Enclosure 2

GE Hitachi Nuclear Energy

**(NEDO-33543, Revision 0, Class I,
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Non-Proprietary Information

NEBRASKA PUBLIC POWER DISTRICT
COOPER NUCLEAR STATION
SAFETY RELIEF VALVE CAPACITY AND SETPOINT EVALUATION

HITACHI

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Revision 0
Class I
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ACRONYMS AND ABBREVIATIONS

Term	Definition
ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
ASP	Analysis Setpoint
BIIT	Boron Injection Initiation Temperature
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CNS	Cooper Nuclear Station
DBA	Design Basis Accident
EOC	End of Cycle
FWCF	Feedwater Controller Failure
HPCI	High Pressure Coolant Injection
HX	Heat Transfer
IBA	Intermediate Break Accident
ICF	Increased Core Flow
IORV	Inadvertent Opening of Relief Valve
LAR	Licensing Amendment Request
LLS	Low–Low–Set
LOAP	Loss-of-Auxiliary Power
LOCA	Loss–of–Coolant Accident
LRNBP	Load Rejection No Bypass
LRWBP	Load Rejection with Bypass
LWL	Low Water Level

Term	Definition
MAX	Indicates the SRV configuration where the NSP, of all SRVs, is at the Maximum value of 1210 psig (as specified by CNS)
MELLL	Maximum Extended Load Line Limit
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 – Flux Scram
NBR	Nuclear Boiler Rated
NMS	Neutron Monitoring System
NPPD	Nebraska Public Power District
NRC	Nuclear Regulatory Commission
NSP	Nominal Setpoint
ODYN	One Dimensional Transient Analysis Program
OOS	Out–of–Service
OLTP	Original Licensed Thermal Power
OPL–3	Form containing plant data for use in Transient Analysis (Reload Licensing)
OPL–3A	Form containing plant data for ATWS analyses
OPP	Over Pressure Protection
PANAC	Three Dimensional Neutronics Analysis Program
PCT	Peak Clad Temperature
PRFO	Pressure Regulator Failure – Open
PSI	Pounds per Square Inch
PUAR	Plant Unique Analysis Report
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RV	Reactor Vessel

Term	Definition
SBA	Small Break Accident
SLCS	Standby Liquid Control System
SRV	Safety Relief Valve (Target Rock Valve)
SRVDL	SRV Discharge Line
SSV	Spring Safety Valve (Dresser Valve)
TS	Technical Specification
TTNBP	Turbine Trip No Bypass
TTWBP	Turbine Trip with Bypass

1.0 INTRODUCTION

1.1 PURPOSE

The intent of this evaluation is to determine the maximum number of safety relief valves (SRVs) that can be declared out-of-service (OOS), coupled with the maximum increase in SRV setpoint, that meets the Cooper Nuclear Station (CNS) specified limits. These limits are (1) the anticipated transient without scram (ATWS) limits [vessel pressure, peak clad temperature (PCT), containment pressure, suppression pool temperature and confirming that the Neutron Monitoring System (NMS) hardware will continue to function, for one hour, in the ATWS environmental conditions], (2) the American Society of Mechanical Engineers (ASME) Over–Pressure limits (vessel pressure and dome pressure) and (3) an analysis of the pressure margin to the spring safety valve (SSV) nominal setpoint, for anticipated operational occurrences (AOOs) and expected events.

CNS has specified analysis goals for the ATWS and ASME vessel and dome pressure limits: [[ ]] for ASME vessel; and [[ ]] for the dome pressure safety limit. These reductions are intended to ensure that this evaluation used reasonable margins to the ASME Code limits.

Additionally, a qualitative evaluation determining the effect of reducing the number of operable SRVs on the CNS Containment analyses bases is included.

These evaluations are presented in this report to formulate sufficient documentation to support a licensing amendment request (LAR) for CNS Technical Specification (TS) 3.4.3.

1.2 OVERALL EVALUATION APPROACH

Three separate evaluations were performed. The three evaluations were: (1) ATWS, (2) ASME and (3) margin to the nominal SSV setpoint. All three evaluations had the SRV setpoints in common.

Table 1-1 below summarizes the methods used for the calculations. The application of these methods is within code application capabilities.

Table 1-1
Computer Codes

Computer Code	Version or Revision	NRC Approved	Comments
ISCOR	09	Yes	Steady–state thermal–hydraulic analysis.[1]
ODYN	10	Yes	Reference 1. One dimensional reactor transient analysis. Approved with water level lowered to TAF + 5 ft and with a conservative bias applied to the user input boron mixing efficiency tables.
PANAC	11	Yes	Reference 2. The physics code PANACEA provides inputs to the transient code ODYN. The use of PANAC Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE), "Amendment 26 to GE Licensing Topical Report NEDE–24011–P–A, GESTAR II Implementing Improved GE Steady–State Methods (TAC NO. MA6481)," November 10, 1999.
TASC	03	Yes	Reference 3. Single channel transient analysis.
STEMP	04	Yes	Accepted by the NRC in previous applications (Reference 4) for Suppression Pool heat–up.

Note: 1. The ISCOR code is not approved by name. However, in the SER supporting approval of NEDE–24011–P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (USNRC) to R. Gridley (GE), the Nuclear Regulatory Commission (NRC) finds the models and methods acceptable for steady–state thermal–hydraulic analysis, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal–hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and loss-of-coolant accident (LOCA) applications is consistent with the approved models and methods.

2.0 SAFETY RELIEF VALVE SETPOINTS

The SSV setpoints are the same as was used in the Cycle 26 licensing analysis (3 valves, all with a nominal setpoint (NSP) of 1240 psig and an analysis setpoint (ASP) of 1277.2 psig). The SSV setpoints do not change in this study. Only the SRV setpoints are varied.

The SRV setpoints are modeled with various levels of pressure increase, above the nominal, and are labeled as +00 (Reference case), +20, +40, +60, +70, +80, +95, +100 and MAX. The number in the label (60 in +60) represents the pressure (psi) increase above the Cycle 26 NSP. With the exception of MAX, the setpoints were determined by simply applying the adder (+20, +40, etc.) to the Cycle 26 NSP and multiplying by 1.03, representing potential setpoint drift. An example would be (1080 + 40)*1.03 = 1153.6 psig. The MAX case had all SRVs set at 1210*1.03 = 1246.3 psig. Table 2-1 shows the SRV setpoints for several of the conditions analyzed.

