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**PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390**

10 CFR 50.90

June 10, 2013

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

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: License Amendment Request – Increase the Safety Relief Valve/Safety Valve Lift Setpoint Tolerance, the Required Number of Operable Safety Relief Valves/Safety Valves, and the Standby Liquid Control System Discharge Pressure Value

In accordance with 10 CFR 50.90, Exelon Generation Company, LLC (Exelon) requests proposed changes that would modify Technical Specification (TS) Section 3.4.3 ("Safety Relief Valves (SRVs) and Safety Valves (SVs)") and Section 3.1.7 ("Standby Liquid Control (SLC) System"). The proposed changes: 1) revise TS Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found SRV and SV lift setpoint tolerance from ±1% to ±3%; 2) revise TS Limiting Conditions for Operation (LCO) 3.4.3 to increase the required number of operable SRVs and SVs from 11 to 12, and; 3) revise TS SR 3.1.7.8 to increase the SLC System pump discharge pressure from 1255 psig to 1275 psig.

The proposed changes have been reviewed by the Peach Bottom Atomic Power Station (PBAPS) Plant Operations Review Committee, and approved by the Nuclear Safety Review Board in accordance with the requirements of the Exelon Quality Assurance Program.

Exelon requests approval of the proposed amendments by June 10, 2014. Once approved, these amendments shall be implemented within 60 days of issuance.

Attachment 1 contains the evaluation of the proposed changes. Attachment 2 provides the marked up TS and Bases pages. The Bases pages are being provided for information only.

Attachment 3 contains information proprietary to General Electric Hitachi Nuclear Energy. General Electric Hitachi Nuclear Energy requests that the document be withheld from public disclosure in accordance with 10 CFR 2.390(b)(4). Attachment 4 contains a non-proprietary version of the General Electric Hitachi Nuclear Energy document. An affidavit supporting this request is contained in Attachment 5.

**Attachment 3 transmitted herewith contain Proprietary Information.**
**When separated from attachments, this document is decontrolled.**

In accordance with 10 CFR 50.91, Exelon is notifying the Commonwealth of Pennsylvania of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no commitments contained in this submittal.

Should you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 10th day of June 2013.

Respectfully,

James Barstow
Director, Licensing & Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1) Evaluation of Proposed Changes
2) Markup of Technical Specifications and Bases Pages
3) "Peach Bottom Atomic Power Station Units 2 and 3 Safety Valve Setpoint Tolerance Increase Safety Analysis Report," NEDC-33533P, Revision 1, May 2013 (Proprietary Version)
4) "Peach Bottom Atomic Power Station Units 2 and 3 Safety Valve Setpoint Tolerance Increase Safety Analysis Report," NEDO-33533, Revision 1, May 2013 (Non-Proprietary Version)
5) Affidavit

cc: USNRC Region I, Regional Administrator
USNRC Senior Resident Inspector, PBAPS
USNRC Senior Project Manager, PBAPS
R. R. Janati, Bureau of Radiation Protection
S. T. Gray, State of Maryland

**ATTACHMENT 1**
**Evaluation of Proposed Changes**

SUBJECT: Increase the Safety Relief Valve/Safety Valve Lift Setpoint Tolerance, the Required Number of Operable Safety Relief Valves/Safety Valves, and the Standby Liquid Control System Discharge Pressure Value

**ATTACHMENT 1**
**Evaluation of Proposed Changes**

## 1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (Exelon) requests an amendment to Renewed Facility Operating License Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The proposed changes: 1) revise Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found Safety Relief Valve (SRV) and Safety Valve (SV) lift setpoint tolerance from ±1% to ±3%; 2) revise TS Limiting Conditions for Operation (LCO) 3.4.3 to increase the required number of operable SRVs and SVs from 11 to 12, and; 3) revise TS SR 3.1.7.8 to increase the SLC System pump discharge pressure from 1255 psig to 1275 psig.

## 2.0 DETAILED DESCRIPTION

The proposed change includes the following TS revisions:

a) TS Section 3.4.3, "Safety Relief Valves (SRVs) and Safety Valves (SVs)," SR 3.4.3.1 – This SR is being revised to change the setpoint tolerance for the SRVs and SVs from ±1% to ±3%. Additionally, a change is being made to add the TS requirement that, following as-left testing, lift settings shall be within ±1%.

b) TS Section 3.4.3, "Safety Relief Valves (SRVs) and Safety Valves (SVs)," LCO – This LCO is being revised to increase the required number of operable SRVs and SVs from 11 to 12.

c) TS Section 3.1.7, "Standby Liquid Control (SLC) System," SR 3.1.7.8 – This SR is being revised to increase the discharge pressure of the Standby Liquid Control (SLC) System pumps from 1255 psig to 1275 psig.

Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Marked-up pages showing corresponding changes to the TS Bases are provided in Attachment 2 for information only. The TS Bases changes will be processed in accordance with the PBAPS, Units 2 and 3 TS Bases Control Program (TS 5.5.10).

## 3.0 TECHNICAL EVALUATION

The proposed change increases the allowable as-found SRV and SV lift setpoint tolerance from ±1% to ±3%. In support of this proposed change, the required number of operable SRVs/SVs will be increased from 11 to 12. This change does not alter the SRV/SV nominal lift setpoints. Additionally, a requirement has been added to ensure that, prior to placing new or refurbished valves into service, the valve as-left setpoints must be adjusted to be within ±1% of its nominal setting. This performance requirement has not changed. As an additional result of this change, the SLC System pump discharge pressure will be increased from 1255 psig to 1275 psig.

The ASME Boiler and Pressure Vessel Code, Section III, requires the reactor pressure vessel to be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs and SVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

**ATTACHMENT 1**
**Evaluation of Proposed Changes**

Eleven (11) valves (any combination of SRVs and SVs) are currently required to be operable by TS 3.4.3. In support of this proposed change, the required number of operable SRVs/SVs will be increased from 11 to 12.

The SRVs are Target Rock three-stage pilot operated safety/relief valves. The SVs are Dresser spring loaded safety valves. The SRVs and SVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The SRVs can actuate by either of two modes: the safety mode or the depressurization mode. In the safety mode, the pilot disc opens when steam pressure at the valve inlet expands the bellows to the extent that the hydraulic seating force on the pilot disc is reduced to zero. Opening of the pilot stage allows a pressure differential to develop across the second stage disc which opens the second stage disc, thus venting the chamber over the main valve piston. This causes a pressure differential across the main valve piston which opens the main valve. The SVs are spring loaded valves that actuate when steam pressure at the inlet overcomes the spring force holding the valve disc closed. This satisfies the Code requirement. The proposed changes do not impact the depressurization mode function of the SRVs.

Most Boiling Water Reactor (BWR) TSs originally required that the safety mode pressure setpoints for SRVs and SVs remain within ±1% tolerance band. It was subsequently identified that nuclear power plant licensees were experiencing difficulty in meeting the ±1% setpoint tolerance criterion. As a result, the BWR Owners' Group (BWROG) developed NEDC-31753P (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 U.S. Nuclear Regulatory Commission (USNRC) issued a safety evaluation approving NEDC-31753P (Reference 2). In the safety evaluation, the USNRC stated that a generic change of lift setpoint tolerance to 3% is acceptable provided that it is evaluated in the analytical bases. The USNRC 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 USNRC approved methods, of abnormal operational occurrences as described in NEDC-31753P utilizing a ±3% lift setpoint tolerance for the safety mode of the SRVs/SVs.

2. Analysis of the design basis overpressure event using the 3% tolerance limit for the SRV/SV 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 SRVs/SVs 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 SRV discharge lines and containment.

In support of the proposed TS changes, General Electric Hitachi Nuclear Energy (GEH) performed the plant specific analyses and evaluations described above, and the results are documented in Attachment 3 ("Peach Bottom Atomic Power Station Units 2 and 3 Safety Valve Setpoint Tolerance Increase Safety Analysis Report," NEDC-33533P, Revision 1, dated May 2013). This analysis addresses the effect of the SRV/SV setpoint tolerance on vessel overpressure, thermal limits, Anticipated Transient Without Scram (ATWS), Emergency Core Cooling Systems (ECCS) and Loss of Coolant Accident (LOCA), High Pressure Systems, and Vessel Thermal Cycle. It determined that the impact of the setpoint tolerance increase was acceptable; however, certain areas required further assessment by Exelon. These areas include:

1. The evaluations were performed or included one SRV/SV out of service. The analyses performed in NEDC-33533P utilized PBAPS, Unit 3 Cycle 18 as a representative operating cycle to evaluate the impact of the change in SRV/SV setpoint tolerance. For current operating cycles for both PBAPS units, the cycle specific reload analyses have been performed with acceptable results utilizing the ±3% setpoint tolerance with 1 SRV out of service, which bounds the current TS configuration (1% SRV/SV tolerance, 2 SRVs out of service).

