Attachment 4 to Entergy Letter 2.10.016

(Non-Proprietary Version)

GE Hitachi Nuclear Energy Report, NEDC-33532,

"Pilgrim Nuclear Power Station Safety Valve Setpoint Increase", Rev. 0, March 2010

(70 Pages)

HITACHI

GE Hitachi Nuclear Energy

NEDO-33532
Revision 0
DRF 0000-0108-5986
Class I
March 2010

NON-PROPRIETARY INFORMATION

Pilgrim Nuclear Power Station Safety Valve Setpoint Increase

Copyright 2010 GE-Hitachi Nuclear Energy Americas, LLC

All Rights Reserved

INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33532P, Revision 0, from which the proprietary information has been removed. Portions of the document that have been removed are identified by white space within double square brackets, 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: A License Amendment Request by Entergy Nuclear Operations, Inc. (Entergy), for an increase in safety valve setpoints in proceedings before the U.S. Nuclear Regulatory Commission. The only undertakings of GEH with respect to information in this document are contained in the contract between GEH and Entergy, Agreement No. 10175939, effective October 01, 2001 and Contract Order No. 10242337, effective June 23, 2009, and nothing contained in this document will 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.

Copyright 2010, GE-Hitachi Nuclear Energy Americas LLC, All Rights Reserved

TABLE OF CONTENTS

ACRONYMS AND ABBREVIATIONS ..... **VIII**

EXECUTIVE SUMMARY ..... **XI**

1 INTRODUCTION **1-1**

1.1 PURPOSE 1-1

1.2 OVERALL EVALUATION APPROACH 1-1

1.3 SUMMARY AND CONCLUSIONS 1-2

2 VESSEL OVERPRESSURE/ANTICIPATED OPERATIONAL OCCURRENCE EVALUATION **2-1**

2.1 ANALYSIS OVERVIEW 2-1

2.2 OVERPRESSURE ANALYSIS 2-2

2.3 REVIEW OF FSAR TRANSIENT AND ACCIDENT EVENTS 2-3

2.4 REVIEW OF EQUIPMENT OUT OF SERVICE OPTIONS 2-8

2.5 MARGIN TO SPRING SAFETY VALVE SETPOINT 2-8

2.6 EFFECTS ON INTERFACING SYSTEMS 2-9

3 ATWS ANALYSIS **3-1**

3.1 METHODOLOGY 3-1

3.1.1 Analytical Models 3-1

3.2 KEY ASSUMPTIONS 3-2

3.3 ANALYSIS OVERVIEW 3-3

3.4 ANALYSIS INPUTS 3-5

3.5 ANALYSIS RESULTS 3-10

3.6 CONCLUSIONS 3-31

4 ECCS/LOCA EVALUATION **4-1**

4.1 ECCS/LOCA PERFORMANCE EVALUATION 4-1

5 CONTAINMENT RESPONSE AND LOADS ANALYSIS 5-1

5.1 CONTAINMENT PRESSURE AND TEMPERATURE FOR DBA LOCA 5-1

5.2 SMALL STEAM LINE BREAKS 5-1

5.3 IBA AND SBA 5-2

5.4 NUREG-0783 LOCAL SUPPRESSION POOL TEMPERATURE 5-2

5.5 DBA LOCA HYDRODYNAMIC LOADS 5-3

5.6 SRV AND SSV DYNAMIC LOADS 5-3

5.7 RIPD EVALUATION 5-3

5.8 ATWS 5-4

6 HIGH PRESSURE SYSTEMS PERFORMANCE 6-1

6.1 HPCI & RCIC – SYSTEM DESCRIPTION AND FUNCTIONS 6-1

6.1.1 Pump Flow and Head 6-1

6.1.2 System Boundary Components 6-2

6.1.3 Pump and Turbine 6-2

6.1.4 Instrumentation 6-2

6.1.5 Motor Operated Valves 6-2

6.1.6 Start Up Transient 6-3

6.1.7 Key Assumptions 6-3

6.1.8 Key Input for RCIC 6-3

6.1.9 Key Input for HPCI 6-4

6.1.10 Key Results 6-4

6.1.11 Recommendations and Observations 6-6

6.2 SETPOINT CALCULATIONS 6-6

6.2.1 Description 6-6

6.2.2 Inputs and Assumptions 6-7

6.2.3 Evaluation 6-7

6.2.4 Conclusion 6-7

6.3 STANDBY LIQUID CONTROL SYSTEM 6-8

7 APPENDIX R ANALYSIS 7-1

8 REFERENCES 8-1

LIST OF TABLES

TABLE	TITLE	PAGE
1-1	Summary of Analyses Presented in this Report	1-2
2-1	SRV and SSV Configuration	2-2
2-2	Pilgrim Cycle 18 Overpressure Results With Current and New SRV and SSV Configuration	2-3
2-3	FSAR Transient and Accident Event Review	2-4
3-1	Summary of ATWS Key Input Parameters	3-5
3-2	Axial Power Shapes	3-6
3-3	Key Equipment Parameters	3-7
3-4	Summary of Key ODYN Parameters for ATWS Calculation	3-10
3-5	Summary of Peak Suppression Pool Temperature, Containment Pressure and Integrated SRV Flow	3-11
3-6	Sequence of Events for MSIVC at BOC	3-12
3-7	Sequence of Events for MSIVC at BOC	3-13
3-8	Sequence of Events for PRFO at EOC	3-14
3-9	Sequence of Events for PRFO at EOC	3-15
3-10	Acceptance Criteria Results	3-16
3-11	Peak Pressures for Other System Evaluations	3-16
6-1	Key Results for RCIC	6-5
6-2	Key Results for HPCI	6-5

LIST OF FIGURES

FIGURE	TITLE	PAGE
3-1a	MSIVC BOC - GE14 and GNF2 Fuel	3-17
3-1b	MSIVC BOC - GE14 and GNF2 Fuel	3-18
3-1c	MSIVC BOC - GE14 and GNF2 Fuel	3-19
3-2a	PRFO BOC – GE14 and GNF2 Fuel	3-20
3-2b	PRFO BOC – GE14 and GNF2 Fuel	3-21
3-2c	PRFO BOC – GE14 and GNF2 Fuel	3-22
3-3a	MSIVC EOC - GE14 and GNF2 Fuel	3-23
3-3b	MSIVC EOC - GE14 and GNF2 Fuel	3-24
3-3c	MSIVC EOC - GE14 and GNF2 Fuel	3-25
3-4a	PRFO EOC – GE14 and GNF2 Fuel	3-26
3-4b	PRFO EOC – GE14 and GNF2 Fuel	3-27
3-4c	PRFO EOC – GE14 and GNF2 Fuel	3-28
3-5	Containment Response MSIVC BOC	3-29
3-6	Containment Response PRFO BOC	3-29
3-7	Containment Response MSIVC EOC	3-30
3-8	Containment Response PRFO EOC	3-30
3-9	Lower Plenum Pressure PRFO BOC	3-31

ACRONYMS AND ABBREVIATIONS

Term	Definition
ADS	Automatic Depressurization System
AFC	Automatic Flow Control
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
ARI	Alternate Rod Insertion
ARTS	Average Power Range Monitor, Rod Block Monitor, Technical Specifications Improvement Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BIIT	Boron Injection Initiation Temperature
BOC	Beginning of Cycle
BPV	Bypass Valve
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CPR	Critical Power Ratio
DBA	Design Basis Accident
DW	Drywell
ECCS	Emergency Core Cooling System
EOC	End of Cycle
EOOS	Equipment Out-of-Service
EQ	Environmental Qualification
FSAR	Final Safety Analysis Report
FW	Feedwater

Term	Definition
FWCF	Feedwater Controller Failure
HPCI	High Pressure Coolant Injection
HSBW	Hot Shutdown Boron Weight
IBA	Intermediate Break Accident
IORV	Inadvertent Opening of Relief Valve
LFWH	Loss-of-Feedwater Heater
LOFW	Loss-of-Feedwater Flow
LOCA	Loss–of–Coolant Accident
LOOP	Loss-of-Offsite Power
LPCI	Low Pressure Coolant Injection
LRNBP	Load Rejection No Bypass
LRWBP	Load Rejection with Bypass
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MFC	Manual Flow Control
M/G	Motor/Generator
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVD	Main Steam Isolation Valve Closure – Direct Scram
MSIVF	Main Steam Isolation Valve Closure with Flux Scram
MW	Million Watts
NBR	Nuclear Boiler Rated
NMS	Neutron Monitoring System
NPSH	Net Positive Suction Head

Term	Definition
NRC	Nuclear Regulatory Commission
NTSP	Nominal Trip Setpoint
OOS	Out–of–Service
PCT	Peak Clad Temperature
PRFO	Pressure Regulator Failure – Open
PNPS	Pilgrim Nuclear Power Station
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RIPD	Reactor Internal Pressure Difference
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSLB	Recirculation Suction Line Break
SBA	Small Break Accident
SLB	Steam Line Break
SLCS	Standby Liquid Control System
SRV	Safety Relief Valve
SSV	Spring Safety Valve
TAF	Top of Active Fuel
TDH	Total Dynamic Head
TS	Technical Specification
TTNBP	Turbine Trip No Bypass
TTWBP	Turbine Trip with Bypass
UFSAR	Updated Final Safety Analysis Report

EXECUTIVE SUMMARY

The Pilgrim Nuclear Power Station (PNPS) safety valve set-point increase evaluation supports Entergy's pursuit of a solution to improve the current operating conditions at PNPS. This solution provides a greater operational simmer margin as well as increasing allowed Safety Relief Valve (SRV)/Spring Safety Valve (SSV) setpoint tolerance.

