ATTACHMENT 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)

HITACHI

GE Hitachi Nuclear Energy

NEDO-33533
Revision 1
DRF 0000-0161-8763
May 2013

Non-proprietary Information - Class I (Public)

Peach Bottom Atomic Power Station
Units 2 and 3
Safety Valve Setpoint Tolerance Increase
Safety Analysis Report

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33533P, Revision 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purposes of supporting Peach Bottom Units 2 and 3 implementation of SRV Setpoint Tolerance Increase. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

REVISION SUMMARY

Rev.	Required Changes to Achieve Revision
0	NA
1	Provided proprietary marking to facilitate creation of public version.

TABLE OF CONTENTS

LIST OF TABLES

ACRONYMS AND ABBREVIATIONS

SHORT FORM	DESCRIPTION
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
DBA	Design Basis Accident
ECCS	Emergency Core Cooling Systems
EIS	Equipment In Service
FWCF	Feedwater Controller Failure
GEH	GE Hitachi Nuclear Energy
HPCI	High Pressure Core Injection
IBA	Intermediate Line Break Analysis
ICF	Increased Core Flow
LOCA	Loss of Coolant Accident
LRNBP	Load Rejection No Bypass
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure Flux Scram
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PBAPS	Peach Bottom Atomic Power Station

PRFO	Pressure Regulator Failure Open to maximum demand
psia	Pounds per square inch absolute
psid	Pounds per square inch difference
psig	Pounds per square inch gauge
RCIC	Reactor Core Isolation Cooling
rpm	Revolutions Per Minute
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RV	Relief Valve
SBA	Small Line Break Analysis
SLB	Small Line Break
SLCS	Standby Liquid Control System
SRV	Safety Relief Valve
SRVDL	Safety Relief Valve Discharge Line
SRV/SV	Safety Relief Valve and/or Safety Valve
SV	Safety Valve
TBV	Turbine Bypass Valve
TDH	Total Dynamic Head
TTNBP	Turbine Trip No Bypass

Executive Summary

This report summarizes the analysis results that support the operation of Peach Bottom Atomic Power Station with a setpoint tolerance increase from 1% to 3% for the function of the Safety Relief Valves and Safety Valves.

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

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

1. INTRODUCTION

1.1 PURPOSE

Reference 1 presents an evaluation of the effects of increasing the setpoint tolerance of Safety Relief Valves (SRV) and Safety Valves (SV) and identifies the specific areas that should be evaluated on a plant-specific basis. These evaluations support the operation of Peach Bottom Atomic Power Station (PBAPS) with an increase in the setpoint tolerance for the safety function of the SRV/SV from 1% to 3%. The increase in setpoint tolerance includes both an increase in the upper limit of the setpoint tolerance as well as a decrease in the lower limit of the setpoint tolerance. The upper limit is defined as +3%, and the lower limit is defined as –3%, with respect to the nominal setpoint.

1.2 PROPOSED PERFORMANCE REQUIREMENT CHANGES

The present and proposed SRV performance requirement changes are presented in Table 1-1.

Table 1-1 Comparison of Present to Proposed Performance Requirements

Performance Requirement	Present Limit	Proposed Limit
Opening pressure at which an SRV is capable of performing its intended function (i.e., operational).	±1%	±3%
Opening pressure at which the licensing basis other than reload analyses have been performed.	±1%	±3%
Opening pressure at which the cycle-specific reload licensing basis analyses have been performed.	±3%	±3%
Tolerance beyond which additional valve testing is required as demonstrated by surveillance testing.	±3%	±3%
Tolerance on the as-left SRV setting prior to the valve being returned to service.	±1%	±1%
The number of SRVs that are assumed to be out-of-service based on the most limiting case.	2	1

1.2.1 SRV Installation Tolerance

Normal maintenance includes periodic refurbishment of SRVs. Prior to placing refurbished valves into service, the valve openings setpoints must be adjusted to be within $\pm$1% of their nominal setting. This performance requirement has not changed.

1.2.2 SRV Out-of-Service

The current PBAPS Technical Specifications allow for up to two SRVs/SVs to be out-of-service with the current $\pm$1% setpoint tolerance. In this report, the allowable number of SRVs out-of-service was re-evaluated using the proposed $\pm$3% tolerance. The conclusion of this study is that only one SRV may be out-of-service when the $\pm$3% tolerance is used.

1.3 OVERALL EVALUATION APPROACH

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

1. Vessel Overpressure;
2. Thermal Limits;
3. Anticipated Transients Without Scram (ATWS);
4. Emergency Core Cooling System (ECCS) and Loss of Coolant Accident (LOCA);
5. High Pressure Systems; and
6. Vessel Thermal Cycle.

Each of these subject areas is affected by the increase in the SRV/SV setpoint tolerance from $\pm$1 to $\pm$3%.

1.4 SUMMARY AND CONCLUSIONS

A summary of the results of the evaluations for each of the subjects of concern is provided in Table 1-2. These evaluations determined that the effect of the setpoint tolerance increase on the operation of PBAPS is acceptable for each of the following subject areas: (1) Vessel Overpressure Analysis, (2) Thermal Limits, (3) ATWS Analysis, (4) ECCS and LOCA Performance, (5) High Pressure Systems Performance, and (6) Vessel Thermal Cycle. There are a few areas, however, that require further evaluation before implementing the setpoint tolerance increase. The subject areas that require additional evaluation are identified in the notes below Table 1-2 and should be addressed by Exelon before its implementation of the setpoint tolerance increase.