2. Although the effect on SRV dynamic loads is acceptable, the effect on T-Quencher bubble pressure loads, discharge line piping loads, and line pressure loads may exceed the available margin. None of these effects were evaluated in terms of the PBAPS Licensing Basis in the Attachment 3 report.

3. Motor-operated valve (MOV) operation will be assessed by Exelon to ensure the requirements described in Section 6 of Attachment 3 are met.

4. High Pressure Coolant Injection (HPCI) System performance is still adequate, but the Reactor Core Isolation Cooling (RCIC) turbine would require a speed increase of 50 rpm to maintain the appropriate level of conservatism. The design documentation for both HPCI and RCIC should be revised to reflect the available analyzed design values, rather than the original assumed values.

5. The Standby Liquid Control (SLC) System performance will be assessed by Exelon to ensure the requirements described are met.

6. Technical Specification Section 3.1.7 will need to be revised to reflect the increased maximum SLC System pump discharge pressure value. A fatigue monitor program should be used to review the number of cycles and accumulated fatigue usage.

## ATTACHMENT 1
## Evaluation of Proposed Changes

1. One SRV/SV Out of Service

   As discussed in Attachment 3, the current PBAPS TS allows for up to two SRVs/SVs to be out-of-service with the current ±1% setpoint tolerance. In this analysis, the allowable number of SRVs/SVs out-of-service was re-evaluated using the proposed ±3% tolerance. The conclusion of this analysis is that only one SRV/SV may be out-of-service when the ±3% tolerance is used in order to meet the regulatory limits. As stated previously, the analyses performed in NEDC-33533P are based on PBAPS Unit 3, Cycle 18. The current cycle specific reload analyses for both PBAPS units utilize the ±3% setpoint tolerance with 1 SRV/SV out of service.

2. Effect on SRV Discharge Line and other Suppression Chamber Loads

   SRV discharge piping has been re-analyzed for the effects of increasing the SRV setpoint tolerance in combination with the existing design basis loads. The resulting stresses remain less than the original ANSI/ASME Code allowable stress. In order to accomplish this analysis, new SRV forcing functions based on the revised lift tolerance are applied to the current piping models. Time-history analysis is performed and the calculated stresses are reviewed against the current design basis load combinations. The SV setpoint tolerance increase does not affect any SRV discharge line or other suppression chamber loads since the SVs discharge directly into the drywell through a Tee connected to the SV outlet.

   Additionally, associated restraint loads and deflections from the revised analyses are evaluated against the current restraint designs and the results indicate that the existing restraints can accommodate the effects of an increase in the SRV tolerance. This portion of the analysis also included the effect of the setpoint tolerance increase on the SRV T-Quenchers, which are considered anchors in all of the piping analyses. The T-Quencher components and supports evaluation also indicates that they maintain their structural qualification for design basis loads when considering the revised SRV forcing functions. Due to a known conservative error in the earlier General Electric computer code RVFOR, which has been corrected, the new SRV forcing functions have not resulted in higher loads at the T-Quenchers. Also, the contribution of SRV discharge load on submerged structures in governing load combinations is relatively small.

3. Motor-Operated Valve Operation

   The impact on MOVs due to the potential for increased reactor vessel and system pressure as a result of the increase in SRV/SV setpoint tolerance has been evaluated in accordance with the Generic Letter 89-10/96-05 Program and USNRC Bulletin 85-03 (NEDC-31322, "BWR Owners' Group Report on the Operational Design Basis of Selected Safety Related Motor-Operated Valves," dated September 1986). It has been determined that the impact on MOV operation is acceptable. The MOV methodology uses the reactor pressure corresponding to the spring setpoint of the SRV/SV with the lowest nominal setpoint for differential pressure determinations. Therefore, the MOV methodology is not impacted by the SRV/SV setpoint tolerance revision. Use of nominal SRV/SV setpoints is consistent with the BWROG methodology.

4. HPCI and RCIC System Design Documentation

As discussed in Attachment 3, the Reactor Core Isolation Cooling (RCIC) turbine may require a speed increase of 50 rpm to maintain the appropriate level of conservatism. However, as discussed below, no maximum speed change is required for the RCIC turbine based on a re-analysis of the system design documentation.