This report specifically addresses several analyses/subject areas that are sensitive to the valve setpoint tolerances, nominal setpoint increases, and valve capacity increases. Other subjects that are insensitive to these valve setpoint and capacity changes are not addressed in this report.

This report also identifies the requirements necessary to implement the setpoint changes and the SSV capacity increase.

1 INTRODUCTION

1.1 PURPOSE

This report supports the following increase in setpoints, setpoint tolerance, SSV capacity increase, Plant Modifications and associated Technical Specification (TS) changes:

- Replace the full complement of four Target Rock 2-Stage SRVs with 3-Stage SRVs (No increase in SRV capacity)
- Replace the full complement of two Dresser Safety Valves with higher capacity SSVs.
- Increase the nominal setpoint of the SRVs and SSVs by 40 psig to 1155 psig and 1280 psig respectively.
- Increase the setpoint tolerance of the SRVs and SSVs from ±1% to ±3%.
- Increase the TS Reactor Pressure Alternate Rod Insertion (ARI) and Recirculation Pump Trip (RPT) Setpoints by 40 psig to 1215 ±5 psig
- Increase the TS Reactor Steam Dome Safety Limit currently set at 1325 psig to 1340 psig.
- Lower the High Reactor Pressure Feedwater Pump Trip Setpoint from 1415 psig to 1315 psig.

Reference 1 presents a generic evaluation of the effects of increasing the setpoint tolerance of the SRVs and identifies specific areas that should be evaluated on a plant specific basis. This report provides the results of the plant specific evaluations performed to assess the effect of the setpoint tolerance increase, nominal setpoint increase, and SSV capacity increase. These evaluations support the operation of PNPS with an increase in the setpoint tolerance for the safety function of the Target Rock Dual Mode SRVs and the Dresser SSVs from 1% to 3%. The increase in setpoint tolerance includes both an increase in the upper limit of the setpoint tolerance as well as a decrease in the lower limit of the setpoint tolerance. The upper limit is defined as +3% and the lower limit is defined as –3%.

1.2 OVERALL EVALUATION APPROACH

The effect of the SRV and SSV setpoint tolerance relaxation, SRV and SSV setpoint increase, and SSV capacity increase, on the following subjects is addressed in this report:

- Vessel Overpressure
- Updated Final Safety Analysis Report (UFSAR) Events
- Anticipated Transients Without Scram (ATWS)

- Emergency Core Cooling (ECCS)/ Loss-of-Coolant Accident (LOCA)
- Containment Response and Loads
- High Pressure Systems Performance
- Appendix R Analysis

These subjects are affected by the changes in valve setpoints, setpoint tolerance, and capacity, as detailed above.

1.3 SUMMARY AND CONCLUSIONS

A summary of the results of the evaluations for each of the subjects of concern is provided in Table 1-1. The evaluation determined that the effect of the setpoint tolerance increase on the following subjects are acceptable: 1) Vessel Overpressure, 2) UFSAR Events, 3) ATWS Analysis, 4) Containment Response and Loads Assessment, and 5) High Pressure Systems Performance. These specific subjects were addressed in detail as described in this report.

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

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

Subject	Section	Result
Vessel Overpressure, Transient Analysis and SSV Margin	2.0	Acceptable[1,4]
ATWS Analysis	3.0	Acceptable[1]
ECCS/LOCA Analysis	4.0	Acceptable
Containment Response	5.0	Acceptable[5]
High Pressure Systems Performance	6.0	Acceptable[2,3]
Appendix R Analysis	7.0	Acceptable

1. These evaluations did not include any SRV or SSV OOS.
2. Motor Operated Valve (MOV) operation will be assessed by Entergy to ensure the requirements described in section 6 are met.
3. The Standby Liquid Control System (SLCS) performance will be assessed by Entergy to ensure the requirements described in section 5 are met.
4. Effects on Interfacing System Piping Design Pressures will be assessed by Entergy.
5. SRV and SSV Dynamic Loads will be evaluated by Entergy.

2 VESSEL OVERPRESSURE/ANTICIPATED OPERATIONAL OCCURRENCE EVALUATION

2.1 ANALYSIS OVERVIEW

Reference 1 presents a generic evaluation of the effect of increasing the setpoint tolerance to ± 3% for safety valves in the pressure relief system. PNPS's configuration for Safety and Relief valves includes four safety-relief valves (SRV) that are piped to the suppression chamber and each of the two SSVs discharge to the drywell. This section presents the results of the plant specific evaluations associated with the increase of both the setpoint and the setpoint tolerance of the SSVs and the SRVs and the increase of the high pressure recirculation pump trip by 40 psi (1180 to 1220 psig) and the reduction of the ATWS high pressure feedwater pump trip by 100 psi (1415 to 1315 psig). The setpoint increases are 40 psi for both the SRVs (1115 to 1155 psig) and SSVs (1240 to 1280 psig). The setpoint tolerance is increased from ± 1% to ±3%. In addition, the existing SSVs will be replaced with new valves having a larger throat diameter, flow capacity, and performance characteristics that are consistent with the analysis presented in this report. The capacity of the SSVs was increased from 644,501 lbm/hr to 1,126,200 lbm/hr. The SSV capacity was increased to provide the analytical margin required to allow the SRV and SSV set pressure to be increased (to increase SRV simmer margin) and to allow the SRV and SSV set pressures tolerances to be increased. Table 2-1 lists the SRV/SSV configurations (current and new). In this section, SSVs are defined as valves that are qualified for use in the American Society of Mechanical Engineers (ASME) overpressure protection analysis. SRVs are valves that function as relief valves but are also qualified for use in ASME overpressure protection analysis. The high reactor pressure recirculation pump trip was also increased 40 psi, consistent with the increase in SRV opening setpoints. This increase retains the margin between the SRV opening setpoint and the high reactor pressure recirculation pump trip and this increase is included in these analyses. It was also necessary to reduce the ATWS high pressure feedwater pump trip setpoint. This was necessary to ensure that all ATWS events, that have a Main Steam Isolation Valve (MSIV) closure, trip the feedwater pumps on high pressure.

In addition to the plant specific overpressure analysis, a plant specific review of the events in the UFSAR was performed to determine if any other events are impacted by change to the safety and relief valve configuration. This review is summarized in Table 2-3. Based on the generic evaluation in Reference 1 and the review of the transient and accident events in Table 2-3, the overpressure analysis was evaluated with the SSVs and SRVs at the +3 % limit and the Loss of Feedwater Event was reviewed with the SSVs and SRVs setpoints at the –3% limit. All other events were determined to be unaffected by the change in setpoint tolerance.

Table 2-1: SRV and SSV Configuration

SRV and SSV Configuration	# SRV	# SSV	Nominal Setpoints (psig)		SRV and SSV Setpoint Tolerance	Capacity[1] (lbm/hr)	
			SRV	SSV		SRV	SSV
Current	4	2	1115	1240	± 1%	870,000	644,501
New	4	2	1155	1280	± 3%	870,000	1,126,200

Notes: 1. The reference pressure, for the capacities, are 1122.7 psig for SRV and 1277.2 psig for SSV. These pressures include 3% accumulation. The flow rate in the transient simulation varies primarily based on the upstream pressure condition relative to the reference pressure.

In addition, this evaluation proposes to increase the Technical Specification dome pressure Safety Limit to 1340 psig. Section 2.2 provides the assessment to increase the dome pressure safety limit.

2.2 OVERPRESSURE ANALYSIS

The most recent Main Steam Isolation Valve Closure with Flux scram (MSIVF) transients (Cycle 18 Reload Licensing), for PNPS, were analyzed with the current SSV/SRV configuration and high pressure recirculation pump trip setpoint. The reload evaluation contains both GE14 and GNF2 fuel. These results and those with the new configuration with the increase to the high pressure recirculation pump trip are provided in Table 2-2 below. These results demonstrate that the dome pressure safety limit (1325 psig) and the peak vessel pressure limit (1375 psig) are met when analyzed with a 3% setpoint tolerance. The overpressure analyses were performed in accordance with the methodologies described in Reference 2.

These results show that the margin to the current dome pressure safety limit is less than 5 psi, but there is still large margin to the ASME overpressure limit of 1375 psig with over 30 psi of margin. This shows that the dome pressure safety limit has excess conservatism. The safety limit is placed on the dome pressure to have a plant measurable parameter to demonstrate compliance with the vessel pressure limit of 1375 psig. These results show that there is less than a 20 psi difference between the peak vessel pressure and the peak dome pressure. The dome pressure safety limit is proposed to increase to assure margin is available for cycle-to-cycle variation in cycle specific overpressure calculations while retaining margin in the vessel to dome pressure difference. Establishing a safety limit at 1340 psig provides margin to the ASME overpressure analysis and provides for a 35 psi pressure difference between the vessel bottom to dome. This 35 psi is 75% higher than the observed pressure difference (~20 psi) for the limiting ASME overpressure event.

