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

Subject: Request for License Amendment to Increase Main Steam Safety Valve Lift Setpoint Tolerance and Standby Liquid Control System Enrichment

References:

1. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990

2. Letter from A. C. Thadani (NRC) to C. L. Tully (BWR Owners' Group), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report' (TAC No. M79265)," dated March 8, 1993

3. Letter from M. Banerjee (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3 – Request for Additional Information Related to License Amendment Request to Revise Technical Specification Surveillance Requirement 3.4.3.1 and 3.1.7.10 (TAC Nos. MD2166 and MD2167)," dated August 10, 2006

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

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2, respectively. The proposed change revises Technical Specification (TS)

Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve (MSSV) lift setpoint tolerance from ± 1% to ± 3%. The proposed change does not alter the TS requirements for the number of MSSVs required to be operable, the nominal lift setpoints, the MSSV testing frequency, or the manner in which the valves are operated. The current TS requirement to adjust the MSSV as-left tolerance to within ± 1% of the nominal lift setpoint, prior to returning a valve to service, is not being changed. In addition, the proposed change revises SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the Standby Liquid Control (SLC) system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

The proposed change is consistent with guidance specified in Boiling Water Reactor Owners' Group (BWROG) document NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report" (i.e., Reference 1), which was developed to support the use of a ± 3% lift setpoint tolerance for MSSVs. In Reference 2, the NRC approved NEDC-31753P.

Dresden Nuclear Power Station (DNPS) has submitted a similar license amendment request, dated June 2, 2006. In response to this request, the NRC issued a formal request for additional information (i.e., Reference 3). The request involved providing information applicable to both DNPS and QCNPS and to a generic issue involving surveillance requirements. The additional information supplied to the NRC in Reference 4 applies directly to QCNPS and is not duplicated here.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.
- Attachment 2 provides the marked-up TS pages, with the proposed change indicated.
- Attachment 3 provides a marked-up copy of the affected TS Bases pages. The TS Bases pages are provided for information only and do not require NRC approval.
- Attachment 4 provides a summary of the analysis results that support QCNPS operation with a MSSV lift setpoint tolerance change from ± 1% to ± 3%.

Attachment 4 contains proprietary information as defined in 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." General Electric (GE), as the owner of the proprietary information, has executed the affidavit provided within Attachment 4, which identifies that the information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. Accordingly, it is requested that the proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 10 CFR 9.17, "Agency records exempt from public disclosure." A non-proprietary version of the information contained in Attachment 4 is provided in Attachment 5.

The proposed change has been reviewed by the QCNPS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed change by November 2, 2007. Once approved, the amendment for QCNPS Units 1 and 2 shall be implemented prior to MSSV testing during the next refueling outage for each unit respectively. This will allow adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

In accordance with 10 CFR 50.91(b), "Notice for public comment," EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Any actions discussed in this letter represent intended or planned actions by EGC. They are described for the NRC's information and are not regulatory commitments. Should you have any questions related to this letter, please contact Mr. Timothy Byam at (630) 657-2804.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 7th day of November 2006.

Respectfully,

Darin M Benyak

Darin M. Benyak
Manager – Licensing

Attachment 1: Evaluation of Proposed Change
Attachment 2: Markup of Proposed Technical Specifications Pages
Attachment 3: Markup of Technical Specification Bases Pages (For Information Only)
Attachment 4: GE-NE-0000-0053-8435-R1P, "Dresden 2 & 3 and Quad Cities 1 & 2 Safety Valve Setpoint Tolerance Relaxation," May 2006 (PROPRIETARY)
Attachment 5: GE-NE-0000-0053-8435-R1NP, "Dresden 2 & 3 and Quad Cities 1 & 2 Safety Valve Setpoint Tolerance Relaxation," May 2006 (NON-PROPRIETARY)

ATTACHMENT 1
Evaluation of Proposed Change

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2, respectively. The proposed change revises Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve (MSSV) lift setpoint tolerance from ± 1% to ± 3%. In addition, the proposed change revises SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the Standby Liquid Control (SLC) system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

Each QCNPS unit is designed with nine safety valves. Eight (8) of these valves are spring safety valves and are used to perform the safety function of the safety relief valves (S/RVs) as discussed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report" (i.e., Reference 1). The remaining valve is a dual function Target Rock safety/relief valve (S/RV). The term MSSV is used throughout this attachment, and is intended to include both the eight safety valves and the Target Rock S/RV.

2.0 PROPOSED CHANGE

The proposed change revises the lift setpoint tolerances for the MSSVs that are listed in SR 3.4.3.1 of QCNPS TS 3.4.3, "Safety and Relief Valves." The proposed revision implements a wider MSSV lift setpoint tolerance to better match the TS performance requirements with the installed valve capabilities. The intended change increases the allowable MSSV lift setpoint tolerance from ± 1% of the nominal lift setpoint to ± 3% of the nominal lift setpoint. This change only applies to the as-found tolerance and not to the as-left tolerance, which will remain unchanged at ± 1% of the nominal lift setpoint. The proposed change does not alter the TS requirements for the number of MSSVs required to be operable, the nominal lift setpoints, the MSSV testing frequency, or the manner in which the valves are operated.

The proposed change also revises QCNPS TS 3.1.7, "Standby Liquid Control (SLC) System," SR 3.1.7.10 to increase the required enrichment of sodium pentaborate used in the SLC system. SR 3.1.7.10 currently states:

> "Verify sodium pentaborate enrichment is ≥ 30.0 atom percent B-10."

The proposed change revises SR 3.1.7.10 to read:

> "Verify sodium pentaborate enrichment is ≥ 45.0 atom percent B-10."

Attachment 2 provides marked up TS pages indicating the proposed change. Attachment 3 provides marked up TS Bases pages. The TS Bases pages are provided for information only and do not require NRC approval.

3.0 BACKGROUND

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. Each QCNPS unit is designed with nine safety valves, one of which also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV). The safety valves and S/RV are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. All nine MSSVs are required to be operable by TS 3.4.3, "Safety and Relief Valves."

The safety valves actuate in the safety mode (i.e., spring mode of operation). In this mode, the safety valve opens when the inlet steam pressure reaches the lift set pressure. At that point, the vertical upward force generated by the inlet pressure under the valve disc balances the downward force generated by the spring.

The S/RV is a dual function Target Rock valve that can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (i.e., spring mode of operation), the S/RV spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. In the relief mode (i.e., power actuated mode of operation), automatic or manual switch actuation energizes a solenoid valve, which pneumatically actuates a plunger located within the main valve body. Actuation of the plunger allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve.

The S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The eight safety valves discharge directly to the drywell.

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position). For the purpose of the analyses, all nine MSSVs are assumed to operate in the safety mode. The relief function of the S/RV is not credited to function during this event. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). LCO 3.4.3 helps to ensure that the acceptance limit of 1375 psig is met during the design basis event.

The safety function of all nine MSSVs is required to be operable to satisfy the ASME overpressure analysis. The setpoints are established to ensure that the ASME Code limit for peak reactor pressure is satisfied. This transient evaluation is based on these setpoints, but also includes an additional lift setpoint tolerance uncertainty to provide an added degree of conservatism.

The use of a limited ± 1% allowable as-found MSSV lift setpoint tolerance in plant TSs was a generic industry issue. Nuclear power plant licensees have experienced difficulty in meeting the typical 1% lift setpoint tolerance for MSSVs. As a result, the BWR Owners' Group (BWROG) developed NEDC-31753P (i.e., Reference 1) to support the use of a 3% lift setpoint tolerance, which is consistent with the ASME OM Code requirements (formerly Section XI requirements). On March 8, 1993, the NRC issued a safety evaluation approving NEDC-31753P (i.e., Reference 2).

In the safety evaluation, the NRC stated that a generic change of lift setpoint tolerance to 3% is acceptable provided that it is evaluated in the analytical bases. Specific analyses required to be provided are transient analysis, design basis overpressurization event, re-evaluation of high pressure systems (i.e., motor operated valves, reactor vessel instrumentation, and piping), alternate operating modes, containment response during a loss-of-coolant accident (LOCA), and hydrodynamic loads on MSSV discharge lines. These plant specific analyses have been performed for QCNPS, and the results are discussed in Section 4.0 and Attachment 4.

The SLC system is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory to a subcritical condition with the reactor in the most reactive, xenon-free state without taking credit for control rod movement. The SLC system satisfies the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." The SLC system consists of a boron solution tank, two positive displacement pumps, two explosive valves that are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel. The borated solution is discharged near the bottom of the core shroud, where it then mixes with the cooling water rising through the core.

Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Action to verify the actual boron-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper boron-10 atom percentage is being used. The sodium pentaborate enrichment is selected to ensure that the SLC system is capable of bringing the reactor to a subcritical condition in the event of a postulated ATWS event where the control rods cannot be inserted to maintain subcritical conditions. Attachment 4 describes the impact of the setpoint tolerance increase on the ATWS analysis. In order to ensure that the SLC pump discharge relief valve does not lift during an ATWS event, EGC plans to implement a design modification that will ensure that the requirement of 10 CFR 50.62 is exceeded using a single SLC pump at a nominal 40 gpm. The modification involves an increase in the enrichment of sodium pentaborate used in the SLC system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

4.0 TECHNICAL ANALYSIS

Reference 1 was reviewed and approved by the NRC as documented in a safety evaluation issued by Reference 2. The NRC determined that it was acceptable for licensees to submit TS amendment requests to revise the safety function lift setpoint tolerance to ± 3%, provided that the setpoints for those valves are restored to within ± 1% prior to reinstallation. The NRC also indicated in its safety evaluation that licensees planning to implement TS changes to increase the lift setpoint tolerances should provide the following plant specific analyses.

1. Transient analysis, using NRC approved methods, of abnormal operational occurrences as described in NEDC-31753P utilizing a ± 3% lift setpoint tolerance for the MSSVs.

2. Analysis of the design basis overpressure event using the ± 3% tolerance limit for the MSSV setpoints to confirm that the vessel pressure does not exceed ASME pressure vessel code upset limits.

3. Plant specific analysis described in Items 1 and 2 should assure that the number of MSSVs included in the analysis corresponds to the number of valves required to be operable in the TS.

4. Re-evaluation of the performance of high pressure systems (e.g., pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping considering the ± 3% tolerance limit.

5. Evaluation of the ± 3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.).

6. Evaluation of the effects of the ± 3% tolerance limit on the containment response during LOCAs and the hydrodynamic loads on the MSSV discharge lines and containment.

In support of the proposed TS changes, General Electric performed the plant specific analyses and evaluations described above, and the results are documented in Attachment 4. Attachment 4 determined that the impact of the setpoint tolerance increase was acceptable; however, certain areas required further assessment by EGC. These areas include S/RV dynamic loads, motor-operated valve (MOV) operation, and SLC system performance. These items are addressed below.

<u>S/RV Dynamic Loads</u>

Since a broadened S/RV setpoint tolerance can increase the S/RV safety mode opening pressure, the S/RV dynamic loads are expected to increase. Therefore, the impact of the changed setpoint tolerance with regard to S/RV discharge loads was examined. The setpoint upper bound resulted in a 1.66% increase in pressure and a 2.1% increase in flow over the existing analysis. The impact of crediting two items, which had not previously been credited, was evaluated. Crediting these two items, as discussed below, offsets the increased S/RV dynamic loads resulting from the broadened setpoint tolerance.

First, the increased S/RV dynamic loads are offset by the fact that the discharge line clearing loads are reduced by slower valve opening times. The S/RV loading most significantly affected by the main disk stroke time is the transient wave thrust load on the tail piping. Shorter stroke time results in higher loading. The General Electric RVFOR computer code is used to define the blowdown force-time histories. In the benchmarking and validation of that code, an opening time of 0.02 seconds was used to model a 0.05 second S/RV opening time. The RVFOR code is sensitive to valve opening times and was validated using 0.02 seconds as compared to the actual valve time for the benchmarked plant of 0.05 seconds.

The actual valve opening time used in the QCNPS analysis is 0.25 seconds. When a similar adjustment to the opening time is applied, this results in an opening time of 0.10 seconds to be used in for the RVFOR modeling. Application of this longer opening time reduces the load approximately 2%.