The design documents for the HPCI and RCIC Systems have been reviewed and revised to reflect changes in the SRV/SV tolerance values. As stated in Attachment 3, HPCI System performance is adequate and does not require any changes. For RCIC, the required RCIC pump Total Dynamic Head (TDH) for the revised reactor pressure of 1185 psia was compared to the current design basis TDH of 2790 ft when the RCIC discharge flow to the reactor is 600 gpm. A hydraulic analysis of the RCIC System was performed for both units at the reactor pressure of 1185 psia, which resulted in a required RCIC pump TDH of 2809 ft. Maintaining the current RCIC pump TDH margin of 26 ft, the revised design basis TDH is 2835 ft. As discussed in Attachment 3, since the proposed RCIC pump TDH of 2835 (required TDH plus margin) is below the bounding and conservative RCIC pump maximum TDH per Attachment 3, Section 6.1.2, no maximum speed change is required on the turbine driving the RCIC pump.

The current PBAPS TS 3.5.1 Basis (HPCI) and TS 3.5.3 Basis (RCIC) specify that HPCI and RCIC Systems are designed to provide core cooling over a wide range of reactor pressures from 150 psig to 1150 psig. The 1150 psig injection pressure is based on a +1% margin above the lowest SRV setpoint. Therefore, this pressure will be increased by 20 psi to 1170 psig to incorporate a +3% margin per the proposed changes to TS 3.4.3. The HPCI and RCIC pumps' performance to achieve required flow at pressures greater than 1200 psig is routinely demonstrated during periodic system testing. A mark-up of the TS Bases revision is included in Attachment 2.

As a result, no physical plant changes are required to the HPCI or RCIC Systems.

5. SLC System Performance

The SLC System consists of a boron solution storage 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 SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations.

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 Reactor Pressure Vessel (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.

As discussed in Attachment 3, the required SLC System pump discharge pressure must be increased an additional 20 psi as a result of the SRV/SV lift tolerance increase from 1% to 3%. Based on this relaxation, the maximum SLC System discharge pressure will be 1275 psig.

PBAPS performed a review of the impact of increasing SLC System pump discharge pressure and determined the change to be acceptable. The SLC System pump relief valve setpoint margin is based on the discharge pressure during an ATWS. USNRC Information Notice 2001-13 identifies the need to include a margin of 75 psi to prevent inadvertent actuation of the SLC System relief valve. This margin accounts for pressure pulsations from the positive displacement pump and tolerance for the SLC System discharge relief valve. The maximum RPV lower plenum pressure without the SLC System relief valve lifting is the SLC System relief valve setpoint (1450 psig) minus the 75 psi margin to prevent inadvertent actuation, minus the SLC System piping losses (28 psi), and the pressure loss through the SLC System/reactor nozzle through the SLC sparger (25 psi). This results in a maximum RPV lower plenum pressure without the SLC System relief valve lifting of 1322 psig. Therefore, the additional pressure margin to relief valve lift is the maximum RPV lower plenum minus the maximum RPV lower plenum pressure when SLC is credited during ATWS conditions, or 1322 psig – 1245 psig = 77 psi. This represents an additional 77 psi margin greater than the 75 psi (152 psi total) margin required by Information Notice 2001-13 maintained above the maximum RPV lower plenum pressure when the SLC System is credited during ATWS conditions. This margin included a re-analysis of the SLC system piping to determine the current piping system losses.

The PBAPS, Units 2 and 3 SLC System pumps are positive displacement pumps, designed to provide a constant flow rate, independent of discharge pressure. Based on the characteristics of the SLC System positive displacement pumps, this small increase has no effect on the flow rate of the system.

6. <u>SLC System TS Revision and Monitoring</u>

As described above, it is acceptable to increase the SLC System pump discharge pressure from 1255 psig to 1275 psig. A mark-up revision to TS SR 3.1.7.8 is included in Attachment 2.

The SLC System piping is pressure tested once per inspection period which occurs approximately once per three years in accordance with the Inservice Inspection (ISI) program. This system does not experience high fatigue due to the low usage of the system. Therefore, this system will not undergo additional fatigue monitoring.