Table 2-2
PNPS Cycle 18 Overpressure Results
With Current and New SRV and SSV Configuration

Power (% Rated)	Flow (% Rated)	# SSVs Credited	#SRVs Credited	Peak Dome Pressure (psig)	Dome Pressure Safety Limit (psig)	Peak Vessel Pressure (psig)	Vessel Pressure Limit (psig)	SRV/SSV Configuration
102	107.5	2	4	1279	1325	1298	1375	Current
	76.7	2	4	1280	1325	1296	1375	Current
102	107.5	2	4	1322	1325	1341	1375	New
	76.7	2	4	1323	1325	1340	1375	New

2.3 REVIEW OF FSAR TRANSIENT AND ACCIDENT EVENTS

The following table describes the effect of increasing both the SRV and SSV setpoints and setpoint tolerance and increasing the high pressure recirculation pump trip setpoint (high reactor pressure recirculation pump trip). In the event discussions in Table 2-3, note that if there is no SRV actuation, there will be no high reactor pressure recirculation pump trip. This results because the high reactor pressure recirculation pump trip was moved above the new SRV setpoint to be consistent with the current setpoints. The most limiting event for overpressure is the MSIVF and this event was analyzed in Section 2.2.

This evaluation considered the effect of lowering the ATWS high pressure feedwater pump trip. The feedwater pump trip has the potential to reduce the peak pressure due to the reduction in vessel level. No Anticipated Operational Occurrence (AOO) has a peak dome pressure high enough to reach the new ATWS high pressure feedwater pump trip setpoint. Thus, these events are not impacted by lowering that setpoint. The only event that may reach the high pressure feedwater pump trip is the MSIVF. The MSIVF event is not an AOO and is not expected to occur due to the additional failure of the MSIV position scram. For the MSIVF, the peak vessel and dome pressure occur so quickly that the feedwater pump trip has no beneficial effect for this event.

Table 2-3: FSAR Transient and Accident Event Review

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

Turbine Trip	
With Bypass (TTWBP)	Severity varies with BPV capacity and the results are bounded by the Turbine Trip without Bypass event. [[ ]]
Without Bypass (TTNBP)	This transient results in a large vessel pressurization and increase in reactor power and is analyzed on a reload specific basis for CPR as well as for pressure margin to the SSV setpoints. [[ ]] [[ ]] Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
MSIV Closure	
Direct Scram (MSIVD)	This transient is not limiting from a CPR perspective because of the slow steam flow shutoff rate associated with the MSIV stroke times compared to turbine valves. This transient is analyzed on a cycle specific basis to determine the pressure margin to SSV setpoints. [[ ]]
Flux Scram (MSIVF)	This transient is analyzed on a cycle specific basis to ensure that the ASME boiler code requirements and dome pressure TS safety limits are met. The peak vessel pressure increases as the SSV opening setpoints are increased. This transient has been analyzed using the upper bound of the 3 % tolerance for the SSV opening setpoints and the safety mode of the dual mode relief valve opening setpoints.
Single MSIV Closure	This event is bounded by the MSIVD transient for peak pressure and is a non-limiting MCPR transient compared to other analyzed pressurization events. While the peak pressure is expected increase by a few psi, this transient is not materially impacted by the changes to the SSV capacity, safety and relief setpoints, setpoint tolerance and high reactor pressure recirculation pump trip. [[ ]]
Loss of Condenser Vacuum	This event is similar to a Turbine Trip event with no bypass, but there is a period of time where bypass valve flow is available. The duration of the bypass valve flow depends on the rate of loss of vacuum. Because of the limited bypass flow, the event is less severe than a turbine trip without bypass. [[ ]]
Loss of Auxiliary Power	This is a delayed turbine trip following a recirculation pump trip. This event is bounded by other pressurization events and the SRV configurations change does not change this conclusion. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip. [[ ]]
Loss of Feedwater Flow (LOFW)	This transient results in a low level scram followed by a low-low level isolation. The transient is not limiting from a CPR perspective and because of the time delay between the scram and the MSIV closure, this event is far from limiting from an overpressure concern. The low reactor water level scram setpoint is not impacted by the setpoint tolerance change because the low reactor water level scram setpoint is reached before any valve actuation occurs.
Decrease in Reactor Coolant System Flow Rate	
Trip of One Recirculation Pump	

Motor-Generator (M/G) Set Field Breaker Trip	This event results in a pump coast-down and power decrease. No safety or relief valve actuations occur during this transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
M/G Set Drive Motor Breaker Trip	This event results in a pump coast-down and power decrease. No safety or relief valve actuations occur during this transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Trip of Two Recirculation Pumps	
M/G Set Field Breaker Trip for both pumps	This event results in a flow coast-down and power decrease and may result in high Level Turbine Trip after a significant power decrease. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
M/G Set Drive Motor Breaker Trip for both pumps	This event results in a flow coast-down and power decrease and may result in high Level Turbine Trip after a significant power decrease. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Recirculation Flow Controller Malfunction	
Decreasing Flow	This event is similar to field breaker trip and results in a power decrease. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Shaft Seizure	
Two Loop Operation	This event results in a rapid flow decrease, which causes reactor power to decrease. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Single Loop Operation	This event results in a rapid flow decrease, which causes reactor power to decrease. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Reactivity and Power Distribution Anomalies	
Control Rod Withdrawal Error	
During Startup	This transient results in a power increase from very low powers. The increase in reactor power can occur at a high rate, but the neutron monitoring system is designed to limit the peak power achieved during the transient. The peak powers achieved are sufficiently low such that no safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
At Power	This transient results in a power increase due to increased reactivity associated with the control rod withdrawal. The increase in reactor power occurs at a moderate rate. The pressure regulator maintains vessel pressure and no safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Startup of an Inactive Recirculation Loop	[[ ]] The pressure regulator maintains vessel pressure and no safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.

Flow Controller Failure – Increasing Flow	The rapid flow increase results in a power increase that occurs at a moderate rate. The pressure regulator maintains vessel pressure and no safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Slow Flow Run-out	The slow recirculation flow run-out transient is not an original Final Safety Analysis Report (FSAR) event, however it is the basis for the flow dependent off-rated limits. This event assumes a slow increase in recirculation flow rate in both loops from the minimum core flow to the maximum core flow. This analysis is a conservative process for evaluating flow run-out events. The slow increase in core flow causes an increase in reactor power and corresponding increase in steam flow. The pressure regulator maintains vessel pressure and no safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Mislocated Fuel Assembly Accident	This scenario is modeled with a 3 dimensional core simulator code. The event does not result in increased pressure or SSV actuation. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Misoriented Fuel Assembly Accident	This scenario is modeled with lattice physics codes in the bundle design process. The event does not result in increased pressure or SSV actuation. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Control Rod Drop Accident	This results in a very rapid increase in neutron flux and a corresponding increase in fuel temperature. A reactor scram terminates the transient. The pressure regulator maintains vessel pressure. No safety or relief valve actuation occurs during the transient. Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Increase in Coolant Inventory	
Inadvertent High Pressure Coolant Injection (HPCI)	This event is analyzed on a reload specific basis for CPR and margin to SSV. This is an event where the HPCI system is inadvertently initiated. The increased core subcooling causes power to increase. It is possible that the inadvertent HPCI initiation could cause water level to increase to the high reactor water level turbine trip setpoint resulting in a turbine trip. This event is similar to the FWCF. As with the FWCF, the MCPR occurs prior to any SRV actuation. [[ ]] Therefore, this transient is not impacted by the changes to the SSV capacity, safety and relief valve setpoints, setpoint tolerance change and high reactor pressure recirculation pump trip.
Decrease in Reactor Coolant Inventory	
One SRV Opening	This event is not limiting with respect to MCPR because the event results in a very small power change. Because there are no changes to the SRV capacity, this transient is not impacted by the changes to the SSV capacity, safety and relief valves setpoint, setpoint tolerance and high reactor pressure recirculation pump trip.
Instrument Line Break	These events are considered in the Loss of Coolant Analysis section of this report.
Steam Line Break (SLB) Outside Containment	
LOCA Inside Containment	
Radioactive Release from a Subsystem or Component	
Liquid Release due to Tank Failure	These events are evaluated for radiological consequences and are not affected by changes to the SSV capacity, safety and relief valves setpoint, setpoint tolerance and high reactor pressure recirculation pump trip.
Fuel Handling Accident	

2.4 REVIEW OF EQUIPMENT OUT OF SERVICE OPTIONS

In addition to the events in the UFSAR, the following equipment Out-of-Service (OOS) options were considered when determining the effect of the setpoint tolerance change:

1. Turbine Bypass OOS
2. Final Feedwater Temperature Reduction / Feedwater Heater(s) OOS
3. MSIV OOS
4. Single Loop Operation

Various combinations of equipment OOS options are allowed as described in Reference 2. These flexibility options are considered when performing critical power ratio and peak vessel pressure analyses. The minimum CPR for these events occurs before SRV and SSV lift. Therefore, changes to the SRV and SSV setpoint, setpoint tolerance and the high reactor pressure recirculation pump trip, does not affect the critical parameters for the equipment OOS options listed above. The Turbine Bypass OOS option considers the effects of not meeting the fast response performance analyzed for the reload. It does not remove the ability of the pressure regulator to open the bypass valves in an attempt to maintain vessel pressure for slow events such as the rod withdrawal error or loss of feedwater heating where core power and steam flow may increase above the rated value.

