APPENDIX 15D

SUSQUEHANNA STEAM ELECTRIC STATION UNIT 2 FINAL SAFETY ANALYSIS REPORT - CYCLE SPECIFIC DATA

15D.1 Appendix D Contents

15D.1.1 Content Discussion

This Section presents results that are typical of cycle-specific analyses. Actual cycle-specific results may be found in, or calculated from, Reference 15D.1.2-1.

15D.1.2 References

15D.1.2-1 ANP-3741(P), Revision 2, “Susquehanna Unit 2 Cycle 20 Reload Licensing Analysis,” Framatome Inc., June 2019. (General Reference per NEI 98-03)

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 (TYPICAL)

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psia	Maximum Vessel Pressure psia	Maximum Steam line Pressure psia	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.1		DECREASE IN REACTOR COOLANT TEMPERATURE									
15.1.1		Loss of Feedwater Heater	NOTE 5	NOTE 5	NOTE 5	NOTE 5	NOTE 5	0.14	Moderate	0	0 sec
15.1.2	15D.1.2-1	Feedwater Controller Failure (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	~350	1236	1236	1233	~125	A11 – 0.34 A10 – 0.35	Moderate	14	4 sec estimate
15.1.3	15D.1.3-1	Pressure Regulator Failure - Open	100	1051	1094	1043	100	A11 – 0.01 A10 – 0.01	Moderate	2	See Text
15.1.4		Inadvertent Opening of Safety or Relief Valves	See Text						Moderate		
15.1.6		RHR Shutdown Cooling Malfunction	See Text						Moderate		
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1		Pressure Regulator Failure – Closed	See Text						Moderate		
15.2.2		Generator Load Reject – Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.2	15D.2.2-1	Generator Load Reject- Without Bypass (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	~400	1246	1272	1260	~125	A11 – 0.44 A10 – 0.43	Moderate	14	10 sec estimate
15.2.3		Turbine Trip - Bypass Operable	See Text and Appendix 15E						Moderate		
15.2.3	15D.2.2-1	Turbine Trip – Without Bypass (100% Power, 108 Mlb_m/hr, Maximum Allowable Scram Time)	~400	1246	1272	1260	~125	A11 – 0.44 A10 – 0.43	Moderate	14	10 sec estimate
15.2.4		Inadvertent MSIV Closure	See Text and Appendix 15E						Moderate		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 (TYPICAL)

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psia	Maximum Vessel Pressure psia	Maximum Steam line Pressure psia	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.2.5		Loss of Condenser Vacuum	See Text and Appendix 15E						Moderate		
15.2.6		Loss of Auxiliary Power Transformer	See Text and Appendix 15E						Moderate		
15.2.6		Loss of All Grid Connections	See Text and Appendix 15E						Moderate		
15.2.7		Loss of All Feedwater Flow	See Text and Appendix 15E						Moderate		
15.2.8		Feedwater Piping Break	See Section 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		<u>DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE</u>									
15.3.1		Trip of One Recirculation Pump Motor	See Text and Appendix 15E						Moderate		
15.3.2		Trip of Both Recirculation Pump Motors	See Text and Appendix 15E						Moderate		
15.3.3	15.D.3.3-6	Seizure of One Recirculation Pump (Single Loop Operation)	67	959	973	958	67	0.33– A10 0.77 – A11	Limiting Fault		
15.3.4		Recirculation Pump Shaft Break	See Text						Limiting Fault		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 (TYPICAL)

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psia	Maximum Vessel Pressure psia	Maximum Steam line Pressure psia	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.4		REACTIVITY AND POWER ANOMALIES									
15.4.1.1		RWE – Refueling	See Text						Infrequent		
15.4.1.2		RWE – Startup	See Text						Infrequent		
15.4.2		RWE - At Power, 108 Mlbs/hr, Bypass Operable	See Text	Note 5	Note 5	Note 5	Note 5	0.21	Moderate		
15.4.3		Control Rod Maloperation	See Subsections 15.4.1 and 15.4.2								
15.4.4		Startup of Idle Recirculation Loop	See Text and Appendix 15E						Moderate		
15.4.5		Recirculation Flow Controller Failure[(3)]	See Text	Note 5	Note 5	Note 5	Note 5	0.17	Moderate		
15.4.7		Misplaced Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		
15.4.7		Rotated Bundle Accident	See Text	Note 5	Note 5	Note 5	Note 5	See Text	Infrequent		

TABLE 15D.0-1
RESULTS SUMMARY OF TRANSIENT EVENTS
UNIT 2 (TYPICAL)

Section	Figure	Description[1]	Maximum Neutron Flux,% of Rated	Maximum Dome Pressure psia	Maximum Vessel Pressure psia	Maximum Steam line Pressure psia	Maximum Core Average Surface Heat Flux, %	ΔCPR	Frequency Category	Number of Valves - 1st Blowdown	Duration of Blowdown
15.5		INCREASE IN REACTOR INVENTORY									
15.5.1		Inadvertent HPCI Pump Start @ limiting Power	See Text and Appendix 15E					A11 – 0.52 A10 – 0.51	Moderate		
15.5.3		BWR Transients That Increase Reactor Coolant Inventory	See Sections 15.1 and 15.2								

Notes

1. Unless otherwise stated, the plant initial condition listed in this table for transients is: 102% Power, 108 Mlbs/hr Flow, EOC-Reactor Pump Trip Operable, Bypass Operable, Realistic Scram Time.
2. Minimum MCPR operating limit for Single Loop Operation, see Text.
3. Recirculation Flow Controller Failure analyses are initiated from low power/low flow conditions.
4. Steam line pressure is at the turbine stop valve for events in which the turbine trips. For other transients the steam line pressure is assumed to be no higher than the reactor vessel dome pressure.
5. These Anticipated Operational Occurrences are analyzed as steady-state events.