7. <u>Appendix R Review</u>

The Appendix R analysis is based on nominal SRV//SV lift pressures. Accordingly, the SRV/SV tolerance change will not impact the Appendix R analysis. This increase in SRV/SV lift tolerance has been reviewed to ensure Appendix R requirements have been met. The overall RPV inventory loss for Appendix R scenarios is primarily governed by core decay

heat converting reactor coolant to steam. The rate of steam generation will remain essentially unchanged. Increased SRV/SV setpoint tolerance to ±3% may cause one or more SRV/SV to actuate at a higher/lower pressure. This does not have any significant effect on the Appendix R analysis because the Appendix R analysis is based on nominal values. For example, if an SRV/SV opening was delayed due to the setpoint drifting high, another SRV/SV may open instead, minimizing the effect on the RPV. A slightly higher SRV/SV lift pressure due to expanded tolerance results in a slightly delayed opening, but when the valve does open, it passes slightly higher enthalpy steam at the higher pressure, making the overall mass and energy transfer essentially the same.

The suppression pool temperature is affected by energy transferred to the pool through the SRVs and the heat removal rate of suppression pool cooling. Energy transferred to the suppression pool is a function of the decay heat of the core. For example, if an SRV/SV opened at a higher pressure, this would be offset by the higher flowrate at that pressure, higher enthalpy of the steam at that pressure, and less frequent cycling of the valve. The net result is similar heat transported to the suppression pool. Once controlled depressurization is initiated (cooldown to cold shutdown), the operator controls the cooldown rate via controlled operation of the SRVs. Once the operator takes positive control of RPV pressure, SRV/SV setpoint tolerance is no longer a factor. Thus, the SRV/SV 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.

## 4.0 REGULATORY EVALUATION

### 4.1 Applicable Regulatory Requirements/Criteria

The current ±1% tolerance band on the SRV/SV 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 original ±1% lift setpoint tolerance. As a result, the BWROG developed NEDC-31753P (Reference 1) to support the use of the ±3% SRV/SV lift setpoint tolerance. NEDC-31753P was reviewed and approved by the USNRC as documented in Reference 2. The USNRC determined that it is acceptable for licensees to submit TS amendment requests to revise the SRV/SV lift setpoint tolerance to ±3%, provided that the setpoints for those SRVs/SVs tested are restored to ±1% prior to reinstallation. The USNRC also indicated in its safety evaluation that licensees planning to implement TS changes to increase the SRV/SV setpoint tolerances should provide a plant specific analysis. The plant specific analysis for PBAPS, Units 2 and 3 is provided in Attachment 3.

The existing SRVs/SVs are tested in accordance with the ASME OM Code, "Code for Operation and Maintenance of Nuclear Power Plants." The PBAPS, Units 2 and 3 fourth ten-year interval Inservice Testing (IST) program implements the 2001 Edition through 2003 Addenda of the ASME OM Code, Appendix I, "Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants," and requires that a sample of valves from each 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 proposed to revise the owner-established set pressure acceptance criteria to ±3%.

As discussed in the PBAPS, Units 2 and 3 TS Bases, Section 3.1.7 ("Standby Liquid Control (SLC) System"), the SLC System satisfies the requirements of 10 CFR 50.62 on Anticipated Transient Without Scram (ATWS) event using enriched boron.

Demonstrating that each SLC System pump develops the required flow rate at the required discharge pressure ensures that pump performance has not degraded below design values during the fuel cycle. This test is indicative of overall performance in meeting 10 CFR 50.62. Accordingly, increasing the SLC System pump discharge pressure from the current value of 1255 psig to 1275 psig will ensure compliance with 10 CFR 50.62.

### 4.2 Precedent

The USNRC has approved similar license amendments related to increasing the main steam SRV/SV lift setpoints. Recent examples for BWRs include:

1) Pilgrim Nuclear Power Station (License Amendment No. 235 issued by USNRC letter dated March 28, 2011 - ADAMS Accession No. ML110650009). As part of this change, this plant revised the setpoint in addition to the tolerances also resulting in more extensive analysis required.

2) Quad Cities Nuclear Power Station, Units 1 and 2 (License Amendment No. 235 and 230 issued by USNRC letter dated November 1, 2007 - ADAMS Accession No. ML072290179). It is noted that this plant contains fuel from more than one fuel vendor which required more extensive transient evaluation. PBAPS, Units 2 and 3 has only one fuel vendor that performs the reload analysis. This change also revised the SLC System boron enrichment which required more extensive ATWS analysis.

3) Limerick Generating Station, Units 1 (License Amendment No. 137 issued by USNRC letter dated November 10, 1999 – ADAMS Accession No. ML993250099) and 2 (License Amendment No. 98 issued by USNRC letter dated May 17, 1999 – ADAMS Accession No. ML011560845).

