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MAY 0 7 2001

LRN-01-0148

PSEG
Nuclear LLC

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

**RESPONSE TO APRIL 9, 2001 AND APRIL 26, 2001
REQUEST FOR ADDITIONAL INFORMATION IN REGARDS TO REQUEST FOR LICENSE AMENDMENT
INCREASED LICENSED POWER LEVEL
HOPE CREEK GENERATING STATION UNIT NO. 1
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354**

On April 9, 2001 and April 26, 2001, the NRC issued requests for additional information (RAI) to support the staff's review of the request for license amendment submitted by PSEG Nuclear LLC on December 1, 2000 requesting an increase in licensed power level for Hope Creek Generating Station Unit No. 1. The response to the April 9, 2001, request for additional information is contained in Attachment 1. The response to the April 26, 2001, request for additional information is contained in Attachment 2.

In a telephone conversation conducted on May 1, 2001, the NRC staff identified two additional areas that required some further clarification. The additional information is provided in Attachment 3.

Should you have any questions regarding this request, please contact Mr. Brian Thomas at (856)339-2022.

Sincerely,

G. Salamon
Manager – Nuclear Safety and Licensing

Attachments (3)

A001

MAY 0 7 2001

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ATTACHMENT 1
HOPE CREEK GENERATING STATION UNIT NO. 1
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INCREASED LICENSED POWER LEVEL

On April 9, 2001, the NRC issued a request for additional information (RAI) concerning PSEG Nuclear's request for amendment to increase the licensed power level for Hope Creek Unit No. 1. This attachment provides the response to the RAI questions.

NRC Question:

1. Attachment 1, Section 9.1, of the submittal provides the justification for the requested power uprate with respect to the design of the fuel pool cooling and cleanup system (FPCCS). The FPCCS is designed to remove heat and impurities from the spent fuel pool. The licensee has indicated that the FPCCS heat removal function will not be affected by the power uprate, but its cleaning function was not addressed. Describe how the removal of impurities from the water in the spent fuel pool will be affected by the power uprate.

   The regulatory basis for this question is that the cleanup portion of the FPCCS conforms to the requirements of General Design Criteria (GDC) 61 of Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (10 CFR Part 50) as it relates to appropriate filtering systems for fuel storage.

PSEG Nuclear response to Q1:

Fuel Pool (FP) Cleanup portion of the FP Cooling and Cleanup System consist of the FP filter demineralizer which is designed to limit the fission product and activated corrosion product concentrations and maintain the clarity of the water in the spent fuel pool to permit continuous occupancy of the refueling area by plant personnel. The impurities consist of ionic and particulate constituents that are introduced primarily from makeup water, crud on the fuel or failed fuel. The proposed 1.4% power uprate does not impact any one of these. Therefore, the ionic impurities in the spent fuel pool are not expected to increase. Particulate contribution from the FP cooling piping system is not expected to change since the system operating flow, temperature and pressure is not changed. Also, the amount of impurities introduced into the pool from spent fuel is not expected to change significantly since the reactor coolant cleanup systems (reactor water cleanup and condensate pre-filter systems) will control the normal operational impurities. In addition, FP filter demineralizer operating parameters (flow, temperature and pressure) are not being changed. Therefore, the removal of

impurities from the water in the spent fuel pool will not be affected in maintaining the pool purity and clarity.

NRC Question:

2. Attachment 1, Section 9.3, of the submittal provides the justification for the requested power uprate with respect to the design of the Standby Liquid Control System (SLCS). Provide justification for why the concentration of sodium pentaborate in the SLCS is not changed after the power uprate.

   The regulatory basis for this question is that the SLCS conforms to the reactivity control requirements of 10 CFR 50.62(c)(4).

PSEG Nuclear response to Q2:

The Standby Liquid Control System (SLCS) shall be capable of shutting the reactor down from the most reactive reactor operating state at any time in cycle life. The design evaluation acceptance criteria is a calculated reactivity that demonstrates that the reactor is shutdown for the most reactive moderator temperature at any time during the cycle.

For future cycles, the core design is constrained by the requirement that the reactor is shutdown for the most reactive moderator temperature at any time during the cycle for a boron concentration of 660 ppm in the reactor core. If this requirement is not met in a future reload cycle design effort, the core design would be modified to meet the requirement, or at that time, appropriate design or licensing activities would be implemented to change the 660 ppm boron concentration to meet the SLCS shutdown reactivity requirements.

The 1.4% uprate will be implemented for the current cycle after the most reactive reactor operating state for which the Cycle 10 reload specific evaluations demonstrated that the reload core would be shutdown after a SLCS injection that achieves a boron concentration of 660 ppm in the reactor core.

NRC Question:

3. Attachment 1, Section 10, of the submittal provides the justification for the requested power uprate with respect to the design of the Steam and Power Conversion Systems. The submittal states that the power conversion systems and their support systems were designed for 105 percent of rated steam flow and that the proposed 1.4 percent power uprate will increase the rated steam and feedwater flow by about 1.8 percent. Therefore, the proposed power uprate has no impact on the power conversion systems since the increased flow is bounded by the design conditions. Does the design analysis also bound the turbine

overspeed and associated missile production for the 1.8 percent increase in steam flow?

The regulatory basis for this question is that the turbine generator system conforms to the requirements of GDC 4 of Appendix A to 10 CFR Part 50 as it relates to the protection of structures, systems, and components important to safety from the effects of turbine missiles.

PSEG Nuclear response to Q3:

Yes, the turbine overspeed and the associated missile production evaluation was performed at Turbine Valves Wide Open (VWO) condition which bounds the 1.4% power uprate conditions.

NRC Question:

4. Attachment 5 of the submittal provides PSEG's justification for an exemption request associated with the use of American Society of Mechanical Engineers (ASME) Code Case N-588. In a telephone conversation on March 30, 2001, the NRC staff questioned if the exemption was needed for HCGS. Specifically, the staff stated that Code Case N-588 does not appear to provide any benefit since the HCGS reactor pressure vessel is not limited by circumferential weld material in the vessel. The NRC staff requested that PSEG either withdraw the exemption request or provide additional information that demonstrates the need for the exemption. Your staff indicated that the exemption was needed with respect to procedures for determining stress intensity factors and stated that additional information would be provided to justify the exemption request.

PSEG Nuclear response to Q4:

The main intent of Code Case N-588 is to provide an exception to the requirement that axial flaws be assumed in reactor pressure vessel circumferential welds. This code case was used in developing new pressure-temperature (P-T) curves for Hope Creek. Flaw orientation for these welds was not specifically considered during P-T curve development. However, since the Hope Creek vessel circumferential welds are not limiting, we agree that use of the Code Case does not have any affect on the P-T curve calculations with respect to flaw orientation in welds. Code Case N-588 also includes improved thermal stress intensity factor relationships. Those relationships were used in the development of the pressure-temperature curves providing additional operating margin. Therefore PSEG believes that this is a reduction in unnecessary burden and requests that the exemption be granted.

NRC Question:

5. In order to assist in the evaluation of the effects of the proposed change on the Updated Final Safety Analysis Report Chapter 15 analyses, please provide a copy of the fuel vendor's supplemental reload analysis report (or similar documentation as discussed in a telephone conversation on March 28, 2001) for the current fuel cycle. This information is required to assure that proposed changes conform to the requirements of:

    a) GDC 10 of Appendix A to 10 CFR Part 50 as it relates to the reactor coolant system being designed with appropriate margin to ensure that specified fuel design limits are not exceeded during normal operations including anticipated operational occurrences;

    b) GDC 15 of Appendix A to 10 CFR Part 50 as it relates to the reactor coolant system and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including anticipated operational occurrences;

    c) GDC 20 of Appendix A to 10 CFR Part 50 as it relates the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences; and

    d) GDC 26 of Appendix A to 10 CFR Part 50 as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded, including anticipated operational occurrences.

PSEG Nuclear response to Q5:

The following information is supplied for the current operating Cycle 10.

Reload Fuel Bundles

Assembly Type	Number of Assemblies	Cycle Loaded	Description
HB07	48	7	GE9B 8X8-4 3.25 w/o U-235
HC07	52	7	GE9B 8X8-4 3.24 w/o U-235
HD08	176	8	GE9B 8X8-4 3.27 w/o U-235
HE08	60	8	GE9B 8X8-4 2.98 w/o U-235
HF09	196	9	GE9B 8X8-4 2.79 w/o U-235
PA10	184	10	ABB SVEA-96+ 3.25 w/o U-235
PB10	48	10	ABB SVEA-96+ 3.25 w/o U-235

Cycle 10 Design Assembly Type Distribution

I/J	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30
16	HE08	PA10	HF09	HE08	HE08	PA10	HF09	HE08	HE08	PA10	HF09	HD08	HD08	PB10	HC07
17	PA10	HD08	PA10	HD08	PA10	HE08	PA10	HF09	PA10	HF09	PA10	HF09	PB10	HD08	HB07
18	HF09	PA10	HF09	PA10	HF09	PA10	HF09	PA10	HF09	PA10	HD08	PA10	HF09	PB10	HB07
19	HE08	HD08	PA10	HE08	HE08	HF09	PA10	HD08	HD08	HF09	PA10	HD08	HD08	HF09	HC07
20	HE08	PA10	HF09	HE08	HD08	PA10	HD08	HD08	HD08	PA10	HF09	HD08	HD08	HF09	HB07
21	PA10	HE08	PA10	HF09	PA10	HD08	PA10	HF09	PA10	HD08	PA10	HF09	PB10	HF09	HC07
22	HF09	PA10	HF09	PA10	HD08	PA10	HD08	PA10	HF09	PA10	HF09	PA10	HF09	HD08	HC07
23	HE08	HF09	PA10	HD08	HD08	HF09	PA10	HE08	HD08	HF09	PB10	HF09	HD08	HC07	
24	HE08	PA10	HF09	HD08	HD08	PA10	HF09	HD08	HD08	PB10	HF09	HB07	HC07		
25	PA10	HF09	PA10	HF09	PA10	HD08	PA10	HF09	PB10	HD08	HD08	HB07	HC07		
26	HF09	PA10	HD08	PA10	HF09	PA10	HF09	PB10	HF09	HD08	HB07				
27	HD08	HF09	PA10	HD08	HD08	HF09	PA10	HF09	HB07	HB07					
28	HD08	PB10	HF09	HD08	HD08	PB10	HF09	HD08	HC07	HC07					
29	PB10	HD08	PB10	HF09	HF09	HF09	HD08	HC07							
30	HB07	HB07	HB07	HC07	HB07	HC07	HC07								

Reference Core Loading Pattern

Assumed nominal previous cycle core average exposure	25408 Mwd/Mtu
Assumed nominal reload cycle core average exposure at beginning of cycle	14241 Mwd/Mtu
Assumed nominal reload cycle core average exposure at end of cycle	25852 Mwd/Mtu

Calculated Core Effective Multiplication – No Voids, 20 °C

Beginning of Cycle, Keffective	
Fully Controlled	0.9604
Strongest control rod out	0.9848
R, Maximum increase in cold core reactivity with exposure into cycle, Δk	0.0

Standby Liquid Control System Shutdown Capability

Boron (ppm)	Shutdown Margin (%Δk/k), 20 °C
660	4.66

Selected Analysis Options:

- Recirculation Pump Trip
- Improved scram speed
- Exposure Dependent Limits
- Single-Loop Operation
- Extended load line limit
- Increased Core Flow (105%)
- EOC RPT Out of Service
- Thermal Power Monitor

Core wide AOO results: Analyzed with Improved Scram Speed and RPT Operable at Full Power