Table 2-1
Examples of SRV and SSV Setpoints
(psig)

Valve	NSP	Reference (+00)	+40	+70	+95	MAX
		Analysis Setpoint[1]				
SRVs 1 & 2	1080	1112.4	1153.6	1184.5	1210.3	1246.3
SRVs 3, 4 & 5	1090	1122.7	1163.9	1194.8	1220.6	1246.3
SRVs 6 , 7 & 8	1100	1133.0	1174.2	1205.1	1230.9	1246.3
SSVs 1, 2 & 3	1240	1277.2	1277.2	1277.2	1277.2	1277.2

Note: 1. For all cases, the ASP is calculated by adding the offset to the NSP and multiplying by 1.03, e.g., for SRV #1 with "+70" SRV setpoints, the ASP is equal to (1080 + 70)*1.03 = 1184.5 psig. Note that the SSV setpoints (ASP) do not change.

SRV setpoints, generated with this methodology, were used in all analyses. Additionally, in the ATWS evaluation, a statistical spread was applied to the SRV setpoints, about the mean, to each group. The average setpoint was not changed. The statistical spread is illustrated in Table 2-2.

Table 2-2
Example of ATWS SRV Statistical Spread
(for +70 Setpoints)

SRV #	NSP (psig)	ASP (non–ATWS events) (psig)	ASP with ATWS Statistical Spread (psig)
1	1150	1184.5	[[
2	1150	1184.5	
3	1160	1194.8	
4	1160	1194.8	
5	1160	1194.8	
6	1170	1205.1	
7	1170	1205.1	
8	1170	1205.1	]]

3.0 ASME OVER-PRESSURE PROTECTION EVALUATION

The limiting event for the ASME over pressure protection (OPP) is the MSIVF[1] (main steam isolation valve (MSIV) closure, scram on high flux). The licensing and analysis limits are listed in Table 3-1.

Table 3-1
Licensing and Analysis Limits
(ASME OPP)

Parameter	Limits (psig)	
	Licensing	Analysis[1]
Peak Vessel Pressure	1375	[[
Dome Pressure Safety Limit	1337	]]

Notes: 1. The analysis goal is taken to be the licensing limit minus [[ ]] was judged to be a reasonable margin to ensure that the cycle specific ASME OPP analysis does not violate the licensing limit of 1375 psig.

3.1 INITIAL CONDITIONS – ASME

Table 3-2 lists the core initial conditions used in the ASME evaluation.

Table 3-2
ASME Core Initial Conditions

Power (MWt)	Flow (% Rated)	Cycle 26 Exposure
2428.6[1]	105	EOC
2428.6	76.8[2]	EOC

Notes: 1. 1.02*OLTP (2381 MWt)
2. 76.8% flow is meant to represent the flow at the intersection of the maximum extended load line limit (MELLL) rod line and 2428.6 MWt. In reality, the flow would be slightly higher. However, since we are trying to determine the impact of the low flow condition, using a slightly lower flow will produce a conservative result.

The ASME Over–Pressure event is most limiting at end of cycle (EOC). At EOC, the worth of the early portion of the scram is minimized, i.e., it takes longer for the scram to become effective. This condition is more limiting because the EOC power shapes are typically peaked toward the top.

---

[1] The MSIVF is the limiting single failure pressurization event. The single failure is assumed to be the first scram signal, which is the position switches on the MSIV. The next scram signal results from the APRM flux exceeding the high flux scram setpoint. The MSIV is assumed to close in the minimum time (3 seconds) allowed by the TS.

3.2 RESULTS

Analyses were performed for the conditions with 6, 5 and 3 SRVOOS (4 SRVOOS was skipped because the ATWS analysis was proceeding in parallel with the ASME OPP analysis, and it became obvious that 4 SRVOOS would not meet ATWS limits. The results for 6, 5 and 3 SRVOOS are shown in Tables 3-3, 3-4 and 3-5, respectively.

Table 3-3
ASME Over–Pressure Protection Summary
(6 SRVOOS and High Pressure Recirculation Pump Trip)

SRV Setpoints	Core Flow (% rated)	Peak Heat Flux (% initial)	Peak Pressure (psig)	
			Dome	Vessel
[[				
				]]

For the case of 6 SRVOOS, note that even with the lowest set SRV (+00), both the peak vessel and dome pressures exceed the analysis goals [[ ]].

Table 3-4
ASME Over–Pressure Protection Summary
(5 SRVOOS and High Pressure Recirculation Pump Trip)

SRV Setpoints	Core Flow (% rated)	Peak Heat Flux (% initial)	Peak Pressure (psig)	
			Dome	Vessel
[[				
				]]

Considering 5 SRVOOS, for all of the SRV setpoints analyzed, the dome pressure safety [[ ]] is exceeded. While it does appear that setpoints lower than the +40 (lowest setpoints evaluated for 5 SRVOOS) would probably meet the dome pressure limit, concurrent knowledge about the ATWS results not meeting limits for 4 SRVOOS ended the ASME analysis with 5 SRVOOS.

Table 3-5
ASME Over–Pressure Protection Summary
(3 SRVOOS and High Pressure Recirculation Pump Trip)

SRV Setpoints	Core Flow (% rated)	Peak Heat Flux (% initial)	Peak Pressure (psig)	
			Dome	Vessel
[[				
				]]

In general, the 3 SRVOOS setpoints show substantial margin to both the dome pressure safety and the peak vessel analysis goals [[ ]].

Figures 3-1 and 3-2 show the ASME MSIVF transient traces for both +40 and MAX SRV setpoints, respectively, with 3 SRVOOS and the high pressure recirculation pump trip.

Tables 3-6 and 3-7 list the sequence of events for the ASME OPP MSIVF event, with 3 SRVOOS and +80 setpoints, for both EOC increased core flow (ICF) and EOC MELLL conditions, respectively.