### 4.3 No Significant Hazards Consideration

Exelon Generation Company, LLC (Exelon) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below.

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

   Response: No

   The proposed changes: 1) revise Technical Specification (TS) Surveillance Requirement (SR) 3.4.3.1 to increase the allowable as-found Safety Relief Valve (SRV) and Safety Valve

(SV) lift setpoint tolerance from ±1% to ±3%; 2) revise TS Limiting Conditions for Operation (LCO) 3.4.3 to increase the required number of operable SRVs and SVs from 11 to 12; and; 3) revise TS SR 3.1.7.8 to increase the SLC System pump discharge pressure from 1255 psig to 1275 psig. As analyzed in Attachment 3 ("Peach Bottom Atomic Power Station Units 2 and 3 Safety Valve Setpoint Tolerance Increase Safety Analysis Report," NEDC-33533P, Revision 1, dated May 2013), increasing the SRV/SV tolerance results in a change to the TS requirements for the number of SRVs/SVs required to be operable. However, this change does not alter the manner in which the valves are operated. Consistent with current TS requirements, the proposed change continues to require that the SRVs/SVs 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 pump discharge pressure. The proposed change to increase the SLC System pump pressure 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 USNRC in a safety evaluation dated March 8, 1993. General Electric Hitachi Nuclear Energy (GEH) has completed plant-specific analyses to assess the impact of the setpoint tolerance increase on Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3. The plant specific evaluations, required by the USNRC'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 (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 Updated Final Safety Analysis Report (UFSAR).

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

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

   Response: No

The proposed changes: 1) revise TS SR 3.4.3.1 to increase the allowable as-found SRV and SV lift setpoint tolerance from ±1% to ±3%; 2) revise TS Limiting Conditions for Operation (LCO) 3.4.3 to increase the required number of operable SRVs and SVs from 11 to 12; and; 3) revise TS SR 3.1.7.8 to increase the SLC System pump discharge pressure from 1255 psig to 1275 psig. The proposed change to increase the SLC System pump pressure will ensure that the requirements of 10 CFR 50.62 continue to be met. The proposed change to increase the SRV/SV tolerance was developed in accordance with the provisions contained in the USNRC safety evaluation for NEDC-31753P. Additionally, Attachment 3 analyzes the tolerance increase which results in the increase in the required number of SRVs/SVs necessary to remain operable. SRVs/SVs 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 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 changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes 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. Additionally, this change will ensure that the reactor steam dome pressure shall be ≤1325 psig as discussed in Safety Limit 2.1.2 ("Reactor Coolant System Pressure SL"). The proposed change to increase the SLC System pump discharge pressure will ensure that the requirements of 10 CFR 50.62 continue to be met.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation, Exelon 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.

## 4.4 Conclusions

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 the health and safety of the public.

## 5.0 ENVIRONMENTAL CONSIDERATION

Exelon 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 to be prepared in connection with the proposed amendment.

## 6.0 REFERENCES

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

2 ) Letter from A. C. Thadani (U.S. Nuclear Regulatory Commission) 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

**ATTACHMENT 2**

Markup of Technical Specifications and Bases Pages

Revised Pages (Units 2 and 3)

3.1-23
3.4-8
3.4-9
B 3.1-46
B 3.4-16
B 3.4-17
B 3.5-3
B 3.5-24

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.7 | Verify the quantity of B-10 stored in the SLC tank is ≥ 162.7 lbm. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.1.7.8 | Verify each pump develops a flow rate ≥ 43.0 gpm at a discharge pressure ≥ ~~1255~~ 1275 psig. | In accordance with the Inservice Testing Program |
| SR 3.1.7.9 | Verify flow through one SLC subsystem from pump into reactor pressure vessel. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.1.7.10 | Verify sodium pentaborate atom percent B-10 enrichment is within the limits of Table 3.1.7-1. | Once within 8 hours after addition to SLC tank |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