Table 1-2 Summary of the Analyses Presented in this Report

Subject	Section	Result
Vessel Overpressure	2.	Acceptable[1]
Thermal Limits	3.	Acceptable[1]
ATWS Mitigation Analysis	4.	Acceptable[1]
ECCS and LOCA Performance Evaluation, and Containment Pressure and Temperature	5.	Acceptable[2]
High Pressure Systems Performance	6.	Acceptable[3, 4, 5]
Vessel Thermal Cycle Assessment	7.	Acceptable[6]

1. These evaluations were either performed or included a parametric study to evaluate one SRV that is 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 remaining after 5% power uprate. None of these effects were evaluated in terms of the PBAPS Licensing Basis in this report.

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

4. High Pressure Coolant Injection (HPCI) 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 System (SLCS) 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 SLCS discharge pressure value. A fatigue monitor program should be used to review the number of cycles and accumulated fatigue usage.

2. VESSEL OVERPRESSURE EVALUATION

2.1 ANALYSIS OVERVIEW

This section describes the effect of the increased setpoint tolerance on the American Society of Mechanical Engineers (ASME) required analysis for the reactor vessel overpressure protection.

The ASME code requires peak vessel pressures to be less than the upset code limit of 1375 psig during transient events. The limiting overpressure event for the PBAPS is the Main Steamline Isolation Valve Closure Flux Scram (MSIVF) event. The reactor is shutdown by the backup high neutron flux scram due to the vessel pressurization and the resulting collapse of voids.

Modeling all SRV/SV opening setpoints 3% above the nominal lift setpoint represents the greatest challenge to the ASME upset code limit. Increased core flow (ICF) conditions are conservatively assumed for the MSIVF event vessel overpressure calculation.

The GEH transient analysis code ODYN is used to obtain the system response and peak vessel pressure. ODYN is a best estimate, one-dimensional core model that combines neutron kinetics, thermal-hydraulic, and plant-specific characteristics to evaluate BWR rapid pressurization events. The use of this computer code is consistent with the PBAPS reload licensing analysis methodology (Reference 2).

2.2 ANALYSIS RESULTS

The MSIVF overpressure analysis results are applicable to PBAPS Unit 3 Cycle 18 (Reference 3). This event is re-analyzed on a cycle-specific basis for both Units 2 and 3, and the results are cycle-specific. However, the conclusions reached from the Unit 3 Cycle 18 calculations are applicable to reload dependent calculations for both units. The system response is simulated with no SRV out-of-service and the SRV/SV opening setpoints at +3% of the nominal setpoint. Presented below is a sensitivity study with one SRV out-of-service, which was performed based on the initial reload analysis.

The event is initiated with the Main Steam Isolation Valves (MSIVs) on all four steam lines rapidly closing. As a result of the reactor vessel isolation, the reactor pressure rises and collapses the core coolant voids, which in turn, increases the neutron flux, causing the reactor to scram on high neutron flux. System pressure subsequently reaches the ATWS Recirculation Pump Trip (RPT) setpoint. The reduction in coolant flow increases the core void fraction and accelerates the power reduction. The vessel pressure continues to rise until the SRV/SV opening pressures are reached, and the SRV/SV actuations terminate the pressurization transient.

With the SRV/SV configuration at a +3% tolerance setting, the resulting peak vessel pressure is [[ ]] to a margin of [[ ]] to the ASME upset code limit. The peak dome pressure reached is [[ ]], with psig, corresponding [[ ]] margin to the Tech Spec limit. These pressure margins are adequate to account for any differences in the unit- and cycle-specific reload transient analyses. Therefore, with respect to the vessel overpressurization requirement, PBAPS can operate with the proposed SRV/SV performance changes.

2.3 CONCLUSIONS

The analyses documented in this section demonstrate that the PBAPS Unit 2 and 3 operation with one SRV out-of-service and a ±3% SRV/SV setpoint tolerance is acceptable. Compliance with the ASME upset code limit for vessel overpressure protection is still ensured with this SRV/SV performance change.

3. THERMAL LIMITS

3.1 ANALYSIS OVERVIEW

This section describes the effect of the increased setpoint tolerance on the PBAPS transient fuel thermal limits analysis. The minimum critical power ratio (MCPR) is the most significant thermal limit for this evaluation. For the analyzed transient events, MCPR must be maintained above the MCPR safety limit.

The most limiting events for MCPR considerations are: the Load Rejection No Bypass (LRNBP), the Turbine Trip No Bypass (TTNBP), and the Feedwater Controller Failure (FWCF) with increasing flow event. These events form the basis for the cycle-specific MCPR operating limits.

Changing the SRV setpoint tolerance and/or the number of SRVs out-of-service could only effect the protection of the MCPR safety limit if it worsened the reactor pressure increase before the peak surface heat flux and the minimum MCPR occur.

3.2 ANALYSIS RESULTS

The limiting transient MCPR analysis results are applicable to PBAPS Unit 3 Cycle 18 (Reference 3). These events are re-analyzed on a cycle-specific basis for both Units 2 and 3, and the results are cycle-specific. However, the conclusions reached from the Unit 3 Cycle 18 calculations are applicable to reload-dependent calculations for both units. The system response is simulated with no SRV out-of-service and the SRV/SV opening setpoints at +3% of the nominal setpoint.