Transient (Mwd/Mtu values represent cycle exposure)	SVEA-96+ OLMCPR	8x8-4 OLMCPR
Generator Load Rejection No Bypass, ≤7701 Mwd/Mtu	1.21	1.27
Turbine Trip No Bypass, ≤7701 Mwd/Mtu	≤1.21	≤1.27
Feedwater Controller Failure, ≤7701 Mwd/Mtu	1.20	1.27
Generator Load Rejection No Bypass, >7701 Mwd/Mtu and ≤11732 Mwd/Mtu	1.35	1.44
Turbine Trip No Bypass, >7701 Mwd/Mtu and ≤11732 Mwd/Mtu	≤1.35	≤1.44
Feedwater Controller Failure, >7701 Mwd/Mtu and ≤11732 Mwd/Mtu	1.29	1.36
Loss of Feedwater Heating, BOC10 to EOC10	1.20	1.28
Inadvertent HPCI Startup, BOC10 to EOC10	≤1.20	≤1.28

Local Rod Withdrawal Error Summary

Rod Block Monitor Setting (%)	Distance Withdrawn (ft)		ΔCPR	
	SVEA-96+	8x8-4	SVEA-96+	8x8-4
104	4.0	4.0	0.14	0.18
105	4.5	4.5	0.16	0.20
106	5.0	5.0	0.19	0.20
107	5.5	5.5	0.21	0.21
108	6.0	7.5	0.24	0.25

Rod Block Monitor Setting Selected: 106%

Initial Control Rod Pattern for SVEA-96+ RWE Analysis at the Limiting Burnup

				--	--	--	--	--	--	--				
			--	--	--	--	--	--	--	--	--			
		--	0	--	--	--	0	--	--	--	0	--		
	--	--	--	--	--	--	--	--	--	--	--	--	--	
--	0	--	0	--	8	--	4	--	8	--	0	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	0	--	0	--	0	--	0	--	0	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	0	--	0	--	0	--	0	--	0	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	0	--	8	--	4	--	8	--	0	--	0	--
	--	--	--	--	--	--	--	--	--	--	--	--	--	
		--	0	--	--	--	00#	--	--	--	0	--		
			--	--	--	--	--	--	--	--	--			
				--	--	--	--	--	--	--				

Location of error rod

Initial Control Rod Pattern for 8x8-4 RWE Analysis at the Limiting Burnup

				--	--	--	--	--	--	--				
			0	--	0	--	0	--	0	--	0			
		--	--	--	--	--	--	--	--	--	--	--		
	0	--	0	--	0	--	10	--	0	--	0	--	0	
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	0	--	12	--	00#	--	12	--	0	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	10	--	0	--	32	--	0	--	10	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
--	0	--	0	--	12	--	0	--	12	--	0	--	0	--
--	--	--	--	--	--	--	--	--	--	--	--	--	--	--
	0	--	0	--	0	--	10	--	0	--	0	--	0	
		--	--	--	--	--	--	--	--	--	--	--		
			0	--	0	--	0	--	0	--	0			
				--	--	--	--	--	--	--				

Location of error rod

MCPR Fuel Cladding Integrity Safety Limit

For SVEA-96+ fuel assemblies, the MCPR Safety Limit (including channel bow effects) is 1.10 for two-loop operation and 1.13 for one-loop operation. For 8x8-4 fuel assemblies, the MCPR Safety Limit (including channel bow effects) is 1.10 for two-loop operation and 1.12 for one-loop operation.

Misplaced Assembly Accident

The Mislocated Fuel Assembly OLMCPRs for Cycle 10 are shown below:

Burnup Range	OLMCPR	
	SVEA-96+	8x8-4
BOC10 to EOC10	1.22	1.26

The Misoriented Fuel Assembly OLMCPRs for Cycle 10 are shown below:

Burnup Range	OLMCPR	
	SVEA-96+	8x8-4
BOC10 to EOC10	1.25	1.25

Overpressurization Analysis Summary: MSIV Closure (Flux Scram)

Parameter	Value	Units
Maximum Vessel Pressure	1261	psia
Maximum Steam Dome Pressure	1235	psia
Maximum Steam Line Pressure	1238	psia

Control Rod Drop Accident

The Control Rod Drop Accident was evaluated for HCGS Cycle 10. The evaluation considered all exposures, A1 and A2 control rod withdrawal sequences, and implementation of a 10% low power setpoint. Based on the analysis of control rod worth and post-drop nodal power peaking conditions, the Control Rod Drop Accident maximum peak fuel enthalpy, including analysis uncertainties, will be less than 90 cal/g.

Core Thermal-Hydraulic Stability

Stability Interim Corrective Action

GE SIL-380 recommendations have been included in HCGS operating procedures. NRC approval for deletion of a cycle specific stability analysis is documented in NEDE-24011-P-A-US. Hope Creek recognizes the issuance of NRC Bulletin No. 88-07, Supplement 1, Power Oscillations in Boiling Water Reactors (BWRs), and will comply with the recommendations contained within.

Stability Long Term Solution Option III

In support of the installation of stability Long Term Solution Option III hardware and software at HCGS, ABB has confirmed the following results to be applicable

to Cycle 10 if stability Long Term Solution Option III were to be implemented in Cycle 10.

Analytical Oscillation Power Range Monitor (OPRM) Peak/Average Amplitude Setpoint (S_p)	**Two-Loop Operation Minimum OLMCPR**	
	SVEA-96+	**8x8-4**
1.083	1.27	1.27

Analytical Oscillation Power Range Monitor (OPRM) Peak/Average Amplitude Setpoint (S_p)	**≤75% Power SLO Minimum OLMCPR**	
	SVEA-96+	**8x8-4**
1.083	1.31	1.30

The listed analytical value of S_p is valid since the corresponding minimum OLMCPRs are bounded by all OLMCPRs throughout the power / flow operating regime throughout Cycle 10.

HCGS LOCA ECCS ANALYSIS RESULTS

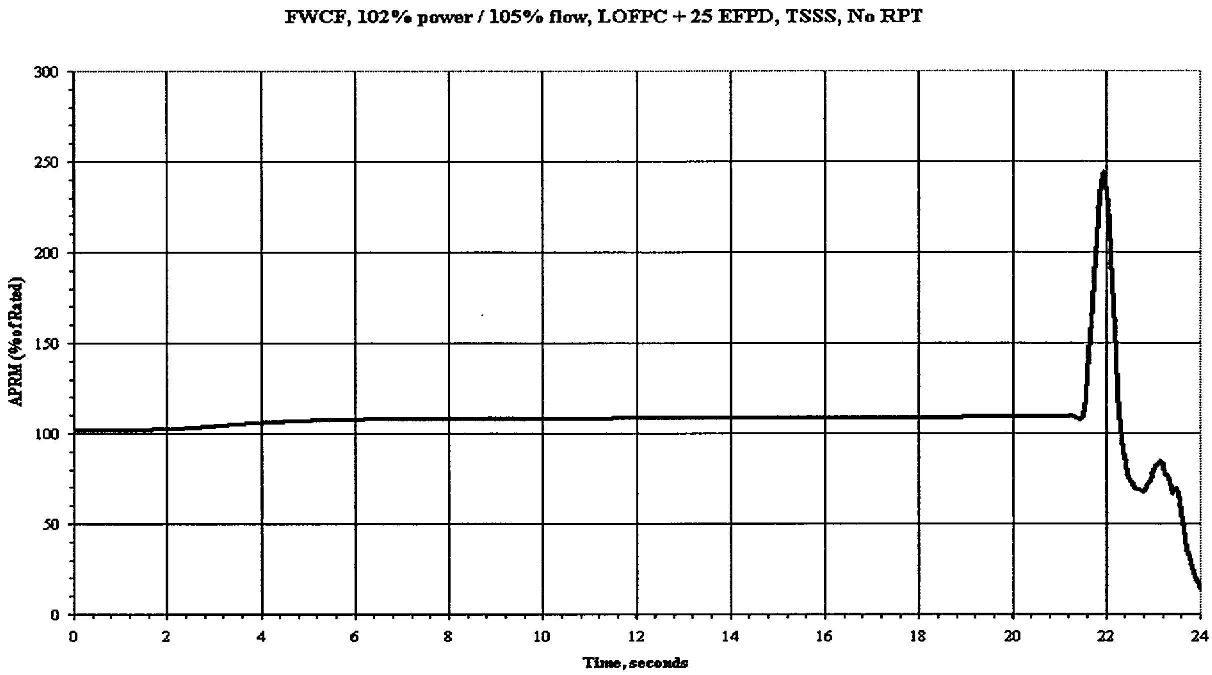
Parameter	**Value**		**Units**
	Two Loop Operation	**Single Loop Operation**	
Thermal Power (including 2% power Uncertainty)	104.2	75	% Rated Power
Core Flow	105	50	% Rated Flow
Peak Cladding Temperature	< 2051#		°F
Maximum Cladding Oxidation	< 9.0#		%
Total Core Hydrogen Generation	< 0.3		%

\# Values are the maximum for all SVEA-96+ fuel types and for Two Loop and Single Loop Operation.

See the Hope Creek Generating Station, Core Operating Limits Report, Cycle 10 / Reload 9, Revision 2, effective September 22, 2000, and submitted by letter LR-N00-0438 dated November 16, 2000 for the MAPLHGR limits applicable to Cycle 10.

Reload Analysis Results Summary

The following figures copied from the HCGS-UFSAR, Revision 11, November 24, 2000, provide Cycle 10 reload analysis results for selected limiting events.



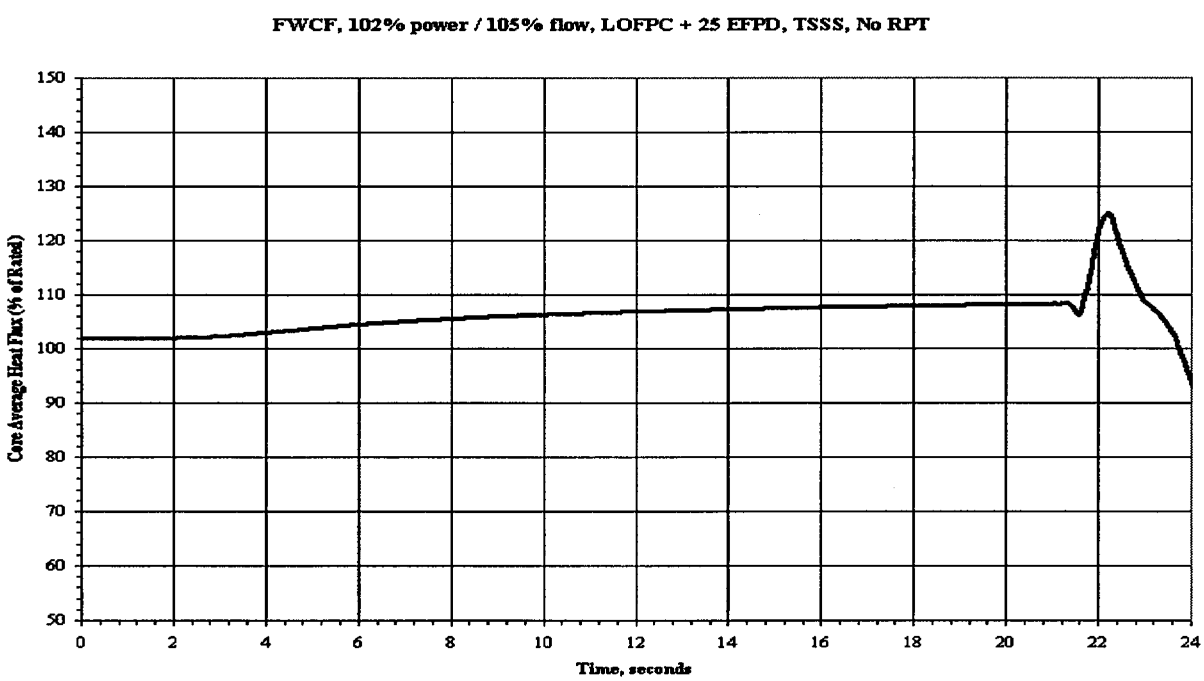
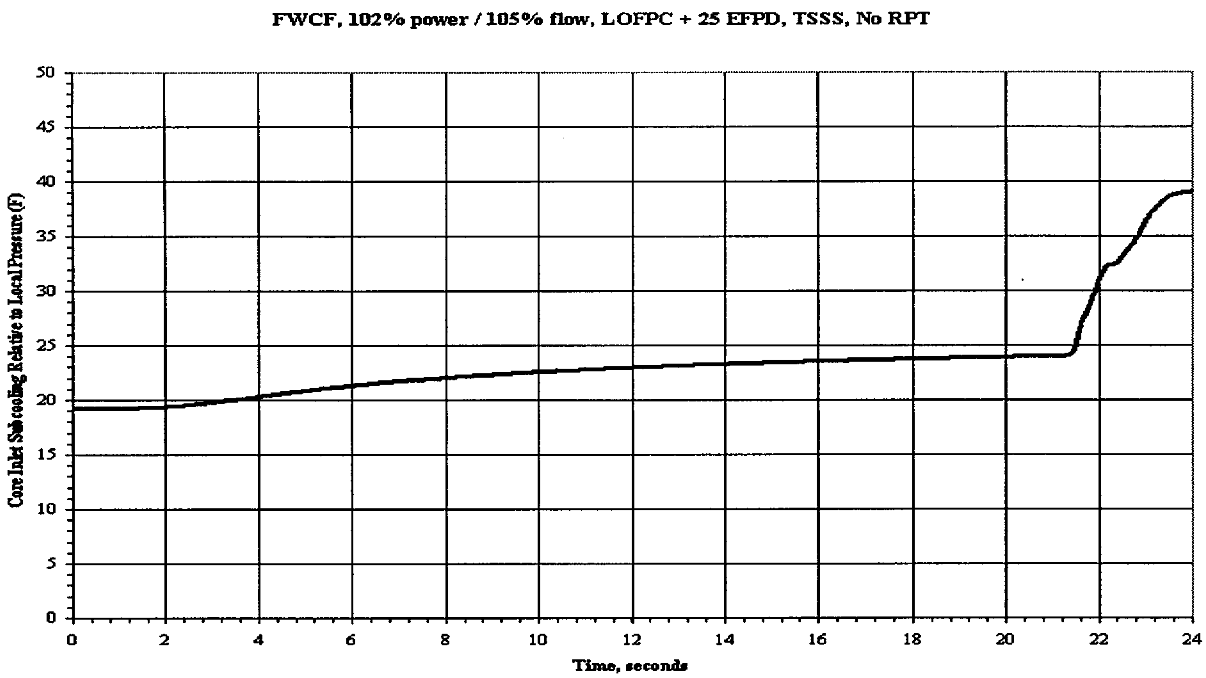