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

Single loop operation is at low power reduced flow condition. Transients initiated from these low power conditions have a reduced overpressure and are bounded by the limiting overpressure events analyzed at high power.

Therefore, the equipment OOS options listed above are not impacted by the valve setpoint tolerance increase.

2.5 MARGIN TO SPRING SAFETY VALVE SETPOINT

[[

]]

2.6 EFFECTS ON INTERFACING SYSTEMS

The piping systems connected to the Reactor Pressure Vessel (RPV) require evaluation for a Peak Vessel Bottom Pressure of 1375 psig, as appropriate. These systems include Reactor Recirculation, HPCI, Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR), Low Pressure Coolant Injection (LPCI), Core Spray, SLCS, Reactor Water Cleanup, Main Steam, Feedwater, Reactor Vents & Drains, and the RPV Flange Leakoff System. Entergy shall ensure that the piping systems affected by the proposed SRV and SSV upgrade meet the requirements of the applicable piping code (ASME or B31.1) for a maximum reactor vessel dome pressure of 1340 psig and maximum vessel bottom pressure of 1375 psig.

3 ATWS ANALYSIS

3.1 METHODOLOGY

3.1.1 Analytical Models

The GE computer model ODYN is used for the reactor transient analysis. The Nuclear Regulatory Commission (NRC) has approved the application of ODYN to ATWS evaluations. The major features of the model are:

- One-dimensional reactor core neutron kinetics and thermal hydraulics.
- Drift-flux upper plenum and bulk water modeling.
- Multi-node steam line for modeling acoustic phenomena.
- Balance-of-plant control systems and equipment models.
- Emergency protection systems, including boron injection.
- Boron transport and reactivity consistent with one-dimensional nodalization.

The GE computer model STEMP04 is used for the suppression pool heatup analysis. The STEMP analytical models have been accepted by the NRC in previous applications and other ATWS analyses. STEMP calculates the temperature rise of the suppression pool due to SRV and SSV discharge. The temperature is calculated using a mass and energy balance for the suppression pool, including the effects of steam entering the pool and the heat removal capability of the RHR heat exchangers. The integrated steam flow from the SRVs and SSVs, calculated by ODYN, is input to STEMP to calculate the Boron Injection Initiation Temperature (BIIT) time and the peak suppression pool temperature.

The May-Witt decay heat model is used in both the ODYN code for reactor transient analysis, and the STEMP code for suppression pool energy balance analysis. The May-Witt correlation bounds the ANSI/ANS-5.1-1979 + 2σ decay heat curves during the ATWS events.

3.2 KEY ASSUMPTIONS

Item	Parameter	Reference/Basis
1	Core and Fuel Design	The ATWS analyses are performed using a representative core and fuel design based on the reference loading pattern for PNPS Cycle 18.
2	Reactor Water Level	The ATWS analyses are performed with reactor water level controlled at Top of Active Fuel (TAF) + 5 feet due to limitations of the ODYN computer program. This limitation was factored into NRC approval for application of ODYN to ATWS.
3	Reactor Depressurization	Reactor depressurization cannot be modeled with ODYN. In the ATWS analyses, Automatic Depressurization System (ADS) operation is inhibited, and the vessel cycles on SRV setpoints until the reactor is shutdown.
4	Decay Heat	The May-Witt decay heat correlation is used in the suppression pool temperature calculation following reactor shutdown. The May-Witt decay heat correlation is known to be conservative relative to the 1979 ANS 5.1 + 2σ curve.
5	ATWS Parameters	The ATWS analyses are generally a best estimate calculation at the maximum power and the lowest core flow at the Maximum Extended Load Line Limit Analysis (MELLLA) boundary. The NRC has approved the use of less conservative parameters in the ATWS analyses. Equipment that is not OOS is assumed to function per its design.
6	ATWS Mitigation Equipment Performance	• Alternate Rod Insertion is assumed to fail. • ATWS Recirculation Pump Trip (RPT) and SLCS are assumed to function as designed. The SLCS pump discharge relief valves are assumed to not lift during the ATWS events. The acceptability of SLCS during ATWS is confirmed in the SLCS system evaluation. • Both loops of RHR are assumed to function with all pumps operational during the Pressure Regulator Failure-Open (PRFO) and MSIVC ATWS events.

Item	Parameter	Reference/Basis
7	SRV Operating Parameters	The SRV valve opening/closing setpoints are statistically spread around the upper Analytical Limit (AL) setpoint. The statistical spread is adequate for the SRV setpoint drift tolerance.
8	Fuel Cladding Oxidation	Cladding oxidation is insignificant if the Peak Clad Temperature (PCT) remains below 1600 °F. Consequently, no cladding oxidation calculation is performed if the calculated PCT is below 1600 °F.
9	Operator Action	Operator is assumed to initiate boron injection within 120 sec after high pressure ATWS recirculation pump trip setpoint has been reached.

3.3 ANALYSIS OVERVIEW

This section describes the effect of the setpoint tolerance increase, setpoint increase and SSV capacity increase on the PNPS ATWS analysis.

The ATWS analysis is performed in order to demonstrate that reactor integrity, containment integrity, and fuel integrity are maintained for scenarios where an automatic SCRAM fails to occur. Reactor integrity is demonstrated by ensuring that peak reactor vessel pressure is within the ASME Service Level C limit of 1500 psig. Containment integrity is demonstrated by ensuring that the peak suppression pool temperature is below the maximum bulk suppression pool temperature limit of 185°F and containment pressure is less than the containment design pressure limit of 56 psig. Fuel integrity is demonstrated by ensuring that peak cladding temperature is below the 10CFR50.46 limit of 2200°F and fuel local cladding oxidation is below the 10CFR50.46 limit of 17 % total clad thickness. Because the cladding temperature increase for ATWS is of short duration and limited magnitude, cladding oxidation is not explicitly calculated in the ATWS analysis.

[[

]].

Because the LOOP event, for PNPS, does not result in a reduction in the number of Residual Heat Removal (RHR) cooling loops, the LOOP event is not potentially limiting for the long-term suppression pool temperature or containment pressure response.

For the IORV event, the availability of the main condenser reduces the severity of the peak suppression pool temperature increase. The absence of reactor vessel isolation avoids the vessel pressurization; therefore, the peak vessel pressure remains at or below the initial value. A reactor power excursion does not occur and the fuel does not experience boiling transition. The IORV is not explicitly analyzed due to the non-limiting nature of the event for all acceptance criteria.

Both the MSIVC and PRFO ATWS events were evaluated for both Beginning of Cycle (BOC) and End of Cycle (EOC) exposure conditions and at rated core power (2028 MWt) and minimum core flow (76.7% of rated)

ATWS analyses were performed for the proposed SRV and SSV configuration changes, which include:

1. SRV setpoint increase of 40 psi and tolerance relaxation to +/- 3%
2. SSV setpoint increase of 40 psi and tolerance relaxation to +/- 3%
3. High reactor pressure recirculation pump trip setpoint increase of 40 psi
4. High reactor pressure feedwater pump trip setpoint decrease of 100 psi
5. SSV reference capacity increase from 644,501 lbm/hr to 1,126,200 lbm/hr at 1240 psig plus 3% accumulation.

The increased SRV, SSV and high reactor pressure recirculation pump trip setpoints would be expected to result in an increase of the ATWS peak vessel pressure. However, the replacement SSVs were selected to provide a large increase in SSV capacity and substantial decrease in the peak pressure. The Feedwater (FW) pump trip setpoint was reduced to ensure the peak dome pressure of all ATWS events, that have a MSIV isolation, will reach the FW pump trip setpoint. The feedwater pump trip provides both protection (1) against the overpressurization of the reactor coolant pressure boundary (lowers peak vessel pressure) and (2) against the suppression pool exceeding both pressure and temperature limits.

Both the upper tolerance limit (+3%) and the lower tolerance limit (-3%) were considered. The MSIVC and PRFO transients were re-evaluated using the design inputs summarized in Table 3-1 through 3-3, which include the configuration changes described above. The MSIVC and PRFO events were reanalyzed at beginning and end of cycle exposure points (i.e., BOC and EOC)

3.4 ANALYSIS INPUTS

Table 3-1 summarizes the initial conditions assumed for the ATWS event. These conditions are consistent with the initial conditions assumed for the ATWS analysis performed for the implementation of Thermal Power Optimization (TPO) (Reference 3).

