PSVs conservatively represents the actual PSV lift setpoint. Any changes in the current test method will be evaluated to ensure an appropriate accounting for the associated instrument measurement uncertainty. Similarly, the as-left lift setting will be revised to appropriately account for the calculated instrument measurement uncertainty.

Loss of Coolant Accidents Evaluation

A loss of coolant accident (LOCA) will result in a decrease in RCS pressure and therefore, will not challenge the PSV opening pressure lift setting. Consequently, UFSAR LOCA analyses do not model PSV actuation.

Steam Generator Tube Rupture Evaluation

The steam generator tube rupture (SGTR) event results in a decreasing RCS pressure and therefore, does not challenge the PSV opening pressure lift setting. Since actuation of the PSVs does not occur in the SGTR analysis, the proposed change has no effect on the SGTR analysis.

Non-LOCA Accidents Analyses/Evaluation

The proposed change does not affect the maximum opening pressure assumed in the non-LOCA analyses since the proposed change in maximum PSV opening pressure is insignificant and in the conservative direction (i.e., 2509 psig compared to 2510 psig). Therefore, only those transients for which it is conservative to minimize the RCS pressure, e.g., departure from nucleate boiling (DNB) related events and pressurizer overfill concerns, are potentially impacted by the proposed change.

For peak pressure cases, the current licensing basis analyses model the PSVs with positive tolerance. The assumed PSV lift pressure in these analyses bounds the proposed change. That is, the current licensing basis models the PSV opening at 2485 psig plus 1% (to account for tolerances), plus an additional 1%, (to account for a pressure shift due to water-filled pressurizer loop seals, (see Reference 6)), or 2534.7 psig, (i.e., 2485 psig + 0.01(2485 psig) + 0.01(2485 psig) = 2534.7 psig). Modeling the proposed PSV lift setting change would result in the PSV opening at 2460 psig plus 2% (to account for tolerances) plus an additional 1% (to

account for pressure shift) or 2533.8 psig, (i.e., 2460 psig + 0.02(2460 psig) + 0.01(2460 psig) = 2533.8 psig), which is bounded by the current analysis. Thus, the peak pressure cases are not impacted.

The affected transients, listed below, assume PSV opening with negative tolerance modeled; therefore, the following reanalyses/evaluations were performed:

- reanalysis of loss of load (LOL) / turbine trip (TT), i.e., DNB limiting cases with revised PSV setpoint.
- reanalysis of rod withdrawal at power (RWAP), i.e., DNB limiting cases with revised PSV setpoint.
- reanalysis of loss of normal feedwater (LONF) limiting cases with revised PSV setpoint.
- reanalysis of loss of alternating current (LOAC) limiting cases with revised PSV setpoint.
- reanalysis of LOAC with reactor coolant pump (RCP) seal injection with revised PSV setpoint.
- evaluation of the impact of the revised PSV setpoint on the feedline break (FLB) analysis.
- evaluation of the potential impact of revised PSV setpoint on spurious safety injection (SI) pressurizer overfill.

For the LOL /TT and RWAP departure from nucleate boiling ratio (DNBR) calculations, the negative tolerance modeling results in a lower PSV opening pressure, causing RCS pressure to be lowered which can impact DNBR results. For the LONF, LOAC, LOAC with RCP seal injection, and spurious SI at power events, current analyses are performed to address a pressurizer water solid condition. PSV opening with negative tolerance is modeled in these analyses to decrease the time it takes to fill the pressurizer. For the FLB event, RCS pressure is minimized to minimize the saturation pressure. A lower PSV opening pressure can potentially further reduce the RCS pressure and hence saturation pressure.

In summary, these reanalyses/evaluations concluded that all of the subject transients continue to have acceptable results. A brief description of the results for each transient is presented below.