Figure 15D.1: APRM Flux and Core Average Heat Flux (FWCF at 102% Power / 105% Flow)



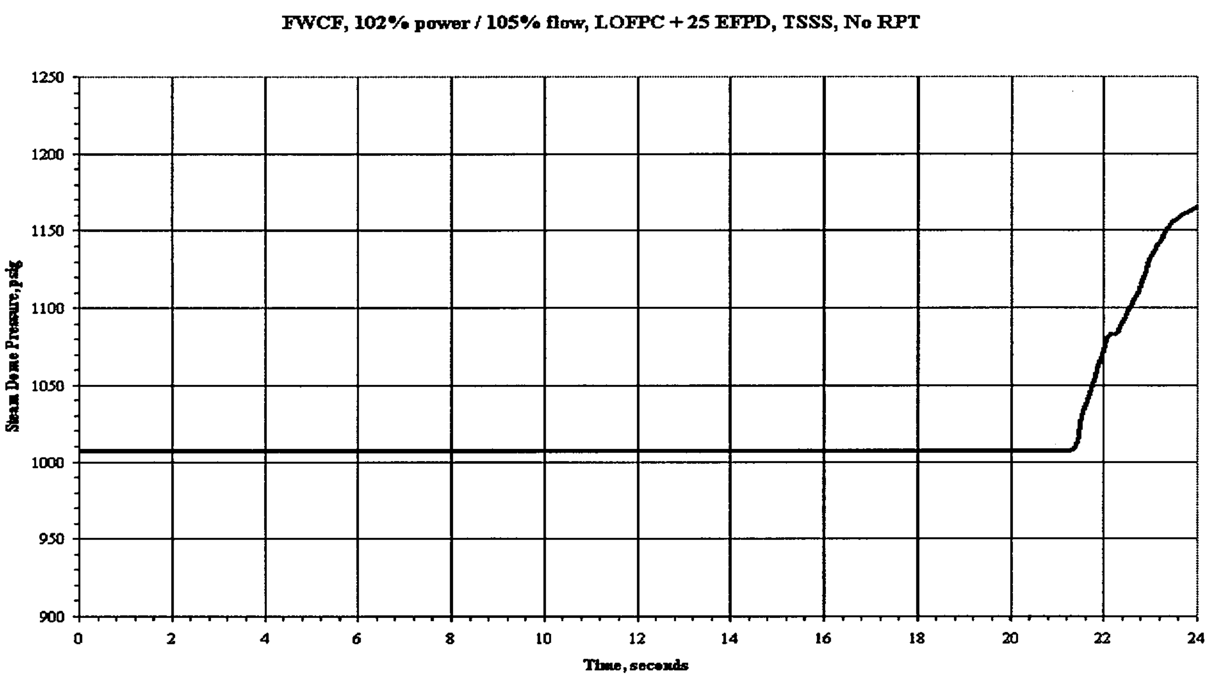
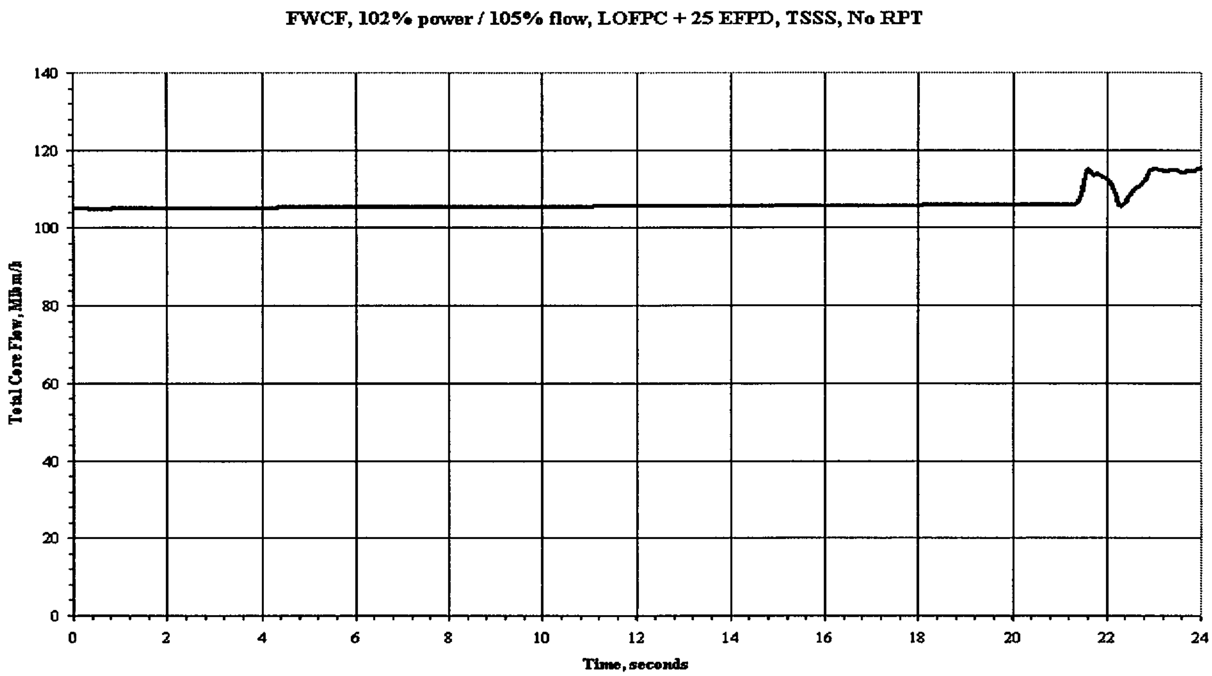


Figure 15D. 2: Core Inlet Subcooling and Steam Dome Pressure (FWCF at 102% Power / 105% Flow)



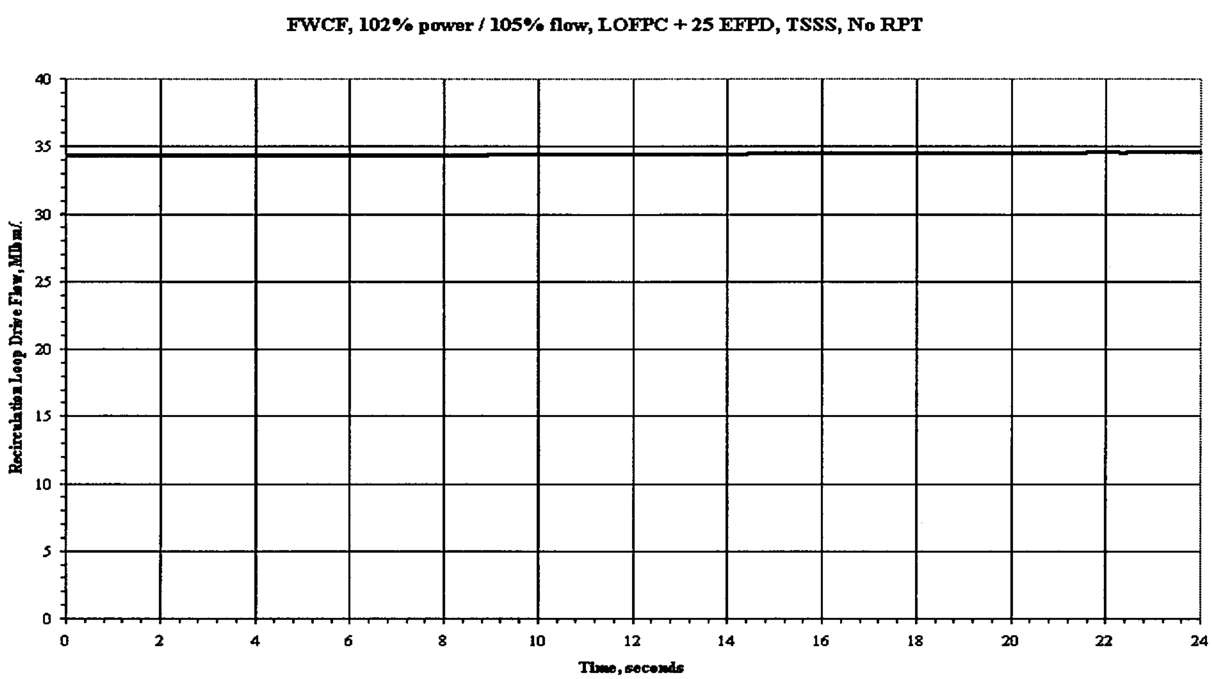
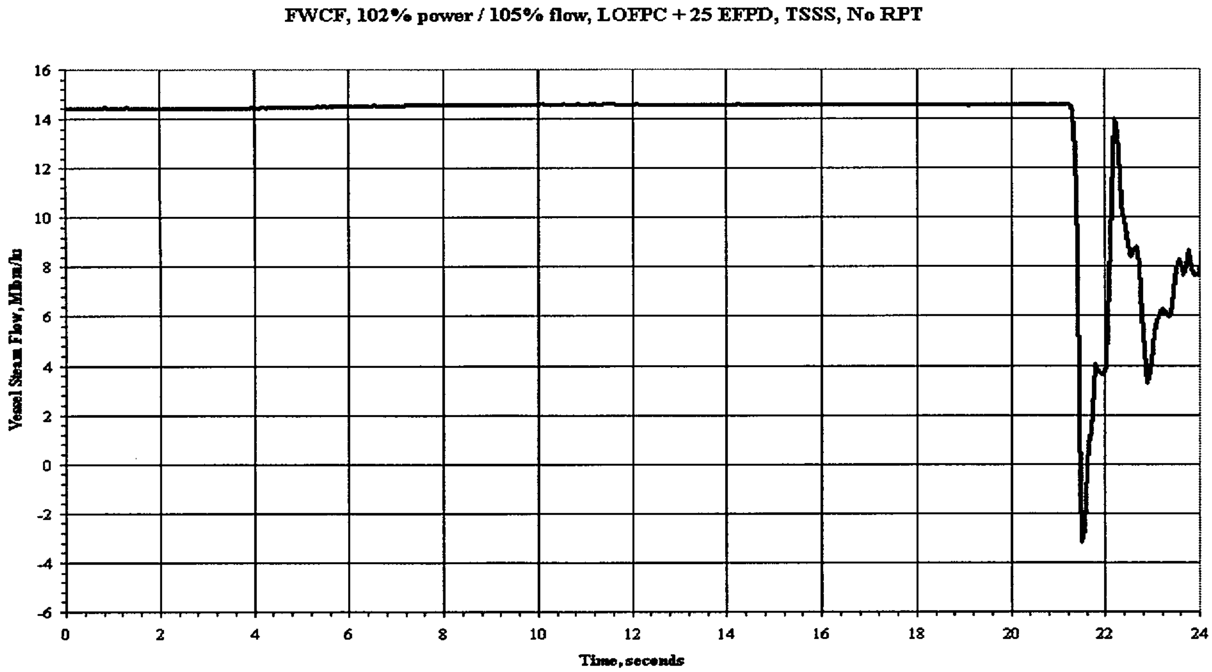