For both the Equipment In-Service (EIS) and the RPT out-of-service application conditions, the LRNBP is the limiting event. The FWCF event is limiting in the Turbine Bypass Valve (TBV) out-of-service condition. The Table 3-1 below provides the timing of the minimum MCPR and the first SRV lift for these limiting events.

Table 3-1 Timing of the Minimum MCPR and the First SRV Lift

Application Condition	Transient Event	MCPR (sec)	SRV Lift (sec)
[[			
			]]

[[
]]

An SRV opening pressure 3% below the nominal setpoint would cause a premature actuation of the SRVs. An earlier actuation of SRVs will reduce the rate of vessel pressurization and decrease the rate of void collapse. If SRV actuation occurs at or before the time of the MCPR, the decreased void reactivity could produce lower neutron and surface heat fluxes, and therefore a smaller ΔCPR. Thus, lowering the SRV setpoint will either not affect the MCPR or will produce a less limiting MCPR.

3.3 CONCLUSIONS

The analyses documented in this section demonstrate that the PBAPS operation with one SRV out-of service and a ±3% SRV/SV setpoint tolerance is acceptable. Compliance with the MCPR safety limit is still ensured with this SRV/SV performance change.

4. ATWS MITIGATION ANALYSIS

4.1 ANALYSIS OVERVIEW

This section describes the effect of the increased setpoint tolerance on the PBAPS ATWS analysis.

The evaluation of the PBAPS response to postulated ATWS events is not a design basis requirement. However, the evaluation is necessary to demonstrate that the implementation of the SRV/SV opening setpoint tolerance increase does not have an unacceptable effect on the mitigation capability of the plant during postulated ATWS events.

The peak vessel pressure analysis is provided because that is the ATWS acceptance criteria significantly affected by the SRV/SV setpoint tolerance increase. The peak cladding temperature (PCT) and suppression pool temperature acceptance criteria are not affected by this setpoint tolerance increase. Long-term suppression pool heatup is driven by the core power generation and the relief capacity of the SRVs and SVs, and is insensitive to small changes to the lift setpoints.

Calculations are performed to determine the peak vessel pressure during an ATWS event to demonstrate compliance with the ASME Code Service Level C limit of 1500 psig (Emergency Condition). This analysis is performed using the ODYN transient analysis code with the reactor operating at the limiting Maximum Extended Load Line Limit Analysis (MELLLA) core flow conditions.

Two events are considered: Main Steam Isolation Valve Closure (MSIVC) and Pressure Regulator Failure Open (PRFO) to maximum demand. Each of these events has the potential to yield the maximum vessel overpressure result. In the MSIVC event, the MSIVs on all four steam lines close simultaneously, while the normal direct scram on full MSIV closure fails. Scram on high flux and high vessel pressure are also assumed to fail. In the PRFO event, the pressure regulator failure produces the maximum steam flow demand. Reactor pressure drops and the MSIVs close on a low steamline pressure signal. Scram on full MSIV closure, high flux and high pressure again fail. For these events, it is conservatively assumed that the reactor scram does not take place on any reactor protection system signals. The ultimate shutdown of the plant is accomplished through the actuation of the SLCS. The initial reduction in power occurs as a result of the ATWS high dome pressure RPT. After the ATWS RPT and following the opening of the SRV/SVs, the event is terminated for overpressure considerations.

4.2 ANALYSIS RESULTS

The results of the ATWS analyses with one SRV out-of service and opening setpoint tolerances of [[ ]] for the SRVs and [[ ]] for the SVs are based on a nominal setpoint of 1260 psig increased by 3%. The PRFO event is the limiting ATWS case for the peak vessel pressure with a resulting peak pressure of [[ ]], which is below the 1500 psig limit.

With respect to the SLCS actuation, the vessel dome pressure at the point of interest is bounded by the second pressure peak shown for the MSIVC event [[ ]]. After the large initial peak, vessel pressure exhibits a saw tooth behavior as a function of the relief valves cycle to remove the core heat generation. Although only one such cycle is considered, the peaks in subsequent cycles will be no greater than the first, as the reactor power gradually decreases. The PFRO event pressure profile, following a slightly higher and delayed peak, is essentially the same as that of the MSIVC event. Both of these assertions are supported by the long-term dome pressure response curves for the two limiting ATWS events shown in Figures 3-8 and 3-9 of Reference 4.

4.3 CONCLUSION

During the most limiting ATWS event, the peak vessel pressure does not exceed the licensing basis criterion of 1500 psig. Therefore, with respect to ATWS overpressure, technical justification that both PBAPS Units 2 and 3 may operate with one SRV out-of-service and a ±3% SRV/SV setpoint tolerance is demonstrated.

5. ECCS/LOCA PERFORMANCE EVALUATION

The LOCA analysis was performed for PBAPS using the SAFER/GESTR LOCA application methods approved by the US Nuclear Regulatory Commission (NRC).

The effect of SRV/SV setpoint relaxation on the ECCS-LOCA performance for BWR 2-6 plants has been evaluated on a generic basis in the BWR Owners Group (BWROG) report approved by the NRC in Reference 1. The ECCS conclusions contained in Reference 1 apply to this LOCA analysis.