Second, correction of an error in the General Electric computer code RVFOR results in an additional reduction. On May 25, 1984, General Electric informed the Mark I owner's group of a program error that was discovered for the RVFOR code that defines the blowdown force-time histories. The final disposition of the error concluded that the clearing thrust load calculated by RVFOR could be over-predicted by as much as 50%. Since existing RVFOR analyses for QCNPS predate the error discovery, the current plant unique S/RV load analysis include the additional conservatism afforded by this error. Correction of this error further offsets the increased S/RV dynamic loads resulting from the higher setpoint tolerance.

Therefore, based on the information above, crediting these two items offsets the increased S/RV dynamic loads resulting from the broadened setpoint tolerance.

MOV Operation

The impact of changing the MSSV setpoint tolerance on MOVs was reviewed by QCNPS Engineering. MOVs in the Main Steam, Reactor Core Isolation Cooling (RCIC), and High Pressure Coolant Injection (HPCI) systems were affected by the increased differential pressure. This review found some reductions of MOV margin, but not below acceptable levels. No MOV's margin fell below their current rank of high margin (i.e., ≥ 10%) as a result of the proposed MSSV tolerance change.

The scope identified above is limited to steam-side valves. Water-side valves, such as the HPCI, RCIC, or Safe Shutdown Makeup Pump injection valves have their maximum pressure differential defined by the discharge pressure of their associated pumps. Therefore, their worst differential pressure case is at low reactor pressure and they are not impacted by this change.

SLC System Performance

As part of these plant specific analyses and evaluations, it was identified that a change to the sodium pentaborate enrichment in the SLC system was necessary. 10 CFR 50.62 requires the SLC system to deliver 86 gpm of 13 weight percent (wt%) (minimum) sodium pentaborate solution or equivalent, at the natural boron-10 isotopic enrichment. Currently, QCNPS exceeds this requirement, using a performance objective for the SLC system that provides a system flowrate, using both SLC pumps, of 80 gpm at a minimum concentration of 14 wt% sodium pentaborate solution at 30.0 atom percent boron-10 isotopic enrichment (Reference 3).

An increase to the allowable MSSV lift setpoint tolerance results in a higher peak reactor vessel pressure during an ATWS event. EGC has evaluated the increase and determined that the pressure in the SLC system needed to overcome the higher peak reactor vessel pressure is such that the SLC pump discharge relief valve could potentially lift. As described in NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," the lifting of the SLC pump discharge relief valve would cause the sodium pentaborate solution to be recycled to the pump suction and, therefore, prevent the system from meeting the equivalent flow capacity required by 10 CFR 50.62.

In order to ensure that the SLC pump discharge relief valve does not lift during an ATWS event, as analyzed in Attachment 4, EGC plans to implement a design modification that will ensure that the requirement of 10 CFR 50.62 is exceeded using a single SLC pump at a nominal 40 gpm. The modification involves an increase in the enrichment of sodium pentaborate used in the SLC system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

In Reference 3, the NRC issued an amendment to the QCNPS TS to add SR 3.1.7.10, which requires verification that the sodium pentaborate enrichment is ≥ 30.0 atom percent boron-10. Although this amendment has been issued, it has only been implemented for QCNPS Unit 2; it has not yet been implemented for QCNPS Unit 1, since implementation is tied to the upcoming refueling outage for Unit 1. The change to SR 3.1.7.10 that is currently being proposed will increase the required sodium pentaborate enrichment specified in SR 3.1.7.10 from a minimum of 30.0 atom percent boron-10 to a minimum of 45.0 atom percent boron-10. This change will ensure that sodium pentaborate solution added to the SLC tank meets the requirement of 10 CFR 50.62 using a single SLC pump at ≥ 35.2 gpm. The TS SR 3.1.7.7 will continue to verify that each SLC system pump will develop a flow rate of ≥ 40 gpm.

In Reference 4, the NRC issued an amendment to the QCNPS TS that adopts an alternative source term (AST) in accordance with 10 CFR 50.67, "Accident source term." The supporting analyses for AST assume the pH of the suppression pool is controlled to prevent the re-evolution of iodine following a design basis loss of coolant accident (i.e., DBA LOCA). This is accomplished by injecting SLC (i.e., boron solution) following a DBA LOCA to ensure pH is controlled to a value greater than 7.0. The changes proposed herein have no impact on the chemical properties of the SLC boron solution and therefore, do not impact the assumptions of the supporting AST analyses.

Optima2 Fuel

In Reference 3, the NRC approved operation with Westinghouse SVEA-96 Optima2 fuel at QCNPS. The proposed change to 3% setpoint tolerance is supported by Westinghouse analyses of events which credit the MSSVs for introduction of SVEA-96 Optima2 fuel at QCNPS. Specifically, the impact of a MSSV tolerance of 3% has been analyzed by Westinghouse for ATWS containment response due to Westinghouse SVEA-96 Optima2 fuel at QCNPS. The analysis demonstrated that the containment response is acceptable. Also, the Westinghouse analysis demonstrated that the transition to SVEA-96 Optima2 fuel results in no significant change in the loads on reactor internals components and the respective design criteria are met for the response of the SVEA-96 Optima2 fuel assemblies. As part of the QCNPS Unit 2 Cycle 19 reload analyses, the overpressure analysis was performed assuming an MSSV tolerance of 3%. Similarly, QCNPS Unit 1 Cycle 20 reload analyses, currently in progress, will be analyzed using the same assumption.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-29 and DPR 30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2. The proposed change revises Technical Specifications (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found main steam safety valve (MSSV) lift setpoint tolerance from ± 1% to ± 3%. The proposed change does not alter the TS requirements for the number of MSSVs required to be operable, the nominal lift setpoints, the MSSV testing frequency, or the manner in which the valves are operated. The current TS requirement to adjust the MSSV as-left tolerance to within ± 1% of the nominal lift setpoint, prior to returning a valve to service, is not being changed. In addition, the proposed change revises SR 3.1.7.10 to increase the enrichment of sodium pentaborate used in the Standby Liquid Control (SLC) system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

According to 10 CFR 50.92 (c), "Issuance of amendment," 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.

EGC has evaluated the proposed change to the TS for QCNPS Units 1 and 2, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change increases the allowable as-found MSSV lift setpoint tolerance, determined by test after the valves have been removed from service, from ± 1% to ± 3%. The proposed change does not alter the TS requirements for the number of MSSVs required to be operable, the nominal lift setpoints, the allowable as-left lift setpoint tolerance, the MSSV testing frequency, or the manner in which the valves are operated.

Consistent with current TS requirements, the proposed change continues to require that the MSSVs be adjusted to within ± 1% of their nominal lift setpoints following testing. Since the proposed change does not alter the manner in which the valves are operated, there is no significant impact on reactor operation.

The proposed change does not involve a physical change to the valves, nor does it change the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components, with the exception of the SLC system enrichment change. The proposed change to increase the enrichment of sodium pentaborate used in the SLC system will ensure that the requirements of 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," continue to be met. The SLC system is not an initiator to an accident; rather, the SLC system is used to mitigate an ATWS event. Therefore, these changes will not increase the probability of an accident previously evaluated.

Generic considerations related to the change in setpoint tolerance were addressed in NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," and were reviewed and approved by the NRC in a safety evaluation dated March 8, 1993. General Electric Company (GE) completed plant-specific analyses to assess the impact of the setpoint tolerance increase on Dresden Nuclear Power Station Units 2 and 3 and QCNPS Units 1 and 2. The impact of the MSSV setpoint tolerance increase, as addressed in this analysis, included vessel overpressure, Updated Final Safety Analysis Report (UFSAR) Chapter 15 events, ATWS, Loss of Coolant Accident (LOCA), containment response and loads, high pressure systems performance, Appendix R fire protection, vessel thermal cycle, operating mode and equipment out of service review, and extended power uprate evaluation review. The proposed change to 3% setpoint tolerance is supported by Westinghouse SVEA-96 Optima2 fuel analysis of events that credit the MSSVs.

The plant specific evaluations, required by the NRC's safety evaluation and performed to support this proposed change, show that there is no change to the design core thermal limits and adequate margin to the reactor vessel pressure limits using a ± 3% lift setpoint tolerance. These analyses also show that operation of Emergency Core Cooling Systems is not affected, and the containment response following a LOCA is acceptable. The plant systems associated with these proposed changes are capable of meeting applicable design basis requirements and retain the capability to mitigate the consequences of accidents described in the UFSAR. The accident analyses that credit the initiation of SLC as a dose mitigation feature are not impacted by the proposed change because the chemical properties of the SLC boron solution are not affected. Therefore, these changes do not involve an increase in the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change increases the allowable as-found lift setpoint tolerance for the QCNPS MSSVs, and increases the required enrichment of sodium pentaborate used in the SLC system. The proposed change to increase the enrichment of sodium pentaborate used in the SLC system will ensure that the requirements of 10 CFR 50.62 continue to be met.

   The proposed change to increase the MSSV tolerance was developed in accordance with the provisions contained in the NRC safety evaluation for NEDC-31753P. MSSVs installed in the plant following testing or refurbishment will continue to meet the current tolerance acceptance criteria of ± 1% of the nominal setpoint. The proposed change does not affect the manner in which the overpressure protection system is operated; therefore, there are no new failure mechanisms for the overpressure protection system.

   The proposed change to allow an increase in the MSSV setpoint tolerance does not alter the nominal MSSV lift setpoints or the number of MSSVs currently required to be operable by QCNPS TS. The proposed change does not involve physical changes to the valves, nor does it change the safety function of the valves. There is no alteration to the parameters within which the plant is normally operated. As a result, no new failure modes are being introduced.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change does not modify the safety limits or setpoints at which protective actions are initiated, and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

   Establishment of the ± 3% MSSV setpoint tolerance limit does not adversely impact the operation of any safety-related component or equipment. Evaluations performed in accordance with the NRC safety evaluation for NEDC-31753P have concluded that all design limits will continue to be met.

The proposed change to increase the enrichment of sodium pentaborate used in the SLC system will ensure that the requirements of 10 CFR 50.62 continue to be met.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements and Criteria

The current ± 1% tolerance band on the MSSV opening setpoints stems from the original acceptance criterion defined by the ASME for inservice performance testing. Nuclear power plant licensees have experienced difficulty in meeting the typical ± 1% lift setpoint tolerance. As a result, the BWROG developed NEDC-31753P to support the use of the ± 3% MSSV lift setpoint tolerance.

NEDC-31753P was reviewed and approved by the NRC as documented in Reference 2. The NRC determined that it is acceptable for licensees to submit TS amendment requests to revise the MSSV lift setpoint tolerance to ± 3%, provided that the setpoints for those MSSVs tested are restored to ± 1% prior to reinstallation. The NRC also indicated in its safety evaluation that licensees planning to implement TS changes to increase the MSSV setpoint tolerances should provide a plant specific analysis. The plant specific analysis for QCNPS is provided in Attachment 4.

The existing MSSVs are tested in accordance with the ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants." The QCNPS fourth ten year inservice testing (IST) program implements the 1998 Edition through 2000 Addenda of the ASME OM Code. Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," Section I-1300, "Guiding Principles," of the ASME OM Code requires that a sample of valves from each valve group be periodically tested. The as-found acceptance criteria for those valves tested is either the ± tolerance limit of the owner-established set-pressure acceptance criteria (i.e., currently ± 1%) or ± 3% of the valve nameplate set-pressure.

Since the ASME OM Code allows a ± 3% limit to be used, no relief from the ASME OM Code is required with regard to the setpoint tolerance change. However, a change to the TS is required to revise the owner-established set-pressure acceptance criteria to ± 3%.

10 CFR 50.62 requires the SLC system to deliver 86 gpm of 13 wt% (minimum) sodium pentaborate solution or equivalent, at the natural boron-10 isotopic enrichment. Currently, to satisfy this requirement for QCNPS, a performance

objective of the SLC system is to provide a system flowrate, using both SLC pumps, of 80 gpm at a minimum concentration of 14 wt% sodium pentaborate solution at 30.0 atom percent boron-10 isotopic enrichment.