LCO 3.4.3 The safety function of ~~11~~ 12 valves (any combination of SRVs and SVs) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more required SRVs or SVs inoperable. | A.1 Be in MODE 3. | 12 hours |
| | AND | |
| | A.2 Be in MODE 4. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.3.1 | Verify the safety function lift setpoints of the required SRVs and SVs are as follows:<br><br>Number of SRVs — Setpoint (psig)<br>4 — ~~1135 ± 11.0~~ 1135 ± 34.1<br>4 — ~~1145 ± 11.0~~ 1145 ± 34.4<br>3 — ~~1155 ± 12.0~~ 1155 ± 34.7<br><br>Number of SVs — Setpoint (psig)<br>2 — ~~1260 ± 13.0~~ 1260 ± 37.8<br><br>Following testing, lift settings shall be within ± 1%. | In accordance with the Inservice Testing Program |
| SR 3.4.3.2 | Verify each required SRV actuator strokes when manually actuated in the depressurization mode. | In accordance with the Surveillance Frequency Control Program. |

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 (continued)

cooldown in the normal manner. The required quantity contains an additional amount of B-10 equal to 25% of the minimum required amount of B-10 necessary to shutdown the reactor, to account for potential leakage and imperfect mixing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.7.8

1275

Demonstrating that each SLC System pump develops a flow rate ≥ 43.0 gpm at a discharge pressure ≥ ~~1255~~ psig ensures that pump performance has not degraded below design values during the fuel cycle. This test is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. In addition, the test results for each pump are used to determine that the limits of Table 3.1.7-1 are satisfied for each SLC subsystem. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.9

This Surveillance ensures that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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) (Ref. 1). For the purpose of the analyses, ~~11~~ 12 SRVs and SVs are assumed to operate in the safety mode. The analysis results demonstrate that the design SRV and SV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 2 discusses additional events that are expected to actuate the SRVs and SVs.

SRVs and SVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The safety function of any combination of ~~11~~ 12 SRVs and SVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). Regarding the SRVs, the requirements of this LCO are applicable only to their capability to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety mode).

The SRV and SV 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%~~ + 3% of the nominal setpoint 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, all required SRVs and SVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The SRVs and SVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The SRV and SV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required SRVs or SVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more required SRVs or SVs 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 required SRVs and SVs will open at the pressures assumed in the safety analyses of References 1 and 2. The demonstration of the SRV and SV 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 and be verified with insulation installed simulating the in-plant condition. The SRV and SV setpoint is ~~± 1%~~ for OPERABILITY.

± 3%

Prior to placing new or refurbished valves into service, the valve openings setpoints must be adjusted to be within ±1% of their nominal setting.

(continued)

BASES

BACKGROUND (continued)

The two LPCI subsystems can be interconnected via the LPCI cross tie valve; however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (if offsite power is available, A and B pumps in approximately 2 seconds and C and D pumps in approximately 8 seconds, and, if offsite power is not available, all pumps immediately after AC power is available). Since one DG supplies power to an RHR pump in both units, the RHR pump breakers are interlocked between units to prevent operation of an RHR pump from both units on one DG and potentially overloading the affected DG. RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the corresponding recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for the four LPCI pumps to route water to the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

1170

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to ~~1150~~ psig,). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow

(continued)

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

## B 3.5.3 RCIC System

BASES

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BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures 150 psig to ~~1150~~ 1170 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water back to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.7 | Verify the quantity of B-10 stored in the SLC tank is ≥ 162.7 lbm. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.1.7.8 | Verify each pump develops a flow rate ≥ 43.0 gpm at a discharge pressure ≥ ~~1255~~ 1275 psig. | In accordance with the Inservice Testing Program |
| SR 3.1.7.9 | Verify flow through one SLC subsystem from pump into reactor pressure vessel. | In accordance with the Surveillance Frequency Control Program. |
| SR 3.1.7.10 | Verify sodium pentaborate atom percent B-10 enrichment is within the limits of Table 3.1.7-1. | Once within 8 hours after addition to SLC tank |

## 3.4 REACTOR COOLANT SYSTEM (RCS)

### 3.4.3 Safety Relief Valves (SRVs) and Safety Valves (SVs)

LCO 3.4.3 The safety function of ~~11~~ 12 valves (any combination of SRVs and SVs) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more required SRVs or SVs inoperable. | A.1 Be in MODE 3. | 12 hours |
| | AND | |
| | A.2 Be in MODE 4. | 36 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.4.3.1 | Verify the safety function lift setpoints of the required SRVs and SVs are as follows:<br><br>Number of SRVs — Setpoint (psig)<br>4 — ~~1135 ± 11.0~~ → 1135 ± 34.1<br>4 — ~~1145 ± 11.0~~ → 1145 ± 34.4<br>3 — ~~1155 ± 12.0~~ → 1155 ± 34.7<br><br>Number of SVs — Setpoint (psig)<br>2 — ~~1260 ± 13.0~~ → 1260 ± 37.8<br><br>Following testing, lift settings shall be within ± 1%. | In accordance with the Inservice Testing Program |
| SR 3.4.3.2 | Verify each required SRV actuator strokes when manually actuated in the depressurization mode. | In accordance with the Surveillance Frequency Control Program. |