Table 3-6
ASME Over Pressure Protection
Sequence of Events for MSIVF at EOC and ICF (3 SRVOOS and MAX Setpoints)

Time (sec)	Events
0.0	MSIVs start to close
0.3	Direct scram (MSIV position switches) fails
1.7	Scram signal on high neutron flux
1.9	Dome pressure reaches Recirc Pump Trip Setpoint
2.0	Peak Neutron Flux
3.0	MSIVs fully closed
3.4	SSVs start to open (0.0 sec delay)
3.5	Lowest set SRVs start to open (0.4 sec delay)
4.0	Peak Vessel Pressure
4.2	Peak Dome Pressure
4.3	Recirculation pump speed = 50% of initial (ICF case)
5.6	Control rods fully inserted

Table 3-7
ASME Over Pressure Protection
Sequence of Events for MSIVF at EOC and MELLL (3 SRVOOS and MAX Setpoints)

Time (sec)	Events
0.0	MSIVs start to close
0.3	Direct scram (MSIV position switches) fails
1.7	Scram signal on high neutron flux
1.9	Dome pressure reaches Recirc Pump Trip Setpoint
2.0	Peak Neutron Flux
3.0	MSIVs fully closed
3.3	SSVs start to open (0.0 sec delay)
3.4	Lowest set SRVs start to open (0.4 sec delay)
4.0	Peak Vessel Pressure
4.2	Peak Dome Pressure
5.1	Recirculation pump speed = 50% of initial (MELLL case)
5.6	Control rods fully inserted

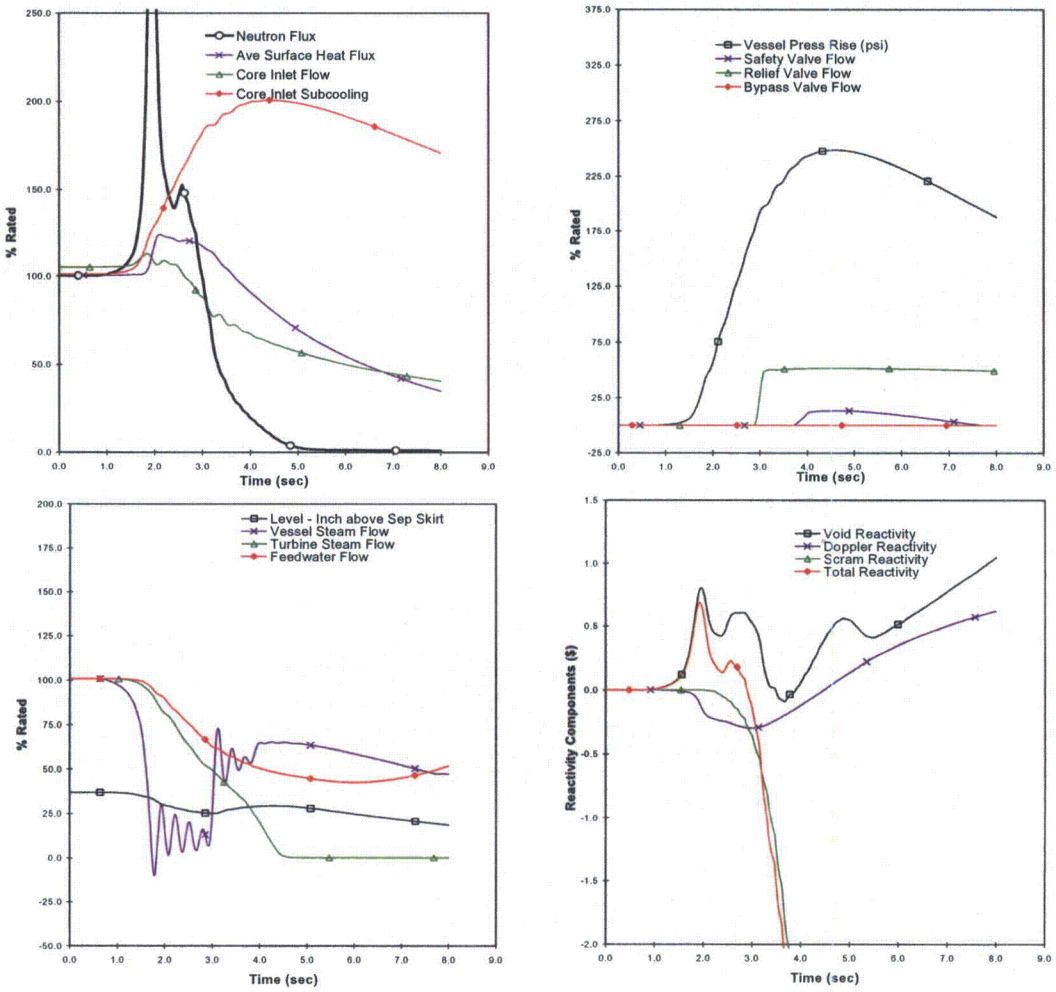

Figure 3-1
ASME MSIVF EOC 105% Flow
3 SRVOOS SRV Setpoints +40, with Pump Trip[2]

[2] Unless noted, units are in % Rated. For example, "Vessel Pressure Rise (psi)" is in units of psi.