In order to ensure that the SLC pump discharge relief valve does not lift during an ATWS event, EGC plans to implement a design modification that will ensure that the requirement of 10 CFR 50.62 is exceeded using a single SLC pump at a nominal 40 gpm. The modification involves an increase in the enrichment of sodium pentaborate used in the SLC system from ≥ 30.0 atom percent boron-10 to ≥ 45.0 atom percent boron-10.

In Reference 3, the NRC issued an amendment to the QCNPS TS to add SR 3.1.7.10, which requires verification that the sodium pentaborate enrichment is ≥ 30.0 atom percent boron-10. Although this amendment has been issued, it has only been implemented on Unit 2, since implementation for Unit 1 is tied to its upcoming refueling outage. The change to SR 3.1.7.10 that is currently being proposed will increase the required sodium pentaborate enrichment specified in SR 3.1.7.10 from a minimum of 30.0 atom percent boron-10 to a minimum of 45.0 atom percent boron-10. This change will ensure that sodium pentaborate solution added to the SLC tank exceeds the requirement of 10 CFR 50.62 using a single SLC pump at a nominal 40 gpm.

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change 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." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9). Therefore, in accordance with 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990

2. Letter from A. C Thadani (NRC) to C. L. Tully (BWR Owners' Group), "Acceptance for Referencing of Licensing Topical Report NEDC-31753P, 'BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report' (TAC No. M79265)," dated March 8, 1993

3. Letter from M. Banerjee (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 – Issuance of Amendments Re: Transition to Westinghouse Fuel and Minimum Critical Power Ratio Safety Limits (TAC Nos. MC7323, MC7324, MC7325 and MC7326)," dated April 4, 2006

4. Letter from M. Banerjee (U. S. NRC) to C. Crane (Exelon Generation Company, LLC), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments Re: Adoption of Alternative Source Term Methodology," dated September 11, 2006 [SER correction letter: D. Collins (U. S. NRC) to C. Crane (Exelon Corporation), "Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station, Units 1 and 2 - Correction of Safety Evaluation for Amendment Dated September 11, 2006," dated September 28, 2006].

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

<u>REVISED TECHNICAL SPECIFICATIONS PAGES</u>

3.1.7 -3
3.4.3 -2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 40 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ ~~30.0~~ 45.0 atom percent B-10.	Prior to addition to SLC tank

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the safety valves are as follows: Number of Safety Valves — Setpoint (psig) 1 — ~~1135 ± 11.3~~ 2 — ~~1240 ± 12.4~~ 2 — ~~1250 ± 12.5~~ 4 — ~~1260 ± 12.6~~	In accordance with the Inservice Testing Program
SR 3.4.3.2	Verify each relief valve actuator strokes when manually actuated.	24 months
SR 3.4.3.3	-------------------NOTE-------------------- Valve actuation may be excluded. --- Verify each relief valve actuates on an actual or simulated automatic initiation signal.	24 months

Following testing, lift settings shall be within ± 1%.

1135 ± 34.1
1240 ± 37.2
1250 ± 37.5
1260 ± 37.8

ATTACHMENT 3
Markup of Technical Specifications Bases Pages
(For Information Only)

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 and 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

<u>REVISED TECHNICAL SPECIFICATIONS BASES PAGES</u>

B 3.1.7 -2
B 3.1.7 -3
B 3.4.3 -3
B 3.4.3 -5

BASES

APPLICABLE SAFETY ANALYSES (continued)	such that the required concentration is achieved accounting for dilution in the RPV with reactor water level at the high alarm point, including the water volume in the residual heat removal shutdown cooling piping, the recirculation loop piping, and portions of other piping systems which connect to the RPV below the high alarm point. This quantity of borated solution represented is the amount that is above the bottom of the boron solution storage tank. However, no credit is taken for the portion of the tank volume that cannot be injected. The SLC System satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the SLC System provides backup capability for reactivity control independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE; each contains an OPERABLE pump, an explosive valve, and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path. ~~With one subsystem inoperable the requirements of 10 CFR 50.62 (Ref. 1) cannot be met, however, the remaining subsystem is still capable of shutting down the unit.~~
APPLICABILITY	In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE when only a single control rod can be withdrawn.

(continued)

BASES (continued)

ACTIONS

A.1

and meet the requirement of Reference 1

because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability

If one SLC subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to shutdown the unit. However, the overall capability is reduced ~~since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1~~. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of shutting down the reactor and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) System to shut down the reactor.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

C.1

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1, SR 3.1.7.2, and SR 3.1.7.3

SR 3.1.7.1 through SR 3.1.7.3 are 24 hour Surveillances verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure that the proper borated solution volume and temperature, including the temperature of the pump

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

valves as well as the S/RV are assumed to function. [The opening of the relief valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events.] In these events, the operation of four of the five relief valves are required to mitigate the events. Reference 4 discusses additional events that are expected to actuate the safety and relief valves.

Safety and relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of nine safety valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 1). The safety valve requirements of this LCO are applicable to the capability of the safety valves to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The safety valve setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the UFSAR are based on these setpoints, but also include the additional uncertainties of
3% ± ~~1%~~ of the nominal setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

The relief valves, including the S/RV, are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

of one or more safety valves is inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the safety valves, including the S/RV, will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the safety valve and S/RV safety lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The safety valve and S/RV setpoints are ± ~~1%~~ 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

SR 3.4.3.2

The actuator of each of the Electromatic relief valves (ERVs) and the dual function safety/relief valves (S/RVs) is stroked to verify that the pilot valve strokes when manually actuated. For the S/RVs, the actuator test is performed by energizing a solenoid that pneumatically actuates a plunger located within the main valve body. The plunger is connected to the second stage disc. When steam pressure actuates the plunger during plant operation, this allows pressure to be vented from the top of the main valve piston, allowing reactor pressure to lift the main valve piston, which opens the main valve disc. The test will verify movement of the plunger in accordance with vendor recommendations. However, since this test is performed prior to establishing the reactor pressure needed to overcome main valve closure forces, the main valve disc will not stroke during the test.

(continued)

ATTACHMENT 5

GE-NE-0000-0053-8435-R1NP, "Dresden 2 & 3 and Quad Cities 1 & 2 Safety Valve Setpoint Tolerance Relaxation," May 2006 (NON-PROPRIETARY)

GE Nuclear Energy

GE-NE-0000-0053-8435-R1NP
Revision 1
DRF 0000-0042-5222
Class I
May 2006

Dresden 2 & 3 and Quad Cities 1 & 2 Safety Valve Setpoint Tolerance Relaxation

INFORMATION NOTICE

This is a non-proprietary version of the document GE-NE-0000-0053-8435-R1NP, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]]

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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ABSTRACT

This report summarizes the analysis results that support the operation of Dresden Units 2 and 3, and Quad Cities Units 1 and 2, with a setpoint tolerance increase from 1% to 3% for the Safety function of the Target Rock Dual Mode Safety Relief Valves and the Dresser Spring Safety Valves.

This report specifically addresses several analyses/subject areas that are sensitive to the valve setpoint tolerances. Other subjects that are insensitive to the valve setpoint tolerance change are not addressed in this report.

Several requirements were identified in this report in order to implement the setpoint tolerance increase. These requirements are summarized in the introduction section of the report.

1 INTRODUCTION

1.1 PURPOSE

Reference 1 presents a generic evaluation of the effects of increasing the setpoint tolerance of the Safety Relief Valves and identifies specific areas that should be evaluated on a plant specific basis. This report provides the results of the plant specific evaluations performed to assess the impact of the setpoint tolerance increase. These evaluations support the operation of Dresden Units 2 and 3 and Quad Cities Units 1 and 2 with an increase in the setpoint tolerance for the safety function of the Target Rock Dual Mode Safety Relief Valves (SRV) and the Dresser Spring Safety Valves (SSV) from 1% to 3%. The increase in setpoint tolerance includes both an increase in the upper limit of the setpoint tolerance as well as a decrease in the lower limit of the setpoint tolerance. The upper limit is defined as +3% and the lower limit is defined as –3%.

1.2 OVERALL EVALUATION APPROACH

The impact of the SRV setpoint tolerance increase on the following subjects is addressed in this report:

- Vessel Overpressure
- UFSAR Chapter 15 Events
- Anticipated Transients Without Scram (ATWS)
- LOCA
- Containment Response and Loads
- High Pressure Systems Performance
- Appendix R Fire Protection
- Vessel Thermal Cycle
- Operating Mode and Equipment Out Of Service (EOOS) Review
- Extended Power Uprate (EPU) Evaluation Review

These subjects are affected by the increase in valve setpoints associated with the setpoint tolerance change from 1 to 3%.

1.3 SUMMARY AND CONCLUSIONS

A summary of the results of the evaluations for each of the subjects of concern is provided in Table 1-1. The evaluation determined that the impact of the setpoint tolerance increase on the following subjects are acceptable: 1) Vessel Overpressure, 2) USAR Chapter 15 Events, 3) ATWS Analysis, 4) ECCS/LOCA Performance, 5) Containment Response and Loads Assessment, 6) High Pressure Systems Performance, 7) Appendix R Fire Protection, 8) Vessel Thermal Cycle, 9) Plant Operating Modes and EOOS, and 10) EPU Project Impact. These specific subjects were addressed in detail as described in this report.

Based on the results of the different analyses described in this report, several areas require further evaluation for implementation of the setpoint tolerance increase. The subjects that require additional evaluation are identified in Table 1-1 and will be addressed by Exelon before the implementation of the setpoint tolerance increase.

Table 1-1: Summary of Analyses Presented in this Report

Subject	Section	Result
Vessel Overpressure, Transient Analysis and Spring Safety Valve Margin	2.0	Acceptable[1]
ATWS Analysis	3.0	Acceptable[1]
ECCS/LOCA Evaluation	4.0	Acceptable
Containment Response and Loads Analysis	5.0	Acceptable[2]
High Pressure Systems Performance	6.0	Acceptable[3,4]
Appendix R Analysis	7.0	Acceptable
Vessel Thermal Cycle Assessment	8.0	Acceptable
Operating Modes and Equipment Out of Service Review	9.0	Acceptable
Emergent Extended Power Uprate Issues Review	10.0	Acceptable

1. These evaluations did not include any SRV OOS.
2. SRV Dynamic Loads will be assessed by Exelon to ensure the requirements described in section 5 are met.
3. MOV operation will be assessed by Exelon to ensure the requirements described in section 6 are met.
4. The Standby Liquid Control System performance will be assessed by Exelon to ensure the requirements described in section 6 are met.

2 VESSEL OVERPRESSURE/ANTICIPATED OPERATIONAL OCCURRENCE EVALUATION

2.1 ANALYSIS OVERVIEW

Reference 1 presents a generic evaluation of the effect of increasing the setpoint tolerance to +/- 3% for safety valves in the pressure relief system. This section presents the results of the plant specific evaluations associated with the increase of the setpoint tolerance of the safety valves from +/- 1 to +/- 3%. In this section Safety Valves (SV) are defined as valves that are qualified for use in the ASME overpressure protection analysis and include the spring safety valves (SSV) and the safety function of the Target Rock Dual Mode Safety Relief Valve (DSRV). In addition to the plant specific overpressure analysis, a plant specific review of the events in Chapter 15 of the FSARs was performed to determine if any other events are impacted by the setpoint tolerance increase. This review is summarized in Table 2-2. Based on the generic evaluation in Reference 1 and the review of the Chapter 15 events in Table 2-2, the overpressure analysis was evaluated with the safety valves at the +3 % limit and the Loss of Feedwater Event was reviewed with the safety valve setpoints at the –3% limit. All other events were determined to be unaffected by the change in setpoint tolerance.

2.2 OVERPRESSURE ANALYSIS

The most recent MSIVF transients, for Dresden Units 2 and 3 and Quad Cities Units 1 and 2, were analyzed with a 3% safety valve setpoint tolerance. The results of these analyses are provided in Table 2-1 below. These results demonstrate that the dome pressure safety limit (1345 psig) and the peak vessel pressure limit (1375 psig) are met when analyzed with a 3% setpoint tolerance. The Overpressure analyses were performed in accordance with the methodologies described in Reference 2.