Table 3-1: Summary of ATWS Key Input Parameters

Parameter	Value
Dome Pressure, psig	1035
Rated Core Flow, Mlbm/hr	69.0
Core flow, Mlbm/hr / % of Rated	52.9/76.7
Rated Power, MWt	2028
Power, MWt / % of Rated	2028/100
Steam Flow, Rated, Mlbm/hr	8.12
Feedwater Temperature, °F	364.3
Initial Dynamic Void Reactivity Coefficient, ¢/%	–12.6 (BOC) –11.0 (EOC)
Core Average Void Fraction, %	51.2 (BOC) 39.2 (EOC)
Initial Doppler Coefficient, ¢/°F	–0.13 (BOC) –0.13 (EOC)
Initial Suppression Pool Liquid Volume (ft^3)	84,000
Initial Suppression Pool Temperature (°F)	80
Initial Suppression Pool Mass, Mlbm	5.225
Initial Inventory in Condensate Storage Tank, lbm	883,000

Table 3-2 shows the initial axial power shapes for the beginning of cycle and end of cycle analyses. The ATWS analysis results are based on GE14 and GNF2 fuel and the analysis is cycle independent. These analyses are applicable to the current PNPS core with GNF2 reloads.

Table 3-2 Axial Power Shapes

Node Location (From Bottom of Active Fuel)	BOC (2028 MWt & 76.7% Flow)	EOC (2028 MWt & 76.7% Flow)
1	0.42	0.15
2	1.36	0.45
3	1.64	0.55
4	1.66	0.64
5	1.61	0.74
6	1.53	0.86
7	1.45	0.97
8	1.38	1.07
9	1.31	1.15
10	1.23	1.22
11	1.18	1.30
12	1.14	1.35
13	1.09	1.39
14	1.04	1.41
15	0.93	1.33
16	0.89	1.35
17	0.84	1.38
18	0.76	1.37
19	0.70	1.35

Node Location (From Bottom of Active Fuel)	BOC (2028 MWt & 76.7% Flow)	EOC (2028 MWt & 76.7% Flow)
20	0.62	1.27
21	0.52	1.13
22	0.41	0.92
23	0.19	0.42
24	0.10	0.22

Table 3-3 summarizes key equipment parameters and input values used in the ATWS analysis.

Table 3-3: Key Equipment Parameters

Parameter	Analysis Value
Nominal Closure Time of MSIV, sec	4.0
SRV System Capacity, % Nuclear Boiler Rated (NBR) Steam Flow at 1122.7 psig / Number. of Valves	42.9/4
SRV Nominal Opening Setpoint, psig	1155[1]
SRV Setpoint Tolerance, %	3
Safety/Relief Valve Closing Setpoint, % of Opening Setpoint	97
SRV Time Delay On Opening Signal, sec	0.4
SRV Opening Stroke Time, sec	0.15
SRV Closure Time Delay, sec	0.4
SRV Closure Stroke Time, sec	0.15
SSV System Capacity, % NBR Steam Flow at 1240 psig / Number of Valves	27.7/2

Parameter	Analysis Value
SSV Nominal Opening Setpoint, psig	1280
SSV Setpoint Tolerance, %	3
SSV Closing Setpoint, % of Opening Setpoint	94
Recirculation Pump Trip Logic Delay, sec	0.53
ATWS High Pressure Recirculation Pump Trip Setpoint, psig	1220[2]
ATWS High Pressure Feedwater Pump Trip Setpoint, psig	1315[3]
Feedwater Pump Trip Logic Delay, sec	0.6
SLCS Injection Location	Lower Plenum Standpipe
SLCS Injection Time, sec after High Pressure ATWS High Pressure Recirculation Pump Trip	120
SLCS Injection Rate per Pump, gpm	39
Minimum Boron–10 Enrichment, %	54.5
Sodium Pentaborate Concentration, %	8.42
BIIT, °F	110
SLCS Liquid Transport Time, sec	60
SLCS Liquid Solution Enthalpy, Btu/lbm	78
Time to Inject Hot Shutdown Boron Weight, sec	1538
HPCI Flow Rate, gpm	4250
Enthalpy of the HPCI Flow, Btu/lbm	81.0
RCIC Flow Rate, gpm	400

Parameter	Analysis Value
Enthalpy of the RCIC Flow, Btu/lbm	81.0
Low Pressure MSIV Isolation Setpoint, psig	782.3
Number of RHR Loops	2
Number of RHR Loops for LOOP event	2
RHR Service Water Temperature °F	75.0
RHR Heat Exchanger K–Factor per Loop in Containment Cooling Mode, Btu/sec– °F	175.0
RHR Heat Exchanger K–Factor per Loop during LOOP, Btu/sec– °F	175.0

Notes:

1. The analysis setpoints were determined by (1) adding the 3% tolerance (1155 to 1190 psig) and (2) then applying a statistical spread (about 1190 psig). The resulting opening setpoints were then used in the ATWS evaluations.
2. The current ATWS high pressure recirculation pump trip setpoint upper limit is 1180 psig. The proposed ATWS high pressure recirculation pump trip setpoint Allowable Value (AV) used in this analysis is 1220 psig. The increase of 40 psi is consistent with the increase in nominal setpoint proposed for the SRVs and SSVs. This 40 psi increase provides the same level of protection to avoid tripping both recirculation pumps during expected transients like the TTWBP, LRWBP, and MSIVD.
3. The feedwater pump trip setpoint was originally set at 1415 psig. However, with the inclusion of the large capacity SSVs it became necessary to lower the setpoint to ensure that all ATWS isolation events will reach the feedwater trip setpoint. Less limiting ATWS events, with the turbine bypass available, are also expected to experience a high pressure feedwater pump trip. The new setpoint (1315 psig) is high enough to ensure only ATWS events or the ASME Over-Pressure Protection event (MSIVF) will trip the feedwater pump on high pressure.

In general, nominal operating and equipment parameters are utilized for the evaluation of ATWS. For key input such as the ATWS High Pressure RPT, Technical Specification AVs have been conservatively used in the analysis. The AVs are summarized below.

Function	Current AV	New AV
ATWS High Pressure RPT	1180 psig Upper	1220 psig Upper

3.5 ANALYSIS RESULTS

The ATWS analysis yielded similar results to previous ATWS analyses. The ODYN results from this analysis are summarized in Table 3-4 below. The suppression pool temperature, suppression pool airspace pressure and integrated valve flows are shown in Table 3-5. A sequence of key events was developed for each of the transients analyzed. These are provided in Tables 3-6 through 3-9. Table 3-10 shows the ATWS acceptance criteria and the applicable limiting results, and Table 3-11 describes the peak pressures for other system evaluations. Plots of key ODYN outputs were generated for each of the transient analyzed and these are provided in Figures 3-1a through 3-4c. Plots of suppression pool temperature and suppression pool airspace pressure verses time are provided for the MSIVC and PRFO transients at beginning and end of cycle in Figures 3-5 through 3-8. Finally, Figure 3–9 shows the lower plenum pressure for the limiting event (PRFO BOC).

Table 3-4: Summary of Key ODYN Parameters[1] for ATWS Calculation

Event	Power (MWt) /Flow (%)	Exposure	Peak Neutron Flux (%)[2]	Peak Heat Flux (%)[2]	Peak Vessel Press. (psig)
MSIVC	[[				
MSIVC					
PRFO					
PRFO					]]

Notes: 1. Values enclosed in parentheses represent the time of peak values in seconds.
2. The peak neutron and heat fluxes are normalized to their respective initial power.

Table 3-5: Summary[1] of Peak Suppression Pool Temperature, Containment Pressure and Integrated SRV Flow

Event	Power (MWt) /Flow (%)	Exposure	Peak Suppression Pool Temperature (°F)	Peak Suppression Pool Airspace Pressure (psig)	Integrated SSV and SRV Flow at Hot Shutdown[2] (10^5 lbm)
MSIVC	[[				
MSIVC					
PRFO					
PRFO					]]

Notes: 1. Values enclosed in parentheses represent the time of peak values in seconds.

2. Hot shutdown, in the ODYN ATWS evaluation, is defined as when the neutron flux is less than 0.1% for more than 100 sec.

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

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

Table 3-7: Sequence of Events for PRFO at BOC

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

Table 3-8: Sequence of Events for MSIVC at EOC

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

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

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

Table 3-10: Acceptance Criteria Results

Acceptance Criteria	Allowed Value	Limiting Result	ATWS Event and Conditions
Peak Vessel pressure (psig)	1500	1478	[[
Peak Cladding Temperature (°F)	2200	Note 1	
Peak Suppression Pool temperature (°F)	185	175.9	
Peak Suppression Pool pressure (psig)	56	10.4	]]

Notes: 1. Fuel integrity is demonstrated by ensuring that peak cladding temperature is below the 10CFR50.46 limit of 2200°F and fuel local cladding oxidation is below the 10CFR50.46 limit of 17 % total clad thickness. The peak cladding temperature for ATWS has been confirmed to be less than 1600°F for GNF2 and GE14. Because the cladding temperature increase for ATWS is of short duration and limited magnitude, cladding oxidation is not explicitly calculated in the ATWS analysis.

Table 3-11: Peak Pressures for Other System Evaluations

Parameter	Value	Comments
Lower Plenum Pressure	1212 psig	The lower plenum pressure for all transients was reviewed and compared to the initiation time of the SLCS pumps. 1212 psig is the highest lower plenum pressure that occurs after the initiation of the SLCS pumps.