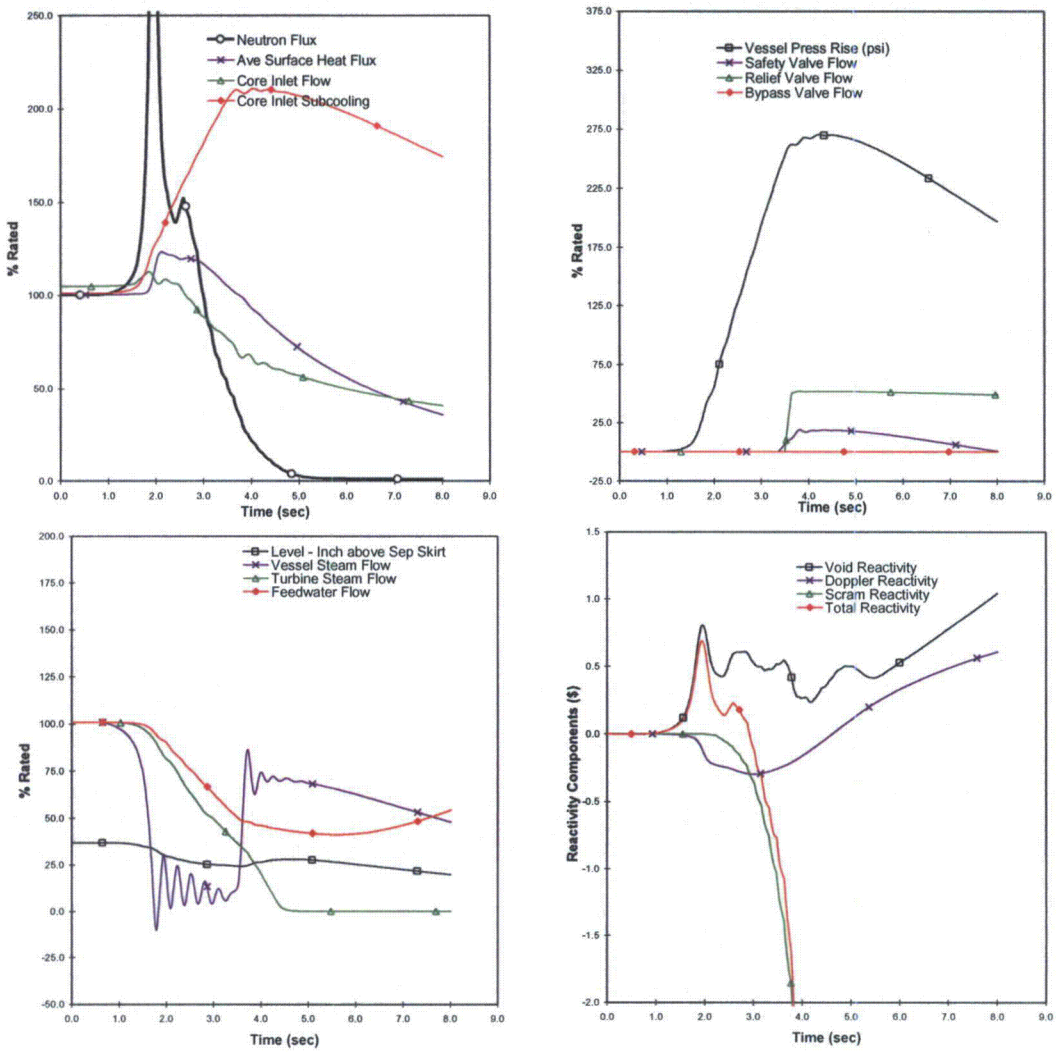

Figure 3-2
ASME MSIVF EOC 105% Flow
3 SRVOOS SRV Setpoints MAX, with Pump Trip[3]

[3] Unless noted, units are in % Rated.

4.0 ATWS EVALUATION

4.1 INITIAL CONDITIONS

This ATWS evaluation will only consider the PRFO[4] and the MSIVC[5]. The analysis of record showed that the other ATWS events (loss-of-auxiliary power (LOAP) and inadvertent opening of relief valve (IORV)) are less limiting than the PRFO and MSIVC. Nothing has changed between this evaluation and the previous analysis that will change that conclusion, i.e., the LOAP and IORV will not challenge any ATWS limits. The Licensing and Analysis ATWS limits are listed in Table 4-1.

Table 4-1
Licensing and Analysis Limits
(ATWS Events)

Parameters (Peak Value)	**Limits**	
	Licensing	**Analysis**
Vessel Pressure (psig)	1500[1]	[[ ]][2]
Cladding Temperature (°F)	2200[3]	2200
Local Cladding Oxidation (%)	17[3]	17
Containment Pressure (psig)	56	56
Suppression Pool Temperature (°F)	208	208
Functionality of Neutron Monitoring System, in ATWS environment, time (hrs)	1[4]	1
Demonstrating the SLCS pump bypass valve does not open	(not evaluated)	(not evaluated)
HPCI and RCIC Operability	(not evaluated)	(not evaluated)

Note: 1. ASME Service Level C Limit.
2. [[ ]], between the licensing and the analysis limits [[ ]] is judged to be a reasonable amount to account for future changes (fuel and hardware).
3. 10CFR50.46 limits.
4. Limit specified in Reference 5. Functionality determined by comparing the CNS ATWS results with bounding evaluation.

[4] The pressure regulator failure-open (PRFO) is an event where the Pressure Regulator is assumed to fail open. That is, the failure results in the regulator opening the TCV and/or the TBV, which results in a depressurization of the reactor. Once the pressure drops to the low pressure isolation setpoint, the MSIVs start closing. For the ATWS condition, scram fails.

[5] The main steam isolation valve closure (MSIVC) is similar to the PRFO except that the isolation occurs at rated power and pressure. The MSIVs are assumed to close at the nominal speed (4 sec). All scram attempts are assumed to fail.

The evaluation process is essentially identical to that used in the analysis of record. The only differences are in the boundary conditions (power, SRV setpoints and availability, some initial water temperatures and the initial nuclear condition – see Table 4-3). All inputs have been approved by CNS. The core nuclear condition is based on the Cycle 26 licensing basis. The operating history and differences in the fuel design (enrichment and gadolinia concentration) create some differences (as compared to the analysis of record) in the nuclear statepoint. The major difference between the analysis of record and this analysis is that this analysis considers SRVOOS.

Table 4-2 lists the core initial conditions used in the ATWS evaluation.

Table 4-2
ATWS Core Initial Conditions

Power (MWt)	Flow (% Rated)	Cycle 26 Exposure
2419[1]	76.8	BOC
2419	105	BOC
2419	76.8	EOC

Note: 1. 2419=1.016*OLTP.

Differences between this ATWS analysis and the analysis of record are summarized in Tables 4-3 and 4-4. In addition, this analysis only evaluates the PRFO and MSIVC. The LOAP and IORV were not evaluated because it was previously shown that the PRFO and MSIVC were substantially more limiting. The results of the short–term {peak pressure (ODYN) and PCT (ODYN/ISCOR/TASC)} analyses are summarized in Tables 4-5 and 4-6. The results of the long–term peak suppression pool temperature and the associated peak containment pressures (ODYN/STEMP) are listed in Table 4-7.

Table 4-3
Comparisons of Initial Conditions
(EOC)

Parameters	Value	
	Analysis of Record	**Current Analysis**
Core Power (MWt)	2381	2419
Core Flow (Mlb/hr / %rated)	55.13 / 75% (MELLL) 73.5/100% (Rated)	56.45 / 76.8% (MELLL) 77.18/105% (ICF)[2]
Steam Flow (Mlb/hr)	9.56[1]	9.72[1]
Nominal Water Level (AVZ, inches)	551.8	551.8
Feedwater Enthalpy, BTU/lbm	339.4[1]	340.8[1]
Initial Dynamic Void Reactivity Coefficient (Cents/%), (MELLL, ODYN calc, typical value EOC)	–12.05	–12.80
Core Average Void Fraction, % (MELLL, ODYN calc, typical value, EOC)	46.7	44.1
Core Exposure	BOC and EOC	BOC and EOC

Note: 1. ODYN calculation.
2. A limited set of cases was evaluated at EOC ICF.