Table 2-1: Overpressure Results with 3 % Setpoint Tolerance

Plant	Power (% Rated)	Flow (% Rated)	# SSVs Credited	#DSRVs Credited	Peak Dome Pressure (psig)	Peak Vessel Pressure (psig)	Basis
Dresden 2	102	108	8	0	1339	1365	Cycle 20 Reload Licensing Results
		95.3	8	0	1339	1361	
Dresden 3	102	108	8	0	1323	1351	Cycle 19 Reload Licensing Results
		95.3	8	0	1323	1348	
Quad Cities 1	102	108	8	1	1342	1366	Cycle 19 Reload Licensing Results *
		95.3	8	1	1340	1362	
Quad Cities 2	102	108	8	1	1339	1362	Cycle 18 Reload Licensing Results *
		95.3	8	1	1339	1360	

* The Quad Cities 1 and Quad Cities 2 results include the effects of the Acoustical Side Branch Modification.

2.3 REVIEW OF CHAPTER 15 EVENTS

The following table describes the impact of the setpoint tolerance on the events in Chapter 15 of the FSAR.

Table 2-2: Chapter 15 Event Descriptions

Increase in Heat Removal by the Reactor Coolant System	
Loss of FW Heater (LFWH)	
Manual Flow Control (MFC)	This transient results in a power increase due to increased core inlet subcooling. The increase in reactor power occurs at a moderate rate. No safety or relief valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Auto Flow Control (AFC)	MFC is more severe than AFC because AFC would limit the power increase. No safety or relief valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Feedwater Controller Failure	
Maximum Demand (FWCF)	This transient is similar to a Turbine Trip, however it is initiated at a higher power. This transient is analyzed on a reload specific basis for CPR as well as for pressure margin to the unpiped safety valve setpoints. [[ ]] Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Increase in Steam Flow	
Pressure Regulator Failure	
Upscale	This event results in a decrease in vessel pressure followed by a low pressure isolation. [[ ]] The vessel pressure increase is bounded by the Main Steam Isolation Valve Closure with direct scram which does not result in safety valve actuation. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Decrease in Heat Removal by the Reactor Coolant System	
Pressure Regulator Failure	
Downscale	Backup pressure regulator controls pressure. This event results in a small pressure change and power perturbation. No safety actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Load Rejection	
With Bypass (LRWBP)	Severity varies with BPV capacity and the results are bounded by the Load Rejection without Bypass event.
Without Bypass (LRNBP)	This transient results in a large vessel pressurization and increase in reactor power and is analyzed on a reload specific basis for CPR as well as for pressure margin to the unpiped safety valve setpoints. [[ ]] Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Turbine Trip	
With Bypass (TTWBP)	Severity varies with BPV capacity and the results are bounded by the Turbine Trip without Bypass event.

Without Bypass (TTNBP)	This transient results in a large vessel pressurization and increase in reactor power and is analyzed on a reload specific basis for CPR as well as for pressure margin to the unpiped safety valve setpoints. [[ ]] Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
MSIV Closure	
Direct Scram (MSIVD)	This transient is not limiting from a CPR perspective because of the slow steam flow shutoff rate associated with the MSIV stroke times. This transient is analyzed on a cycle specific basis to determine the pressure margin to unpiped spring safety valve setpoints. [[ ]] Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Flux Scram (MSIVF)	This transient is analyzed on a cycle specific basis to ensure that the ASME boiler code requirements and dome pressure tech spec. safety limits are met. The peak vessel pressure increases as the Safety Valve opening setpoints are increased. This transient has been analyzed using the upper bound of the 3 % tolerance for the spring safety valve opening setpoints and the safety mode of the dual mode relief valve opening setpoints.
Single MSIV Closure	This event is bounded by the MSIVD transient for peak pressure and is a non-limiting MCPR transient compared to other analyzed pressurization events. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Loss of Condenser Vacuum	This event is similar to a Turbine Trip event with no bypass, but there is a period of time where bypass valve flow is available. The duration of the bypass valve flow depends on the rate of loss of vacuum. Because of the limited bypass flow, the event is less severe than a turbine trip without bypass. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Loss of Auxiliary Power	This is a delayed turbine trip with recirculation pump trip. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Loss of Feedwater Flow (LOFW)	This transient results in a low level scram followed by a low-low level isolation. The transient is not limiting from a CPR perspective and because of the time delay between the scram and the MSIV closure, this event is far from limiting from an overpressure concern. This event does not result in safety valve actuations so the increased setpoint tolerance does not result in higher peak pressures. The Bernoulli effect on the L3 setpoint is not impacted by the setpoint tolerance change because the L3 setpoint is reached before any valve actuation occurs. The effect of the increased setpoint tolerance on the initiation of flow to the isolation condenser was also evaluated.
Decrease in Reactor Coolant System Flow Rate	
Trip of One Pump Motor	
Field Breaker	This event results in a pump coastdown and power decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Line Breaker	This event results in a pump coastdown and power decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Trip of All Recirc Loops	
Drive Motors	This event results in a flow coastdown and power decrease and may result in high Level Turbine Trip after a significant power decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.

Pump Motors	This event results in a flow coastdown and power decrease and may result in high Level Turbine Trip after a significant power decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Recirculation Flow Controller Malfunction	
Decreasing Flow	This event is similar to field breaker trip and results in a power decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Shaft Seizure	
Two Loop Operation	This event results in a rapid flow decrease which causes reactor power to decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Single Loop Operation	This event results in a rapid flow decrease which causes reactor power to decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Recirculation Pump Shaft Break	This event results in a rapid flow decrease which causes reactor power to decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Jet Pump Malfunction	The event results in very small change (decrease) to core flow which causes reactor power to decrease. No safety valve actuation occurs during the transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Reactivity and Power Distribution Anomalies	
Control Rod Withdrawal Error	
During Startup	This transient results in a power increase from very low powers. The increase in reactor power can occur at a high rate, but the neutron monitoring system is designed to limit the peak power achieved during the transient. The peak powers achieved are sufficiently low such that no safety valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
At Power	This transient results in a power increase due to increased reactivity associated with the control rod withdrawal. The increase in reactor power occurs at a moderate rate. The pressure regulator maintains vessel pressure and no safety valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Startup of an Inactive Recirc Loop	[[ ]] The pressure regulator maintains vessel pressure and no safety valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Flow Controller Failure – Increasing Flow	The rapid flow increase results in a power increase that occurs at a moderate rate. The pressure regulator maintains vessel pressure and no safety valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Slow Flow Runout	The slow flow runout transient is not an original Chapter 15 FSAR event, [[ ]] This event assumes a slow increase in recirculation flow rate in both loops from the minimum core flow to the maximum core flow. This analysis is a conservative process for evaluating flow runout events. The slow increase in core flow causes an increase in reactor power and corresponding increase in steam flow. The pressure regulator maintains vessel pressure and no safety valve actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Mislocated Fuel Assembly Accident	This scenario is modeled with a 3 dimensional core simulator code. The event does not result in increased pressure or safety valve actuation. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.

Misoriented Fuel Assembly Accident	This scenario is modeled with a 3 dimensional core simulator code. The event does not result in increased pressure or safety valve actuation. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Control Rod Drop Accident	This results in a very rapid increase in neutron flux and a corresponding increase in fuel temperature. A reactor scram terminates the transient. The pressure regulator maintains vessel pressure. No safety actuation occurs during this transient. Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Increase in Coolant Inventory	
Inadvertent HPCI	This event is analyzed on a reload specific basis for CPR and margin to unpiped SSV. This is an event where the HPCI system is inadvertently initiated. The increased core subcooling causes power to increase. [[ ]] It is possible that the inadvertent HPCI initiation could cause water level to increase to the Level 8 setpoint resulting in a turbine trip. This event is similar to the FWCF. In either case, no safety valve actuation occurs. [[ ]] Therefore, this transient is not impacted by the safety valve setpoint tolerance change.
Decrease in Reactor Coolant Inventory	
One RV/SV Opening	Event is not limiting with respect to MCPR or fuel duty because the event results in a very small power change. The event is analyzed for the highest single valve capacity and the highest single valve capacity is not changed with the safety valve setpoint tolerance change.
Instrument Line Break	These events are considered in the Loss of Coolant Analysis section of this report.
Steam Line Break Outside Containment	
LOCA Inside Containment	
Radioactive Release from a Subsystem or Component	
Liquid Release due to Tank Failure	These events are evaluated for radiological consequences and are not affected by the safety and relief valve setpoint tolerance increase.
Fuel Handling Accident	
Spent Fuel Cask Drop Accident	

2.4 REVIEW OF EQUIPMENT OUT OF SERVICE OPTIONS

In addition to the events in Chapter 15 of the FSAR, the following equipment out-of-service options were considered when determining the impact of the setpoint tolerance change:

1. Turbine Bypass OOS
2. Final Feedwater Temperature Reduction / Feedwater Heater(s) OOS
3. TCV(s) Slow Closure OOS
4. Single Recirc Loop
5. Power Load Unbalance OOS
6. TCV or TSV Stuck Closed
7. Pressure Regulator OOS
8. ADS OOS
9. MSIV Out of Service

Various combinations of equipment out-of-service options are allowed as described in Reference 3. These flexibility options are considered when performing critical power ratio and peak vessel pressure analyses. [[

]] Therefore, the increase in the safety valve setpoint tolerance does not impact the critical powers for the equipment out of service options listed above. The Turbine Bypass Out-of-Service option considers the effects of not meeting the fast response performance analyzed for the reload. It does not remove the ability of the pressure regulator to open the bypass valves in an attempt to maintain vessel pressure for slow events such as the rod withdrawal error or loss of feedwater heating where core power and steam flow may increase above the rated value.

For vessel overpressure calculations, the limiting event is the Main Steam Isolation Valve Closure with flux scram. This transient is evaluated from 102% of rated power at the high and low end of the rated power licensed core flow. The overpressure results are bounding for the equipment out-of-service options listed above.

The ADS system relies on the relief valves and is not impacted by the safety valve setpoint tolerance increase.

Therefore, the equipment out-of-service options listed above are not impacted by the valve setpoint tolerance increase.

2.5 MARGIN TO SPRING SAFETY VALVE SETPOINT

[[

]]

2.6 ISOLATION CONDENSER AND LOSS OF FEEDWATER FLOW

The loss of feedwater event relies on the reactor core isolation cooling system or the isolation condenser in order to maintain sufficient coolant inventory to ensure water level remains above Top of Active Fuel (TAF). This event was analyzed during the implementation of EPU. The results of the analysis were used to ensure that flow would initiate to the isolation condensers during the Loss of Feedwater Analysis. During the Loss of Feedwater event, only the lowest set of relief valves actuate. If additional relief valves opened, the reactor vessel pressure profile would be affected and the initiation of flow to the isolation condenser could also be affected.

The increase in setpoint tolerance only applies to the spring safety valves and the safety mode of the Target Rock Dual Mode S/RV. The relief valve setpoint tolerances remain unaffected. The increase in setpoint tolerance lowers the low end of the setpoint tolerance band for the Target Rock Dual Mode S/RV. Based on the nominal setpoint of 1135 psig for the Target Rock Dual Mode S/R, the low end of the 3 % setpoint tolerance is 1101 psig or 1115.7 psia. This is higher than the peak Reactor Pressure of 1099.9 psia in the previous LOFW analysis. Therefore, the Target Rock Dual Mode S/RV will not lift during the LOFW event with the 3% setpoint tolerance. Based on this information, the Loss of Feedwater event and associated initiation of flow to the isolation condenser is not impacted by the increase in setpoint tolerance.

3 ATWS EVALUATION

3.1 ANALYSIS OVERVIEW

This section describes the impact of the setpoint tolerance increase on the Dresden and Quad Cities ATWS analysis.

The ATWS analysis is performed in order to demonstrate that reactor integrity, containment integrity, and fuel integrity are maintained for scenarios where an automatic SCRAM fails to occur. Reactor integrity is demonstrated by ensuring that peak reactor vessel pressure is within the ASME Service Level C limit of 1500 psig. Containment integrity is demonstrated by ensuring that the peak suppression pool temperature is below the maximum bulk suppression pool temperature limit of 202°F and containment pressure is less than the containment design pressure limit of 62 psig. Fuel integrity is demonstrated by ensuring that peak cladding temperature is below the 10CFR50.46 limit of 2200°F and fuel local cladding oxidation is below the 10CFR50.46 limit of 17 % total clad thickness. Because the cladding temperature increase for ATWS is of short duration and limited magnitude, cladding oxidation is not explicitly calculated in the ATWS analysis.