Figure 3–1a: MSIVC BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–1b: MSIVC BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–1c: MSIVC BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–2a: PRFO BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–2b: PRFO BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–2c: PRFO BOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–3a: MSIVC EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–3b: MSIVC EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–3c: MSIVC EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–4a: PRFO EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–4b: PRFO EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–4c: PRFO EOC – GE14 and GNF2 Fuel

[[

]]

Figure 3–5: Containment Response MSIVC BOC

[[

]]

Figure 3–6: Containment Response PRFO BOC

[[

]]

Figure 3–7: Containment Response MSIVC EOC

[[

]]

Figure 3–8: Containment Response PRFO EOC

[[ ]]

Figure 3–9: Lower Plenum Pressure PRFO BOC

[[ ]]

3.6 CONCLUSIONS

The ATWS evaluation incorporating the SSV capacity increase, SRV/SSV setpoint increase, setpoint tolerance increase to 3% and the increase in the ATWS recirculation pump trip setpoint confirms that all ATWS acceptance criteria are met. Therefore, the implementation of these increases in SRV/SSV setpoints, setpoint tolerance and ATWS pump trip setpoint at the PNPS is acceptable. The ATWS evaluations are based on a 39 gallon per minute Standby Liquid Control System injection rate with a 8.42% weight concentration of Sodium Pentaborate solution containing 54.5% enriched Boron-10. The peak lower plenum pressure during the operation of the Standby Liquid Control System is 1212 psig. The Standby Liquid Control system is required to attain an equivalent Boron injection rate with a lower plenum pressure up to 1212 psig in order for these analyses to remain valid. The Standby Liquid Control System performance is addressed in Section 5.3. The high pressure feedwater pump trip setpoint is reduced and is a required function for ATWS mitigation.

4 ECCS/LOCA EVALUATION

4.1 ECCS/LOCA PERFORMANCE EVALUATION

The most recent SAFER/GESTR-LOCA analysis for PNPS is reported in References 4 and 5. The analysis was performed using the SAFER/GESTR-LOCA application methodology approved by the NRC.

The impact of safety valve setpoint relaxation on the ECCS-LOCA performance for Boiling Water Reactor (BWR) 2-6 plants has been evaluated on a generic basis in the Boiling Water Reactor Owners Group (BWROG) report approved by the NRC (Reference 1). The ECCS conclusions contained in Reference 1 apply to Reference 4.

[[

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

It is also noted that SSV actuation is not predicted for the LOCA analysis and therefore, the SSV modifications do not impact the ECCS SAFER/GESTR-LOCA Analyses.

5 CONTAINMENT RESPONSE AND LOADS ANALYSIS

This section presents the results of the various containment related evaluations in support of the SRV and SSV 40 psi setpoint increase and a setpoint tolerance increase from 1% to 3%. This evaluation also addresses an increase in the capacity of the SSVs from 644,501 lbm/hr to 1,126,200 lbm/hr @ 1240 psig +3% overpressure accumulation.

5.1 CONTAINMENT PRESSURE AND TEMPERATURE FOR DBA LOCA

The Design Basis Accident (DBA)-LOCA short-term containment analyses includes the large Recirculation Suction Line Break (RSLB) which is used to establish the peak drywell pressure and also used to establish the containment conditions used to evaluate the DBA-LOCA hydrodynamic loads. The DBA-LOCA analysis does not predict actuation of SSVs or SRVs because the break size is large enough to cause reactor vessel depressurization without actuation of the SRVs. Therefore, the current short-term containment analysis for PNPS remains unaffected for the SRV and SSV changes. The effects on the peak suppression pool temperature and wetwell pressure for the long-term DBA-LOCA were considered. Changes to the SSV and SRV safety valve setpoints, tolerance and SSV capacity have no effect on the DBA-LOCA event because the vessel depressurizes without any SRV or SSV actuations. 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 DBA-LOCA containment pressure and suppression pool temperature used to demonstrate available Net Positive Suction Head (NPSH) margins are also unaffected.

5.2 SMALL STEAM LINE BREAKS

Small steam line break spectrum established in Reference 6 (0.05, 0.10, 0.35, 0.50 and 1.0 ft^2 breaks) and Reference 7 (0.01 and 0.02 ft^2 breaks) was evaluated to determine the impact on the drywell temperature for generating the Environmental Qualification (EQ) curve. The SLBs that generally produce the most limiting peak drywell temperature are large enough to maintain the initial vessel pressure below the SRV setpoints and also large enough to depressurize through the break without requiring SRV actuation. Therefore, an increase in SRV opening setpoint and tolerance has no effect on the larger SLBs that do not have SRV actuation. The drywell temperature response for smaller SLBs that require SRV actuation may be slightly affected. For these breaks, the peak drywell temperature is well below that of the larger limiting SLB. Furthermore, the peak drywell temperature for the smaller SLBs occurs later in the event at the time the drywell sprays are actuated. Since this time occurs after many 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 setpoint and tolerance increase. 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 defined vessel depressurization rate, the initial vessel liquid inventory, and decay heat. These factors are not affected by the changes to the SRV's. The break steam flow to the drywell is controlled by the vessel pressure response, which is determined by the assumed vessel depressurization rate. This parameter is also unaffected by the change in the SRV setpoint and tolerance. Since the steam break flow and drywell spray temperature response for the smaller SLBs are not impacted by the SRV changes, the drywell temperature response for the smaller SLBs is also not impacted. The SRV's maintain the vessel pressure below the SSV setpoint for even the smallest break analyzed so there is no SSV actuation. Therefore, the subject SRV and SSV changes have no impact on the bounding drywell temperature response and the EQ curve remains valid.

SLB's from 1.0 ft^2 to as small as 0.01 ft^2 were also evaluated in the NPSH assessment where the impact of the setpoint tolerance increase was determined to have negligible impact on the long term suppression pool temperature.

5.3 IBA AND SBA

The effect on Intermediate and Small Break Accidents (i.e., IBA and SBA) was also evaluated. The containment pressure and temperature response for the IBA and the SBA, were originally evaluated as part of the Mark I Containment Program and documented in the Plant Unique Load Definition report (PULD – Reference 8). PNPS profiles from Reference 8 are calculated in accordance with the generic Mark I Load Definition Report (Reference 9) and are based on endpoint type calculations, which are controlled by the amount of initial stored energy in the primary system and decay heat. There is no increase in the initial primary system stored energy or decay heat due to an increase in the SRV safety valve setpoint and tolerance. Therefore, there is no effect on the IBA and SBA event results presented in Reference 8. Additionally, for the SBA the drywell temperature response is taken to be bounding, constant value of 340°F. This bounding drywell temperature value would not change due to an increase in SRV setpoint tolerance.

5.4 NUREG-0783 LOCAL SUPPRESSION POOL TEMPERATURE

The Reference 10 plant drawings identified that the PNPS SRV quencher elevation is above the Emergency Core Cooling System (ECCS) pump suction elevation and therefore, per the NRC conditional acceptance of NEDC-30832 (Reference 11), PNPS is no longer required to meet NUREG-0783 local suppression pool temperature limits for events with SRV discharge.

5.5 DBA LOCA HYDRODYNAMIC LOADS

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

5.6 SRV AND SSV DYNAMIC LOADS

The SRV discharge loads are defined by parameters which include:

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

Since an SRV setpoint increase and the setpoint tolerance relaxation will increase the SRV safety valve opening pressure, the SRV discharge dynamic loads are expected to increase. Entergy will need to evaluate the factors affecting the SRV dynamic loads.

The SSV discharge loads are affected by parameters which include the SSV flow capacity and SSV opening pressure.

Since an SSV setpoint increase and setpoint tolerance relaxation will increase the SSV safety valve opening pressure and an increase in the SSV throat size will increase the SSV flow capacity, the SSV dynamic loads are expected to increase. Entergy will need to evaluate the factors affecting the SSV dynamic loads..

5.7 RIPD EVALUATION

During normal operation, there is no SRV or SSV actuation. Therefore, the SRV setpoint and tolerance change or SSV modification have no effect on the Reactor Internal Pressure Differences (RIPDs) at normal conditions.

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

]] In addition, SSV does not actuate due to higher setpoint than SRV and SRV actuation to maintain pressure below SSV setpoint. Therefore, the RIPD results at upset conditions remain valid for the SRV setpoint and tolerance change.

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

]] In addition, SSV is not part of ADS. Thus, the RIPD results at emergency conditions are still applicable for the SRV setpoint and tolerance increase or SSV modification.

The limiting faulted event for RIPD is an instantaneous circumferential break of one main steam line, for which SRV or SSV does not actuate. Therefore, the SRV setpoint and tolerance change or SSV modification has no effect on the RIPD results at faulted conditions.

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

5.8 ATWS

Suppression Pool temperatures during an ATWS are provided as reported in section 3 of this report.

The increase in SSV capacity with its discharge directly to the Drywell (DW) has the potential to increase the DW temperature and challenge the DW temperature for qualification of the Neutron Monitoring System equipment. This was evaluated as a BWROG activity in response to RG 1.97 and is addressed in GE-NE-C5100121-01 (Reference 12).