Loss of Load / Turbine Trip

Reanalyses of the LOL / TT DNB case were performed for both Unit 1 (with Babcock and Wilcox International (BWI) steam generators) and Unit 2 (with Westinghouse Model D5 steam generators). Results for both units showed that, although the minimum DNBR values for these cases remains well above the safety analysis limit, they are slightly lower than those calculated in the current licensing basis analyses. Therefore, the revision of the PSV setpoint results in a more limiting DNBR for this transient. However, the loss of RCS flow transient remains the most limiting DNBR case. The appropriate revisions will be made to reflect this change in the UFSAR.

The results of the analysis also show that the pressurizer does not reach a water solid condition. The current licensing basis analyses for the maximum pressure case remain applicable; therefore, the applicable RCS and secondary side pressure limits continue to be met.

Consequently, all acceptance criteria for the LOL / TT accident are met considering the revised PSV setpoint.

Rod Withdrawal at Power

Reanalyses of the most limiting RWAP DNB cases were performed for both Units 1 (with BWI steam generators) and Unit 2 (with D5 steam generators). Results showed that Unit 2 remains limiting and that the minimum DNBR reached with the revised PSV setpoint remains above that calculated in the current licensing basis analysis. Therefore, the current licensing basis analysis remains applicable and all acceptance criteria are met.

Loss of Normal Feedwater

Analyses were performed for both Unit 1 (with BWI steam generators) and Unit 2 (with D5 steam generators). For the LONF event, the results show that the Unit 2 analysis is more limiting and is presented below.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to reduction of the steam generator void fraction and because steam flow through the main steam safety valves continues to dissipate the stored and generated heat. Approximately one minute following the initiation of the steam generator low-low level trip, the motor-driven auxiliary feedwater (AFW) pump automatically starts, consequently reducing the rate at which the steam generator water level is decreasing. The capacity of the motor-driven AFW pump enables sufficient heat transfer from the four steam generators receiving auxiliary feedwater to dissipate the core residual heat.

Although this case results in a slightly higher pressurizer water volume than the previous limiting case, the pressurizer never reaches a water solid condition. The calculated peak pressurizer volume is 1863.0 ft^3, which is 1.40 ft^3 below the maximum limit; therefore, no water relief from the pressurizer occurs.

With respect to DNB and overpressure concerns, the LONF event is bounded by the LOL / TT event which demonstrated that the applicable acceptance criteria are met.

In summary, the results of this analysis show that the pressurizer does not reach a water solid condition; therefore, the LONF event with the revised PSV setpoint does not adversely affect the core, the RCS, or the main steam system.

Loss of AC Power to the Plant Auxiliaries

Analyses were performed for both Unit 1 (with BWI steam generators) and Unit 2 (with D5 steam generators). However, for the LOAC power to the plant auxiliaries event, the Unit 1 analysis is more limiting and is presented below.

The first few seconds after the LOAC power to the RCPs, the flow transient for a LOAC power event closely resembles the complete loss of RCS flow transient, where core damage due to rapidly increasing core temperatures is prevented by the reactor trip. After the reactor trip, stored and residual heat must be removed to prevent damage to the core and the reactor coolant and main steam systems. The LOFTRAN computer code results show that the natural

circulation and available AFW flow is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

Although this case results in a slightly higher pressurizer water volume than the previous limiting case, the pressurizer never reaches a water solid condition. The peak pressurizer volume calculated is 1721.5 ft^3, which is approximately 143 ft^3 below the limit; therefore, no water relief from the pressurizer occurs.

With respect to DNB, the LOAC power event is bounded by the complete loss of RCS flow event which demonstrated that the minimum DNBR is greater than the safety analysis limit value. With respect to overpressure concerns, the LOAC power event is bounded by the LOL / TT event which demonstrated that the RCS and secondary side pressure limits are met.

In summary, the results of this analysis show that the pressurizer does not reach a water solid condition; therefore, the LOAC power event with the revised PSV setpoint does not adversely affect the core, the RCS, or the main steam system.