[[

]]
As such, plant-specific evaluations of ECCS performance and the effect of SRV/SV set point relaxation on the LOCA Licensing Basis peak cladding temperature are not required.

5.1 CONTAINMENT PRESSURE AND TEMPERATURE FOR DBA LOCA

The effects on the peak containment pressure and temperature response for the short-term Design Basis Accident (DBA) LOCA event and on the peak suppression pool temperature and wetwell pressure for the long-term DBA LOCA were considered. Relaxation of the SRV/SV setpoint tolerance or operation with a single SRV out-of-service [[

]]. Therefore, there is no effect on the DBA LOCA containment pressure and temperature and on the DBA LOCA suppression pool temperature and wetwell pressure. The inputs of containment pressure and suppression pool temperature to the available Net Positive Suction Head (NPSH) analysis are also unaffected.

5.2 SMALL STEAM LINE BREAKS

Small Steam Line Breaks (SLBs) were evaluated to determine the drywell temperature for generating the equipment qualification curve. In that analysis, the effects of the SRV setpoint pressure tolerance relaxation on the small steam line break analyses were considered.

The SLBs that generally produce the most limiting peak drywell temperatures [[

]] Therefore, an increase in SRV/SV opening setpoint has no effect on the smaller SLBs that do not have SRV actuation.

For some SLB events, the SRVs can actuate. The drywell temperature response for smaller SLBs that require SRV actuation may be slightly affected. However, for these breaks, the peak drywell temperature is well below that of the larger limiting SLB. Furthermore, the peak drywell temperature for the smaller SLBs occurs later in the event at the time the drywell sprays are actuated. Because this time occurs after many SRV actuations the peak temperature is controlled by the integrated steam flow to the drywell, which is not affected by the change in the SRV/SV setpoint tolerance or operation with a single SRV out-of-service. The long-term drywell temperature, after the sprays are initiated, is controlled by the break steam mass flow to the drywell and the spray temperature. The drywell spray temperature is controlled by the suppression pool temperature that is mainly governed by energy transferred to the suppression pool through the SRVs. The rate of SRV energy transfer to the suppression pool is controlled by [[

]] The break steam flow to the drywell is controlled [[

]] This parameter is also unaffected by the change in the SRV/SV setpoint tolerance or operation with a single SRV out-of-service. [[

]], the drywell temperature response for the smaller SLBs is also not affected. Therefore, an increase in SRV/SV setpoint tolerance and/or operation with a single SRV out-of-service will have no affect on the drywell temperature response and the equipment qualification curve remains valid.

5.3 INTERMEDIATE AND SMALL LINE BREAK ACCIDENTS

The containment pressure and temperature response for the Intermediate Line Break Accident (IBA) (a liquid line break of 0.1 ft^2) and for the Small Line Break Analysis (SBA) (a line break of 0.01 ft^2) were originally evaluated as part of the Mark I Containment Program and documented in the plant unique load definition report (Reference 5). The results for the IBA and SBA documented in Reference 5 are based on an endpoint-type calculation, which is controlled by the amount of initial stored energy in the primary system and decay heat. There is no increase in the initial primary system stored energy or decay heat due to an increase in SRV/SV setpoint tolerance or operation with a single SRV out-of-service. Therefore, there is no change for either the IBA and SBA event results as presented in Reference 5. Additionally for the SBA, the Reference 5 drywell temperature response is taken to be bounding at a constant value of 340°F. This bounding drywell temperature value would not change due to an increase in SRV setpoint tolerance or operation with a single SRV out-of-service.

5.4 NUREG-0783 LOCAL SUPPRESSION POOL TEMPERATURE

The NUREG-0783 for the PBAPS local pool temperature analysis is documented in NEDC-24380-P (Reference 6). Additional evaluations have been performed as part of the 5% Power uprate project (Reference 7). In the NUREG-0783 analysis for PBAPS, the limiting event is the stuck open relief valve at power with one residual heat removal loop. The limiting event does not experience a vessel pressurization transient that results in automatic SRV actuation. Neither the SRV setpoint tolerance increase nor operation with a single SRV out-of-service will affect the limiting NUREG-0783 event.

For the non-limiting NUREG-0783 events that include automatic SRV actuations, the transients are long-term pool heatup transients that are driven primarily by a mass and energy balance between the reactor pressure vessel and suppression pool. Because the SRV setpoint tolerance does not affect the reactor pressure vessel's mass and energy, the SRV setpoint increase or operation with a single SRV out-of-service will have a negligible effect on the peak local pool temperatures for the cases that include automatic SRV actuations. The SRV setpoint tolerance increase or operation with a single SRV out-of-service will not result in any of the non-limiting NUREG-0783 events to become a limiting event for PBAPS Units 2 and 3.

In addition to automatic valve actuations the NUREG-0783 analysis models manual SRV actuations to simulate operator actions to cool the vessel. The cases evaluated in the NUREG-0783 analysis address a range of cool down rates using different numbers of SRVs, ranging from one to five SRVs. PBAPS Units 2 and 3, each have a total of 11 SRVs with five of the SRVs are also serving as Automatic Depressurization System (ADS) valves. Because the single SRV out-of-service cannot be an ADS valve, operators will always have an adequate number of valves to perform the vessel cool down as modeled in the NUREG-0783 analysis. The single SRV out-of-service will therefore not affect the manual cool down portion of the NUREG-0783 analysis.

