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10 CFR 50.90

RS-06-087

June 2, 2006

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

Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

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 – Issuance of Amendments for Main Steam Safety Valve Technical Specifications (TAC Nos. MB6537 and MB6538)," dated July 30, 2004

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-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3. 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

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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 system from $\geq$ 30.0 atom percent boron-10 to $\geq$ 45 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.

This license amendment request satisfies one of the additional conditions (i.e., item 5) added to the Reference 3 amendment, which states "Regarding the issue of the main steam safety valve and safety/relief valve test data potentially exceeding the current Technical Specification SR 3.4.3.1 allowed tolerance limit, Exelon shall submit a Technical Specification amendment request to change this tolerance value to one derived from item 4 above, if necessary, and the results of revisions to all applicable design basis analyses, within six months of NRC approval of item 4."

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 DNPS 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 DNPS 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 June 4, 2007. Once approved, the amendment for DNPS Unit 3 shall be implemented within 60 days. This will allow adequate time for the affected station documents to be revised using the appropriate change control

mechanisms. For DNPS Unit 2, the amendment shall be implemented prior to unit startup from the Fall 2007 refueling outage, since implementation of the modification to transition to enriched sodium pentaborate will be completed during this refueling outage.

In accordance with 10 CFR 50.91(b), 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. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 2nd day of June 2006.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:
- Attachment 1: Evaluation of Proposed Change
- Attachment 2: Markup of Proposed Technical Specifications Pages
- Attachment 3: Markup of Technical Specifications Bases Pages
- 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

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-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3. 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 atom percent boron-10.

Each DNPS unit is designed with nine safety valves. Eight 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 report, 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 DNPS 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 as-found tolerances are used for determining operability. 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 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 DNPS 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."

In addition to the safety valves and S/RV, each unit is designed with four relief valves, which actuate in the relief mode to control reactor coolant system pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell.

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 relief valves and S/RV discharge steam through discharge lines to a point below the minimum water level in the suppression pool. The 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 safety valves are assumed to operate in the safety mode. The relief valves and the relief function of the S/RV are 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 safety valves is required to be operable to satisfy the assumptions of the safety analysis. The safety valve setpoints are established to ensure that the ASME Code limit for peak reactor pressure is satisfied. The transient evaluations in the Updated Final

Safety Analysis Report are based on these setpoints, but also include the additional lift setpoint tolerance uncertainties to provide an added degree of conservatism.

The use of +/- 1% allowable as-found MSSV lift setpoint tolerance in plant TSs was a generic issue in the industry. 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 DNPS, 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 met using a single SLC pump. 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 analyses described in Items 1 and 2 should assure that the number of MSSVs included in the analyses 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.

S/RV Dynamic Loads

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 2.2% 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 of 0.05 seconds.

The actual valve opening time used in the DNPS 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 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 owners' group of a program error that was discovered in 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 DNPS predate the error discovery, the current plant unique S/RV load analyses 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.

<u>MOV Operation</u>

The impact of changing the MSSV setpoint tolerance on MOVs was reviewed by DNPS Engineering. MOVs in the Main Steam Line Drain, Reactor Water Clean-Up, Isolation Condenser, and High Pressure Coolant Injection systems were impacted by the increased differential pressure. This review found some reduction 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.

<u>SLC System Performance</u>

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, to satisfy this requirement for DNPS, 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 the natural boron-10 isotopic enrichment.

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 met using a single SLC pump. 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 DNPS 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 not yet been implemented for either DNPS unit, since implementation is tied to upcoming refueling outages for each unit. 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.

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-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3. 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 atom percent boron-10.

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

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

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

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

EGC has evaluated the proposed change to the TS for DNPS Units 2 and 3, 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 a postulated anticipated transient without scram (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. 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 loss-of-coolant accident 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 Updated Final Safety Analysis Report. 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 DNPS 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 DNPS 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 a 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/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 DNPS 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 DNPS 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 the owner-established acceptance criteria 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 DNPS, 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 the natural 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 met using a single SLC pump. 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 DNPS TS to add SR 3.1.7.10, which requires verification that the sodium pentaborate enrichment is $\geq$ 30.0 atom percent boron-10. Although this amendment has been issued, it has not yet been implemented for either DNPS unit, since implementation is tied to upcoming refueling outages for each unit. 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.

In conclusion, based on the considerations discussed above, (1) there is 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 Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment needs 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

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

<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 <u>AND</u> 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 ± 11.4~~ 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

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

<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)

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.

ACTIONS

A.1

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

(continued)

BASES

ACTIONS

A.1 (continued)

and meet the requirements of Reference 1

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.

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

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 suction piping, are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure (Refs. 2 and 3, respectively), the relief 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 all 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 ~~± 1%~~ of the nominal setpoint drift to provide an added degree of conservatism.

± 3%

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.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With less than the minimum number of required safety valves OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the relief function of the inoperable relief valves cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, or if the relief function of two or more relief valves are inoperable, or if the safety function 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,

(continued)

General Electric Company

AFFIDAVIT

I, Louis M. Quintana, state as follows:

(1) I am Manager, Licensing, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GE proprietary report GE-NE-0000-0053-8435-R1P, *Dresden 2 & 3 and Quad Cities 1 & 2 Safety Valve Setpoint Tolerance Relaxation,* Revision 1, Class III (GE Proprietary Information), dated May 2006. The proprietary information is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions regarding GE processes and methodology supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability for relaxation of the safety valve setpoint tolerance for a GE BWR, utilizing analytical models and methods, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several hundred thousand dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 23rd day of May 2006

Louis M. Quintana

Louis M. Quintana
Manager, Licensing