Figure 15D.3: Total Core and Drive Flow (FWCF at 102% Power / 105% Flow)



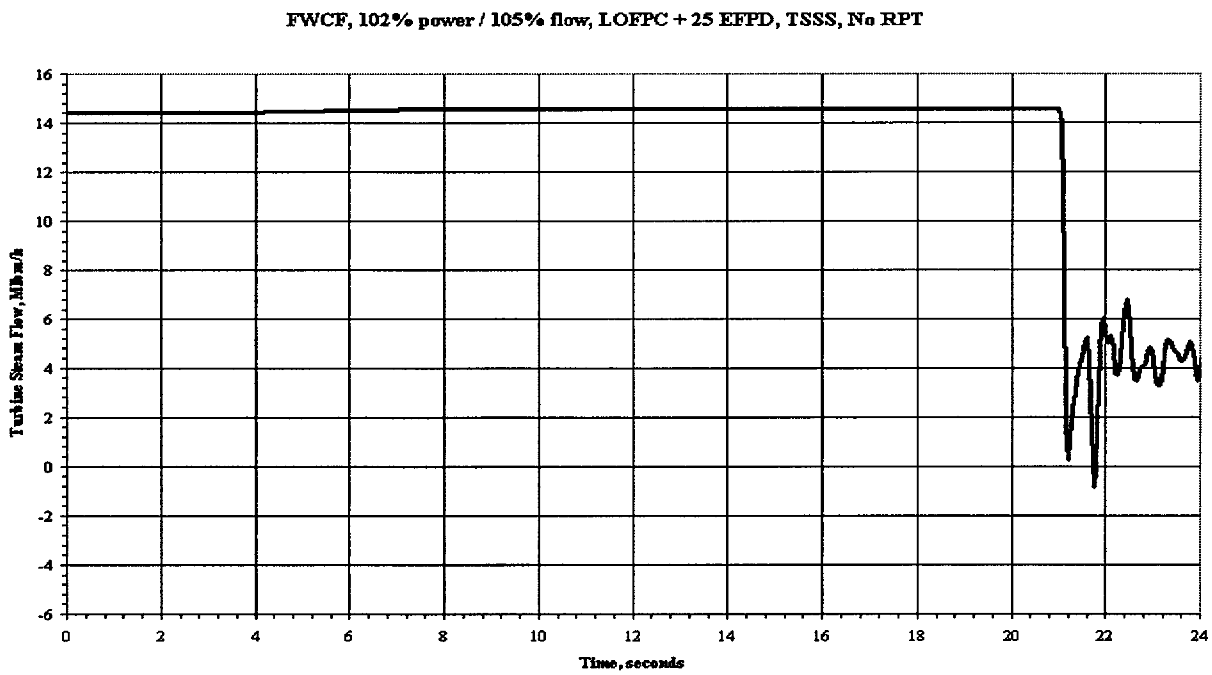
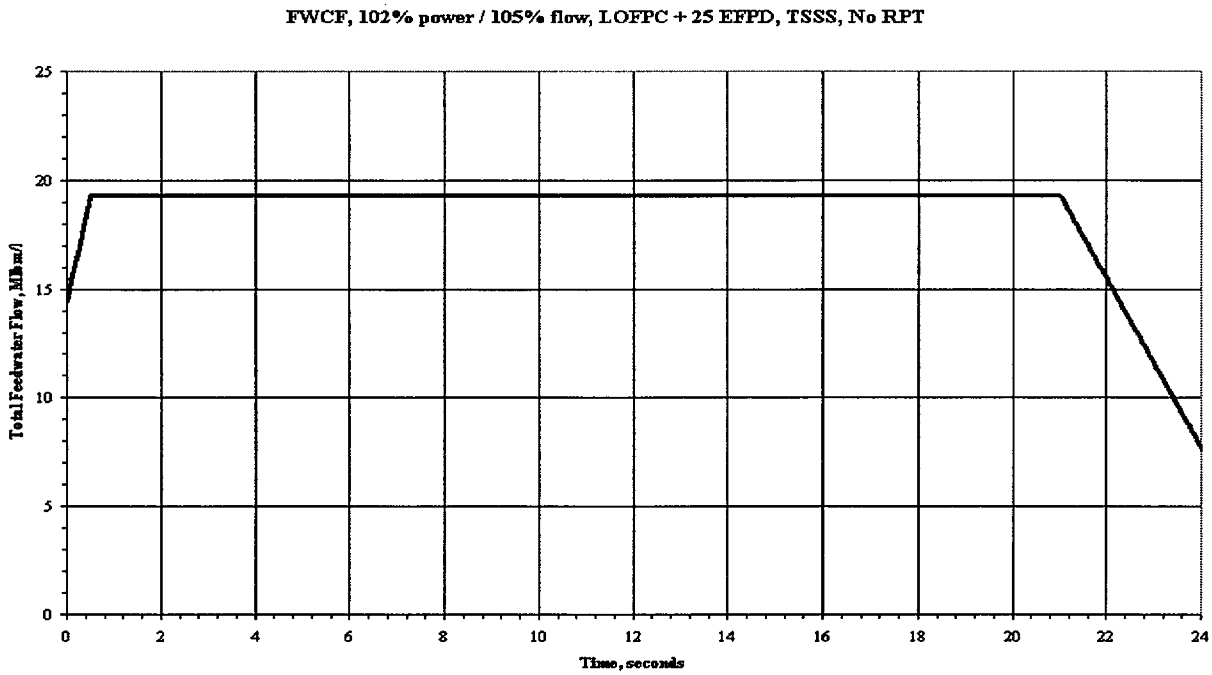


Figure 15D.4: Vessel Steam Flow and Turbine Steam Flow (FWCF at 102% Power / 105% Flow)



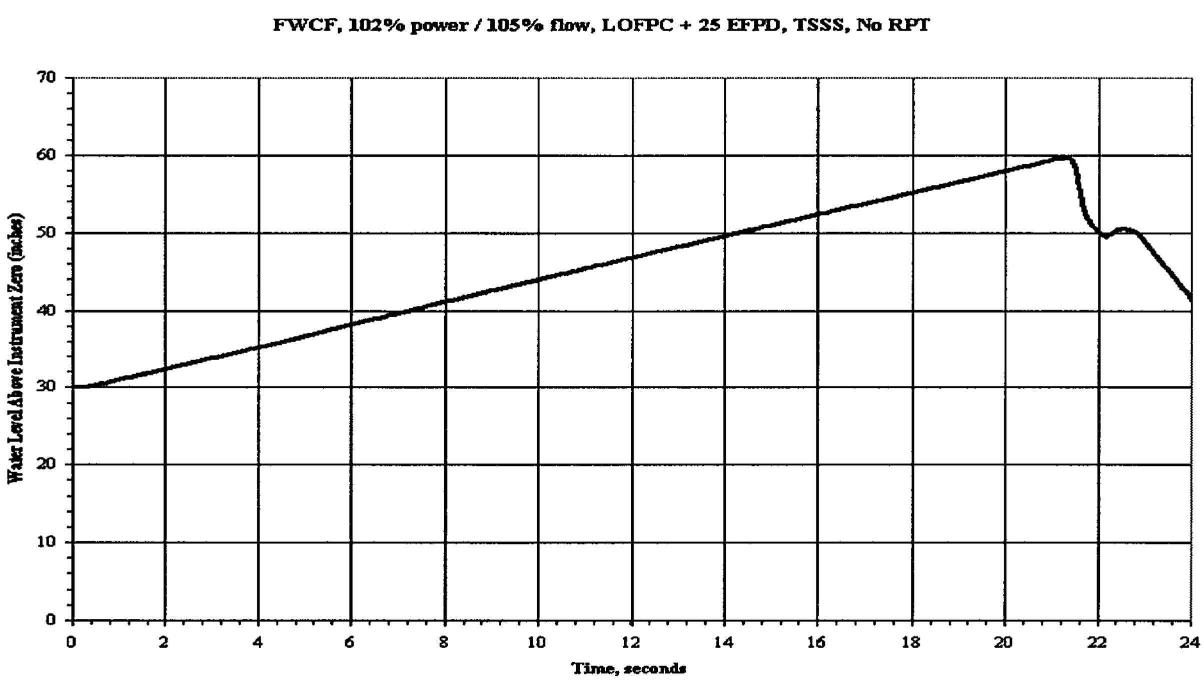


Figure 15D.5: Feedwater Flow and Water Level (FWCF at 102% Power / 105% Flow)