Because the SRV setpoint tolerance increase does not affect the limiting NUREG-0783 event, and/or because the SRV setpoint tolerance increase will not result in a nonlimiting NUREG-0783 event becoming the limiting event, it is concluded that the SRV setpoint tolerance increase will not affect the current NUREG-0783 analysis, even with a single SRV out-of-service.

5.5 DESIGN BASIS ACCIDENT LOCA HYDRODYNAMIC LOADS

The DBA LOCA hydrodynamic loads, such as pool swell, vent thrust, condensation oscillation and chugging are dependent on the containment pressure and temperature response during the DBA LOCA. [[

]], the DBA LOCA hydrodynamic loads are also unaffected.

5.6 SRV DYNAMIC LOADS

An increase in the SRV pressure setpoint tolerance from 1% to 3% may increase the pressure at which the SRVs open. The resulting increase in SRV opening pressure could increase the SRV flow rate, which could result in an increase in the internal SRV dynamic loads pressure, the thrust loads and the external T-quencher bubble pressure loads.

The SRV dynamic load analysis performed for the PBAPS SRV setpoint relaxation evaluation uses the same methods that were used to evaluate the effect of a 30 psi SRV setpoint increase for the stretch uprate project (Reference 7). In that report, the stretch uprate SRV setpoint increase evaluations were determined as a result of a 30 psi increase in the SRV setpoint on internal and external SRV Discharge Lines (SRVDL).

The SRV dynamic loads evaluation is divided into two main parts:

1) External loads on the torus submerged boundaries and submerged structures.

2) Internal loads on the SRV T-quencher and SRVDL; and the SRVDL supports is an internal loads evaluation that is split between loads in the drywell and loads in the wetwell.

5.6.1 External Loads on the Torus Submerged Boundaries and Submerged Structures

The effect of the SRV setpoint tolerance relaxation on external air bubble pressure loads is evaluated using the same methods that were used to evaluate the 30 psi SRV setpoint increase evaluated as part of the 5% power uprate project (Reference 7). The evaluation is based on the following two key assumptions:

[[

]]

For the 5% uprate project, the SRV loads analysis was based on the nominal setpoint pressure. Therefore in order to conservatively account for the potential 3% increase in SRV lift pressure, the nominal setpoint pressure must be increased by 3%.

Because the reference pressures and capacities at the reference pressure are identical for all SRVs the following equation can be written for the percent increase in SRV flow associated with a 3% increase in setpoint pressure.

[[ ]]

Using the nominal SRV setpoints, the percentage increases in SRV flow associated with the full 3% tolerance are set out in the following Table 5-1.

Table 5-1 Percentage Increases in SRV Flow Associated with Full 3% Tolerance Increase

Group	Set pressure (psig)	Percent Increase in SRV flow with a Full 3% SRV Lift Pressure Tolerance
I	1135	[[
II	1145	
II	1155	]]

[[

]]

The 5% power uprate evaluation, documented in Reference 7, determined that there is an approximate [[ ]] margin in the external bubble load definition after the implementation of the 30 psi SRV setpoint increase associated with the 5% power uprate. Therefore, a [[ ]] load increase associated with the SRV setpoint tolerance relaxation project is greater than the available [[ ]] remaining margin of conservatism documented in Reference 7. This [[ ]] load increase would apply to all SRV loads associated with the external T-quencher bubble pressure load, including single and multiple valve bubble pressure loads on the submerged pool boundary (torus shell), submerged structures, and piping attached to the torus shell.

Based on this evaluation, additional reviews would be required to justify the increased SRV load associated with a SRV setpoint tolerance increase from 1% to 3%.

5.6.2 Internal Loads

As documented in Section 5.6.1, the potential 3% increase in SRV lift pressure associated with the SRV setpoint tolerance relaxation project will result in an increase in the SRV flow rate of [[ ]] Based on studies documented in Reference 8, [[ ]] Therefore, the potential 3% increase in SRV lift pressure will result in a [[ ]] increase in SRVDL loads. The [[ ]] increase in SRVDL loads applies to both the wetwell and drywell portions of the SRVDL loads.

The [[ ]] increase also applies to second actuation loads, as [[ ]] While a decrease in the SRV setpoint will result in decreases in the time between valve closure for the first actuation and the second actuation, the change will be [[

]] In addition any reduction in SRV setpoint pressure will result in a reduction of the SRVDL loads due to the reduction in SRV mass flow rate.

5.6.2.1 Wetwell Portion of the Internal Loads

As described in Reference 9 for the SRV T-quencher and the wetwell portion of the SRVDL, [[

]] A conservatism, which is greater than 20%, has been demonstrated in the GE methods (RVFOR) used to calculate this load for PBAPS (Reference 10) is sufficient to bound the [[ ]] due to the 30 psi increase in SRV setpoint associated with the stretch uprate and the additional [[ ]] load increase associated with the SRV setpoint tolerance relaxation project.

5.6.2.2 Drywell Portion of the Internal Loads

The SRV setpoint pressure tolerance increase will result in a [[ ]] increase in SRV discharge line piping loads. This assessment is based on the percentage increase in flow, calculated above, and the results of a prior GEH study of the effect of SRV discharge mass release rate changes on SRV dynamic loads.