In general, the plant response to an ATWS is worse from the lowest core flow, because a low core flow minimizes the effect of the high pressure ATWS pump trip. However, this ATWS evaluation includes the EOC ICF (Increased Core Flow) condition to ensure that the PCT (Peak Clad Temperature) is not bounding in that condition. The ICF condition maximizes the top peaked power shape, which may impact the PCT evaluation more than the reduction in peak power that results from a pump trip from ICF.

Table 4-4
Comparisons of Equipment Performance Characteristics

Parameter	Analysis of Record	Current Analysis
Nominal Closure Time of MSIVs, sec	4.0	4.0
Recirc Pump Trip, on Vessel Pressure, Time Const, sec	0.1	0.1
Recirc Pump Trip Delay, sec	0.5	0.5
SRV Capacity, %Nuclear Boiler Rated (NBR) Steam flow at 1105 psia / # of Valves active	73/8	72/8[1] 45/5
SRV Opening Setpoint Range after Statistical Spread, psia	1127 / 1178	1186[2]/ 1238 (for SRV +95 case)
SRV Closing Setpoint as Fraction of Opening Setpoint	0.97	0.97
SRV Time Delay on Opening Signal, sec	0.3	0.3
SRV Opening Duration, sec	0.3	0.3
SRV Closure Delay, sec	0.3	0.3
SRV Closure Duration, sec	0.3	0.3
SSV Capacity, % NBR Steam Flow at 1255 psia / # of Valves	20/3	20/3
SSV Opening Setpoint, psia	1292	1292
SSV Closing Setpoint, Fraction of Opening Setpoint	0.96	0.96
SSV Opening Duration, sec	0.2	0.2
SSV Closure Duration, sec	0.2	0.2
SLCS Injection Location	Lower Plenum Standpipe	Lower Plenum Standpipe
SLCS Injection Rated, gpm	76.4	76.4
Sodium Pentaborate Solution Concentration, % by Weight	11.5	11.5
Nominal Boron–10 Enrichment, %	19.8	19.8
Boron Injection Initiation Temp (BIIT), °F	110	110

Parameter	Analysis of Record	Current Analysis
SLCS Initiation Method: Manual at time of BIIT or time of ATWS Trip + 120 sec, whichever is greater	Yes	Yes
Control Liquid Transport, sec	[[	
3–D Mixing Time Delay for Standpipe Boron Injection, sec		
Total Boron Injection Delay Time, sec		]]
Control Liquid Solution Enthalpy, BTU/lbm	90.9	90.9 (123 °F)
RCIC (Reactor Coolant Isolation Cooling) Flow Rate (relaxed), gpm	360	360
Enthalpy of the RCIC Flow, BTU/lbm	90.9	68 (100 °F)
HPCI Flow Rated (relaxed), gpm	3825	3825
Enthalpy of the HPCI Flow, BTU/lbm	90.9	68 (100 °F)
ATWS High Pressure Setpoint, psig	1120	1120
Min. Suppression Pool Water Volume at low water level (LWL), ft^3	87,650	87,650
Initial Suppression Pool Temperature, °F	95	95
Residual Heat Removal (RHR) Service Water Temperature, °F	90	95
RHR Heat Exchanger K–Factor per heat transfer (HX) in Containment Cooling Mode, BTU/sec–°F	177	177

Notes: 1. The rated steam flow has increased. The SRV capacity has not decreased from previous analysis.
2. The 3 lowest set SRVs are assumed to be OOS.

4.2 RESULTS

The short–term (ODYN analysis) ATWS results are summarized in Tables 4-5 and 4-6. The results of the long–term (ODYN/STEMP analysis) peak suppression pool temperatures and the associated peak containment pressures are listed in Table 4-7. The results of the PCT (ODYN/ISCOR/TASC) calculations are listed in Table 4-8. A sequence of events table (short and long term) for the PRFO, with 3 SRVOOS and +70 SRV setpoints, is in Table 4-9.

Table 4-5
Summary of Short–Term ATWS Peak Values
(3 SRVOOS)

Event	Fuel Exposure	Core Flow (% rated)	SRV Setpoints	Peak Values (Key Parameters) and Time of Occurrence (sec)			
				Neutron Flux (%)	Heat Flux (%)	Vessel Pressure (psig)	Lower Plenum Pressure (after start of SLCS)[1,2] (psig)
PRFO	[[						
PRFO							
PRFO							
PRFO							
PRFO							
PRFO							
MSIVC							
MSIVC							
PRFO							
PRFO							
PRFO							

Event	Fuel Exposure	Core Flow (% rated)	SRV Setpoints	Peak Values (Key Parameters) and Time of Occurrence (sec)			
				Neutron Flux (%)	Heat Flux (%)	Vessel Pressure (psig)	Lower Plenum Pressure (after start of SLCS)[1,2] (psig)
PRFO							
PRFO							
PRFO							
MSIVC							
MSIVC							
PRFO							
MSIVC							]]

Notes: 1. The Standby Liquid Control System (SLCS) start time is 124 sec (MSIVC) and ~148 sec (PRFO).
2. The peak lower plenum pressure is input to the SLC relief valve setpoint evaluation.

Table 4-6
Summary of Short–Term ATWS Peak Values
(2 SRVOOS)

				Peak Values (Key Parameters) and Time of Occurrence (sec)			
Event	**Fuel Exposure**	**Core Flow (% rated)**	**SRV Setpoints**	**Neutron Flux (%)**	**Heat Flux (%)**	**Vessel Pressure (psig)**	**Lower Plenum Pressure (after start of SLCS)[1] (psig)**
PRFO	[[						]]

Notes: 1. The SLCS start time is ~148 sec (PRFO).

Table 4-7
Summary of Long–Term ATWS Peak Values
(3 SRVOOS)

Event	Fuel Exposure	Core Flow (%)	SRV Setpoints	Peak Values	
				Suppression Pool Temp (°F & sec)	Containment[1] Pressure (psig)
PRFO	[[				
PRFO					
PRFO					
PRFO					
PRFO					
PRFO					
MSIV					
MSIV					
PRFO					
PRFO					
PRFO					
PRFO					
PRFO					
PRFO					
MSIV					
MSIV					
PRFO					
MSIV					]]

Note: 1. With regards to pressure calculations, "containment" refers to the combined air spaces of the drywell and the suppression pool.