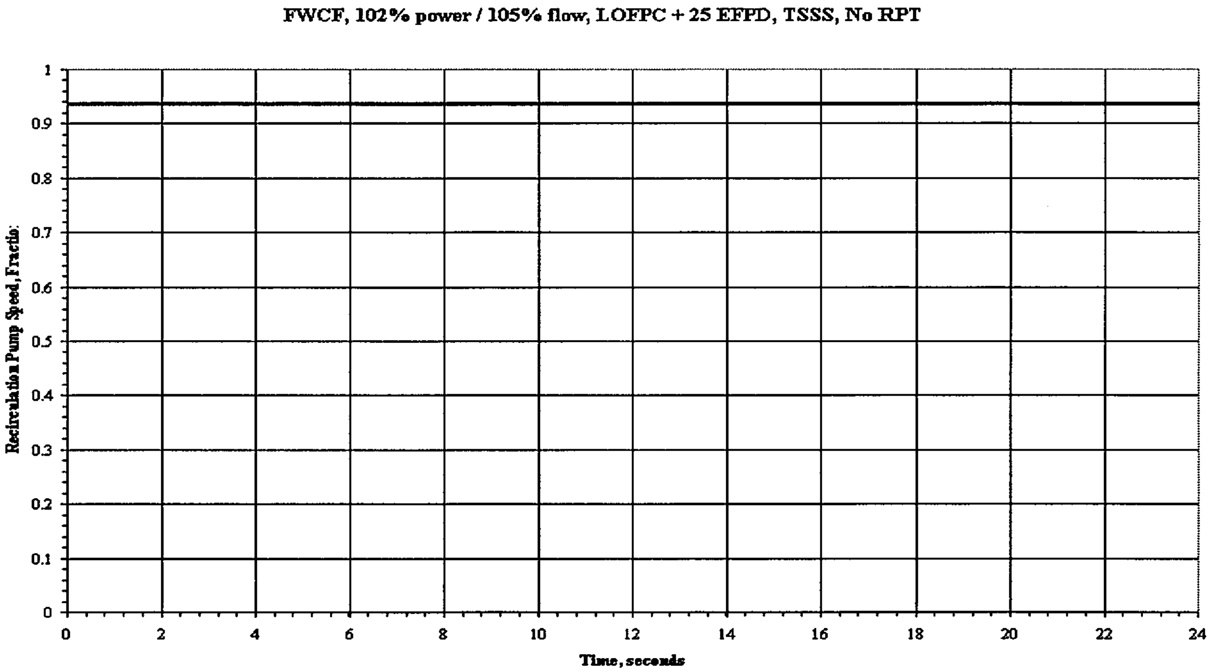
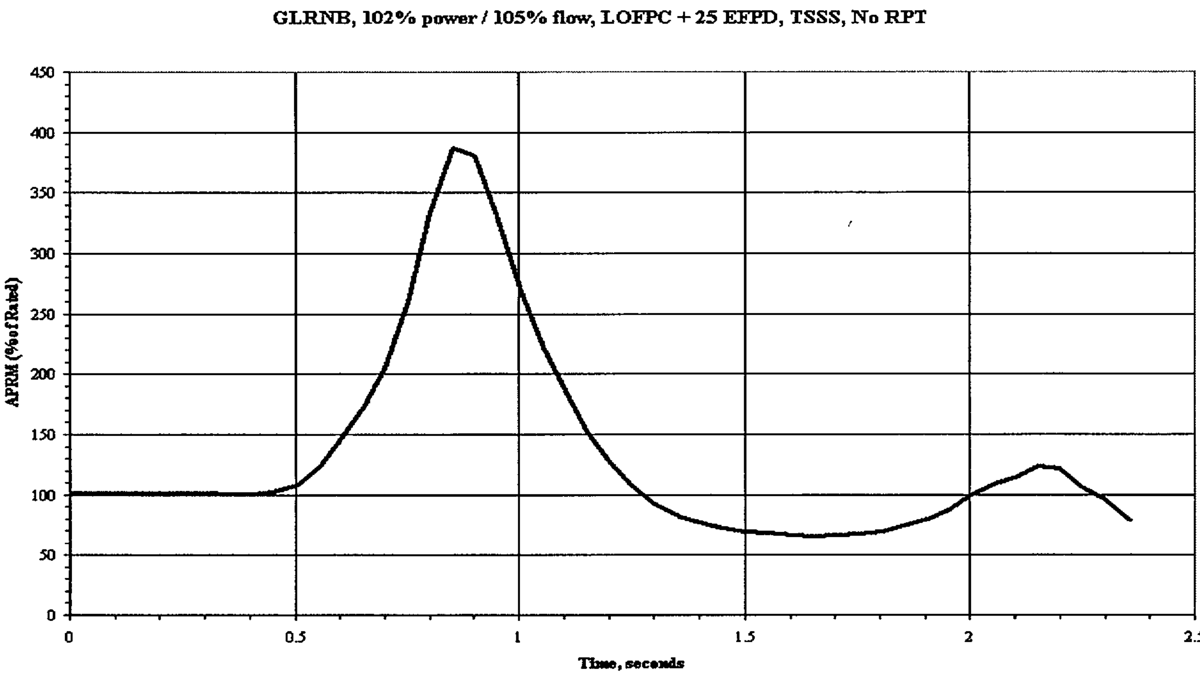


Figure 15D.6: Recirculation Pump Speed (FWCF at 102% Power / 105% Flow)



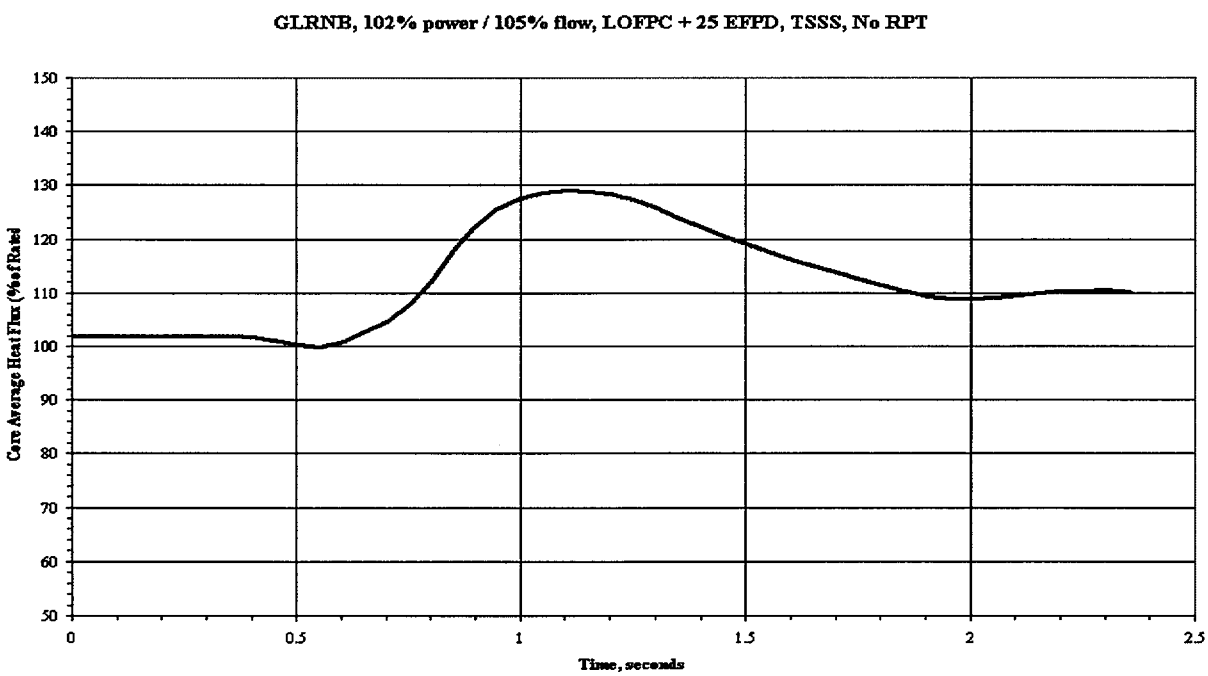
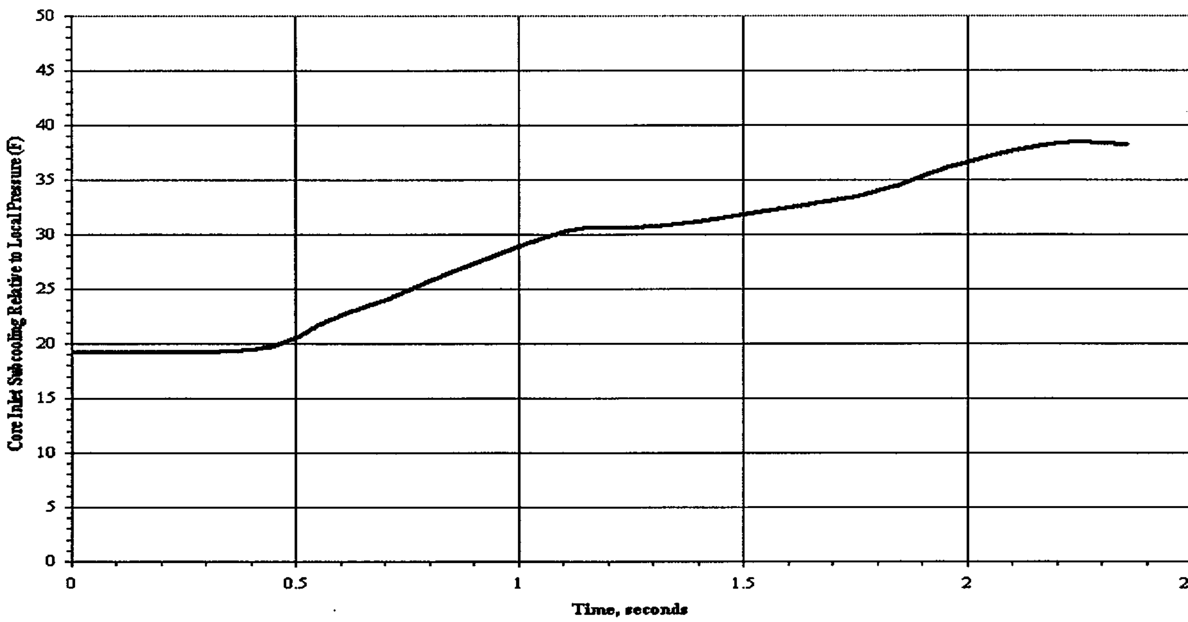


Figure 15D.19: APRM Flux and Core Average Heat Flux (GLRNB 102% Power / 105% Flow)

GLRNB, 102% power / 105% flow, LOFPC + 25 EFPD, TSSS, No RPT



GLRNB, 102% power / 105% flow, LOFPC + 25 EFPD, TSSS, No RPT

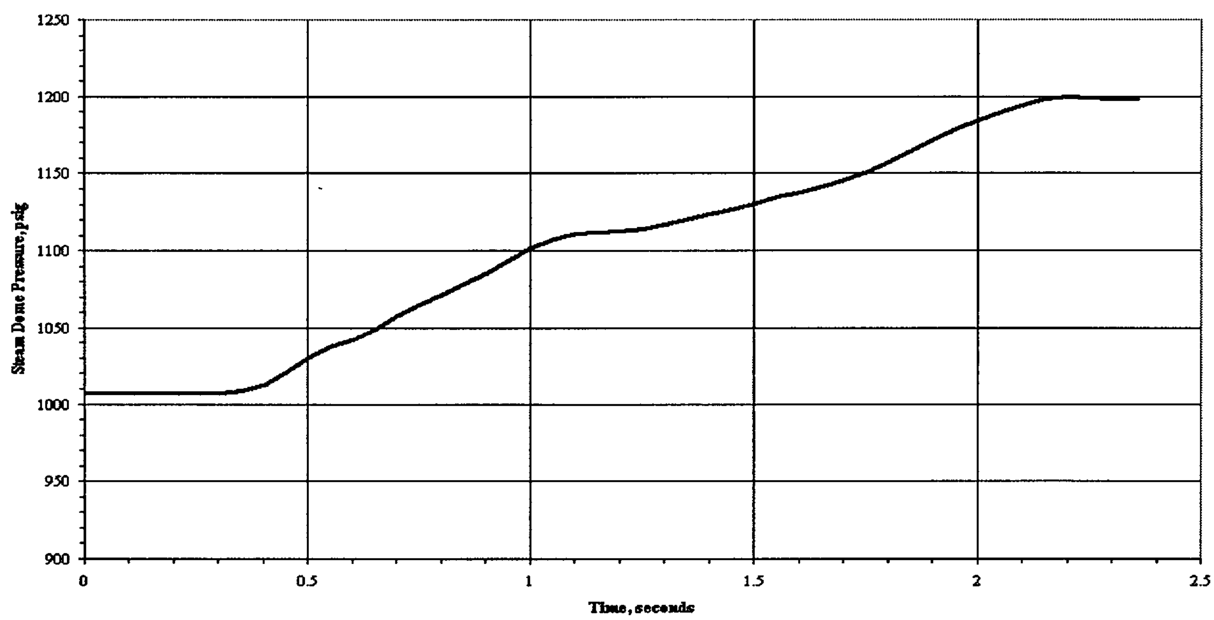
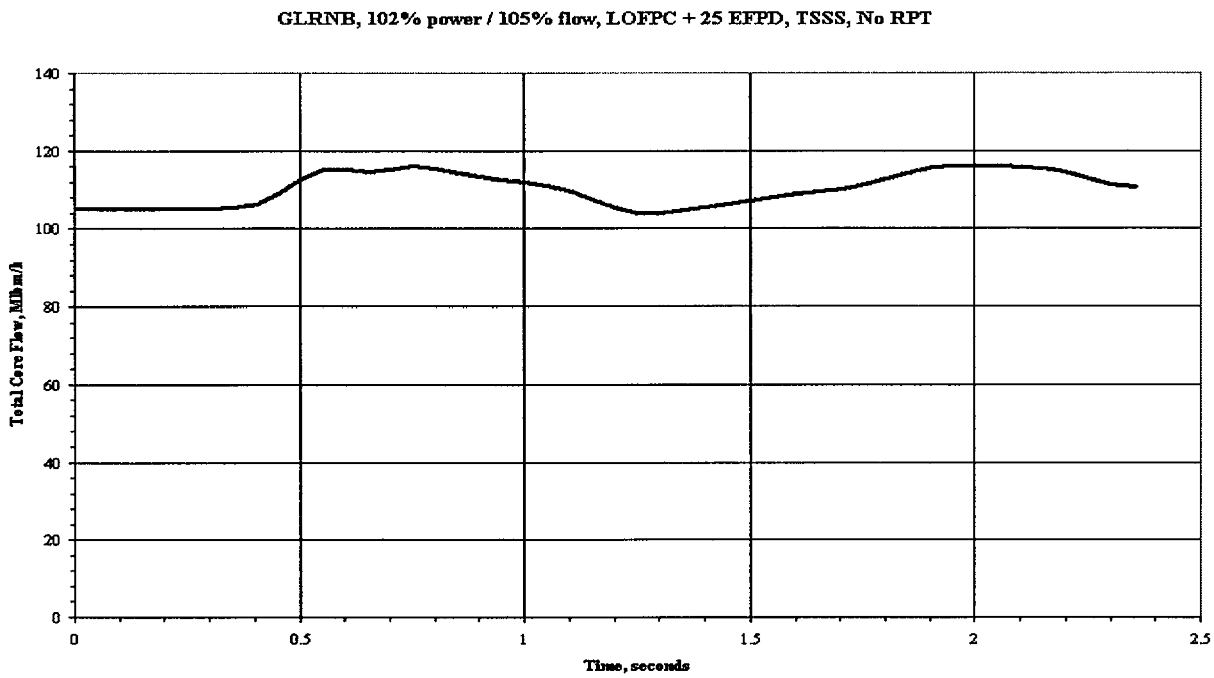