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2

1.	Thermal Power Level, MWT Rated Value Analysis Value	3952 (100%) 4031 (102%)
2.	Steam Flow, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.624
3.	Maximum Core Flow, Mlbs/hr	108.0[(3)]
4.	Feedwater Flow Rate, Mlbs/hr (At 100% Power and 100 Mlbs/hr)	16.592
5.	Feedwater Temperature,°F (At 100% Power and 100 Mlbs/hr)	403.3
6.	Vessel Dome Pressure, psig (At 100% Power and 100 Mlbs/hr)	1035.7
7.	Vessel Core Pressure, psig at Channel Exit (At 100% Power and 100 Mlbs/hr)	1047.4
8.	Turbine Bypass Capacity, % Rated	21.5%
9.	Core Coolant Inlet Enthalpy, BTU/lb (At 100% Power and 100 Mlbs/hr)	523.6
10.	Turbine Inlet Pressure, psia	976.3
11.	Fuel Types	ATRIUM-10 ATRIUM-11
12.	Core Average Gap Conductance, BTU/hr-ft^2-°F	500 to 1700[(1)]
13.	Core Leakage Flow,%	~10%[(2)]
14.	Required MCPR Operating Limit	See Unit 2 COLR (FSAR section 16.3 – TRMs)
15.	MCPR Safety Limit	See Table 15D.0-3
16.	Doppler Coefficient	See Note 4

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS UNIT 2

17.	Void Coefficient	See Note 4
18.	Core Average Rated Void Fraction	See Note 4
19.	Scram Reactivity Analysis Data	See Note 4
20.	Control Rod Scram Times	Table 15D.0-5
21.	Jet Pump Ratio	2.1
22.	Safety Relief Valve Capacity (16 Valves) Percent of Rated Steam Flow	87%
23.	Relief Function Delay, sec	0.1
24.	Relief Function Response, sec	≤0.15
25-a.	Relief Mode Set Points for Safety/Relief Valves, psig	2 @ 1106 3 @ 1136 4 @ 1116 3 @ 1146 4 @ 1126
25-b.	Safety Mode Set Points for Safety/Relief Valves, psig	2 @ 1175 6 @ 1195 8 @ 1205
26.	Number of Valve Groups Simulated	3
27.	High Flux Trip, % Rated Analysis set point	122
28.	High Pressure Trip, Analysis Set Point, psig	1105
29.	Vessel Level Trips, Nominal Setpoints Inches Above(+), Below (-) Dryer Skirt Bottom, (See Note 5)	High Level (L8)≤ 54 (L4)≤ 30 Low Level (L3)≥ 13 Low Low Level (L2)≥ -38 Low Low Low Level (L1)≥-129
30.	APRM Thermal Trip, Analytical Set Point,% Rated	118

TABLE 15D.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS
UNIT 2

31.	Recirculation Pump Trip Delay, sec	0.175
32.	Recirculation Pump Trip Inertia for Analysis, lb_m-ft^2	16,800

NOTES

1. Gap conductance for reactor system behavior is determined for the fuel types within the core as a function of power and exposure. The hot bundle gap conductance is based on the fuel type that is expected to be limiting. It is also determined based on the initial hot bundle power and exposure.

2. Inlet enthalpy and leakage flow are determined for each initial condition analyzed.

3. Core flow shown is the maximum. It is varied depending on the initial conditions being analyzed.

4. The physics characteristics are based on initial conditions determined from a 3-D simulation of the core over a range of power, flow, and pressure conditions. For certain transient analyses this data is transferred and collapsed for use in a 1-D reactor core/system transient simulation model of SSES unit 2.

5. Analytical limits for level setpoints include drift and uncertainty allowances.

TABLE 15D.0-3

MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT (ALL FUEL)
UNIT 2

<table>
<tr><th>MCPRSL for
Two Loop
Operation</th><th>MCPRSL for
Single Loop
Operation</th></tr>
<tr><td colspan="2">Refer to TS 2.1.1.2</td></tr>
</table>

TABLE 15D.0-4

UNIT 2

MINIMUM MCPR REQUIREMENT FOR SINGLE LOOP OPERATION

MCPR Safety Limit	1.11
ATRIUM-10 Minimum MCPR Requirement	1.44
Typical ATRIUM-11 Minimum MCPR Requirement*	1.88

(Based on Analysis of Pump Seizure Accident in Single Loop Operation)

MINIMUM MCPR REQUIREMENT FOR TWO LOOP OPERATION

MCPR Safety Limit	1.08
ATRIUM-10 Minimum MCPR Requirement	1.30
Typical ATRIUM-11 Minimum MCPR Requirement*	1.37

(Based on Analysis of Pump Seizure Accident in Two Loop Operation)

* ATRIUM 11 results are typical and based on cycle-specific delta-CPR results assuming two-loop and single-loop MCPR safety limits of 1.08 and 1.11, respectively.