Loss of AC Power to the Plant Auxiliaries with RCP Seal Injection

Analyses were performed for both Unit 1 (with BWI steam generators) and Unit 2 (with D5 steam generators). This event is analyzed to demonstrate that sufficient time is available for the appropriate operator actions to be taken before the PSVs are damaged due to water relief. Operator action will be taken within one hour to terminate the RCS inventory addition and the PSVs will remain operable following water relief. The parameters most important to the continued operability of the PSVs are the number of water relief cycles that the valves will experience in the one hour period and the minimum water temperature to which the valves will be subjected (see the "Spurious Safety Injection (SI)" section below for additional discussion on passing water through the PSVs). The results of the analyses showed that the revised PSV setpoint did not have a significant impact on these parameters (i.e., no more than one additional PSV cycle and less than 0.5 °F temperature reduction). The results were evaluated by the Westinghouse Systems and Equipment Engineering Group, who provided an assessment that the PSVs will remain operable following a LOAC event with water relief through the PSVs; thus, the acceptance criterion was met.

Feedline Break

The current licensing basis FLB analysis was reviewed to determine whether the proposed change in the PSV setpoint would adversely impact this event. The licensing basis FLB analysis for Byron and Braidwood Stations assumes the pressurizer PORVs are operable. Availability of PORVs minimizes the RCS pressure and therefore the saturation pressure, and is a conservative assumption for Byron and Braidwood Stations because the shutoff head for SI is below the PORV opening pressure. A review of the current licensing basis FLB analysis shows that the opening of the PORVs is sufficient to prevent the pressurizer pressure from reaching the new PSV setpoint (i.e., 2460 psig) assuming negative uncertainties. Therefore, the licensing basis analysis is not affected by the change in the PSV setpoint and tolerance, and remains applicable and valid.

Spurious Safety Injection (SI)

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 analysis is documented in a letter from R. M. Krich (Commonwealth Edison Company (now EGC)), to the NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Station," dated July 5, 2000, and a subsequent letter from R. M. Krich (EGC) 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. In a 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, the NRC approved the subject analysis. Based on the documented analysis, 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."

The spurious safety injection (SI) at power event was evaluated 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, as summarized above, 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 a decrease in PSV lift setpoint 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 larger number of PSV water cycles could result for each transient.

In the LOAC with RCP injection event, the PSVs open and relieve steam before the pressurizer fills; therefore, the reduction in the PSV setpoint affects the RCS conditions until the time that the pressurizer fills and water is relieved through the PSVs. Steam relief occurs early in the LOAC with RCP injection event due to the RCS heating up as a result of the LOAC. Subsequently, PSV water relief occurs later in the LOAC with RCP injection transient as compared to the spurious SI at power event since the pressurizer fills more slowly. Since the rate of pressurizer level increase is much slower in the LOAC with RCP injection event as compared to the spurious SI at power event, there is more time to affect the RCS water temperature in the LOAC with RCP injection event.

The spurious SI at power event fills the pressurizer faster than the LOAC with RCP injection event; however, the RCS pressure does not increase significantly until the level nears the top of the pressurizer. Once the level approaches the top of the pressurizer, the RCS pressure increases very quickly and the PSVs open. Consequently, the impact on RCS temperature is less for the spurious SI at power event than the LOAC with RCP injection event.

Using conservative assumptions, the following evaluation can be made. The water injection rate to the RCS assumed in the spurious SI at power event is approximately eight times larger than the water injection rate assumed in the LOAC with RCP seal injection event. The results of the LOAC with RCP seal injection event at the lower PSV lift setting showed an increase of no more than one PSV water cycle and less than 0.5°F temperature reduction. An evaluation has determined that the spurious SI event would demonstrate similar results in terms of the change in the number of PSV water cycles and PSV discharge water temperature. Therefore, it is concluded that the spurious SI at power event, considering the proposed PSV lift setting and increased tolerance, would yield similar results as 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 LOCA) consistent with the conclusion of the existing evaluation.