[[

]] The ATWS analysis performed during the implementation of EPU demonstrated that all acceptance criteria listed above were met.

The increased setpoint tolerance associated with the Spring Safety Valves increases the upper analytical limit of the Spring Safety Valve setpoints. This increase in setpoint tolerance alone will tend to increase the peak vessel pressure during the ATWS events as well as the subsequent pressure peaks as Spring Safety Valves cycle to assist in maintaining vessel pressure. Both the upper limit (+3%) and the lower limit (-3%) of the setpoint tolerance band were considered. [[

]] The MSIVC and PRFO transients were re-evaluated using the updated design inputs summarized in Table 3-3. The MSIVC and PRFO events were re-analyzed at BOC and EOC

As part of the EPU ATWS analysis, the drywell temperature response was evaluated and a peak drywell airspace temperature of 311°F was calculated. The drywell temperature of 311°F is well below the equipment qualification temperature limit of 340°F. The drywell temperature analysis resulted in a peak drywell airspace pressure of 34.2 psig and a corresponding peak wetwell pressure of 38.4 psig at the bottom of the wetwell. This pressure is well below the 62 psig containment design limit and demonstrates that the peak containment pressure design limit is met. The EPU drywell temperature analysis resulted in a peak containment shell temperature of 280°F which is below the shell temperature limit of 281°F. [[

]] therefore, the peak drywell temperature analysis is not affected by an increase in the setpoint tolerance of the unpiped spring safety valves. Additionally, the electromatic relief valve delay times have been reduced from the values used during the EPU analysis. This will not impact the drywell temperature evaluation significantly, but would tend to increase the percentage of steam flow to the suppression pool. Therefore the drywell temperature evaluation performed in conjunction with the ATWS analysis during the implementation of EPU remains bounding for the increased setpoint tolerance. The suppression pool temperatures were evaluated as part of the analysis for the increased setpoint tolerance.

3.2 ANALYSIS INPUTS

Table 3-1 summarizes the initial conditions assumed for the ATWS event. These conditions are consistent with the initial conditions assumed for the ATWS analysis performed for the implementation of EPU.

Table 3-1: Summary of ATWS Key Input Parameters

Parameter	Value
Dome Pressure, psia	1020
Rated Core Flow, Mlbm/hr	98.0
Core flow, Mlbm / % of Rated	93.4 / 95
Rated Power, MWt	2957
Power, MWt / % of Rated	2957/100
Steam Flow, Rated, Mlbm/hr	11.71
Feedwater Temperature, °F	356
Initial Dynamic Void Reactivity Coefficient (EOC Value), ¢/%	-11.7 (BOC) -10.3 (EOC)
Core Average Void Fraction (EOC Value), %	49.5 (BOC) 36.2 (EOC)
Initial Doppler Coefficient (EOC Value), ¢/°F	-0.13 (BOC) -0.14 (EOC)
Initial Suppression Pool Liquid Volume (ft^3)	111,500
Initial Suppression Pool Temperature (°F)	98
Initial Suppression Pool Mass, Mlbm	6.916
Initial Inventory in CST, lbm	740,000
Initial Inventory in Condenser/Hotwell, lbm	476,000

Table 3-2 shows the initial axial power shapes for the beginning of cycle and end of cycle analyses. These axial power shapes are consistent with the initial axial power shapes in the ATWS analysis performed for the implementation of EPU. The ATWS analysis results are based on GE14 fuel. These analyses are applicable to the current Dresden and Quad Cities cores with GE14 reloads. A small amount of Legacy fuel remains in some cores, however GE14 fuel is the dominant fuel type.

Table 3-2: Axial Power Shapes

Node Location (From Bottom of Active Fuel)	BOC (2957 MWt/95% Flow)	EOC (2957 MWt/95% Flow)
1	0.37	0.14
2	1.28	0.43
3	1.60	0.50
4	1.68	0.56
5	1.66	0.65
6	1.60	0.76
7	1.53	0.88
8	1.46	1.00
9	1.40	1.11
10	1.34	1.20
11	1.28	1.27
12	1.21	1.33
13	1.14	1.37
14	1.05	1.40
15	0.88	1.28
16	0.81	1.32
17	0.77	1.40
18	0.70	1.44
19	0.63	1.45
20	0.55	1.41
21	0.46	1.29
22	0.36	1.08
23	0.15	0.47
24	0.08	0.26

Table 3-3 summarizes key equipment parameters and input values used in the ATWS analysis. For comparison, Table 3-3 also shows the input values used in the ATWS analysis performed for the implementation of EPU. In addition to the design inputs summarized in Table 3-3, the replacement steam dryer parameters were incorporated into the ATWS analysis as well as the Acoustic Side Branch Modifications. The inclusion of the replacement steam dryer and acoustic side branch modifications is conservative for the ATWS analyses presented in this section. The updated dryer parameters were based on a steam dryer D/P of 0.10 psid and a dryer weight of 100,200 lbm consistent with Reference 4 and the Acoustic Side Branch Modifications were based on a SRV inlet piping pressure drop of 11 psid for a flow rate of 644,543 lbm/hr.

Table 3-3: Key Equipment Parameters

Parameter	Original EPU Analysis	Re-Analysis
Nominal Closure Time of MSIV, sec	4.0	4.0
Relief Valve System Capacity, % NBR Steam Flow at 1120 psig / No. of Valves	18.4 / 4	18.4 / 4
Relief Valve Nominal Opening Setpoint Range, psig (Note 1)	1112, 1112, 1135, 1135	1115, 1115, 1135, 1135
Relief Valve Closing Setpoint, % of Opening Setpoint	96	93.2
Relief Valve Time Delay On Opening Signal, sec	1.85	0.677
Relief Valve Opening Stroke Time, sec	0.25	0.25
Relief Valve Closure Time Delay, sec	4.0	4.0
Relief Valve Closure Stroke Time, sec	10.0	10.0
Opening Delay for the 2 lowest setpoint relief valves on subsequent valve cycling, sec.	10.0	15.0
Safety/Relief Valve System Capacity, % NBR Steam Flow at 1125 psig / No. of Valves	5.3 / 1	5.3 / 1
Safety/Relief Valve Nominal Opening Setpoint, psig (Note 2)	1135	1135

Parameter	Original EPU Analysis	Re-Analysis
Safety/Relief Valve Closing Setpoint, % of Opening Setpoint	96	93.2
Safety/Relief Valve Time Delay On Opening Signal, sec	0.4	0.4
Safety/Relief Valve Opening Stroke Time, sec	0.15	0.25
Safety/Relief Valve Closure Time Delay, sec	0.4	0.4
Safety/Relief Valve Closure Stroke Time, sec	10.0	10.0
Safety Valve System Capacity, % NBR Steam Flow at 1240 psig / No. of Valves	44.1 / 8	44.1 / 8
Safety Valve Nominal Opening Setpoint, psig	1240, 1240, 1250, 1250, 1260, 1260, 1260, 1260	1240, 1240, 1250, 1250, 1260, 1260, 1260, 1260
Safety Valve Setpoint Tolerance, %	1	3
Safety Valve Closing Setpoint, % of Opening Setpoint	96	96
Safety Valve Opening Stroke Time, sec	0.3	0.3
Safety Valve Closure Stroke Time, sec	0.3	0.3
Recirc Pump Trip Logic Delay and Time Constant, sec	0.60	0.60
SLCS Injection Location: Lower Plenum Standpipe	Yes	Yes
Number of SLCS Pumps	2	2
SLCS Injection Rate per Pump, gpm	40	40
Nominal Boron-10 Enrichment, %	19.8	19.8
Sodium Pentaborate Concentration, %	14	14
Boron Injection Initiation Temperature (BIIT), °F	110	110
SLCS Liquid Transport Time, sec	60	60

Parameter	Original EPU Analysis	Re-Analysis
SLCS Liquid Solution Enthalpy, Btu/lbm	78	78
Time to Inject Hot Shutdown Boron Weight, sec	1073	1138
HPCI Flow Rate, gpm	5000	5000
Enthalpy of the HPCI Flow, Btu/lbm	103	103
ATWS High Pressure Setpoint, psig	1250	1250
Low Pressure Isolation Setpoint, psig	825	785
Number of RHR Loops	2	2
Number of RHR Loops for LOOP event	1 (Note 3)	1
RHR Service Water Temperature °F	98	98
RHR Heat Exchanger K-Factor per Loop in Containment Cooling Mode, Btu/sec- °F	343	343
RHR Heat Exchanger K-Factor per Loop during Loss of Offsite Power, Btu/sec- °F	343 (Note 3)	343(Note 4)

Notes:

[[

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3. EPU analyses assumed that one RHR heat exchanger was available with K-Factor of 343 Btu/Sec-°F. If suppression pool temperature limit was exceeded, the number of RHR loops available becomes 2 with a reduced K Factor of 252 Btu/Sec-°F.

4. The Re-Analysis for the SRV Tolerance Change assumes only one RHR heat exchanger is available with a K factor of 343 Btu/Sec-°F.

3.3 ANALYSIS RESULTS

The ATWS analysis yielded similar results to previous ATWS analyses. The ODYN results from this analysis are summarized in Table 3-4 below. The suppression pool temperature, suppression pool airspace pressure and integrated valve flows are shown in Table 3-5. A sequence of key events was developed for each of the transients analyzed. These are provided in Tables 3-6 through 3-9. Table 3-10 shows the ATWS acceptance criteria and the applicable limiting results. Plots of key ODYN outputs were generated for each of the transient analyzed and these are provided in Figures 3-1 through 3-12. Finally, plots of suppression pool temperature and suppression pool airspace pressure verses time are provided for the MSIVC and PRFO transients at end of cycle in Figures 3-13 through 3-14.

Table 3-4: Summary of Key ODYN Parameters for ATWS Calculation

Event	Power (MWt) /Flow (%)	Exposure	Peak Neutron Flux (%)	Peak Heat Flux (%)	Peak Vessel Press. (Psig)
MSIVC	[[				
MSIVC					
PRFO					
PRFO					]]

Notes: Values in the parentheses represent the time of peak values in seconds

The peak neutron and heat fluxes are normalized to the respective initial power of the individual cases.

Table3-5: Summary of Peak Suppression Pool Temperature, Containment Pressure and Integrated SRV Flow

Event	Power (MWt) /Flow (%)	Exposure	Peak Suppression Pool Temperature, °F	Peak Suppression Pool Airspace Pressure, psig	Integrated SSV and RV* Flow at Hot Shutdown, lbm
MSIVC	[[				
MSIVC					
PRFO					
PRFO					]]

Note: Values in the parentheses represent the time of peak values in seconds.

Values in the brackets represent the hot shutdown time in seconds. The hot shutdown in ODYN ATWS evaluation is defined as neutron flux less than 0.1% for more than 100 seconds.