Table Rev. 65

TABLE 15D.0-5

AVERAGE SCRAM INSERTION TIMES
UNIT 2

Refer to COLR Section 5.0

TABLE 15D.1.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATING
UNIT 2

TIME, SECONDS	EVENT
0	Initiate a 100°F temperature reduction into the feedwater system.
2	Initial effect of unheated feedwater starts to raise core power level and steam flow, (Transport delay in feedwater piping is neglected).
≈40 (estimate)	APRM high neutron flux alarm sounds.
≈60 (estimate)	Reactor variable settle into new steady state, (below Scram trip point).
600 (estimate)	Operator begins to reduce power (reduce core flow and/or insert normal sequence control rods) to restore plant operation within normal power-flow conditions.

The above times are estimates. This event is a relatively slow transient and the analysis was performed as a series of steady-state calculations

TABLE 15D.1.2-1

SEQUENCE OF EVENTS FOR FEEDWATER CONTROLLER FAILURE, MAXIMUM DEMAND UNIT 2 (TYPICAL)

TIME, SECONDS	EVENT
0	Initiate simulated failure of 130% upper limit on feedwater flow.
13.79	L8 vessel level setpoint trips main turbine and feedwater pumps.
17.92	Reactor scram trip actuated from main turbine stop valve position switch.
17.95	Bypass Valves actuated
18.03	Recirculation pump trip (RPT) actuated by stop valve position switch.
~21	Activation of safety/relief valves.

Initial Conditions:

Power	= 100%
Flow	= 108 Mlbs/hr
Bypass	= Operable
RPT	= Operable
Scram Time	= Maximum Allowable
Exposure	= EOC

TABLE 15D.1.3-1

SEQUENCE OF EVENTS FOR PRESSURE REGULATOR FAILURE - OPEN
UNIT 2

TIME, SECONDS	EVENT
0	Initial conditions, maximum limit on steam flow to turbine.
0.2	Main turbine bypass valves full open
~7.5	Main steamline isolation trip occurs
~8.0	Initiation of scram trip signal, 0.06 seconds after the Main Steam Isolation Valves reach 85% open position.
~12.0	Pressure in the reactor reaches a minimum and starts to increase.
~12.5	MSIV's are fully closed
48 (est)	Relief valves at lowest setting start to cycle to remove decay heat.

Table Rev. 66

TABLE 15D.2.2-1

SEQUENCE OF EVENTS FOR GENERATOR LOAD REJECTION WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS UNIT 2 (TYPICAL)

TIME, SECONDS	EVENTS
~0	Turbine-generator detection of loss electrical load.
0	Generator lockout relays act to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0.001	Turbine control valves (TCV) close on GLR, (Generator Load Reject)
0.080	Initiate scram on TCV fast closure (Trip oil pressure-low).
~0.100	Turbine control valves closed.
0.185	EOC-Reactor Pump Trip initiated.
~2	Actuation of safety/relief valves

Initial Conditions

Power: 100% Flow: 108 Mlbs/hr
Bypass: Inoperable Scram: Maximum Allowable Time
RPT: Operable

TABLE 15D.3.3-1

PUMP SEIZURE ACCIDENT FROM TWO LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 2

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
~1.0	Jet Pump Diffuser Flow Reverses in Seized Loop
1.30 1.81	Minimum CPR (ATRIUM-10) Minimum CPR (ATRIUM-11)

Note: Figures include a 0.5 second null transient

TABLE 15D.3.3-2

PUMP SEIZURE ACCIDENT FROM SINGLE LOOP OPERATION
SEQUENCE OF EVENTS
UNIT 2

TIME, SEC	EVENT
0.0	Single Pump Seizure was Initiated
N/A	Jet Pump Diffuser Flow Reverses in Seized Loop
1.84 2.50	Minimum CPR (ATRIUM-10) Minimum CPR (ATRIUM-11)

TABLE 15D.4.2-1

SEQUENCE OF EVENTS - RWE IN POWER RANGE
UNIT 2

ELAPSED TIME	EVENT
0	Core is assumed to be at rated conditions.
0	Operator selects and withdraws the maximum worth control rod.
~1 sec	The total core power and the local power in the vicinity of the control rod increase.
~5 sec	The operator ignores warning and continues withdrawal.
~15 sec	The RBM system indicates excessive localized peaking.
~15 sec	The operator ignores warning and continues withdrawal.
~20 sec	The RBM system initiates a rod block inhibiting signal, credit is taken for this signal. Further control rod withdrawal is blocked.
~40 sec	Reactor core stabilizes at higher core power level.
~60 sec	Operator attempts to re-insert control rod to reduce core power level.
~80 sec	Core stabilizes at rated conditions.

TABLE 15D.4.5-1

SEQUENCE OF EVENTS FOR RECIRCULATION FLOW
CONTROLLER FAILURE
UNIT 2

TIME, SECONDS	EVENT
0	Master Flow Controller fails initiating a slow run-up of both reactor recirculation pumps
~220	Two relief valves open at 1120.7 psia.
~220	Reactor high flux scram (analytical setpoint, 122%).
~230	Two relief valves reseat at 1045.7 psia.