For the spurious SI at power event DNBR case, a review of the current licensing basis analysis shows that the DNBR never decreases below its initial value and pressurizer pressure does not reach the new PSV lift setting assuming negative uncertainties. Therefore, it is concluded that the current licensing basis analysis for the spurious SI DNBR case is not impacted by the change to the PSV lift setting and tolerance, and remains applicable and valid.

Overall Conclusions

The LOL / TT, RWAP, LONF and LOAC power events were reanalyzed to address a change in the PSV lift setting and tolerance for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The results of the analyses show that the applicable acceptance criteria continue to be met for all events. For the LOL / TT event, the resulting minimum DNBRs were found to be slightly more limiting than the current licensing basis analysis. The appropriate UFSAR revisions will be made to reflect these changes.

A reanalysis of the LOAC power with RCP seal injection was also performed assuming the revised PSV lift setting. Operator action will be taken within one hour to terminate the RCS inventory addition and the PSVs will remain operable following water relief; thus, the acceptance criterion was met.

In addition, a review of the licensing basis FLB analysis concluded that the PSV lift setting and tolerance change will have no effect on that analysis.

Finally, it is concluded that the proposed PSV lift settings will not have a significant effect on the existing spurious SI at power event evaluation results relative to pressurizer water temperature, number of steam and water relief cycles, pressurizer fill time, and satisfactory termination of the transient. This conclusion is based on the similarity between the spurious SI at power event and the LOAC power with RCP seal injection event for which a satisfactory reanalysis was performed. With respect to the DNBR case for the spurious SI at power event, the current licensing basis analysis remains applicable and valid.

The Overpressure Protection Report will be revised as appropriate.

5.0 REGULATORY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an 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 valve (PSV) lift settings from "≥ 2460 psig and ≤ 2510 psig," to "≥ 2411 psig and ≤ 2509 psig." The existing TS represents a ±1% tolerance band around a lift setting of 2485 psig. The proposed lift setting range of "≥ 2411 psig and ≤ 2509 psig" represents a ±2% tolerance band around a lift setting of 2460 psig.

The current TS Limiting Condition for Operation (LCO) 3.4.10 requires that three PSVs shall be operable with lift settings ≥ 2460 psig and ≤ 2510 psig. As noted above, these values represent a ±1% tolerance around a lift setpoint of 2485 psig. The ±1% setpoint tolerance for "as-found" and "as-left" values, as currently required by TS Surveillance Requirement (SR) 3.4.10.1, is the original nuclear industry TS standard for pressurizer safety valve lift settings. This standard was prescribed by the American Society of Mechanical Engineers (ASME), Section III, "Rules for Construction of Nuclear Facility Components." Industry experience has shown that normally functioning PSVs periodically exceed the ±1% tolerance after a cycle of operation; therefore, many licensees have revised their TS PSV as-found tolerance limits to ±3%, consistent with NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2, dated April 2001, and ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

Braidwood and Byron Stations have PSVs manufactured by Crosby. There have been numerous instances where the PSV lift settings have fallen outside the ±1% tolerance limits during end of cycle inservice testing. These as-found lift settings have prompted the generation of licensee event reports (LERs) in accordance with 10 CFR 50.73(a)(2)(i)(b), "Any operation or condition prohibited by the plant's Technical Specifications..." Most of the as-found lift settings have not exceeded ±2% of the pressure setpoint. Subsequent evaluations have shown that there are no material condition concerns with these valves and they are performing within their design capabilities. Therefore, generating a LER for a PSV that is performing satisfactorily within its design capability becomes an unnecessary burden for both the licensee and the NRC.