GEH has not evaluated the effect of the load increase on the PBAPS Licensing Basis.

5.6.3 Effect of a Single SRV Out-of-Service on SRV Loads

Having a single SRV out-of-service will not have an effect on SRV dynamic loads. SRV loads are driven by SRV opening pressure and the SRV discharge line water level at the time of the second SRV actuation. The SRV discharge line water level is in turn a function of the time between the closure of the valves at the end of the first actuation and the time of the second actuation. Having a single SRV out-of-service will not affect the valve setpoints or the time between the initial and second valve actuations. As such, having a single SRV out-of-service will have no effect on SRV dynamic loads.

6. HIGH PRESSURE SYSTEMS PERFORMANCE

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

6.1 HIGH PRESSURE COOLANT INJECTION

6.1.1 System Description and Functions

The HPCI system, which is part of the ECCS, is designed to provide sufficient core cooling to prevent excessive fuel cladding temperature in the event of a small break LOCA that does not depressurize the reactor quickly enough to permit timely operation of the low pressure ECCS. The HPCI system accomplishes this function by injecting coolant makeup water into the pressure vessel with a turbine driven pump. The HPCI system also serves as a backup to the RCIC system to maintain the nuclear boiler in the standby condition in the event the vessel becomes isolated from the main condenser and feedwater makeup flow.

The RCIC system provides makeup water to the Reactor Pressure Vessel (RPV) whenever the vessel is isolated from the main condenser and feedwater make up. The RCIC system can also provide makeup water during shutdown whenever the normal water supply is unavailable. RCIC uses a turbine driven pump to maintain adequate reactor vessel water level.

The most significant effect of relaxing the SRV tolerance on the HPCI and RCIC systems' operations is the maximum reactor pressure at which they are required to deliver water to the reactor. Both systems are required to provide injection into the reactor pressure vessel at the lowest group of SRV setpoints (including drift). Increasing the allowable setpoint tolerance to 3%, while maintaining the current setpoint of 1135 psig, [[
]]

6.1.2 Pump Flow and Head

Relaxation of the SRV setpoint tolerance, while increasing the required maximum discharge pressure of the pumps, does not change the flow requirement. Because there is no increase in the flow, the NPSH required or available, does not change. Likewise, the calculated pressure losses due to pipe friction remain unchanged.

The original design for both systems made conservative, bounding assumptions for dynamic and static head losses to determine the pump's Total Dynamic Head (TDH) requirement. The original requirement of [[ ]] TDH was increased by [[ ]] during power rerate, resulting in the current requirement of [[ ]]. The increase of [[ ]] was derived from the increase of 30 psi in the SRV setpoint.

PBAPS has performed analyses of the piping configuration for both systems to determine both dynamic and static head losses. Using the more accurate results from these analyses, demonstrates that the [[ ]] TDH requirement remains both bounding and conservative for the HPCI and RCIC systems.

[[

]] If the original conservatism is maintained, HPCI performance is still adequate, but the RCIC turbine would require a speed increase of [[ ]] Rather then modifying the RCIC turbine, the design documentation for both HPCI and RCIC should be revised to reflect the available analyzed design values, rather than the original assumed values.

6.1.3 System Boundary Components

HPCI and RCIC System components that form part of the RPV pressure boundary were evaluated by comparing the design pressures and temperatures for those lines and components affected by the higher system operating conditions. [[

]], there is no change to the design requirements of the piping and components attached to the reactor vessel.

6.1.4 Pump and Turbine

The pump and turbine speeds required to meet the TDH requirement for the HPCI system, using either assumed or analyzed values for head losses, are bounded by the current design. There is no change in turbine or pump power requirements, steam flow to the turbine or system reliability.

The pump and turbine speeds required to meet the TDH requirement for the RCIC system, using analyzed values for head losses, are bounded by the current design. There is no change in turbine or pump power requirements, steam flow to the turbine, or system reliability. If the plant wishes to maintain the current conservatisms with assumed head loss, RCIC turbine speed has to be increased by [[ ]] and the system performance re-evaluated.

6.1.5 Instrumentation

Instrument specifications according to the instrument data sheet have been reviewed for application to the HPCI and RCIC required pressure increase. Because the maximum reactor pressure and temperature design parameters remain unchanged, and exceed those conditions required for the SRV setpoint tolerance increase, there is no change to the design pressure requirements of the instrumentation. In addition, because there is no increase in flow for either make up or steam, no change is required for leak detection or flow instruments. Exhaust trip setpoint and rupture disks do not need to be changed because there is no change in exhaust pressure.

6.1.6 Motor Operated Valves

The system valves that are affected by reactor pressure require evaluation for operability at the increased operating pressures expected at the required maximum vessel pressure conditions. The specified full differential pressure values for the steam supply and pump discharge valves will require adjustment to reflect the increased SRV tolerances and the higher system operating pressures.

6.1.7 Start Up Transient

[[

]] will not affect the start up transient. The original time to full rated speed has been increased for both systems from the original requirement of less than 30 seconds. Therefore, the time to reach the slightly higher turbine speed required to inject at the higher pressure will remain within the design requirement.

6.1.8 Input and Assumptions

The input parameters used in the analyses and the assumptions made are provided in the Tables 6-1 through 6-4 that follow. Design values for input parameters are the same for both units unless specifically noted.