This sequence of events is for the event initiated from:

INITIAL CONDITIONS

Power	=	69%
Flow	=	60M lbs/hr
Bypass	=	Inoperable
Exposure	=	EOC

TABLE 15D.4.7-1

UNIT 2

SEQUENCE OF EVENTS FOR MISLOADED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in the wrong position.
2. Subsequently, the bundle intended for this position is placed in the position of the previous bundle.
3. During core verification procedure, error is not observed.
4. Plant is brought to full power operation without detecting misplaced bundle.
5. Plant continues to operate.

SEQUENCE OF EVENTS FOR ROTATED BUNDLE ACCIDENT

1. During core loading operation, bundle is placed in its proper location but rotated either 90° or 180° from its proper orientation.
2. During core verification procedure this error is not observed.
3. Plant is brought to full power operation without detecting rotated bundle.
4. Plant continues to operate.

TABLE 15D.4.9-1

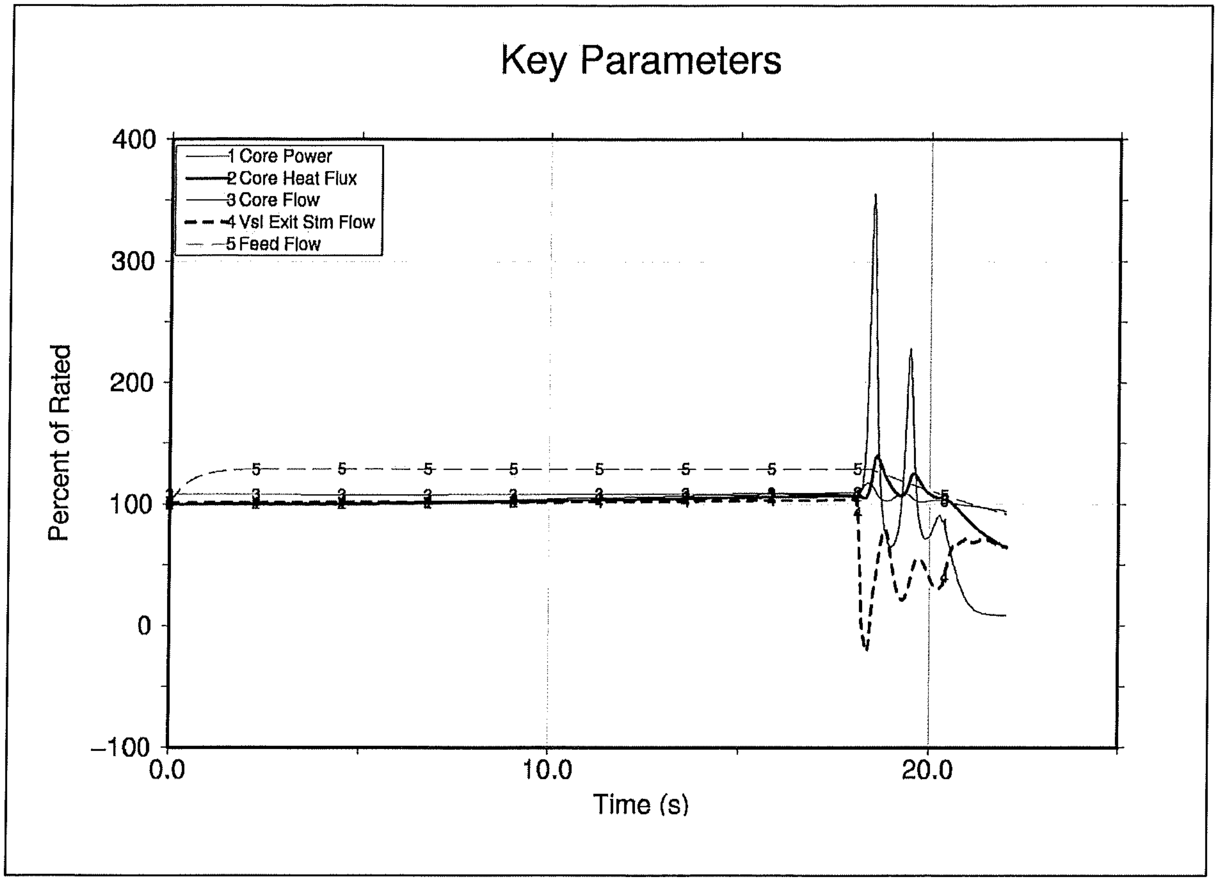
SEQUENCE OF EVENTS FOR CONTROL ROD DROP ACCIDENT UNIT 2

APPROXIMATE ELAPSED TIME	EVENT
	Reactor is operating at a rod density pattern of up to 50%.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod along with the other control rods assigned to the Bank Position Withdrawal Sequence (BPWS).
	Decoupled control rod sticks in the fully inserted or in an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
$<$1 second	Reactor goes on a positive period and initial power increase is terminated by the Doppler effect.
$<$1 second	APRM 120% power signal scrams the reactor
$<$5 seconds	Scram terminates the accident.