Figure 15D.20: Core Inlet Subcooling and Steam Dome Pressure (GLRNB 102% Power / 105% Flow)



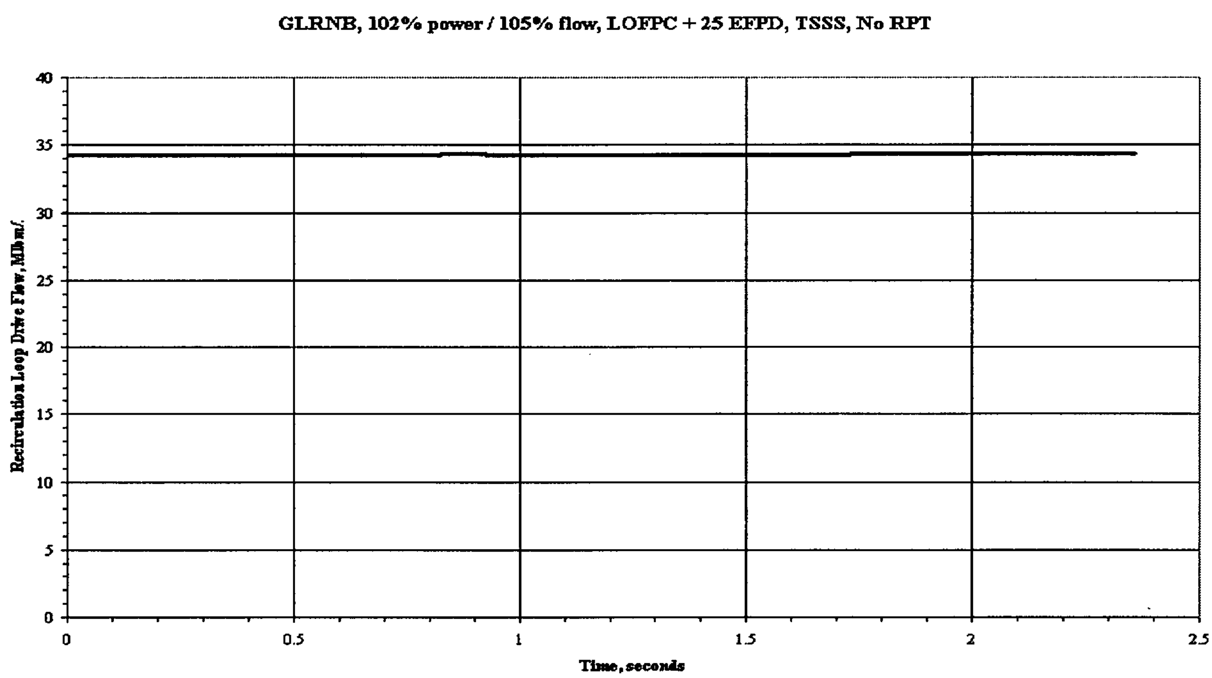
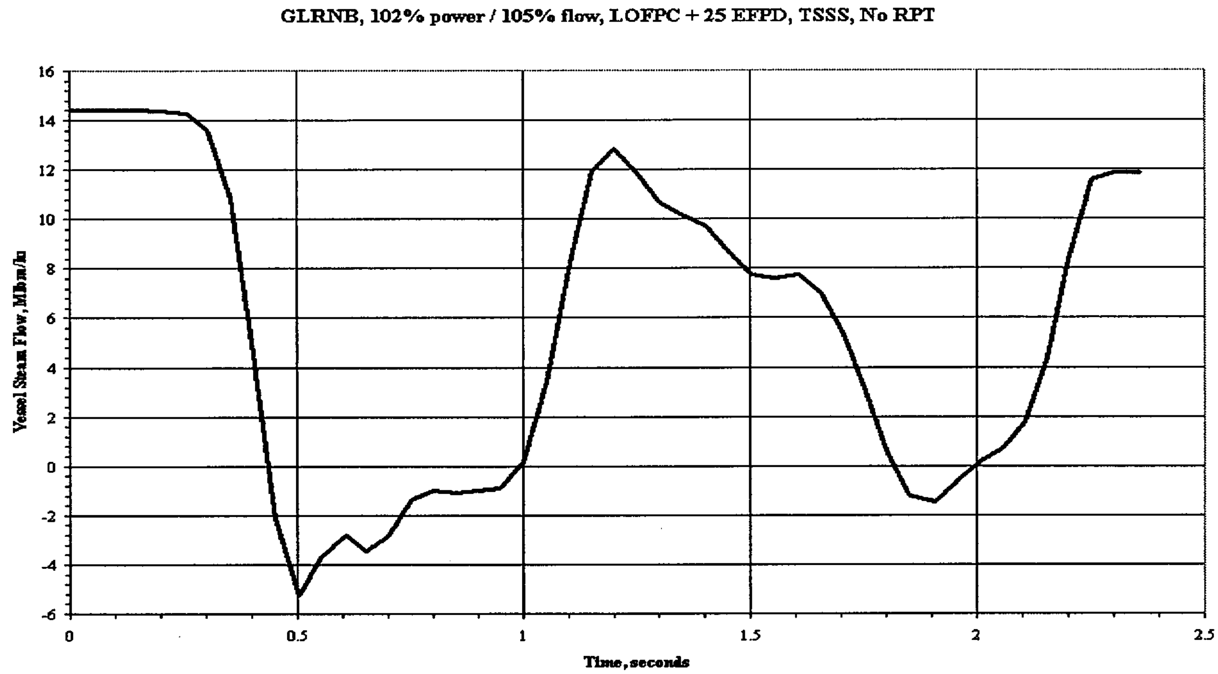


Figure 15D.21: Total Core and Drive Flow (GLRNB 102% Power / 105% Flow)



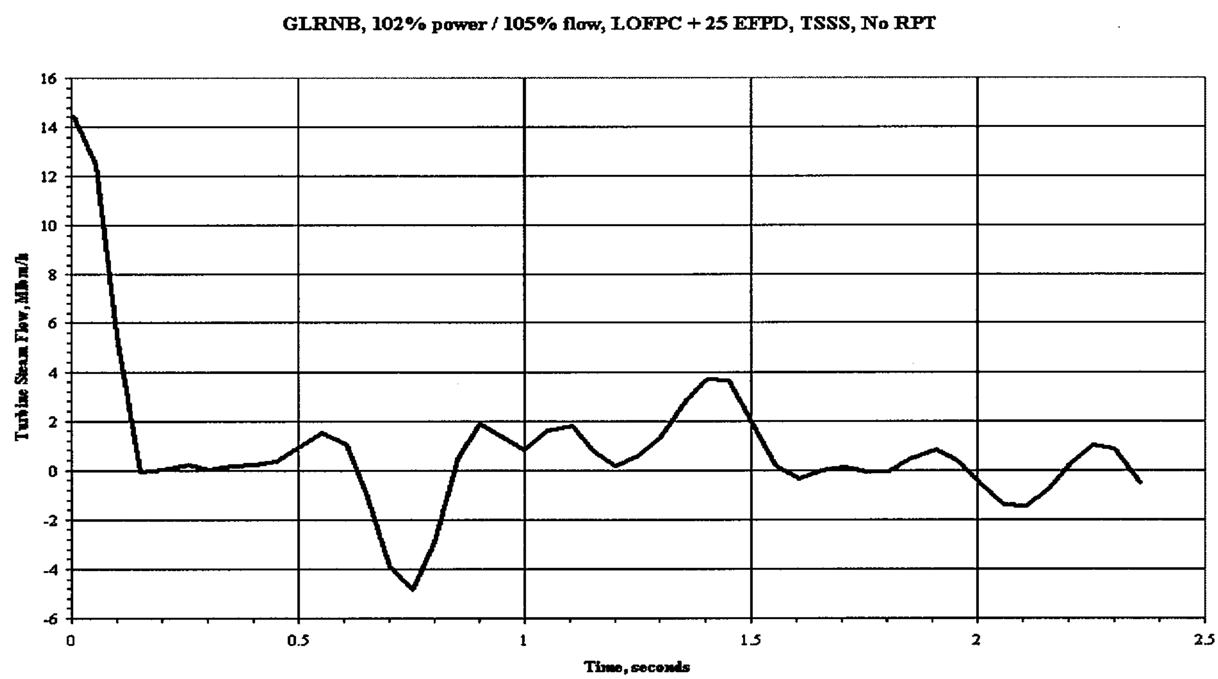


Figure 15D.22: Vessel Steam Flow and Turbine Steam Flow (GLRNB 102% Power / 105% Flow)