Table 4-8
Summary of ATWS Peak Clad Temperature (PCT) (3 SRVOOS)

Event	SRV Setpoint	Exposure	Core Flow (%)	PCT (°F) and Time (sec)
PRFO	[[			
PRFO				
PRFO				
PRFO				
PRFO				
MSIV				
MSIV				
PRFO				
PRFO				
PRFO				
MSIV				
MSIV				
PRFO				
MSIV				]]

Table 4-9
ATWS Sequence of Events
PRFO with 3 SRVOOS, +70 SRV Setpoints, at BOC with 76.8% Core Flow

Time (sec)	Event
[[	Steam demand increased to 125%
	MSIV isolation initiated (low pressure isolation setpoint)
	Peak neutron flux [[ ]]
	High pressure ATWS setpoint
	Start opening first SRV
	Recirculation pumps tripped
	Peak heat flux [[ ]]
	Start opening SSVs
	Peak vessel pressure [[ ]]
	Feedwater pumps tripped
	SLCS pumps start
	Water level increased
	Hot shutdown achieved (Neutron flux below 0.1%)
	Maximum suppression pool temperature [[ ]]
]]	Maximum containment pressure [[ ]]

Table 4-5 shows that the highest SRV setpoint, to meet the peak vessel [[ ]], is the +70 (1488.8 psig at BOC). All other higher setpoints fail to meet this limit.

The long term limits of 56 psig and 208°F, (containment[6] design pressure and peak, post-LOCA suppression pool temperature), are easily met for all setpoints, up to and including the MAX (see Table 4-7).

The PCT results (see Table 4-8), likewise meet the limits of 2200°F and 17% cladding oxidation (not specifically evaluated here because the temperature is below that of the analysis of record [1479 °F], which was judged to result in insignificant oxidation and was less than the 17% limit).

Tables 4-9 and 4-10 show typical event sequences for both a PRFO and MSIV ATWS event.

[6] With regards to pressure calculations, "containment" refers to the combined air spaces of the drywell and the suppression pool.

Table 4-11 summarizes required SRV setpoints for appropriate ATWS limits.

Figures 4-1, 4-2 and 4-3 show the ATWS PRFO transient traces for 3 SRVOOS and SRV setpoints of +60, +70 and +80, respectively. Figure 4-4 shows the ATWS MSIVC transient traces for 3 SRVOOS and SRV setpoints of MAX.

Table 4-12 shows the ATWS limits and the analysis results.

Table 4-10
ATWS Sequence of Events
MSIV Closure with 3 SRVOOS, MAX SRV Setpoints, at BOC with 76.8% Core Flow

Time (sec)	Event
[[	MSIV Isolation Initiates
	MSIVs Closed
	Peak Neutron Flux [[ ]]
	High Pressure ATWS Setpoint Reached
	Recirculation Pumps Tripped
	Start Opening of the First Relief Valve (0.3 sec delay and 0.15 sec to open)
	Peak Heat Flux [[ ]]
	Peak Vessel Pressure [[ ]]
	Feedwater Pumps Trip
	SLCS Pumps Start
	Water Level Increased
	Hot Shutdown Achieved (Neutron flux below 0.1%)
]]	Peak Suppression Pool Temperature [[ ]]

Table 4-11
Allowable SRV Setpoints For ATWS Licensing Limits

ATWS Limits	Value	Maximum SRV Setpoints Allowed[1]
Peak Vessel Pressure (psig)	1500 [[ ]][2]	+70[3]
PCT (°F)	2200	MAX
Cladding Oxidation (%)	17	MAX
Suppression Pool Temperature (°F)	208	MAX
Peak Containment Drywell Design Pressure (psig)	56	MAX

Notes:
1. Only the maximum setpoint condition is shown. All lower setpoints also meet the criteria.
2. [[ ]] The licensing basis is 1500 psig.
3. The +70 setpoints are shown in Tables 2-1 and 2-2.

Table 4-12
Comparisons of Limiting Results to ATWS Acceptance Criteria

Acceptance Criteria	Allowed Value (Licensing/ Analysis)	Limiting Result		ATWS Event and Conditions
		Analysis of Record	SRVOOS Study[1]	
Peak Vessel Pressure (psig)	1500/[[ ]]	1307	1489[2]	[[ ]]
Peak Cladding Temperature (°F)	2200	1479	1442	[[ ]]
Peak Local Cladding Oxidation (%)	17	not calculated[3]	not calculated[3]	N/A
Peak Suppression Pool Temperature (°F)	208	181	187.1	[[ ]]
Peak Containment Pressure (psig)	56	9.0	11.1	[[ ]]

Notes:
1. All results presented here are for 3 SRVOOS and the most limiting of all SRV setpoint conditions evaluated, unless otherwise specified.
2. The SRV setpoint condition was the +70 with 3 SRVOOS.
3. Cladding oxidation is not explicitly calculated because the peak cladding temperature is below that at which significant metal–water reaction begins and is well below the limit.

[[

]]

Figure 4-1a
Short Term BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +60

[[

]]

Figure 4-1b
STEMP Results
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +60

[[

]]

Figure 4-1c
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +60

[[

]]

Figure 4-1d
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +60

[[

]]

Figure 4-2a
Short Term BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +70

[[

]]

Figure 4-2b
STEMP Results
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +70

[[

]]

Figure 4-2c
EOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +70

[[

]]

Figure 4-2d
EOC ATWS PRFO 76.8% Flow
SRVOOS SRV Setpoints +70

[[

]]

Figure 4-3a
Short Term BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +80

[[

]]

Figure 4-3b
STEMP Results
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +80

[[

]]

Figure 4-3c
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +80

[[

]]

Figure 4-3d
BOC ATWS PRFO 76.8% Flow
3 SRVOOS SRV Setpoints +80

[[

]]

Figure 4-4a
BOC ATWS MSIVC 76.8% Flow
3 SRVOOS SRV Setpoint MAX

[[

]]

Figure 4-4b
STEMP Results
BOC ATWS MSIVC 76.8% Flow
3 SRVOOS SRV Setpoint MAX

[[

]]

Figure 4-4c
BOC ATWS MSIVC 76.8% Flow
3 SRVOOS SRV Setpoint MAX

[[

]]

Figure 4-4d
BOC ATWS MSIVC 76.8% Flow
3 SRVOOS SRV Setpoint MAX

4.3 FUNCTIONALITY OF NEUTRON MONITORING SYSTEM

The BWR Owners Group (BWROG) has evaluated the drywell temperature response, for the NMS, to an ATWS event (MSIV closure). The BWROG selected a BWR/3, with 8 SSVs (more than any other BWR design). This plant was assumed to be bounding (henceforth to be called the Bounding Plant). The evaluation demonstrated that the NMS for the Bounding Plant met the NRC requirements (Reference 5). As part of the BWROG work, CNS was shown to be bounded by the Bounding Plant. While it is expected that the Bounding Plant analysis is still limiting, reducing the CNS SRV capacity (3 SRVOOS) requires that the Bounding Plant conclusions be reviewed for the case of only 5 SRVs being declared active, i.e., 3 SRVOOS.