TABLE 15D.4.9-2

CONTROL ROD DROP ACCIDENT
UNIT 2 (TYPICAL)

Cycle Exposure, MWD/MTU	EOC
Control Rod Sequence	B
Rod Group	2
Dropped Rod Location	22-55
Dropped Rod Worth	< 12 mk
Number of Failed Fuel Rods	<2000
Peak deposited Enthalpy, cal/gm	<230

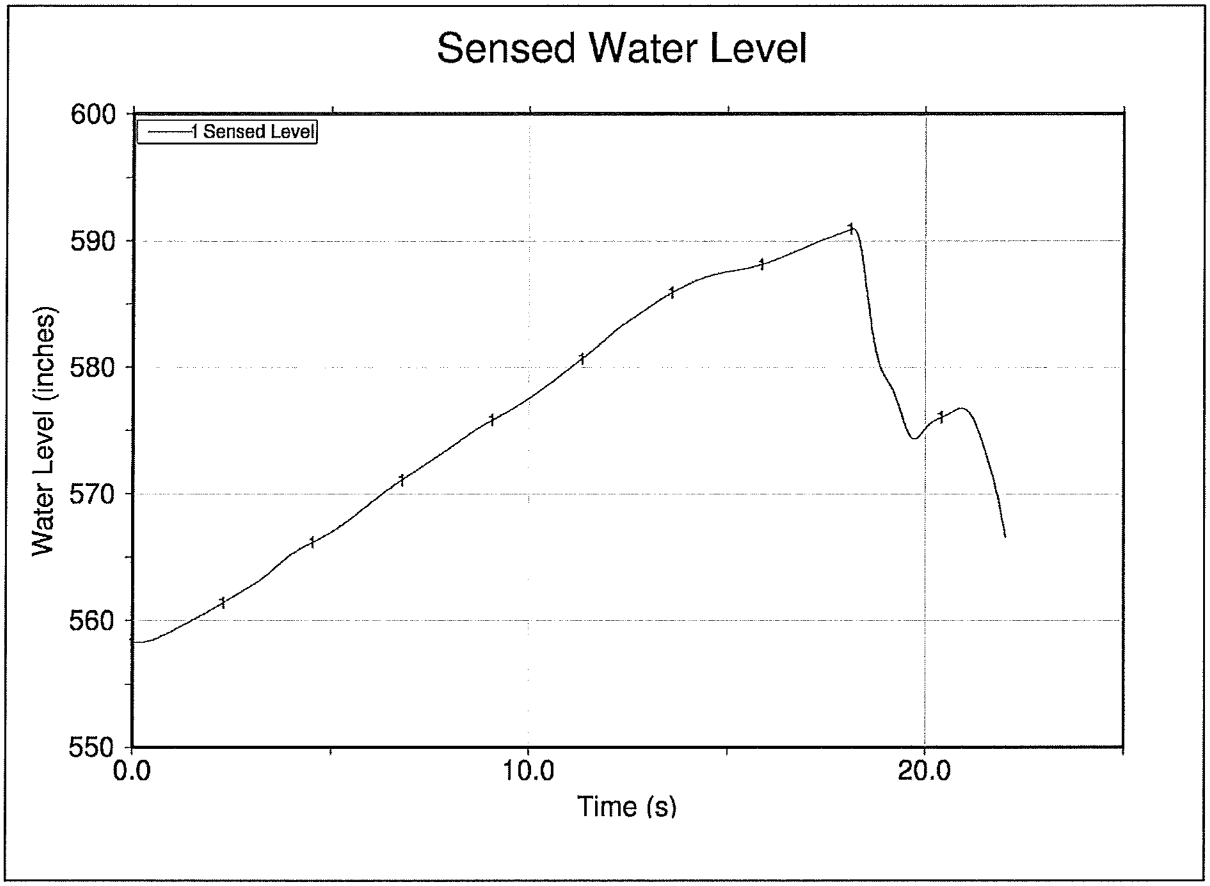


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP
(TYPICAL)