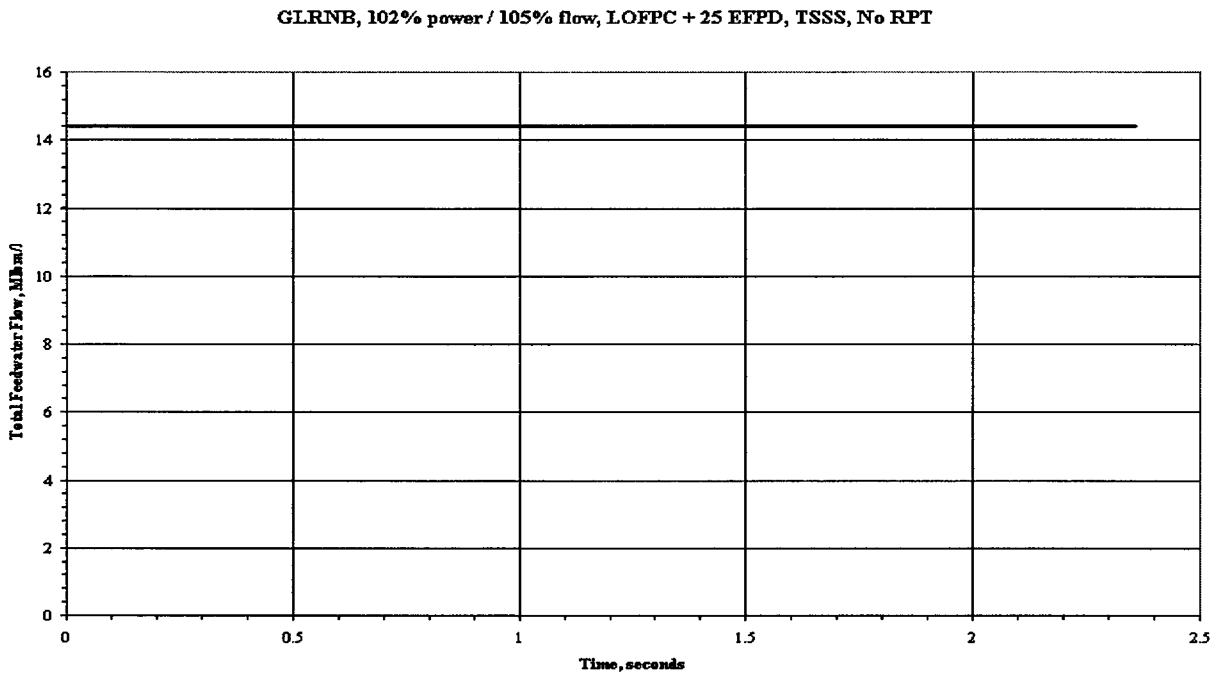


Figure 15D.23: Feedwater Flow (GLRNB 100% Power / 105% Flow)

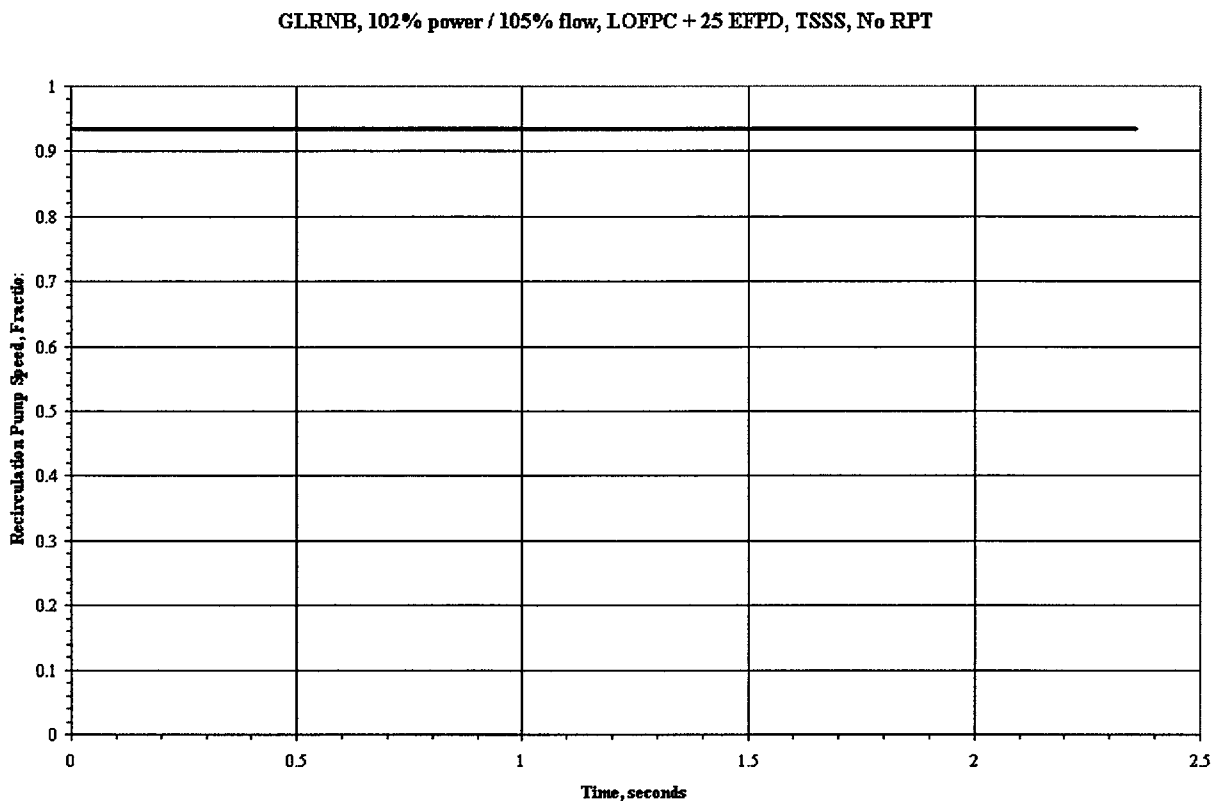


Figure 15D.24: Recirculation Pump Speed (GLRNB 100% Power / 105% Flow)

MCPR Rated, Flow and Power Dependent Limits

See the Hope Creek Generating Station, Core Operating Limits Report, Cycle 10 / Reload 9, Revision 2, effective September 22, 2000, and submitted by letter LR-N00-0438 dated November 16, 2000 for the OLMCPRs applicable to Cycle 10.

NRC Question:

6. Attachment 1, Section 5.5, of the submittal provides the justification for the requested power uprate with respect to the design of the reactor coolant and balance-of-plant (BOP) piping. List the most critical BOP piping systems that were evaluated for the power uprate. Provide a summary of the evaluation used for BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers, and anchorage for pipe supports.

   The regulatory basis for this question is that the BOP piping systems conform to the requirements of GDCs 1, 2, 4, 14, and 15 of Appendix A to 10 CFR Part 50 as they relate to maintaining structural integrity of pressure-retaining components and their supports (reference Standard Review Plan (SRP) Section 3.9.3).

PSEG Nuclear response to Q6:

The BOP piping systems that are affected by the 1.4% power uprate were mainly the Turbine cycle systems. More specifically, the systems, which had a slight change in their operating parameters were Main Steam, Extraction Steam, Turbine By-pass, Condensate and Feedwater.

To review the proposed 1.4% power uprate's impact on these BOP systems a new heat balance was generated. The results of the heat balance were compared to the system piping and component design parameters. These systems and their components were initially designed to operate at Turbine Valves Wide Open (VWO) heat balance parameters. Turbine VWO condition equates to about 105% steam flow at about 104.2% power. Proposed 1.4% power uprate increases the steam flow by about 1.8%, which is bounded by the original design. Original piping design parameters (temperature and pressure) enveloped the VWO parameters. Therefore, piping and component design parameters (temperature, pressure and flow) remained bounded by the original design. Calculations of Record piping stress analyses were also reviewed. The input parameters (temperature and pressure) for the piping stress analyses, which used the original design values, also remained bounding. Thus, no new stress analysis runs were required. Since the existing pipe stress analysis bounds the 1.4% power uprate conditions, the existing design bounds the power uprate conditions for the pipe supports, nozzles, penetrations, guides, or anchorage for pipe supports.

Based on this review the BOP systems will continue to maintain their structural integrity.

NRC Question:

7. Attachment 1, Section 5.11, of the submittal provides the justification for the requested power uprate with respect to the design of the control rod drive hydraulic system. Provide a summary of evaluation for the effects of the 1.4 percent power uprate on the design basis analysis of the control rod drive mechanism (CRDM). Confirm that the CRDMs structural integrity will be adequate for the 1.4 percent power uprate.

   The regulatory basis for this question is that the CRDMs conform to the requirements of GDC 14 of Appendix A to 10 CFR Part 50 as it relates to maintaining the reactor coolant pressure boundary.

PSEG Nuclear response to Q7:

The Control Rod Drive Mechanism (CRDM) is used for positioning the control rod in the reactor core. The CRDM is a hydraulic cylinder using water from the condensate treatment system and/or the condensate storage tank (CST) as its operating fluid. Reactor operating and design pressures are not changed as a result of the proposed 1.4% power uprate. Thus, design and operation of the CRDM is not affected.

The design of the CRDM components which are part of the reactor coolant pressure boundary were also reviewed. The calculated stresses for Control Rod Drive (CRD) ring flange, Indicator Tube, CRD Housing, Control Rod Guide Tube and CRD Housing Support (Beams and Grid Structure) are listed in the HCGS UFSAR Tables 3.9.4aa, 3.9.4cc, 3.9.4.w and 3.94.x. Calculated stress values demonstrate considerable margin to the allowable stress. Since the Reactor Vessel design/operating pressure and temperature and the CRD Hydraulic system pressure, temperature and flow parameters are not changed, it is concluded that the structural integrity of the CRDM is not affected.

NRC Question:

8. Discuss the functionality of safety-related mechanical components (i.e., all safety-related valves and pumps, including air-operated valves (AOV) and power-operated relief valves) affected by the power uprate to demonstrate that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at HCGS will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves

for GL 95-07 and on the evaluation of overpressurization of isolated piping segments for GL 96-06.

The regulatory basis for this question is that the assumptions, analyses, and conclusions of the HCGS programs associated with GL 89-10, GL 95-07, and GL 96-06 remain valid (i.e., consistent with the current licensing basis).

PSEG Nuclear response to Q8:

Safety related systems are designed and analyzed to mitigate the consequences of an accident or a transient, maintain containment integrity and provide long-term decay heat removal capability at 102% power level. The proposed 1.4% power uprate remains bounded by the original design basis for the safety related SSC's. The design parameters (pressure, temperature and flow) of the safety related systems are not impacted by the proposed power uprate. Therefore, the safety related components are expected to perform as designed.

As discussed in section 6.1.4 of the December 1, 2000, request for amendment, the Generic Letter 89-10 MOV program was reviewed for the proposed 1.4% power uprate. Program guidelines require that worst-case pressure, temperature and flow parameters be used for determining the differential pressure and flow conditions. The only systems within the GL 89-10 program where the normal operating pressure, temperature and flow increasing slightly are the Feedwater and the Main Steam System. The Main Steam system will only experience a slight increase in flow but no change in operating pressure and temperature.

For the Main Steam System valves, safety related MOVs include drain line isolation valves, main steam drain valves, steam header downstream drain isolation valve, startup drain valves, main steam stop valve and the pneumatically operated MSIVs. Only the MSIVs will experience a flow rate change. Since closures of MSIV's are flow assisted, the increased flow will help in the closure function. The main steam stop valves are closed under no flow conditions to initiate MSIV sealing system. No other operating parameters are changed for other isolation valves. Therefore there is no impact on the MOV program for the Main Steam System.

For the Feedwater system, safety related MOVs include Feedwater inlet check valves and a crosstie isolation valve. There is no differential pressure across the Feedwater inlet check valves (1AE-HV-FO32A/B) for normal and abnormal operations of the inlet check valve. Differential pressure for the crosstie isolation valve (1AE-HV-4144) has been conservatively calculated using the pump shutoff head and low water temperature. Therefore the MOV program for the Feedwater system is also not impacted.

The 1.4% power uprate does not affect the capability of MOV's to operate per the GL 89-10 program.

GL 96-06 addressed the overpressurization of isolated piping segments as a result of the environmental or internal heat sources. Design Basis LOCA analyses that affect piping segments inside containment, or at containment penetrations have been performed at 102% power which bounds the proposed 1.4% power uprate. Thus, the resultant environmental conditions remain bounding.

GL 95-07 addressed pressure locking (PL) and thermal binding (TB) of safety related power operated gate valves. Evaluations that were performed addressed safety related gate valves that are power operated (motor or air) and are required to open to perform their intended safety function. Potentially susceptible valves were limited to the Residual Heat Removal, Reactor Core Isolation Cooling, High Pressure Coolant Injection and Core Spray systems. The proposed 1.4% power uprate however, does not change any of the process fluid parameters of these systems. The worst-case accident conditions assumed for the environmental conditions at these locations was postulated to occur at 102% power level. Therefore, the conclusions of the GL 95-07 program evaluations are not impacted.