Reanalysis/evaluations were performed to assess all transients that could be potentially impacted by the proposed PSV lift setting and tolerance band change. Loss of coolant accidents (LOCAs) and a steam generator tube rupture accident involve a decrease in reactor coolant system (RCS) pressure and therefore, will not challenge the PSV opening pressure lift setting. These accident analyses are unaffected by the proposed change in PSV lift setting. Non-LOCA accidents that conservatively assume a PSV negative tolerance modeling may potentially be affected by this change and therefore, the following reanalyses/evaluations were performed:

- reanalysis of loss of load (LOL) / turbine trip (TT), i.e., departure from nuclear boiling (DNB) limiting cases with revised PSV setpoint.
- reanalysis of rod withdrawal at power (RWAP), i.e., DNB limiting cases with revised PSV setpoint.
- reanalysis of loss of normal feedwater (LONF) limiting cases with revised PSV setpoint.
- reanalysis of loss of alternating current (LOAC) limiting cases with revised PSV setpoint.
- reanalysis of LOAC with reactor coolant pump (RCP) seal injection with revised PSV setpoint.
- evaluation of the impact of the revised PSV setpoint on the feedline break analysis.
- evaluation of the potential impact of revised PSV setpoint on spurious safety injection (SI) pressurizer overfill.

These reanalyses/evaluations concluded that all of the subject transients continue to have acceptable results.

The NRC has previously approved a similar change for the Wolf Creek Generating Station in Amendment No. 133, issued March 23, 2000.

Criteria

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

In support of this determination, an evaluation of each of the three criteria set forth in 10 CFR 50.92 is provided below regarding the proposed license amendment.

1. **The proposed TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   Reanalysis/evaluations were performed to assess all transients that could be potentially impacted by the proposed PSV lift setting and tolerance band change. The proposed change in the PSV tolerance from ±1% to ±2% with a reduction in the lift setting from 2485 psig to 2460 psig allows a decrease in the valve minimum opening pressure and therefore provides earlier pressurizer relief and a reduced RCS pressure. The proposed change does not affect the maximum opening pressure assumed in the non-LOCA analyses since the proposed change in maximum PSV opening pressure is insignificant and in the conservative direction. Therefore, only those transients for which it is conservative to minimize the RCS pressure (i.e., DNB and pressurizer overfill concerns)

are potentially impacted by the proposed change. The reanalyses/evaluations of all the affected transients demonstrated acceptable results with no significant increase in the probability or consequences.

Further, any evaluations performed on an overpressure transient conservatively assume the upper limit of the PSV tolerance. The proposed change to the lower tolerance limit of the PSV lift setting means that an overpressure transient may be terminated at a pressure that is lower than assumed in the analysis. It has also been determined that the transient analyses are not adversely affected because the limiting transients are not sensitive to the pressure tolerance decrease. Therefore, the primary system pressure boundary is not challenged by the PSV lower tolerance limit change. The assumed maximum PSV lift setting was not changed, and therefore, does not impact analyses performed for overpressure transients. It has been determined that the design relieving capacity of the PSVs can still be met with the reduction in PSV setpoint. Except for the PSV lower lift setting and increased tolerance, the design and operation of the PSVs remains unchanged.

Based on this analysis, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the Byron/Braidwood Stations Updated Final Safety Analysis Report.

2. **The proposed TS change does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed change in the PSV tolerance from ±1% to ±2% with a reduction in the lift setting from 2485 psig to 2460 psig allows a decrease in the valve minimum opening pressure and therefore provides earlier pressurizer relief and a reduced RCS pressure. The proposed change does not affect the maximum opening pressure assumed in the accident analyses since the proposed change in maximum PSV opening pressure is insignificant and in the conservative direction. The pressurizer PORVs serve to minimize challenges to the PSVs. An assessment of the impact of reducing the PSV lift setpoint and increasing the tolerance has determined that the resulting margin is sufficient to ensure that the PORVs will actuate prior to the PSVs. Except for the PSV lower lift setting and increased tolerance, the design and operation of the PSVs remain unchanged.