Figure 15D.1.2-1-1, Rev. 67

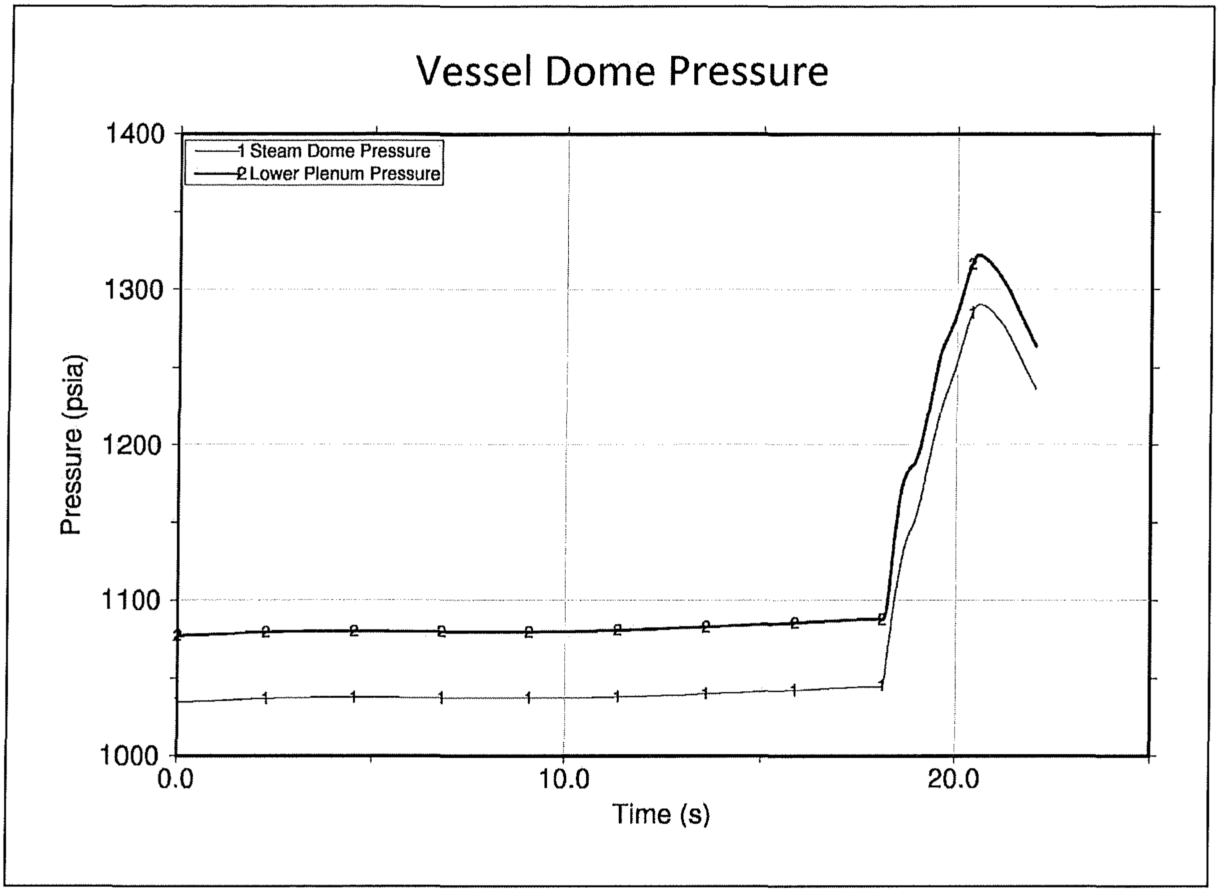


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP
(TYPICAL)

Figure 15D.1.2-1-2, Rev. 66

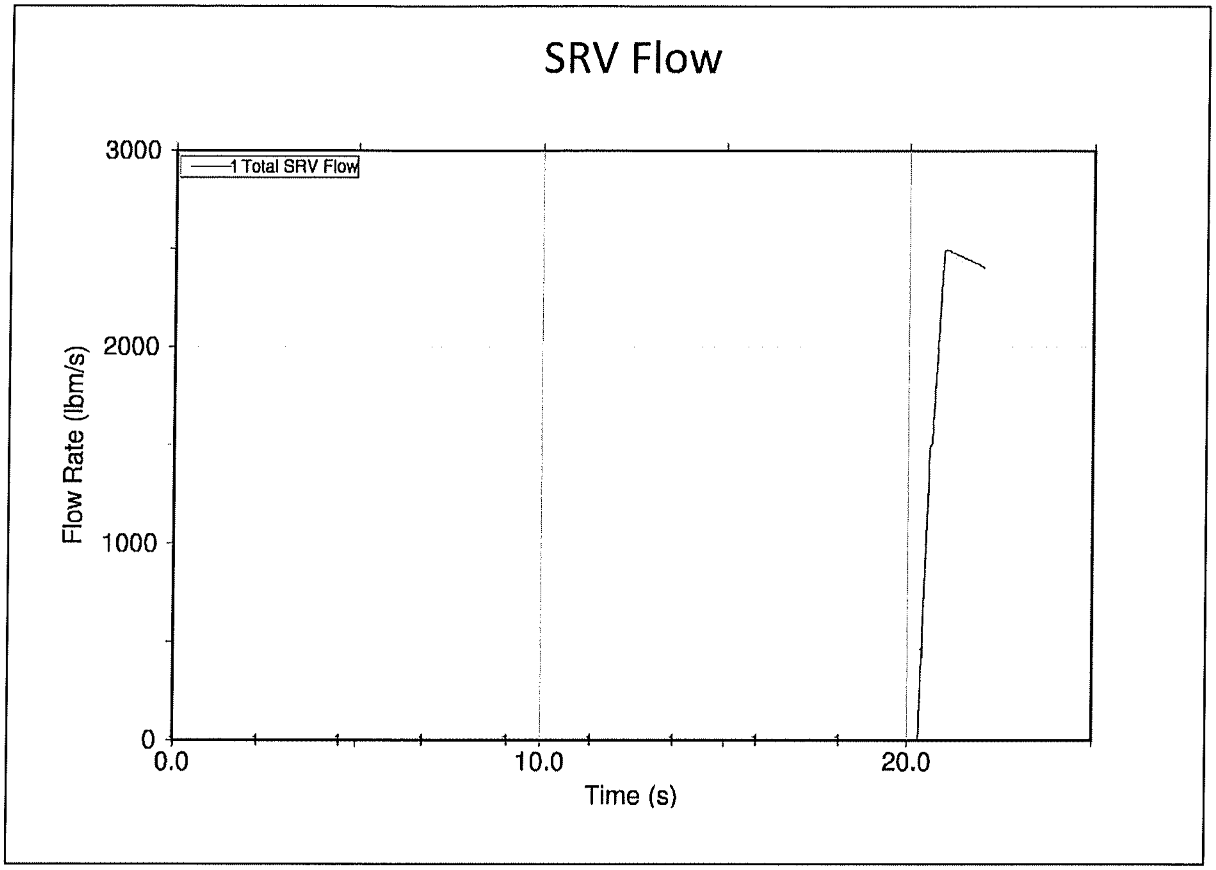


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP
(TYPICAL)