Table 3-6: Sequence of Events for MSIVC at BOC

Event	Time (s)
MSIV Isolation Initiates	[[
MSIVs Closed	
Peak Neutron Flux [[ ]]	
Opening of the First Relief Valve	
High Pressure ATWS Setpoint	
Recirculation Pumps Tripped	
Peak Heat Flux Occurs [[ ]]	
Peak Vessel Pressure [[ ]]	
BIIT Reached	
Feedwater Reduction Initiated	
SLCS Pumps Start	
Boron Solution Reaches Lower Plenum	
HSBW Injected and Water Level Ramped up	
Peak Suppression Pool Temperature [[ ]]	
Water Level Restored to Normal Band	
Hot Shutdown Achieved (Neutron flux below 0.1% for more than 100 seconds)	]]

Table 3-7: Sequence of Events for MSIVC at EOC

Event	Time (s)
MSIV Isolation Initiates	[[
MSIVs Closed	
Peak Neutron Flux [[ ]]	
Opening of the First Relief Valve	
High Pressure ATWS Setpoint	
Recirculation Pumps Tripped	
Peak Heat Flux Occurs [[ ]]	
Peak Vessel Pressure [[ ]]	
BIIT Reached	
Feedwater Reduction Initiated	
SLCS Pumps Start	
Boron Solution Reaches Lower Plenum	
HSBW Injected and Water Level Ramped up	
Peak Suppression Pool Temperature [[ ]]	
Water Level Restored to Normal Band	
Hot Shutdown Achieved (Neutron flux below 0.1% for more than 100 seconds)	]]

Table 3-8: Sequence of Events for PRFO at BOC

Event	Time (s)
Turbine Control and Bypass Valves Start Open	[[
MSIV Closure Initiated by Low Steamline Pressure	
Peak Neutron Flux [[ ]]	
MSIVs Closed	
Opening of the First Relief Valve	
High Pressure ATWS Setpoint Tripped	
Recirculation Pumps Tripped	
Peak Heat Flux Occurs [[ ]]	
Peak Vessel Pressure [[ ]]	
BIIT Reached	
Feedwater Reduction Initiated	
SLCS Pumps Start	
Boron Solution Reaches Lower Plenum	
HSBW Injected and Water Level Ramped up	
Peak Suppression Pool Temperature [[ ]]	
Water Level Restored to Normal Band	
Hot Shutdown Achieved (Neutron flux below 0.1% for more than 100 seconds)	]]

Table 3-9: Sequence of Events for PRFO at EOC

Event	Time (s)
Turbine Control and Bypass Valves Start to Open	[[
MSIV Closure Initiated by Low Steamline Pressure	
MSIVs Closed	
Peak Neutron Flux [[ ]]	
Opening of the First Relief Valve	
High Pressure ATWS Setpoint Tripped	
Recirculation Pumps Tripped	
Peak Heat Flux Occurs [[ ]]	
Peak Vessel Pressure [[ ]]	
BIIT Reached	
Feedwater Reduction Initiated	
SLCS Pumps Start	
Boron Solution Reaches Lower Plenum	
HSBW Injected and Water Level Ramped up	
Peak Suppression Pool Temperature [[ ]]	
Water Level Restored to Normal Band	
Hot Shutdown Achieved (Neutron flux below 0.1% for more than 100 seconds)	]]

Table 3-10: Acceptance Criteria Results

Acceptance Criteria	Allowed Value	Limiting Result	ATWS Event and Conditions
Peak vessel pressure (psig)	1500	1478	[[ ]]
Peak cladding temperature (°F)	2200	Not Calculated (1)	N/A
Peak suppression pool temperature (°F)	202	191	[[
Peak containment pressure (psig)	62	38.4 (2)	]]

Notes:

(1) Not Calculated based on the significant margin to the allowable value for the EPU ATWS analysis.

(2) The peak containment pressure is based on the SHEX calculated pressure at the bottom of the wetwell from the EPU drywell temperature analysis.

Table 3-11: Peak Pressures for Other System Evaluations

Parameter	Value	Elevation	Comments
Lower Plenum Pressure	1301 psig	(Pressure at 152 inches above vessel 0)	The lower plenum pressure for all transients was reviewed and compared to the initiation time of the SLCS pumps. 1301 psig is the highest lower plenum pressure that occurs after the initiation of the SLCS pumps. This pressure is based on an elevation of 152 inches above vessel 0. In addition to the PRFO and MSIVC transients, the LOOP transient was considered for the evaluation of the lower plenum pressure. The LOOP transient resulted in the limiting lower plenum pressure during the time when SLCS was operating. However, it is noted that there was less than 5 psi difference between the peak lower plenum pressure for all events.
Downcomer Pressure	1469 psig - GE14	(Pressure at 309 inches above vessel 0)	The pressure in the downcomer is calculated by ODYN. These pressures represent the peak pressure in the downcomer for all ATWS transients. The enthalpy of the fluid in the downcomer varies around the time of peak downcomer pressure from 535 BTU/lbm to 570 BTU/lbm. The peak pressure value is based on the PRFO transient at BOC.

Figure 3-1: MSIVC - BOC – GE14 Fuel

[[

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Figure 3-2: MSIVC - BOC – GE14 Fuel

[[

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Figure 3-3: MSIVC - BOC – GE14 Fuel

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Figure 3-4: MSIVC - EOC – GE14 Fuel

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Figure 3-5: MSIVC - EOC – GE14 Fuel

[[

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Figure 3-6: MSIVC - EOC – GE14 Fuel

[[

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Figure 3-7: PRFO - BOC – GE14 Fuel

[[

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Figure 3-8: PRFO - BOC – GE14 Fuel

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Figure 3-9: PRFO - BOC – GE14 Fuel

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Figure 3-10: PRFO - EOC – GE14 Fuel

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Figure 3-11: PRFO - EOC – GE14 Fuel

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Figure 3-12: PRFO - EOC – GE14 Fuel

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Figure 3-13: Containment Response MSIVC EOC

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Figure 3-14: Containment Response PRFO EOC

[[

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3.4 CONCLUSIONS

The ATWS evaluation incorporating the 3% setpoint tolerance confirms that all ATWS acceptance criteria are met. Therefore, based on the current Dresden and Quad Cities core loadings, the implementation of this increased tolerance at the Dresden and Quad Cities units is acceptable. The ATWS evaluations are based on an 80 gallon per minute Standby Liquid Control System injection rate with a 14% weight concentration of Sodium Pentaborate solution containing naturally enriched Boron. The peak lower plenum pressure during the operation of the Standby Liquid Control System is 1301 psig. The Standby Liquid Control system is required to attain an equivalent Boron injection rate with a lower plenum pressure up to 1301 psig in order for these analyses to remain valid for the increased setpoint tolerance. The Standby Liquid Control System performance is addressed in Section 6.4. The following recommendation remains applicable:

> It is necessary for HPCI suction to be from the CST when suppression pool temperature exceeds the HPCI qualification limit even if suppression pool water level exceeds the automatic high level alarm setpoint

4 ECCS/LOCA EVALUATION

4.1 ECCS/LOCA PERFORMANCE EVALUATION

Reference 5 provides the results of the Loss-of-Coolant-Accident Analysis (LOCA) performed by GE Nuclear Energy for Dresden and Quad Cities Station (D/Q). The analysis was performed using the SAFER/GESTR-LOCA application methodology approved by the Nuclear Regulatory Commission (NRC).

The impact of safety valve setpoint relaxation on the ECCS-LOCA performance for BWR 2-6 plants has been evaluated on a generic basis in the BWROG report approved by the NRC [Reference 1, NEDC-31753P]. The ECCS conclusions contained in Reference 1 apply to Reference 5.

[[

]] As such, plant-specific evaluations of ECCS performance and the impact of safety valve set point relaxation on LOCA Licensing Basis PCT are not required.

5 CONTAINMENT RESPONSE AND LOADS ANALYSIS

This section presents the results of the various containment related evaluations in support of the safety relief valve setpoint tolerance increase from 1% to 3% for Dresden Units 2 & 3 and Quad Cities Units 1 & 2.

5.1 CONTAINMENT PRESSURE AND TEMPERATURE FOR DBA LOCA

The effects on the peak containment pressure and temperature response for the short-term DBA LOCA event and on the peak suppression pool temperature and wetwell pressure for the long-term DBA LOCA from implementation of EPU and replacement dryer Reference 4 for Quad Cities were considered. Relaxation of the SSV and SRV safety valve setpoint tolerance has no effect on the DBA LOCA event because the vessel depressurizes without any ERV, SRV or SSV actuations. Therefore, there is no impact on the DBA LOCA containment pressure and temperature and on the DBA LOCA suppression pool temperature and wetwell pressure from implementation of EPU. The inputs of containment pressure and suppression pool temperature to the available NPSH analysis from implementation of EPU are also unaffected. The same conclusions as above are applicable for the future dryer replacement at Dresden.

5.2 SMALL STEAM LINE BREAKS

Small steam line break (SLB) spectrum (0.01, 0.10, 0.30 and 0.75 ft^2 breaks) was evaluated for EPU implementation to determine the drywell temperature for generating the EQ curve. The larger SLBs that produce the most limiting peak drywell temperature are however, large enough to maintain the initial vessel pressure below the ERV, SRV and SSV setpoints and also large enough to depressurize through the break without requiring ERV, SRV or SSV actuation. Therefore, an increase in SSV or SRV safety valve opening setpoint has no effect on the larger SLBs that do not have ERV, SRV or SSV actuation. For smaller SLBs events, the ERV and SRVs can actuate, however, the reactor pressure is maintained below the SSV setpoints. The drywell temperature response for smaller SLBs that require ERV and SRV actuation may be slightly affected. For these breaks, the peak drywell temperature is well below that of the larger limiting SLB. Furthermore, the peak drywell temperature for the smaller SLBs occurs later in the event at the time the drywell sprays are actuated. Since this time occurs after many ERV and SRV actuations the peak temperature is controlled by the integrated steam flow to the drywell which is not affected by the change in the SSV and SRV safety valve setpoint tolerance. The long-term drywell temperature, after the sprays are initiated, is controlled by the break steam mass flow to the drywell and the spray temperature. The drywell spray temperature is controlled by the suppression pool temperature that is mainly governed by energy transferred to the suppression pool through the ERV and SRV. The rate of ERV and SRV energy transfer to the suppression pool is controlled by the vessel depressurization rate (assumed at 100°F/hr), the initial vessel liquid inventory, and decay heat. These factors are not affected by the changes to the SRVs. The break steam flow to the drywell is controlled by the vessel pressure response, which is determined by the assumed vessel depressurization rate of 100°F/hr. This parameter is also unaffected by the change in the SSV and SRV safety valve setpoint tolerance. Since the steam break flow and drywell spray temperature response for the smaller SLBs are not impacted by the SRV changes, the drywell temperature response for the smaller SLBs is also not impacted. Therefore, an increase in SSV and SRV safety valve setpoint tolerance has no impact on the drywell temperature response and the EQ curve from EPU implementation remains valid. In addition, the same conclusions as above are applicable for the dryer replacement in Reference 4 for Quad Cities and future dryer replacement for Dresden.

5.3 IBA AND SBA

The impact on intermediate and small break accidents (i.e., IBA and SBA) from EPU implementation was also evaluated. The containment pressure and temperature response for the IBA (a liquid line break of 0.1 ft^2) and the SBA (a steam line break of 0.01 ft^2), were originally evaluated as part of the Mark I Containment Program and documented in the Plant Unique Load Definition reports (PULD – References 6 and 7). The results for the IBA and SBA documented in References 6 and 7 are based on endpoint type calculations which are controlled by the amount of initial stored energy in the primary system and decay heat. There is no increase in the initial primary system stored energy or decay heat due to an increase in the SRV safety valve setpoint tolerance. Therefore, there is no impact on the IBA and SBA event results presented in References 6 and 7. Additionally, for the SBA the References 6 and 7 drywell temperature response is taken to be bounding, constant value of 340°F. This bounding drywell temperature value would not change due to an increase in SRV setpoint tolerance.

The EPU IBA and SBA analyses were performed using the GE SHEX containment code but with assumptions which are consistent with the References 6 and 7 analyses. [[

]] Because of the assumption on vessel depressurization used for the IBA and SBA analyses, the changes to the SRV safety valve setpoint tolerance would have no impact on the results of the EPU calculations for the IBA and SBA.

In addition, the same conclusions as above are applicable for the dryer replacement in Reference 4 for Quad Cities and future dryer replacement for Dresden.

5.4 NUREG-0783 LOCAL SUPPRESSION POOL TEMPERATURE

Quad Cities and Dresden have quenchers on the ERV and SRV discharge lines which, per Reference 9 ensures stable condensation at elevated local suppression pool near saturation. The NRC Safety Evaluation for Reference 9 (see Reference 8), conditionally approved elimination of the NUREG-0783 local pool temperature limits based on the Reference 9 evaluation. Per Reference 8, the local pool temperature limits of NUREG-0783, and associated evaluations, can only be eliminated if plants have pump suction inlets below the elevation of the quencher. This condition was imposed to address NRC concerns regarding steam ingestion of SRV steam into pump suction inlets at high local suppression pool temperature. An evaluation in 2001 determined that steam ingestion into the ECCS suction strainers will not occur for the Dresden and Quad Cities plants with the existing SRVs. This evaluation used the SRV flow capacity for the existing Target Rock SRV. Using the parameters for the Target Rock SRV provided as input by Exelon, and considering a 3% tolerance on the SRV safety valve opening setpoint pressure, [[

]] Therefore, the conclusions from the 2001 evaluation remain valid in that steam ingestion is not predicted.