NRC Question:

9. Nuclear power plants are licensed to operate at a specified power, which, at operating power levels, is indicated in the control room by neutron flux instrumentation that has been calibrated to correspond to core thermal power. Core thermal power is determined by a calculation of the energy balance of the plant nuclear steam supply system. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, temperature, and pressure measurements, which are not safety grade and are not included in the plant technical specifications.

   The uncertainty of calculating values of core thermal power determines the probability of exceeding the power levels assumed in the design basis transient and accident analyses. In this regard, to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties), Appendix K to 10 CFR Part 50, requires loss of coolant accident (LOCA) and emergency core cooling system (ECCS) analyses to assume that the reactor had operated continuously at a power level at least 102 percent of the licensed thermal power. The 2 percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. Later, the NRC concluded that, at the time of the original ECCS rulemaking, the 2 percent power margin requirement appeared to be based solely on considerations associated with power measurement uncertainty.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2 percent margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. In the June 1, 2000, *Federal Register* (Volume 65, Number 106, Rules and Regulations, pages 34913-34921) the Commission published a final rule to reduce an unnecessarily burdensome regulatory requirement by allowing licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation.

The purpose of the proposed changes is to obtain a power uprate on the basis of plant modifications that would result in improved accuracy of feedwater flow rate measurement, which is used in the calculation of reactor thermal power. The improved instrumentation (Crossflow ultrasonic flow measurement system) would allow the licensee to operate HCGS with a reduced margin between the actual power level and the 102 percent margin used in the licensing basis ECCS analyses.

To complete its review of the proposed license changes, the staff requests a description of the programs and procedures that will control calibration of the non-safety-grade instrumentation that affect the total power uncertainty described in the licensee's proposed power uprate license amendment. The licensee has provided this information for the Crossflow system. For the remaining instrumentation the description should include a discussion of the procedures for:

a. Maintaining calibration;
b. Controlling software and hardware configuration;
c. Performing corrective actions;
d. Reporting deficiencies to the manufacturer; and
e. Receiving and addressing manufacturer deficiency reports.

<u>PSEG Nuclear response to Q9:</u>

Maintaining Calibration

Preventive maintenance (PM) is performed on the feedwater measurement instruments as well as the instruments listed below that affect the power uncertainty. The PMs listed and intervals are current practice but may be revised in the future based on the PM program requirements. The PM program is currently controlled by procedure NC.WM-AP.ZZ-0003, "Regular Maintenance Process."

Feedwater Temperature

Feedwater temperature instruments H1AE -1AETT-N602A,B,C,D-B21, are calibrated on a nominal 18-month cycle. Procedure HC.IC-DC.ZZ-0089 currently performs the calibration of the above listed devices.

Reactor Pressure

Reactor Pressure instrument H1BB -1BBPT-N005-C32 is calibrated on a nominal 18-month cycle. Procedure HC.IC-LC.BB-0005 currently performs the calibration of this device.

Control Rod Drive Flow

Control Rod Drive Flow instrument H1BF -1BFFT-N004-C11 is calibrated on a nominal 18-month cycle. Procedure HC.IC-DC.ZZ-0088 currently performs the calibration of this device.

Reactor Water Cleanup

Reactor Water Cleanup Flow instruments H1BG -1BGFT-N036A,D-G33 are calibrated on a nominal 18-month cycle. Procedures HC.IC-SC.BG-0003 and HC.IC-SC.BG-0006 currently perform the calibration of these devices.

Reactor Water Cleanup Temperature

Reactor water cleanup temperature instruments H1BG -1BGTE-N015-G33 and H1BG -1BGTE-N004-G33 will be calibrated every 18 months per procedure HC.IC-GP.ZZ-0004(Q).

Recirculation Pump Watts

Recirculation Pump Watts instrument H1BB -1BBWT-8251A,B-B31 is calibrated on a nominal 18-month cycle. Procedure HC.IC-DC.ZZ-0150 currently performs the calibration of this device.

Controlling Software and Hardware Configuration

The software and hardware configuration of digital plant instrumentation (e.g., Crossflow and plant computer) are controlled by procedure NC.NA-AP.ZZ-0064(Q), "Software Quality Assurance" and the associated implementing procedures. These procedures ensure that the appropriate quality level classifications are identified for the equipment. The quality level classification in turn determines the appropriate software quality assurance program elements that are applied to the equipment.

Performing Corrective Actions

Maintenance and corrective action items are generated through PSEG Nuclear's notification process that is governed by procedure NC.WM-AP.ZZ-0000, "Notification Process." This program is constructed to ensure conditions adverse to quality are dispositioned and corrected in accordance with 10CFR50, Appendix B, Criterion XV, Nonconforming Materials, Parts, or Components, and Criterion XVI, Corrective action.

Reporting Deficiencies to the Manufacturer

Vendors are contacted to assist in the determination of Part 21 reporting for equipment deficiencies that cross the threshold of requiring reporting under 10 CFR Part 21. During the course of maintenance, vendors are routinely contacted to assist in the repair of station equipment, however, there is no formal process for reporting every equipment deficiency to the manufacturer.

Receiving and Addressing Manufacturer Deficiency Reports

Manufacturer deficiency reports are handled through PSEG Nuclear's vendor information process which is governed by procedure NC.NA-AP.ZZ-0043, "Vendor Information Program." When vendor documents are received they are routed to the responsible group for evaluation and disposition. External 10 CFR Part 21 deficiencies submitted by vendors are processed as prescribed in procedure NC.PM-AP.ZZ-0603(Q), "Specification Review, Approval and Processing of Supplier Part 21 Data." Vendor Part 21 items are tracked under PSEG Nuclear's corrective action program for proper disposition.

ATTACHMENT 2
HOPE CREEK GENERATING STATION UNIT NO. 1
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INCREASED LICENSED POWER LEVEL

On April 26, 2001, the NRC issued a request for additional information (RAI) concerning PSEG Nuclear's request for amendment to increase the licensed power level for Hope Creek Unit No. 1. This attachment provides the response to the RAI questions.

NRC Question:

1. Attachment 1, Section 9.3, of the submittal provides the justification for the requested power uprate with respect to the design of the Standby Liquid Control (SLC) system. Attachment 1, Section 14.5 of the submittal provides justification for the requested power uprate with respect to the ability to mitigate the consequences of anticipated transients without scram (ATWS). Your submittal states that the capability of the SLC system to provide its backup shutdown and ATWS functions is not affected by the power uprate. Please provide the following additional information:

    a) With respect to the HCGS reload analysis, confirm that the ATWS analysis is bounded by the uprated power level of 101.4 percent.

    b) Confirm that the MSIV closure transient is the ATWS event for HCGS that yields the maximum pressure during the timeframe for which the SLC system is assumed to inject. Discuss the design basis assumptions and analysis regarding the capability of the SLC system to inject during an ATWS event.

    The regulatory basis for this question is that the proposed power uprate does not affect the ability of SLC system to continue to meet the requirements of 10CFR 50.62(c)(4).

PSEG Nuclear Response:

a) With respect to the HCGS reload analysis, the conclusions of the design basis ATWS analysis and evaluations remain applicable and bounding for uprated power levels up to and including 101.5%.

b) The MSIV closure transient, coupled with a postulated failure to scram is the most limiting ATWS transient from the standpoint of peak vessel pressure, peak heat flux, and peak suppression pool temperature.

Event sensitivity studies have not been performed to determine the event that yields the maximum pressure during the timeframe for which the SLCS is assumed to inject. Primarily this is due to the fact that the SLCS injection is delayed during the ATWS event to allow for the Alternate Rod Insertion (ARI) function to be successful. Consequently, at the time when SLCS injects for all the ATWS events evaluated, the system pressure is below the maximum calculated ATWS pressure due to the actuation of the main steam safety/relief valves (SRVs).

Therefore, from a SLCS design basis assumption viewpoint, the SLCS must be able to inject at the system pressures achievable during the ATWS event when considering the SRVs pressure relieving capability. For HCGS, the maximum SRV lift setpoint is 1130 psig ± 3%. The SLCS pumps are positive displacement pumps with a capability to inject up to 1400 psig (injection capability limited by SLCS relief valve setting of 1400 psig). Therefore, the design capability of the SLCS pumps in conjunction with the SLCS injection delay assures that the boron solution can be delivered to the reactor core during the postulated ATWS events.

The reactor operating pressure and the nominal SRV setpoints are not changed due to the 1.4% uprate. Therefore, the SLCS will continue to function as designed.

ATTACHMENT 3
HOPE CREEK GENERATING STATION UNIT NO. 1
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
INCREASED LICENSED POWER LEVEL

A telephone conversation was held on May 1, 2001, between PSEG Nuclear and the NRC staff concerning some additional clarification needed in response to the NRC's review of the December 1, 2000, Hope Creek Request for Amendment to increase the licensed power level. Below is a summary of the questions and the additional clarifying informing.

NRC Question:

Attachment 1, Section 13.5 of the submittal states that the changes due to the proposed power uprate will have minimal effect on the plant simulator. Are the changes to the simulator being made in accordance with ANSI/ANS 3.5, "Nuclear Power Plant Simulators for Use in Operator Training?" If so, what revision of the standard is being applied? If ANSI/ANS 3.5 is not being used, please provide an explanation.

PSEG Nuclear response:

The changes to the Hope Creek simulator are being made in accordance with ANSI/ANS3.5-1993.

NRC Question:

Attachment 1, Section 5.2 of the submittal discusses reactor overpressure protection analysis and states that the peak vessel bottom pressure is 1331 psig which is below the 1375 psig ASME limit. This section also states that the ASME limit of 1375 psig is 110% of the design value. Attachment 1, Section 15.2 of the submittal discusses fuel related overpressure protection analysis and states that there is more than 100 psi margin to the ASME 110% peak pressure design criteria (i.e., 1375 psig). Please clarify the apparent discrepancy between these two sections and verify that the peak pressure is 1331 psig.

PSEG Nuclear response:

Attachment 1 Section 5.2 of the 1.4% power uprate submittal is discussing reactor overpressure protection in relation to safety/relief valve (SRV) design bases. The discussion relative to the peak vessel bottom pressure of 1331 psig is specifically referring to analyses performed for increasing the Technical Specification limits on the SRV setpoint tolerances. The justification for

increasing the Technical Specification SRV setpoint tolerances from 1% to 3% was based on conservative cycle generic analyses that assumed 13 SRVs available all lifting with a single upper limit setpoint of 1250 psig in addition to an initial core thermal power of 102%. The intent of these analyses was to conservatively and generically demonstrate that the overpressurization consequence was acceptable for an increase of the SRV setpoint tolerances from 1% to 3%. The NRC approved this change to the HCGS Technical Specifications in Amendment No.115. As stated in the 1.4% uprate submittal, since the SRV setpoints and pertinent reactor operating parameters are not being changed by the 1.4% power uprate, the proposed change has no impact on the reactor overpressure protection analyses as discussed in Attachment 1 Section 5.2.

Attachment 1 Section 15.2 of the 1.4% power uprate submittal is discussing fuel related overpressure protection analyses. On a cycle specific basis, reload core analyses are performed to assure that the ASME limits are met when considering the proposed reload core design and the Technical Specification requirements and limits. The reload specific overpressurization analyses utilize the Technical Specification SRV lift settings at the maximum tolerances (lift settings + 3%) which are all lower than the 1250 psig single upper limit lift setting assumed in the SRV setpoint tolerance increase analyses. Consequently, the cycle 10 specific reload analysis resulted in predicted peak pressures with at least 100 psi margin to the ASME 110% peak pressure design criteria.

Therefore, there is no discrepancy between the two sections and the peak pressure is less than 1275 psig as determined by cycle 10 specific reload analysis.