Table 6-1 Key Input for RCIC

Item	Parameter	Value
1	Maximum design value for system start time to rated flow (system in standby line up)	[[
2	Maximum design value for reactor pressure for system operation	
3	Maximum design value for system injection flow rate	
4	Minimum available NPSH	
5	Maximum rated turbine speed	
6	Maximum design value for turbine exhaust pressure	
7	Turbine exhaust high pressure trip	
8	Maximum design pressure value for turbine inlet nozzle	
9	Maximum design pressure value for pump	
10	Maximum steam flow to turbine	]]

Table 6-2 Key Input for HPCI

Item	Parameter	Value
1	Maximum design value for system start time to rated flow (system in standby line up)	[[
2	Maximum design value for reactor pressure for system operation	
3	Maximum design value for system injection flow rate	
4	Minimum available NPSH	
5	Maximum rated turbine speed	
6	Maximum design value for turbine exhaust pressure	
7	Turbine exhaust high pressure trip	
8	Maximum design pressure value for turbine inlet nozzle	
9	Maximum design pressure value for pump	
10	Maximum steam flow to turbine	]]

6.1.9 Key Assumptions

[[

]]

6.1.10 Results

The RCIC and HPCI systems are designed to provide rated flow to the reactor up to maximum reactor dome pressure of 1165 psia. The maximum reactor dome pressure required for injection is equal to the setpoint of the lowest group of SRVs including setpoint drift. The [[ ]] is conservatively higher than the present setpoint of 1135 psig and 1% tolerance for drift. For a setpoint tolerance increase from 1% to 3%, the maximum dome pressure into which the high pressure systems are required to inject rated flow, [[ ]]

The original system design specifications require a pump pressure rise (TDH) of [[ ]] of head in order for the systems to deliver water from the suppression pool to the reactor at the high pressure condition. The [[ ]] of head was based on a conservatively assumed value for piping losses and elevation changes. During a previous power uprate, the SRV setpoint was increased by [[ ]] If the current TDH of [[ ]] for both pumps is [[ ]]

To meet the new requirement of [[ ]] TDH, both systems will require a higher pump speed. The HPCI system maximum evaluated speed for transient events is adequate for the new requirement with the conservatively assumed head losses. The RCIC maximum evaluated speed for transient events is inadequate for the assumed head losses. If the plant desires to maintain the margin inherent in the assumed head losses, then the maximum RCIC speed requires an increase of [[ ]]

Based on design analysis performed by PBAPS, the actual head losses are considerably less than originally assumed. If the NPSH and head losses that were analyzed for PBAPS are used to determine the TDH for both systems, then, even with the [[ ]] increase in discharge pressure, the TDH requirement remains below the present requirement of [[ ]], and no changes are required to either system.

[[ ]], there is no effect on the systems' startup times or the severity of the startup transients.

The NPSH available for the RCIC pump does not change [[ ]] In addition, [[ ]], do not change.

The increase in maximum vessel pressure does require a corresponding change to pump discharge pressure. While the existing design requirements remain conservative, changes to the in-service testing program will be required to demonstrate pump operability.

If the current TDH remains at [[ ]] (using analyzed head losses), no changes are required to the steam line break detection instrumentation set points [[ ]] If the current, assumed head losses are used for a required TDH of [[ ]], no changes to the HPCI setpoints are required [[ ]] If the RCIC turbine and pump speed is increased to meet the [[ ]] TDH requirement, the steam line break detection instrumentation setpoints will require evaluation.

The steam supply and pump discharge motor operated valves will require re-evaluation for operability at the new, higher pressures.

There is no effect to the start up transient due to the increase in reactor pressure.

Table 6-3 Key Results for RCIC

Item	Parameter	Current Design	With 3% SRV Setpoint tolerance	Change
1	Setpoint for lowest group SRV	1135 psig	[[	
2	Maximum vessel pressure	1165 psia		
3	TDH with original assumed head losses	2870 feet		
4	TDH with analyzed head losses	Not calculated or given		
5	Maximum rated pump speed with analyzed head losses	4550 rpm		
6	Pump speed required with original head loss margin	Not available		]]

Table 6-4 Key Results for HPCI

Item	Parameter	Current Design	With 3% SRV Setpoint tolerance	Change
1	Setpoint for lowest group SRV	1135 psig	[[	
2	Maximum vessel pressure	1165 psia		
3	TDH with original assumed head losses	2870 feet		
4	TDH with analyzed head losses	Not calculated		
5	Maximum rated pump speed with analyzed head losses	4100 rpm		
6	Pump speed required with original head loss margin	4100 rpm		]]

6.1.11 Recommendations

[[

]]

6.2 STANDBY LIQUID CONTROL SYSTEM

The SLCS is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. The SLCS is also required to inject the sodium pentaborate solution to control reactor power in the event of an ATWS event to meet the injection rate requirements of 10 CFR 50.62(c)(4) (Reference 11).

6.2.1 SLCS Relief Valve Discussion

The original SLCS design criterion for the maximum system operating pressure was based on the lowest group setpoint among all SRVs plus a margin for analytical inaccuracy. This criterion has been generally replaced by plant-specific ATWS transient pressure data obtained during the SLCS operating analysis specified in NRC Information Notice 2001-13 (Reference 12). The

SLCS pump relief valve setpoint margin is now based on a nominal SLCS pump relief valve setpoint and the maximum SLCS pump discharge pressure during the limiting ATWS event.