In addition, the same conclusions as above are applicable for the dryer replacement in Reference 4 for Quad Cities and future dryer replacement for Dresden.

5.5 DBA LOCA HYDRODYNAMIC LOADS

The DBA LOCA hydrodynamic loads, such as pool swell, vent thrust, condensation oscillation and chugging are dependent on the containment pressure and temperature response during the DBA LOCA. Because the containment DBA LOCA pressure and temperature response are not affected by an increase in the SSV and SRV safety valve setpoint tolerance, the DBA LOCA hydrodynamic loads are also unaffected.

In addition, the same conclusions as above are applicable for the dryer replacement in Reference 4 for Quad Cities and future dryer replacement for Dresden.

5.6 SRV DYNAMIC LOADS

The SRV safety valve setpoint tolerance increase has no effect on the ERVs since they are not the ASME code safety valves.

The SRV discharge loads are determined by the following controlling parameters:

- SRV discharge line (SRVDL) and containment geometry
- Water leg length in the SRVDL at the time of SRV opening
- SRV flow capacity and SRV opening pressure

Since a relaxed SRV setpoint tolerance can increase the SRV safety valve opening pressure, the SRV discharge dynamic loads are expected to increase. Exelon will need to evaluate the SRV discharge dynamic loads with the SRV setpoint tolerance increase.

5.7 RIPD EVALUATION

During normal operation, there is no SRV actuation. Therefore, the SRV setpoint tolerance change have no effect on the reactor internal pressure differences (RIPDs) at normal conditions in Reference 4 and as shown for EPU.

For upset conditions, any event in which SRVs will actuate would have a faster depressurization due to increased SRV flow as a result of SRV setpoint tolerance change, causing higher DPs across the reactor internals. [[

]] Therefore, the RIPD results at upset conditions in Reference 4 and for EPU remain valid for the SRV setpoint tolerance relaxation.

The limiting emergency event used for RIPD is an inadvertent actuation of all ADS valves. Increased SRV flow capacity as a result of SRV setpoint tolerance change would have a faster depressurization and thus would result in higher DPs for reactor internals. [[

]] Thus, the RIPD results at emergency conditions in Reference 4 and for EPU are still applicable for the SRV setpoint tolerance increase.

The limiting faulted event for RIPD is an instantaneous circumferential break of one main steam line, for which SRV does not actuate. Therefore, the SRV setpoint tolerance relaxation has no effect on the RIPD results at faulted conditions in Reference 4 and for EPU.

As part of RIPD, the analyses for acoustic and flow-induced loads on jet pump, core shroud and shroud support due to recirculation line break are not affected by SRV setpoint tolerance relaxation because the SRVs will not actuate during the event. Therefore, the SRV setpoint tolerance increase does not impact the acoustic and flow-induced load analyses for EPU.

6 HIGH PRESSURE SYSTEMS PERFORMANCE

This section summarizes the evaluation of high pressure systems, as well as the performance of systems such as pressure control and piping.

6.1 HIGH PRESSURE COOLANT INJECTION

The purpose of the High pressure Coolant Injection (HPCI) systems at Quad Cities Units 1 and 2 and Dresden Units 2 and 3 is to provide high pressure emergency cooling water to the reactor to prevent excessive peak fuel clad temperature (PCT) following small line breaks that do not result in rapid depressurization. It operates to perform this function in conjunction with the Core Spray (CS) or Low Pressure Coolant Injection (LPCI) systems, and with credit for operation of the Automatic Depressurization System (ADS). The HPCI system also functions as a backup to the Reactor Core Isolation Cooling (RCIC) system at Quad Cities, or the Isolation Condenser (IC) at Dresden, in case of a failure of those systems following a transient event. To achieve this purpose, the HPCI system is designed to supply makeup water to the reactor at a capacity of 5600 gpm over a reactor pressure range of 1135 psia to 165 psia.

The maximum reactor operating pressure for rated makeup flow for the HPCI system at both Dresden and Quad Cities is based on the upper analytical limit (UAL) of the lowest group of relief valves (RVs), on condition that that this group contains a sufficient number of valves to provide the long-term relief function, even allowing for another independent failure within the RVs.

Dresden and Quad Cities both use two reactor relief valves at the lowest RV group setpoint. It was confirmed during EPU that operation of only one RV is needed for the long-term pressure relief function. Thus, the maximum reactor pressure for HPCI system water makeup operation is based on the upper analytical setpoint for the lowest group of RVs. For EPU, this corresponds to a pressure of 1115 psig.

These RVs are not within the group of valves that are receiving a setpoint tolerance increase. Therefore, the RV setpoints are not changing and there is no effect on the HPCI system maximum reactor injection pressure due to the SRV setpoint tolerance increase.

The HPCI system steam supply line contains break detection instrumentation designed to detect high steam flow, indicative of a break in that line. The isolation setpoints for this instrumentation are based on a differential pressure across the flow sensing device. Because the reactor vessel pressure for HPCI system operation remains the same, there will be no increase in rated steam flow to the turbine, and therefore, no effect on the break detection instrumentation or the trip setpoints.

The HPCI steam line containment isolation motor-operated valves (MOVs) are normally open with the system in standby. At Quad Cities and Dresden they are evaluated to be capable of closing against a differential pressure of approximately 1147 psid. This closing differential pressure is based on the current SRV nominal setpoint of 1135 psig and a 1% setpoint tolerance. A change to a 3% setpoint tolerance will increase the upper analytical limit to 1169.1 psig.

Therefore, Exelon will assure that the HPCI steam line MOVs are evaluated to operate acceptably with a reactor vessel pressure of 1169.1 psig prior to implementation.

The HPCI system injection valve is normally closed and is signaled to open during a system initiation. Since the HPCI system is designed for injection based on the RV setpoint, which is not changing, the injection valve is not affected by the SRV setpoint tolerance change.

It is concluded that the SRV setpoint tolerance increase will have no effect on the capability of the HPCI system to provide makeup water to the reactor vessel. The SRV setpoint tolerance change will affect both the Quad Cities and Dresden HPCI steam line MOVs with respect to the maximum closure differential pressure. The maximum closure pressure will increase to 1169.1 psig.

6.2 REACTOR CORE ISOLATION COOLING

The purpose of the RCIC system is to provide cooling water to the Quad Cities Unit 1 or Unit 2 reactor in the event that the reactor becomes isolated from the main condenser simultaneously with a loss of the feedwater system. To achieve this purpose, the RCIC system is designed to supply makeup water to the reactor at a capacity of 400 gpm over a reactor pressure range of 1135 psia to 165 psia.

The maximum reactor operating pressure for water makeup for the RCIC system at the Quad Cities plant is based on the upper analytical limit of the lowest group of relief valves (RVs), providing that this group includes a sufficient number of valves to provide the long-term relief function and there are allowances for another independent failure within the RVs.

The Quad Cities plant uses two RVs in the lowest group of reactor relief valves. For EPU, it was determined that operation of only one RV is needed for the long-term pressure relief function. Thus, the maximum reactor pressure for RCIC system water makeup operation is based on the upper analytical setpoint for the lowest group of RVs. For EPU, this corresponds to a pressure of 1115 psig.

These RVs are not within the group of valves that are receiving a setpoint tolerance increase. Therefore, the RV setpoints are not changing and there is no effect on the RCIC system maximum reactor injection pressure due to the SRV setpoint tolerance increase.

The RCIC system steam line contains break detection instrumentation designed to detect high flow in the line indicative of a break in that line. The isolation setpoints for this instrumentation are based on a differential pressure across the flow sensing device. Because the reactor vessel pressure for RCIC system operation remains the same, there will be no increase in steam flow and is no effect on the break detection instrumentation and the trip setpoints.

The RCIC steam line containment isolation motor-operated valves (MOVs) are capable of closing against a differential pressure of 1147 psid (MO-1301-16) and 1146 psid (MO-1301-17). The closing differential pressure is based on an SRV nominal setpoint of 1135 psig and a 1% setpoint tolerance (1147 psig). The high energy line break (HELB) maximum differential pressure for the MOVs is also based on an upstream pressure of 1147 psig. The SRV has a nominal setpoint of 1135 psig. A change to a 3% setpoint tolerance will increase the upper analytical setpoint to 1169.1 psig. Therefore, Exelon will assure that the RCIC steam line MOVs are evaluated to operate acceptably with a reactor vessel pressure of 1169.1 psig for normal closure and for the HELB closure analysis prior to implementation.

The RCIC system injection valve is normally closed and is signaled to open during a system initiation. Since the RCIC system is designed for injection based on the RV setpoint, and the RV setpoint tolerance is not changing, the injection valve is not affected by the SRV setpoint tolerance change.

It is concluded that the SRV setpoint tolerance increase will have no affect on the capability of the RCIC system to provide makeup water to the reactor vessel. The SRV setpoint tolerance change will affect the RCIC steam line MOVs with respect to the maximum closure differential pressure. The maximum closure pressure (reactor vessel pressure) will increase to 1169.1 psig.

6.3 STEAM BYPASS PRESSURE CONTROL SYSTEM

6.3.1 Description

The purpose of this section is to evaluate the impact of the proposed main steam safety relief valve opening setpoint tolerance relaxation on the Steam Bypass Pressure Control System functionality and performance at both Dresden Units 2 & 3 and Quad Cities Units 1 & 2. This report will be summarized in an overall evaluation to support a Tech Spec change to increase the set point tolerance of the safety relief valves from 1% to 3%.

For this evaluation, each of the following was reviewed to determine affects (if any) the relaxation of Main Steam Safety Valve (MSSV) setpoint tolerances with respect to the Steam Bypass Pressure Control System function and performance.

- Safety Analysis Reports for both the Quad Cities 1 & 2 and the Dresden 2 & 3
- Extended Power Uprate (References 10 and 11); Sections 5.2, 5.2.1, 5.3.11, 5.3.13, and 7.3; also Figure 1-1 and Table 1-2.
- The most recent reload licensing analysis for Dresden Units 2 & 3 and the Quad Cities Units 1 & 2.

6.3.2 Inputs and Assumptions

Based on the Safety Analysis Reports for the Dresden and Quad Cities extended power uprate projects:

- the normal reactor operating pressure is 1005 psig,
- the rated vessel steam flow is 11.71 Mlb/hr,

- the bypass capacity of each of the Dresden Units is 33.5% of rated reactor steam flow, and
- the bypass capacity of each of the Quad Cities Units is 33.3% of rated reactor steam flow.

Based on the individual Dresden and Quad Cities reload-licensing analysis reviewed:

- the normal reactor operating pressures are 1005 psig,
- the vessel steam flows are 11.71 Mlb/hr,
- the bypass capacity credited in the transient analysis (single BPVOOS) for each of the Dresden Units is 29.8%, and
- the bypass capacity credited in the transient analysis (single BPVOOS) for each of the Quad Cities Units is 29.6%.
- By definition, the safety relief valves are not expected to relieve (lift) within the normal operating range.

6.3.3 Evaluation

The Steam Bypass Pressure Control System (SBPCS) is a normally operating system, which provides fast and stable responses to system disturbances related to steam pressure and flow changes and thereby controls reactor pressure within its normal operating range. SBPCS consists of the pressure regulation system, turbine control valve system and the steam bypass valve system.