GE-NE-C5100121-01 (Reference 12) documents a BWROG activity to evaluate the drywell temperature response during an ATWS for those few Mark I plants that have SSV which discharge directly to the drywell atmosphere, including PNPS. That report identified Dresden as a bounding plant and an analysis was performed, based on the Dresden reactor system and containment configuration which could be applied to all affected plants in this group.

The Dresden plant was identified as the bounding plant based upon a comparison of selected ratios of critical parameters for each plant identified. The bounding plant selection criteria was based upon the following ratios:

- SSV capacity to rated thermal power
- SSV capacity to DW volume

- SRV capacity to rated thermal power

The resulting drywell atmosphere temperature and pressure response was provided in Reference 12 to the participating BWROG members, for their use in qualifying the Neutron Monitoring System (NMS) instrumentation for ATWS drywell atmosphere conditions.

The change to the PNPS SSV capacity was reviewed to assure that PNPS remains bounded by the Dresden analysis results in Reference 12. The proposed changes included an increase in the SSV capacity from 644,501 lbm/hr to 1,126,200 lbm/hr @ 1240 psig +3% overpressure accumulation which with 2 SSVs would increase the total SSV capacity from the previously assumed 1290000 lb/hr to 2252400 lb/hr @ 1240 psig +3% accumulation.

Based upon the minimum DW volume of 132000 ft^3 as identified in Table 5.2-1 of the PNPS FSAR, the resulting SSV capacity to DW volume ratio is increased from the value of 8.8 $lbm/hr/ft^3$ in Table 1 of Reference 12 to a new value of 17.1 $lbm/hr/ft^3$. However, this value remains substantially below the Dresden value of 31.3 $lbm/hr/ft^3$ reported in Table 1 of Reference 12 and therefore still bounded by the Dresden value defined as the bounding plant.

The SRV capacity is not impacted by the proposed changes and therefore the previous PNPS ratio of SRV capacity to rated thermal power of 1606 lbm/hr/MWt will remain well bounded by the Dresden value of 1088 lbm/hr/MWt as reported in Table 1 of Reference 12.

Based upon the above parametric evaluation, the PNPS ATWS drywell temperature response with the proposed changes to the SRV and SSV setpoints, tolerances and capacities is still expected to be bounded by the ATWS drywell temperature profile defined in Reference 12 for qualification of the Neutron Monitoring System.

6 HIGH PRESSURE SYSTEMS PERFORMANCE

6.1 HPCI & RCIC – SYSTEM DESCRIPTION AND FUNCTIONS

The HPCI system, an ECCS, is designed to provide sufficient core cooling to prevent excessive fuel cladding temperature in the event of a small break loss of coolant accident 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. HPCI was designed to pump water at the rate of 4250 gpm into the reactor vessel over a wide range of pressures, from 165 psia to 1135 psia. PNPS requirements are for HPCI to deliver 4250 gpm to the reactor vessel over the range of 150 psig to 1000 psig, and 3000 gpm to a peak reactor pressure of 1126 psig. 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 vessel 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. RCIC was designed to supply make up water to the reactor at a capacity of 400 gpm over a dome pressure range of 165 psia to 1135 psia. In 1982 an SRV setpoint increase to 1115 psig resulted in an increased pressure range of 165 psia to 1141 psia for the RCIC system (Reference 13).

The most significant effect of increasing the SRV setpoint and relaxing the SRV tolerance on the HPCI and RCIC systems operation is the maximum dome 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 SRVs setpoints (including drift). Increasing the allowable setpoint tolerance to 3% while increasing the current setpoint of 1115 psig to 1155 psig increases the maximum vessel pressure for HPCI and RCIC injection by 64 psi to 1205 psia.

6.1.1 Pump Flow and Head

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

If the original conservatism in the assumed head losses is maintained, then the Total Dynamic Head (TDH) for both systems is increased from 2800 to 2962 feet. This 162 feet increase in TDH corresponds to a 70 psi increase in the pressure range (e.g., 1135 psia increases to

1205 psia). HPCI performance remains acceptable. The RCIC pump speed is required to be increased to at least 4628 rpm to meet the new TDH (with no allowance for pump degradation).

6.1.2 System Boundary Components

An assessment of pressure integrity for the HPCI and RCIC System was made by comparing the design conditions for the piping and components at the higher system operating conditions. The maximum reactor pressure and temperature design conditions do not change. There is no change to the design requirements of the piping and components attached to the reactor vessel

6.1.3 Pump and Turbine

The pump and turbine speed required to meet the TDH requirement for the HPCI and RCIC systems is bounded by the current system design. There is no change in the HPCI turbine or pump power requirements, steam flow to the turbine, or system reliability.

To meet the new TDH requirements the RCIC turbine and pump speed, power, and steam flow requirements require an increase to meet the discharge pressure requirements.

6.1.4 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 conditions exceed the conditions required for the SRV setpoint and setpoint tolerance increase, there is no change to the design pressure requirements of the instrumentation. Because there is no increase in make-up flow to the vessel, for either HPCI or RCIC, no change is required for flow instrumentation. Exhaust trip setpoints and rupture disks do not need to be changed because changes in exhaust pressure, for either turbine, are negligible.

There is no increase in steam flow for HPCI. Therefore, there is no change in leak detection setpoints. RCIC steam flow will increase, due to an increase in turbine horsepower, and the leak detection setpoint for high steam flow will require revision.

Because there is no increase in maximum containment pressure during an accident, the high exhaust pressure trip remains unaffected.

6.1.5 Motor Operated Valves

The system valves that are impacted by reactor pressure will require re-evaluation by the PNPS Generic Letter 89-10 program 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 shall be revised to reflect the increased SRV tolerances and the higher system operating pressures.

6.1.6 Start Up Transient

Because for HPCI there is no change in steam flow or maximum turbine speed, the slight increase in reactor pressure will not affect the start up transient. The affect of the increased steam flow to RCIC has been separately evaluated and will not affect the startup transient. Therefore, the time to reach the higher turbine speed required to inject at the higher pressure will remain within existing design requirements.

6.1.7 Key Assumptions

None

Design values for input parameters are the same for both units unless specifically noted.

6.1.8 Key Input for RCIC

Item	Parameter	Unit	Value
1	Maximum design value for system start time to rated flow (system in standby line up)	seconds	75
2	Maximum design value for reactor pressure for system operation	psia	1141
3	Maximum design value for system injection flow rate	gpm	400
5	Rated turbine speed	rpm	4500
6	Maximum design value for turbine exhaust pressure	psia	25
7	Turbine exhaust high pressure trip	psig	46
8	Maximum design pressure value for turbine	psig	1250
9	Maximum design pressure value for pump	psig	1500

6.1.9 Key Input for HPCI

Item	Parameter	Unit	Value
1	Maximum design value for system start time to rated flow (system in standby line up)	seconds	90
2	Maximum design value for reactor pressure for system operation	psia	1141
3	Maximum design value for system injection flow rate to 1000 psig	gpm	4250
4	Maximum design value for system injection flow rate to peak vessel pressure (item 2)	gpm	3000
5	Rated turbine speed	rpm	4000
6	Maximum design value for turbine exhaust pressure	psia	65
7	Turbine exhaust high pressure trip	psig	150
8	Maximum design pressure value for turbine	psig	1250
9	Maximum design pressure value for pump	psig	1500

6.1.10 Key Results

The RCIC and HPCI systems are designed to provide rated flow to the reactor up to maximum reactor dome pressure of 1141 psia. The maximum reactor dome pressure required for injection is the setpoint of the lowest group of SRVs, including setpoint drift. For a setpoint increase to 1155 psig and tolerance increase from 1% to 3%, the maximum dome pressure that the high pressure systems are required to inject rated flow, increases by 64 psi to 1205 psia (Tables 6-1 and 6-2).

The original system design specifications require a pump pressure rise (TDH) of 2800 feet in order for the systems to deliver water from the suppression pool to the reactor at the high pressure condition. The 2800 feet of head was based on a design specification value for piping losses and elevation changes. The new required TDH of 2962 feet is still met for HPCI because the rated flow of 4250 gpm is not required above 1000 psig. An increase in RCIC turbine speed (4628 rpm vs. the original 4500 rpm design specification) is required to provide a TDH of 2962 feet. The RCIC turbine has been evaluated up to a speed of 4700 rpm.

Because the turbine steam supply pressure increase is less than 65 psi, there is no effect on the systems' startup time, or the severity of the startup transient.

The NPSH available for the HPCI and RCIC pumps does not change because there is no change in required flow or allowable suction temperature. Pressure losses from the pump to the vessel, due to piping friction or elevation differences, do not change.

The SRV setpoint increase requires a corresponding change to the pump discharge pressure and TDH. Surveillance test procedures, that verify pump performance, will require revision to demonstrate pump performance. The steam supply and pump discharge MOVs will require re-evaluation for operability at the increased differential pressures.

No changes are required to the SLB detection instrumentation setpoints for the HPCI system, because there is no increase in the maximum steam flow. The RCIC SLB detection instrumentation setpoint will require a revision to maintain the analytical limit of 300% above the maximum required steam flow.