Figure 15D.1.2-1-4, Rev. 66

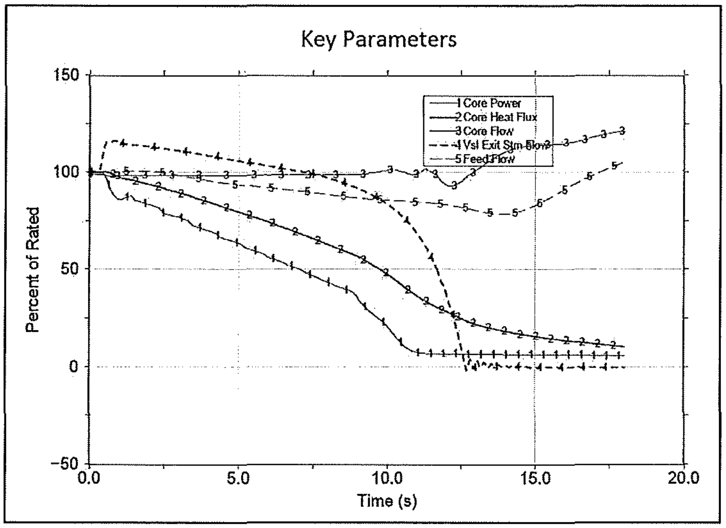


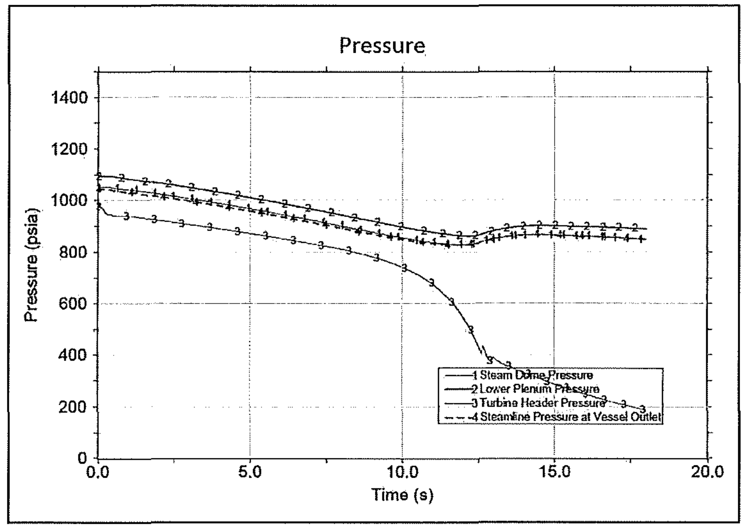
FSAR REV. 70

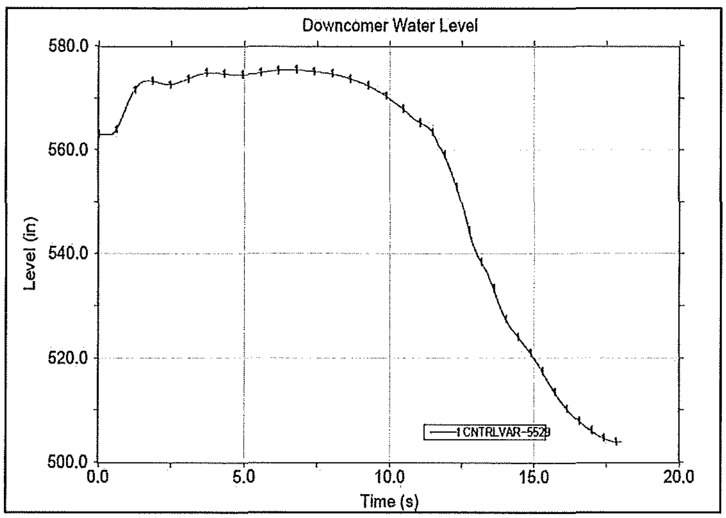
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

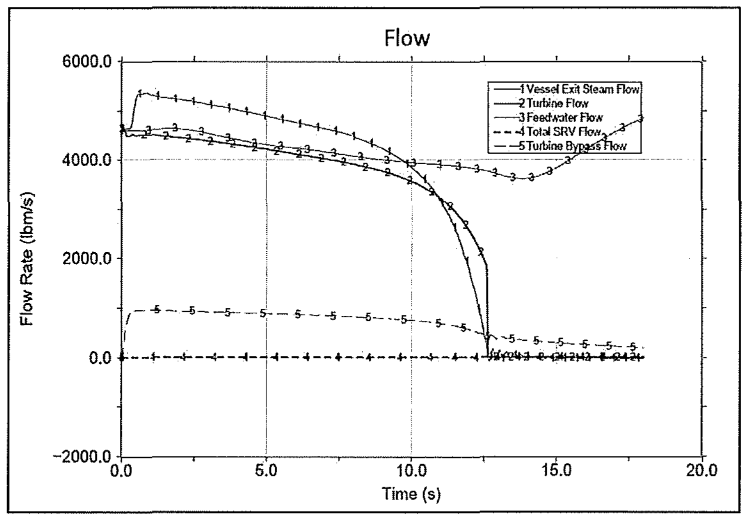
SUSQUEHANNA FEEDWATER CONTROLLER FAILURE,
MAXIMUM DEMAND,
WITH HIGH WATER LEVEL TRIP
(TYPICAL)