The EPU evaluations for the SBPCS, summarized four (4) evaluations performed for the Steam Bypass Pressure Control System as follows:

a) EPU impact to system design basis controlling parameters. The rated steam bypass absolute flow rate does not change, but the increase in steam flow results in the reduced percentage of bypass capacity (i.e., the absolute bypass flow rate as expressed as a percentage of EPU reactor rated steamflow). The bypass capacity is sufficient to support operation of Dresden and Quad Cities at EPU conditions. *Setpoint tolerance relaxation affect on this evaluation – None, since the normal operating steam flow rates used as an input to the original EPU report are the same as the flow rates in the reload transient analyses (11.71 Mlbm/hr).*

b) EPU impact to control room operator instrumentation, setpoint adjustments, indications, alarms, and SBPCS controls. Minimal impact on equipment. Signal ranges and adjustment capabilities are adequate to support EPU. Pressure regulator setpoint adjustment is required (decreased) to maintain 1020 psia (1005 psig) steam dome pressure to account for the increase in main steam line pressure drop. *Setpoint tolerance relaxation affect on this evaluation – None, since the normal operating steam dome pressure is not changed as documented in the most recent reload transient analyses (1005 psig).*

c) Determine if bypass valve inlet pressure conditions are significantly changed due to the changes in the steam line pressure drop to the Turbine Stop Valves (TSV) and steam chest at EPU conditions. Steam passing capabilities of the bypass valves were not significantly impacted by EPU. *Setpoint tolerance relaxation affect on this evaluation - None, since the normal operating steam flow rate and steam dome pressure are not changed as documented in the most recent reload transient analysis (11.71 Mlbm/hr and 1005 psig respectively), the steam line pressure drop does not change.*

d) Determine if the transient performance of the SBPCS, operating under EPU conditions, are impacted for the evaluation of major transients such as main turbine- generator trip or main generator load rejection. The transient evaluations performed for events that require SBPCS operation determined that bypass capacities were adequate for the transient analysis to remain valid at EPU conditions. *Setpoint tolerance relaxation affect on this evaluation – The steam bypass capacity, calculated as 33.3% for Quad Cities and 33.5% for Dresden, was determined to be adequate for the transient analysis to remain valid for EPU conditions. The most recent reload transient analyses for both Dresden and Quad Cities only takes credit for bypass capacity of eight of the nine bypass valves to reflect a single bypass valve out of service (BPVOOS). Therefore, the bypass capacities credited for the transient analysis of 29.6% and 29.8% respectively, which are bound by the EPU capacities, are determined to be adequate.*

6.3.4 Conclusion

This evaluation concludes that the Steam Bypass Pressure Control System functional and performance requirements are not affected by the MSSV setpoint tolerance relaxation.

6.4 STANDBY LIQUID CONTROL SYSTEM

The Standby Liquid Control System (SLCS) is designed to shut down the reactor from rated power condition to cold shutdown in a postulated event in which all or some of the control rods cannot be inserted or during a postulated ATWS event. The SLCS accomplishes this function by pumping a sodium pentaborate solution into the vessel at a prescribed boron injection rate in order to provide neutron absorption and achieve a subcritical reactor condition.

The original performance design basis for the SLCS was that it must be capable of injecting the system design rated flow into the reactor vessel using a single SLC pump at a maximum reactor pressure equal to the SRV group with the lowest setpoint operating in the relief mode. This method has been superseded by the use of the maximum reactor vessel pressure occurring during the limiting ATWS event when the SLCS is in operation in consideration of NRC Information Notice 2001-13.

Exelon will ensure that the 10CFR50.62 requirement to inject 86 GPM of 13% sodium pentaborate solution, or the equivalent, plus the ATWS specific injection requirements stated in Section 3.0 of this report are met for injection against the maximum reactor vessel pressure of 1301 psig at the SLCS sparger occurring during an ATWS event when the SLCS is in operation without opening of the SLCS relief valve.

6.5 SAFE SHUTDOWN MAKUP PUMP SYSTEM

The purpose of the safe shutdown makeup pump (SSMP) system is to provide cooling water to the Quad Cities Unit 1 or Unit 2 reactor in the event that the reactor becomes isolated from the main condenser simultaneously with a loss of the feedwater system. This system was installed as a common backup to the Quad Cities Unit 1 and Unit 2 RCIC systems. To achieve this purpose, the SSMP system is designed to supply makeup water to the reactor at a capacity of 400 gpm over a reactor pressure range of 1135 psia (1120 psig) to 165 psia (150 psig), the same as the RCIC system.

The system consists of a single motor-driven pump designed for a flow rate of 400 gpm at 2885 feet. The system can pump to either Quad Cities Unit 1 or Unit 2. The SSMP injection valves are interlocked to allow injection to only one reactor at a time.

Because this system is installed as a backup to the RCIC system, it shares the same design basis with respect to the maximum reactor vessel pressure for injection. The RCIC system is capable of injecting makeup water to the reactor vessel up to a vessel pressure of 1120 psig (1135 psia). It has been determined that the lowest group of RVs are capable of maintaining reactor vessel pressure below the maximum design injection pressure of 1120 psig for long-term pressure reactor vessel pressure relief.

These RVs are not within the group of SRVs that are receiving a setpoint tolerance increase. Therefore, the RV setpoints are not changing and there is no effect on the SSMP system maximum reactor injection pressure due to the SRV setpoint tolerance increase.

It is concluded that the SRV setpoint tolerance increase will have no affect on the SSMP system.

6.6 ISOLATION CONDENSER SYSTEM

The Isolation Condenser (IC) system design basis is to provide reactor core cooling in the event that the reactor pressure vessel (RPV) becomes isolated from the main condenser by closure of the main steam isolation valves. This event concurrent with the loss of all feedwater flow (LOFW) by the loss of offsite power is the design transient for the IC system. This report evaluates the impact of SRV setpoint tolerance change on the IC system. The IC system applies to the Dresden plants only.

Automatic initiation of IC operation occurs when a high reactor pressure signal of 1068 psig exists for more than 15 seconds. The initiation setpoint and time delay are independent of the SRV setpoint and setpoint tolerance increase. The SRV has a setpoint of 1135 psig. For a 1% setpoint tolerance, the upper analytical setpoint is 1146.4 psig. For a 3% setpoint tolerance, the upper analytical setpoint is 1169.1 psig. Both of these setpoints are above the IC high reactor pressure initiation signal of 1068 psig. This setpoint exceeds the IC initiation setpoint. Thus, the SRV upper analytical setpoint tolerance change does not affect the IC initiation.

The lower analytical setpoint for the 3% tolerance change results in an SRV setpoint of 1101 psig. For the 1% setpoint tolerance, the lower analytical setpoint is 1123.6 psig. Refer to Section 2.6 in this report for discussion of the effect on the IC system initiation.

Those portions of the IC system interfacing directly with the reactor (the RCPB) are designed to 1250 psig and 575°F. The setpoint tolerance change will not increase the maximum reactor pressure following transient and accident events above the current limits. Consequently, the SRV setpoint tolerance increase will not impose changes to the design values for the IC system RCPB components.

The IC system motor-operated valves will not be affected by the SRV setpoint tolerance increase. The steam line isolation valves are maintained open during normal plant operation. The condensate return line isolation valves are maintained closed and must open to allow IC system operation. The reactor operating pressure is not increased; therefore, periodic testing of these valves is not affected. Since the differential pressure across the condensate return line valve will not change (the reactor vessel dome pressure acts equally on both sides of the valve), there is no effect on the valve opening capability due to the SRV setpoint tolerance increase.

The IC system steam line and condensate return line contain break detection instrumentation designed to detect high flow in the line indicative of a break in that line. The isolation setpoints for this instrumentation are based on a differential pressure across the flow sensing device. Because the design flow for the IC system remains the same, there is no effect on the break detection instrumentation and the trip setpoints for the SRV setpoint tolerance increase.

It is concluded that the SRV setpoint tolerance increase will have no affect on the IC system.

7 APPENDIX R ANALYSIS

This section provides an Appendix R fire protection safety evaluation for Quad Cities and Dresden SRV setpoint tolerance increase (safety mode from 1% to 3%).

7.1 VESSEL INVENTORY ASSESSMENT

Increased SRV setpoint tolerance to +3% will cause SRV actuation at higher pressure and thus result in a slight delay in the SRV actuation. Consequently, the instantaneous flow rates out of the SRVs are increased due to the higher critical flow rates in comparison to the case with SRVs at currently analyzed setpoint tolerance. However, the change in the total inventory lost from the vessel due to SRV setpoint tolerance relaxation is negligible. This is because the inventory loss is primarily dependent on the decay heat, which remains unaffected by SRV setpoint tolerance relaxation. In addition, the existing Target Rock SRV with the same capacity was only assumed for stuck open and SRV cycling in EPU evaluations. Therefore, the vessel water level responses and conclusions in the EPU evaluations are still applicable for +/-3% SRV safety valve setpoint tolerance change. Note that the inventory loss as a result of SORV during first 10 minutes is not affected by SRV setpoint relaxation because opening of SRV is caused by a fire at time initiation and closing of SORV is due to manual operator action, not by reaching SRV setpoint.

7.2 CONTAINMENT RESPONSE ASSESSMENT

The suppression pool temperature is mainly governed by energy transferred to the suppression pool through the SRVs. Before depressurization, the similar energy would be transferred to the suppression pool due to a net slightly increased SRV flow as a result of the SRV safety valve setpoint tolerance increase, balanced by less SRV cyclings caused by the +/-3% SRV setpoint tolerance change. After depressurization, the rate of SRV energy transfer to the suppression pool and total energy transfer to the suppression pool are controlled by the vessel depressurization rate (assumed at 100oF/hr), the initial vessel liquid inventory, and decay heat which are unaffected. Thus, the SRV setpoint tolerance change has no adverse impact on the suppression pool temperature, as well as containment temperature and pressure for an Appendix R fire event. Therefore, the containment response in the EPU evaluations are still applicable.

8 VESSEL THERMAL CYCLE ASSESSMENT

8.1 ANALYSIS OVERVIEW

Reactor Pressure Vessel Thermal Cycles: Safety Relief Valve blow down is a "thermal cycle" of the RPV and is counted over the life of the plant. The design basis allowable for SRV blow down is 5 (UFSAR Table 3.9-1). The elevated set point at which the SRV can lift may impact the fatigue usage of the RPV. The number of allowable SRV events was qualitatively reviewed considering the relaxation of their set point tolerance.

8.2 INPUTS AND ASSUMPTIONS

There are two transient pressure rise events and one pressure decrease event considered for the vessel thermal cycle design:

> The pressure rise events are the overpressure 1250 psig event and the overpressure 1375 psig event.
>
> The pressure decrease event is the single relief or safety valve blow down event.

The evaluation is based on a 2% increase in the opening set-point, relaxation from 1% to 3%, and it is assumed that this is within the RPV design pressure.

8.3 ANALYSIS RESULTS

There will be pressure and temperature oscillations during the overpressure events due to the SRV cycling. The temperature oscillation resulting from the SRV opening set-point change from 1% to 3% is within the design temperatures assumed for these events on the thermal cycle diagram.

The SRV opening set-point relaxation from 1% to 3%, will have limited effects on these three events. The set point increase does not have any effect on the pressure decrease event.

8.4 CONCLUSIONS AND RECOMMENDATIONS

It is recommended that the SRV blow-down events be limited to the 5 cycles already reported in the USAR. It is also recommended that the plant use a fatigue monitor program to review the number of cycles and accumulated fatigue usage.

9 OPERATING MODES AND EOOS REVIEW

For the purpose of this task, the review is based on any additional requirements imposed by the SRV setpoint tolerance increase that would affect a specific option. The SRV setpoint tolerance increase is evaluated as to its impact on various analyses that involve conditions or events that actuate the SRVs. All the options include analyses that actuate the SRV and therefore impact the basis for each option. The SRV setpoint tolerance increase is concluded to not impose any additional requirements on the operation and licensing basis for Dresden and Quad Cities, therefore the setpoint tolerance increase is entirely compatible with the Operating Modes and EOOS. Note that an SRV OOS has been previously evaluated, however, that option does not meet the over pressure criteria under EPU. This conclusion remains for the setpoint tolerance increase and the SRV OOS option remains unacceptable for Dresden and Quad Cities.

10 EMERGENT EPU ISSUES REVIEW

The specific areas identified as requiring direct evaluation as a result of the SRV setpoint tolerance increase are addressed by the separate tasks included in the scope. Other tasks are concluded to not be affected on the basis that the SRV setpoint tolerance increase has no impact on normal operating conditions and/or events that do not actuate the SRVs. For this review, the technical and licensing activities corresponding to the most recent EPU project are examined with respect to the SRV setpoint tolerance increase to determine if a risk exists with respect to the Dresden and Quad Cities EPU Basis. No specific areas of concern were identified in this review.

11 REFERENCES

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