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 (continued)

cooldown in the normal manner. The required quantity contains an additional amount of B-10 equal to 25% of the minimum required amount of B-10 necessary to shutdown the reactor, to account for potential leakage and imperfect mixing. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.1.7.8

1275

Demonstrating that each SLC System pump develops a flow rate ≥ 43.0 gpm at a discharge pressure ≥ ~~1255~~ psig ensures that pump performance has not degraded below design values during the fuel cycle. This test is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. In addition, the test results for each pump are used to determine that the limits of Table 3.1.7-1 are satisfied for each SLC subsystem. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.9

This Surveillance ensures that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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) (Ref. 1). For the purpose of the analyses, ~~11~~ 12 SRVs and SVs are assumed to operate in the safety mode. The analysis results demonstrate that the design SRV and SV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 2 discusses additional events that are expected to actuate the SRVs and SVs.

SRVs and SVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The safety function of any combination of ~~11~~ 12 SRVs and SVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). Regarding the SRVs, the requirements of this LCO are applicable only to their capability to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety mode).

The SRV and SV 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%~~ ± 3% of the nominal setpoint 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, all required SRVs and SVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The SRVs and SVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the core heat.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The SRV and SV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required SRVs or SVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the safety function of one or more required SRVs or SVs 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 required SRVs and SVs will open at the pressures assumed in the safety analyses of References 1 and 2. The demonstration of the SRV and SV 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 and be verified with insulation installed simulating the in-plant condition. The SRV and SV setpoint is ~~± 1%~~ for OPERABILITY.

± 3%

Prior to placing new or refurbished valves into service, the valve openings setpoints must be adjusted to be within ±1% of their nominal setting.

(continued)

BASES

BACKGROUND (continued)

The two LPCI subsystems can be interconnected via the LPCI cross tie valve; however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (if offsite power is available, A and B pumps in approximately 2 seconds and C and D pumps in approximately 8 seconds, and, if offsite power is not available, all pumps immediately after AC power is available). Since one DG supplies power to an RHR pump in both units, the RHR pump breakers are interlocked between units to prevent operation of an RHR pump from both units on one DG and potentially overloading the affected DG. RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the corresponding recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for the four LPCI pumps to route water to the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to ~~1150~~ 1170 psig,). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine control valve open and the turbine accelerates to a specified speed. As the HPCI flow

(continued)

# B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

## B 3.5.3 RCIC System

BASES

---

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures 150 psig to ~~1150~~ 1170 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine governor valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water back to the CST to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

**ATTACHMENT 5**

Affidavit

# GE-Hitachi Nuclear Energy Americas LLC

# AFFIDAVIT

**I, Peter M. Yandow,** state as follows:

(1) I am the NPP/Services Licensing Manager, Regulatory Affairs, of GE-Hitachi Nuclear Energy Americas LLC (GEH), 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 GEH proprietary report, NEDC-33533P, "Peach Bottom Atomic Power Station Units 2 and 3 Safety Valve Setpoint Tolerance Increase Safety Analysis Report," Revision 1, dated May 2013. The GEH proprietary information in NEDC-33533P, is identified by dotted underline inside double square brackets. [[This sentence is an example.{3}]]. Figures and large equation objects containing GEH proprietary information 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 that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* (FOIA), 5 U.S.C. Sec. 552(b)(4), and the Trade Secrets Act, 18 U.S.C. 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 qualifies 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, 975 F.2.d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2.d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that 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 GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH or other companies.

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

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, 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 GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to 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 for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH 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 or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains an evaluation of the effects of increasing the setpoint tolerance of safety relief valves and identifies specific areas that should be evaluated on a plant specific basis specific to the BWR. This report provides the results of the plant specific evaluations performed to assess the impact of the setpoint tolerance increase.

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

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH'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.

## GE-Hitachi Nuclear Energy Americas LLC

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. 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. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH 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 GEH 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 GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining 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, 2013.

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