The information notice identifies the need to include the margin for the pressure pulsations of the positive displacement pumps (30 psi) and the tolerance for the SLCS relief valve (typically between 42 and 45 psi) in the calculation to determine if adequate margin exists between the pump discharge pressure and the relief valve setpoint. To provide sufficient margin against inadvertent relief valve lifting, a margin of 75 psi (from 30 psi and 45 psi combined) between the nominal setpoint and the maximum discharge pressure is required.

6.2.2 Parameter Summary

The maximum peak reactor vessel steam dome pressure is [[ ]] Adding [[ ]] to achieve lower plenum pressure and subtracting 14.7 psi gives a value of [[ ]] This value applies to both the current SRV setpoint and the proposed ±3% setpoint.

In order to prevent the SLCS RV from lifting, the lower plenum pressure must not exceed the lift pressure minus the required margin of 75 psi (Reference 12), minus the SLCS piping losses of 75 psi (for one pump running). Therefore, the maximum lower plenum pressure to avoid inadvertent relief valve lifting is [[ ]] Table 6-5 summarizes this information.

Table 6-5 Summary of Pressure Differential Parameters

Parameter	Current	Proposed
SLCS RV Setpoint	**1450 psig**	[[
IN 2001-13 Required Margin	75 psid	
SLC Piping Losses (1 pump running)	**75 psid**	
Max RPV lower plenum pressure w/o RV lifting	**1300 psig**	
Maximum lower plenum pressure when SLC is credited during ATWS conditions	1245 psig *	
Margin to RV lift	**55 psid**	]]

* 3% SRV tolerance is already assumed in the current ATWS analysis.

Based on the results of the ATWS evaluation performed as part of this effort, the SLCS relief valve setpoint does satisfy the requirements of Reference 12 at the current licensed thermal power conditions.

6.2.3 SLCS Pump Discharge and Flow Rate Discussion

Using the Reference 12 guidelines, a pump discharge pressure requirement of 1255 psig is calculated. This value is based on a nominal setpoint of 1135 psig, which is consistent with the lowest setpoints among SRVs at PBAPS, and a setpoint tolerance of 1%, which results in a rounded maximum relief valve setpoint of 1150 psig. By adding the maximum relief valve setpoint of 1150 psig to the 30 psig for the reactor vessel static head, an injection pressure of 1180 psig at the SLCS sparger results. When SLCS line losses are added to 1180 psig (add 75 psig), the result is a maximum discharge pressure of 1255 psig.

Based on the relaxation of setpoint tolerance to 3% (or approximately an increase from 1150 psig to 1170 psig), an additional 20 psi would need to be added. Based on this relaxation, the maximum SLCS discharge pressure would then be approximately 1275 psig. Based on the characteristics of the SLCS positive displacement pumps, this small increase has no affect on the flow rate of the system.

The 1255 psig value is used in the Technical Specification Section 3.1.7. That document will need to be updated to reflect the increased maximum SLCS discharge pressure value.

6.2.4 SLCS Sodium Pentaborate Solution Discussion

No changes are necessary to the boron solution concentration or enrichment as a result of SRV tolerance relaxation. Per the ATWS evaluation, peak cladding temperature and suppression pool temperature analysis is not affected by this setpoint tolerance increase. Therefore, no changes to the boron solution concentration or enrichment are required.

6.2.5 Conclusion

In summary, the SLCS system at PBAPS is expected to meet its design requirements with the SRV tolerance relaxation at the current licensed thermal power conditions.

7. VESSEL THERMAL CYCLE ASSESSMENT

7.1 CONCLUSIONS AND RECOMMENDATIONS

[[

]]

8. REFERENCES

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

2. NEDE-24011-P-A-16, "General Electric Standard Application for Reactor Fuel (GESTAR)," Class III, October 2007.

3. 0000-0094-9487-SRLR, "Supplemental Reload Licensing Report for Peach Bottom Unit 3 Reload 17 Cycle 18," August 2009.

4. NEDC-31984P, "Generic Evaluation of the General Electric Boiling Water Reactor Power Uprate," Supplement 1, October 1991.

5. NEDO-24577, Rev. 2, "Mark I Containment Program, Plant Unique Load Definition Peach Bottom Atomic Power Station, Units 2 and 3," March 1982.

6. NEDC-24380-P, "Peach Bottom Atomic Power Station, Units 2 and 3, Suppression Pool Temperature Response," December 1981.

7. NEDC-32183P, "Power Rerate Safety Analysis Report For Peach Bottom 2 & 3," May 1993.

8. NEDE-21878, "Mark I Containment Program Analytical Model for Computing Air Bubble and Boundary Pressures Resulting from an S/RV Discharge Through a T-Quencher Device," January 1979.

9. Letter, G. V. Kumar (GE) to T. E. Shannon, "Peach Bottom Power Rerate SRV Loads Evaluation," GE93-PEPR-073, April 19, 1993.

10. NEDC-32230P, "Peach Bottom Power Rerate Project Engineering Report," March 1994.

11. 10 CFR 50.62, "Requirements for Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-water-cooled Nuclear Power Plants."

12. NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," August 10, 2001.