The proposed change does not involve the use or installation of new equipment and all currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed change will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **The proposed TS change does not involve a significant reduction in a margin of safety.**

The PSVs provide, in conjunction with the reactor protection system, overpressure protection for the RCS. The PSVs are designed to prevent the system pressure from

exceeding the system safety limit, 2735 psig, which is 110% of the design pressure. The change in the upper limit of the PSV tolerance from +1% to +2% with a reduction in the nominal set point from 2485 psig to 2460 psig does not challenge the upper limit of the overpressure protection. The change in PSV maximum opening lift setting is insignificant and in the conservative direction with respect to overpressure protection, therefore, the proposed change does not impact analyses performed for overpressure transients. For all non-LOCA events, the analyses/evaluations support the change in PSV lift setting and tolerance from 2485 psig ±1% to 2460 psig ±2%. The LOCA analyses are not impacted because the transient results in a decrease in RCS pressure and therefore, will not challenge the PSV opening pressure lift setting. The change in the PSV lift setting and tolerance also has no effect on the reactor protection or engineered safety features systems trip set points. Thus, the proposed change does not involve a significant reduction in any margin of safety.

Based on the above discussions, it has been determined that the requested TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated; or create the possibility of a new or different kind of accident from any accident previously evaluated; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The PSVs must satisfy the requirements of 10 CFR 50.36, "Technical Specifications," paragraph (c)(2)(ii), Criterion 3. This requirement states the following:

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

Criterion 3. A structure, system, or component 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.

NUREG 1431, "Standard Technical Specifications, Westinghouse Plants," Revision 2, dated April 2001, specifies the following for LCO 3.4.10:

"[Three] pressurizer safety valves shall be OPERABLE with lift settings ≥ [2460] psig and ≤ [2510] psig."

In addition, NUREG 1431 specifies the following for TS SR 3.4.10.1:

"Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%."

This SR will remain unchanged in the Byron and Braidwood TS. The Inservice Testing Program supporting documentation will be revised to reflect the PSV lift settings.

The NUREG 1431 Bases for SR 3.4.10.1 states the following:

> "The pressurizer safety valve setpoint is ±[3]% for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift."

ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," references ASME/American National Standards Institute (ANSI) OM-1, Operations and Maintenance of Nuclear Power Plants," 1987 Edition, for testing acceptance criteria for relief valves. ASME/ANSI OM-1, Section 1.3.3, "Test Frequency, Class 1 Pressure Relief Devices," paragraph (e)(2), specifies that the as-found acceptance criteria for Class 1 pressure relief valves is ±3%.

UFSAR Section 5.4.13, "Safety and Relief Valves," provides the design basis for the PSVs.

The proposed change is consistent with all the above criteria.

Impact on Previous Submittals/Precedent

No other license amendment requests currently under review by the NRC are impacted by the information presented in this license amendment request.

The NRC has previously approved a similar change for the Wolf Creek Generating Station in Amendment No. 133, issued March 23, 2000.

6.0 ENVIRONMENTAL CONSIDERATION

Overview

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an 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 valve (PSV) lift settings from "≥ 2460 psig and ≤ 2510 psig," to "≥ 2411 psig and ≤ 2509 psig." The existing TS represents a ±1% tolerance band around a lift setting of 2485 psig. The proposed lift setting range of "≥ 2411 psig and ≤ 2509 psig" represents a ±2% tolerance band around a lift setting of 2460 psig.

Criteria

EGC has evaluated this proposed operating license amendment consistent with the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." EGC has determined that these proposed changes meet the criteria for a categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has

determined that no irreversible consequences exist in accordance with paragraph (b) of 10 CFR 50.92, "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation," or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 5.1, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed change, which revises TS 3.4.10, "Pressurizer Safety Valves," by changing the existing PSV lift settings from "$\geq$ 2460 psig and $\leq$ 2510 psig," to "$\geq$ 2411 psig and $\leq$ 2509 psig," does not result in an increase in power level, does not increase the production nor alter the flow path or method of disposal, of radioactive waste or byproducts; thus, there will be no change in the amounts of radiological effluents released offsite.

Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed change will not result in any changes to the configuration of the facility. The proposed change revises the lift settings of the PSVs and will not cause a change in the level of controls or methodology used for the processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed amendment result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

7.0 REFERENCES

1. Byron/Braidwood Stations Updated Final Safety Analysis Report, Section 5.4.13, "Safety and Relief Valves"

2. ASME, Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components"

3. Updated Final Safety Analysis Report, Chapter 15, "Accident Analyses"

4. WCAP-7769, Revision 1, "Overpressure Protection for Westinghouse PWRs," dated June 1972

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

6. WCAP-12910, Revision 1-A, "Pressurizer Safety Valve Set Pressure Shift, " G. O. Barrett, et al, May 1993

ATTACHMENT 2-A

Markup of Proposed Technical Specifications Page Changes

BRAIDWOOD STATION

REVISED TS PAGES

3.4.10-1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~2460~~ 2411 psig and ≤ ~~2510~~ 2509 psig.

-----NOTE-----

The lift settings are not required to be within the LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

ATTACHMENT 2-B

Markup of Proposed Technical Specification Page Changes

BYRON STATION

<u>REVISED TS PAGES</u>

3.4.10-1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ ~~2460~~ 2411 psig and ≤ ~~2510~~ 2509 psig.

-----------------------------NOTE-----------------------------

The lift settings are not required to be within the LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

ATTACHMENT 3-A

Typed Pages

for

Technical Specification Changes

and

Bases Changes
(for information only)

BRAIDWOOD STATION

<u>REVISED TS PAGES</u>

3.4.10-1

<u>REVISED BASES PAGES</u>

B 3.4.10-1
B 3.4.10-3
B 3.4.10-4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2411 psig and $\leq$ 2509 psig.

---------------------------NOTE---------------------------
The lift settings are not required to be within the LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODES 4 and 5, and in MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 2% tolerance requirement assumed in the safety analysis. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

BASES

LCO

The three pressurizer safety valves are set to open at 2460 psig, slightly below the RCS design pressure (2500 psia), and within the ASME specified tolerance (Ref. 4), to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 2% tolerance requirement assumed in the safety analysis. The limit protected by this Specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

APPLICABILITY

In MODES 1, 2, and 3, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODES 4 and 5, and in MODE 6 with the reactor vessel head on, because Low Temperature Overpressure Protection (LTOP) is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If Required Action A.1 and its associated Completion Time are not met or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 2% of a nominal 2460 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

ATTACHMENT 3-B

Typed Pages

for

Technical Specification Changes

and

Bases Changes
(for information only)

BYRON STATION

<u>REVISED TS PAGES</u>

3.4.10-1

<u>REVISED BASES PAGES</u>

B 3.4.10-1
B 3.4.10-3
B 3.4.10-4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2411 psig and ≤ 2509 psig.

---------------------------NOTE---------------------------
The lift settings are not required to be within the LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODES 4 and 5, and in MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 2% tolerance requirement assumed in the safety analysis. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

BASES

LCO

The three pressurizer safety valves are set to open at 2460 psig, slightly below the RCS design pressure (2500 psia), and within the ASME specified tolerance (Ref. 4), to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 2% tolerance requirement assumed in the safety analysis. The limit protected by this Specification is the Reactor Coolant Pressure Boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The Note allows entry into MODE 3 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame.

APPLICABILITY

In MODES 1, 2, and 3, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is conservatively included, although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODES 4 and 5, and in MODE 6 with the reactor vessel head on, because Low Temperature Overpressure Protection (LTOP) is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

BASES

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If Required Action A.1 and its associated Completion Time are not met or if two or more pressurizer safety valves are inoperable, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 2% of a nominal 2460 psig for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.