Table 6-1 Key Results for RCIC

Item	Parameter	Units	Current Design	Proposed Design	Change
1	Setpoint for lowest group SRV	psig	1115	1155	+40
2	Vessel Pressure at setpoint for lowest group of SRVs with drift	psia	1141	1205	+64
3	Required flow with vessel pressure at setpoint for lowest group SRV with drift	gpm	400	400	0
4	Required TDH	feet	2800	2962	+162
5	Rated pump speed (allowance for pump degradation included)	rpm	4500	4700	+200

Table 6-2 Key Results for HPCI

Item	Parameter	Units	Current Design	Proposed Design	Change
1	Setpoint for lowest group SRV	psig	1115	1155	+40
2	Vessel Pressure at setpoint for lowest	psia	1141	1205	+64

Item	Parameter	Units	Current Design	Proposed Design	Change
	group of SRVs with drift				
3	Required flow with vessel pressure at setpoint for lowest group SRV with drift	gpm	3000	3000	0
4	TDH Required	feet	2800	2962	+162
5	Rated pump speed	rpm	4000	4000	0

6.1.11 Recommendations and Observations

The HPCI and RCIC Systems' steam supply and pump discharge MOVs will require re-evaluation for operability at the new, higher pressures.

The RCIC System requires modification to increase the maximum turbine speed.

The RCIC SLB detection setpoint will require revision for increased steam.

6.2 SETPOINT CALCULATIONS

6.2.1 Description

The purpose of this section is to evaluate the impact of the proposed main steam SRV opening setpoint tolerance relaxation and Tech Specs change to increase the set point tolerance of the SRVs from 1% to 3% on the following setpoint functions at PNPS Nuclear Power Station.

- Anticipated Transient Without Scram (ATWS) High Pressure Recirculation Pump Trip (RPT)
- Reactor Core Isolation Cooling (RCIC) High Steam Flow Isolation

The ATWS High Pressure RPT setpoint function initiates RPT and Alternate Rod Insertion (ARI) when the reactor pressure exceeds the switch setpoint, to provide a back-up method for controlling reactivity in the unlikely event that the Reactor fails to scram when required.

The RCIC High Steam Flow Isolation setpoint function limits the uncontrolled release of radiation to the environment for a RCIC SLB, by isolating (closure) of the RCIC Steam Supply Isolation Valves, tripping the RCIC Turbine, and isolating the Minimum Flow Bypass Valve.

6.2.2 Inputs and Assumptions

Separate documents for RCIC Steam Supply Isolation and ATWS High Pressure RPT itemize the inputs used in the setpoint calculations (References 14 and 15).

6.2.3 Evaluation

The ATWS High Pressure RPT Nominal Trip Setpoint (NTSP) was calculated using GEH Instrument Setpoint Methodology (Reference 16).

In general, nominal operating and equipment parameters are utilized for the evaluation of ATWS. For key input such as the ATWS High Pressure RPT, Technical Specification AVs have been conservatively used in the analysis. The AVs are summarized below.

Function	Current AV	New AV
ATWS High Pressure RPT	1180 psig Upper	1220 psig Upper

The RCIC Steam Supply Isolation AV and NTSP were calculated using GEH Instrument Setpoint Methodology (Reference 14). The ALs are summarized below.

Function	Current AL	New AL
RCIC Turbine Steam Line High Flow	219.9 Inch H_2O 300% Rated RCIC Flow	262.373 Inch H_2O 300 % Rated RCIC Steam

6.2.4 Conclusion

This evaluation concludes that the setpoints for the ATWS High Pressure Recirc Pump Trip and the RCIC High Steam Flow Isolation functions are affected by the Main Steam Safety Valve (MSSV) setpoint tolerance relaxation.

6.3 STANDBY LIQUID CONTROL SYSTEM

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

The original performance design basis for the SLCS was that it must be capable of injecting the system design rated flow into the reactor vessel using a single SLCS pump at all reactor operating pressures up to the reactor design pressure of 1250 psig. This method has been superseded by the use of the maximum reactor vessel pressure occurring during the limiting ATWS event when the SLCS is in operation in consideration of NRC Information Notice 200-13.

Entergy will ensure that the ATWS analyses performed in Section 3 are based on the SLC system delivering 39 gpm of 8.42% sodium pentaborate solution with a minimum B10 enrichment of 54.5%. These SLC system equipment parameters are the minimum required by PNPS Technical Specifications and provide a hot shutdown capability equivalent to 10CFR50.62 requirements to inject 86 gpm of 13% sodium pentaborate solution with a B10 enrichment of 19.8% (natural enrichment). No changes to the SLC performance are necessary because of the proposed changes to the SRV and SSV configuration to comply with 10CFR50.62 requirements for a SLC system.

ATWS specific injection requirements stated in Section 3 of this report are met provided that the SLC system relief valve remains closed against the maximum lower plenum pressure of 1212 psig, which occurs during SLC system operation. Entergy will evaluate the SLC system relief valve setpoint to ensure the relief valve remains closed when the SLC system is in operation so that the system performance assumed in Section 3 is valid.

7 APPENDIX R ANALYSIS

Reference 6 for PNPS demonstrates that no SSV actuation is predicted so the SSV modifications will have no impact on the Appendix R analysis.

The proposed increased SRV setpoint and tolerance will cause SRV actuation at higher pressure and thus result in a slight delay in the SRV actuation. Consequently, the instantaneous flow rates out of the SRVs are increased due to the higher critical flow rates in comparison to the case with SRVs at currently analyzed setpoint and tolerance. However, the change in the total inventory lost from the vessel due to SRV setpoint tolerance relaxation is negligible. This is because the inventory loss is primarily dependent on the decay heat, which remains unaffected by SRV setpoint tolerance relaxation. Therefore, the vessel water level responses and conclusions in the existing evaluations are still applicable for SRV safety valve setpoint tolerance change.

The suppression pool temperature is mainly governed by energy transferred to the suppression pool through the SRVs. Before depressurization, there is negligible change to the energy transferred to the suppression pool because the increased SRV flow which can occur with the SRV safety valve setpoint tolerance increase, is balanced by reduced periods of SRV flow during SRV cycles. After depressurization, the rate of SRV energy transfer to the suppression pool and total energy transfer to the suppression pool are controlled by the vessel depressurization rate, the initial vessel liquid inventory, and decay heat which are unaffected by the SRV changes. Thus, the SRV setpoint tolerance change has no adverse impact on the suppression pool temperature, as well as containment temperature and pressure for an Appendix R fire event. Therefore, the containment response not impacted by the subject changes.

8 REFERENCES

1. GE Nuclear Energy, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," NEDC-31753P, February 1990.
2. GE Nuclear Energy, "General Electric Standard Application for Reactor Fuel (GESTAR), NEDE-24011-P-A-16, October, 2007.
3. GE Nuclear Energy, "Safety Analysis Report for Pilgrim Nuclear Power Station Thermal Power Optimization," NEDO-33050, Rev. 1, October 2002.
4. GE Nuclear Energy, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis for Pilgrim Nuclear Power Station," NEDC-31852P, Rev. 4, January 2008.
5. GE Nuclear Energy, 0000-0095-4071-R0, Pilgrim Nuclear Power Station GNF2 ECCS-LOCA Evaluation, February 2009.
6. GE Nuclear Energy, "Pilgrim Nuclear Power Station Containment Heatup Analysis with ANS 5.1 +2 sigma Decay Heat," GE-NE-T23-00749-01, December 1997
7. GE Nuclear Energy, "Pilgrim Nuclear Power Station Containment Heat Removal Analysis," GE-NE-T23-00732-01, March 1996.
8. GE Nuclear Energy, "Mark I Containment Program Plant Unique Load Definition, Pilgrim Nuclear Power Station: Unit 1," NEDO-24565 Rev 2, May, 1982
9. GE Nuclear Energy, "Mark I Containment Program Load Definition Report," NEDO-21888, November 1981
10. Pilgrim Drawings: C1A198, Rev 0, C1A363, Rev 0, C1A365 Rev 0 and C1A301 Rev 0.
11. GE Nuclear Energy, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," NEDO-30832-A, May 1995.
12. GE Nuclear Energy, "Drywell Temperature Response for Neutron Monitoring Equipment System Assessment During ATWS," GE-NE-C5100121-01, November 1995.
13. GE Nuclear Energy, "General Electric Boiling Water Reactor Increased Safety/Relief Valve Simmer Margin Analysis for Pilgrim Nuclear Power Station Unit 1," NEDO-22159, June 1982.

14. GE Hitachi Nuclear Energy, "Instrument Limits Calculation Entergy Operations Inc. Pilgrim Nuclear Power Station RCIC High Steam Flow Isolation," 0000-0112-0077 Rev 2, March 2010.

15. GE Hitachi Nuclear Energy, "Instrument Limits Calculation Entergy Operations Inc. Pilgrim Nuclear Power Station ATWS High Pressure Recirculation Pump Trip," 0000-0111-7021 Rev 2, March 2010.

16. GE Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, September 1996, and NEDO-31336-A, September 1996.