While all of the Figures–of–Merit (Ratio of SSV Flow Rate to Drywell Volume and Ratio of reactor vessel (RV) Flow Rate to Core Thermal Power) indicated that the Bounding Plant results would continue to bound CNS, with 3 SRVOOS, a more detailed review was performed.

The Bounding Plant SSV flow was plotted (see Figure 4-5) along with the limiting CNS ATWS case (EOC 26 76.8% Flow PRFO with the +80 SRV setpoints).

The integrated flow for the Bounding Plant case was 73,549 lbs. SSV flow ceased by 200 seconds. For the CNS case, the integrated flow was 21,795 lbs. While the Bounding Plant drywell volume is about 20% (158,236/132,465 = 1.19) larger than that of CNS, the total steam flow is more than three times that of CNS (73,549/21,795 = 3.4). The Bounding Plant heat absorption capacity is about 20% higher, but the total energy input is more than three times that of CNS. Thus, it is concluded that, for all SRV setpoints equal to or less than +80, CNS is still bounded by the Bounding Plant results.

[[

]]

Figure 4-5
CNS ATWS SSV Flow Compared to Bounding Plant
(PRFO EOC 76.8% Flow and +80 SRV Setpoints for CNS)

5.0 SSV MARGIN EVALUATION

To determine the likelihood of a SSV lifting during a transient, the pressure margin between the transient peak steam line pressure and the NSP of the lowest set SSV {1240 psig for CNS} is evaluated. GEH recommends that, for expected events[7], this pressure margin meet or exceed [[ ]]. This evaluation is typically performed with the transient analysis in each cycle's reload licensing analyses.

The turbine trip with bypass (TTWBP), Feedwater Controller Failure (FWCF) and main steam isolation valve-direct scram (MSIVD) were evaluated, for 3 SRVOOS with +80 SRV setpoints. Review of the Cycle 26 reload licensing results showed that the turbine trip no bypass (TTNBP) was slightly more limiting than the load rejection no bypass (LRNBP) and thus it was assumed that the TTWBP would also be slightly more limiting than the load rejection with bypass (LRWBP). In addition, the MSIVD was evaluated with +60 setpoints. The analysis conditions were the same as the ASME evaluation (EOC with ICF). The results are shown in Table 5-1.

Table 5-1
Pressure Margin to 1240 psig (lowest set SSV)
(3 SRVOOS and +80 setpoints)

Event	ATWS RPT	SRV Setpoints	MSIV Closure Time (sec)	Pressure Margin to 1240 psig (psi)
TTWBP	Yes	+80	N/A	86.6
FWCF	Yes	+80	N/A	77.2
MSIVD	Yes	+80	3	27.0
MSIVD	Yes	+60	3	43.4
MSIVD	Yes	+80	4	150.3

The [[ ]] recommendation is met for the TTWBP and FWCF cases. However, the MSIVD with the 3 second closure fails this same test. If the MSIVD is evaluated with the nominal closure time (4 sec) the margin is substantial (150.3 psi). If CNS can justify using a slower (than 3 sec closure) MSIV closure time, the recommended margin to SSV setpoint can be met.

[7] "Expected Events" refers to pressurization events with the bypass active (LRWBP, TTWBP and FWCF) or a MSIVD (MSIV closure with scram on valve movement).

6.0 CONTAINMENT EVALUATION

The SRVs can actuate by either of two modes: the safety mode or the relief mode. For the purpose of this analysis only the safety mode as it pertains to TS 3.4.3 is relevant.

The containment analysis bases and results presented in References 6 and 7 were reviewed to determine the extent at which an LAR could be proposed in order to enable CNS to operate with fewer SRVs operable than currently specified in CNS TS 3.4.3 (Reference 8).

Additionally, the results presented in Reference 9 were reviewed to confirm that the effect of an inoperable SRV due to an increased opening pressure has been evaluated to resolve any potential safety concerns associated with an inoperable SRV that fails to lift within ±3% of its nominal lift set point.

The results of these evaluations are presented in the following paragraphs.

6.1 CONTAINMENT SYSTEM PRESSURE AND TEMPERATURE RESPONSE – DESIGN BASIS ACCIDENT (DBA)

The CNS Plant Unique Analysis Report (PUAR) (Reference 6) states that a double-ended guillotine recirculation pump suction line break is the limiting design basis accident (DBA) LOCA for peak containment pressure and peak suppression pool temperature response. The most severe event in terms of peak drywell temperature is typically a steam line break. For both of these postulated break events, the vessel depressurizes without any SRV actuation whatsoever (safety or relief mode). Therefore, the number of inoperable SRVs has no effect on the peak containment pressure and temperature response for large-break LOCAs.

6.2 CONTAINMENT SYSTEM PRESSURE AND TEMPERATURE RESPONSE – INTERMEDIATE BREAK ACCIDENT (IBA) & SMALL BREAK ACCIDENT (SBA)

Less limiting events, (IBA and SBA) also do not require SRV actuation (safety mode). The automatic depressurization system (ADS) reduces reactor system pressure for IBA and SBA if the high pressure coolant injection (HPCI) system is unable to provide adequate makeup. Small steam line breaks can result in high drywell temperature conditions for long periods of time as the vessel remains at high pressure for a longer time. However, for these breaks, the peak drywell temperature is well below that of the limiting steam line break previously addressed in Section 6.1. Furthermore, the peak drywell temperature is primarily governed by the total energy released into the drywell and thus occurs late in the event (10 to 25 minutes, depending on the break size), after many SRV actuations. The SRV Setpoint Tolerance Analysis for CNS (Reference 9) concluded that an increase in the SRV pressure setpoint up to the Upper Limit (1210 psig), established by the vessel overpressure calculations, would have no effect on the peak drywell temperature. This is due to the fact that the total mass and energy available for release is not affected by an increase in the peak vessel transient pressure response resulting from

a higher SRV opening pressure. Although, such an event might result in a slight increase in the initial mass and energy release to the drywell during the first SRV actuation, the timing of the mass and energy released to the containment over the long term would be not be significantly different. This is because CNS's Low-Low-Set (LLS) relief logic function initiates after the first valve actuation to prevent multiple subsequent SRV actuations in rapid succession. Subsequently, the integrated mass and energy release response to the drywell up to and past the time of the peak drywell temperature and pressure will depend primarily upon the automatic safety relief valve control provided by LLS. Therefore, the proposed LAR to enable CNS to operate with fewer SRVs does not invalidate the conclusions made in Reference 9. It does not present any additional safety concerns associated with an inoperable SRV drift to either +3% of its nominal lift set point or the Upper Limit.

6.3 CONTAINMENT HYDRODYNAMIC LOADS

LOCA hydrodynamic loads, such as pool swell, condensation oscillation and chugging, are dependent on the containment pressure and temperature response during the DBA LOCA. As stated in Section 6.1, the containment pressure and temperature response during the DBA LOCA is not affected by a proposed LAR for TS change to enable CNS to operate with fewer SRVs operable than currently specified. Therefore LOCA hydrodynamic loads are also unaffected.

6.4 SRV DISCHARGE RELATED LOADS

Loads associated with the discharge of SRVs, such as maximum water clearing thrust load, torus shell pressure loading, water jet induced loads on the T-quencher, and air bubble induced drag loads on the T-Quencher and submerged portions of the SRV discharge line (SRVDL), documented in Reference 6, can be categorized as either first SRV actuation or subsequent actuation loads. The SRV loads for both initial and subsequent actuations can be further subdivided and categorized as either internal pressure loads or external thrust loads on the SRV discharge line (SRVDL) and T-quencher. Loads due to initial SRV actuation are determined by parameters including the SRV setpoints, SRVDL volume, line lengths and friction losses, and number of turns. Since all these parameters, including the SRV setpoints, will not change, the proposed LAR for TS will not affect loads due to initial SRV actuations.

Loads due to subsequent SRV actuations depend primarily on the maximum SRVDL reflood height at the time of SRV opening and time intervals between openings. The maximum SRVDL reflood height is controlled by the SRVDL geometry and the SRVDL vacuum breaker capacity. The time intervals between SRV openings are controlled by the reactor pressure response, which in turn depends on the reactor power level and to a lesser extent on the ECCS flows and suppression pool temperature (source of ECCS water). Additionally, the LLS relief logic extends the time between SRV actuations to allow the SRV discharge line water legs to return to normal levels after an actuation, thus mitigating the effects of postulated thrust loads on the SRVDL.

As previously mentioned in Section 6.2 the effect of increased SRV opening pressure was considered in Reference 9 to resolve any potential safety concerns associated with SRV drift to the Upper Limit on containment and steam line integrity. No changes are being made to valve mechanism, SRV valve nominal setpoints, or drift assumptions including the Upper Limit. Therefore, the proposed LAR to enable CNS to operate with fewer SRVs operable does not invalidate the conclusions made in Reference 9.

Based on the evidence provided, it is concluded that the containment hydrodynamic loads evaluation results and conclusions documented in the CNS PUAR (Reference 6) remain valid when the proposed LAR for TS change to enable CNS to operate with fewer SRVs operable than currently specified is considered.

7.0 SUMMARY AND CONCLUSIONS

The ATWS results dominated the search for the number of SRV OOS and SRV setpoints that meet required margins. After the ATWS analyses demonstrated that the no more than 3 SRVOOS was acceptable, all other analyses were evaluated with that boundary condition (3 SRVOOS). The ASME main steam isolation valve closure-flux scram (MSIVF) event met all required limits, with SRV setpoints ranging from +00 to MAX, with 3 SRVOOS.

The margin to SSV NSP (1240 psig), was found to be acceptable for SRV setpoints as high as +80 (with 3 SRVOOS), providing that a four second closure time for the main steam isolation valve closure-direct scram (MSIVD) is justified. If is not justified, then a lower SRV setpoint is recommended.

The ATWS results showed that criteria of PCT, containment pressure and suppression pool temperature, were met for all SRV setpoints (+00 to MAX). Only the peak vessel pressure was noticeably affected by changing SRV setpoints. The highest SRV setpoints, that meet the peak vessel pressure analysis limit [[ ]], was the +70.

The proposed change to CNS TS 3.4.3 does not affect the SRV Relief Mode nominal setpoints, or drift assumptions, and no changes are being made to valve mechanisms. Therefore, based on the evaluation contained in Section 6.0, it is concluded that the existing containment and steam line integrity analyses documented in the "Plant Unique Analysis Report (PUAR) for CNS" (Reference 6), "Evaluation of Mark I SRV Load Cases C3.1, C3.2, and C3.3 for CNS" (Reference 7), and "SRV Setpoint Tolerance Analysis for CNS" (Reference 9) are not adversely affected by the proposed Technical Specification change to enable CNS to operate with fewer SRVs in the Safety Mode function than currently specified TS 3.4.3 (Reference 8).

8.0 REFERENCES

1. GE Nuclear Energy, "Qualification of the One–Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," NEDO–24154–A, Vols. 1 – 3, February 1986, NEDC–24154P–A Supplement 1, Volume 4, February 2000.

2. GE Nuclear Energy, "Amendment 26 to GE Licensing Topical Report NEDE–24011–P–A, 'GESTAR' – Implementing Improved GE Steady State Methods (TAC No. MA6481)," S. Richards (NRC) to G. Watford (GE), MFN–035–99, November 10, 1999.

3. GE Nuclear Energy, "TASC–03A, A Computer Program for Transient Analysis of a Single Channel," NEDC–32084P–A, Revision 2, July 2002.

4. GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate," (ELTR–2), NEDC–32523P–A, Supplement 1, Volume I, February 1999 and Supplement 1, Volume II, April 1999.

5. GE Nuclear Energy, "Position on NRC Regulatory Guide 1.97, Revision 3, Requirement for Post–Accident Neutron Monitoring System," NEDC–31558–A, March 1993.

6. Cooper Nuclear Station Plant Unique Analysis Report, Revision 0, NPPD, April 1982.

7. GE Nuclear Energy, "Evaluation of Mark I S/RV Load Cases C3.1, C3.2, and C3.3 for the Cooper Nuclear Station," NEDC–24359, August 1981.

8. CNS Technical Specification 3.4.3, Amendment 234.

9. GE Nuclear Energy, "SRV Setpoint Tolerance Analysis for Cooper Nuclear Station," NEDC–31628P, October 1988.