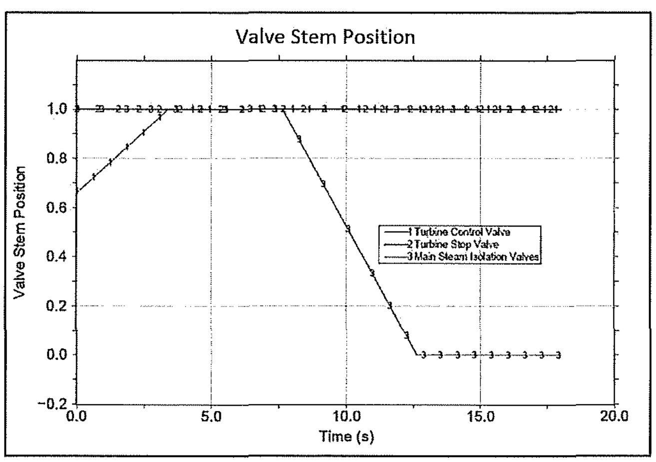
Figure 15D.1.2-1-5, Rev. 66









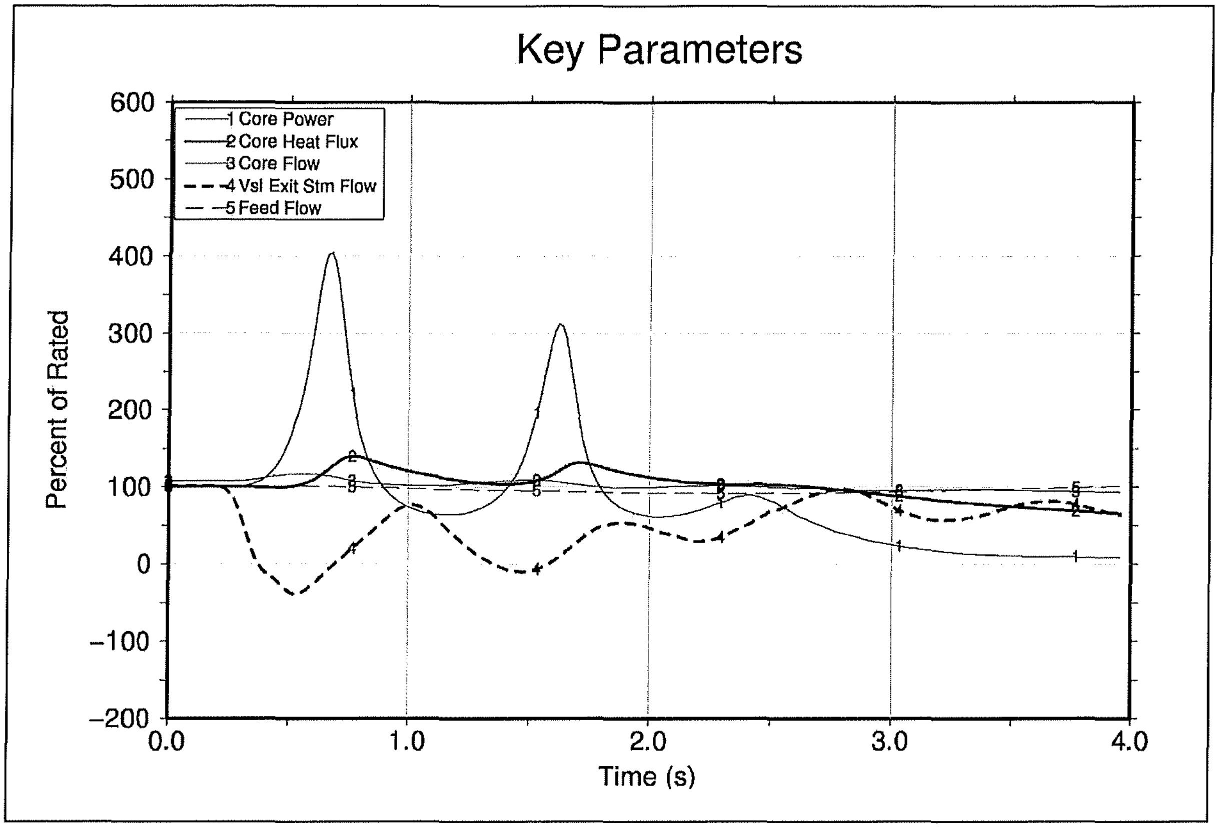


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PRESSURE REGULATOR FAILED OPEN
TYPICAL OF UNIT 2

FIGURE 15D.1.3-1, Rev 0

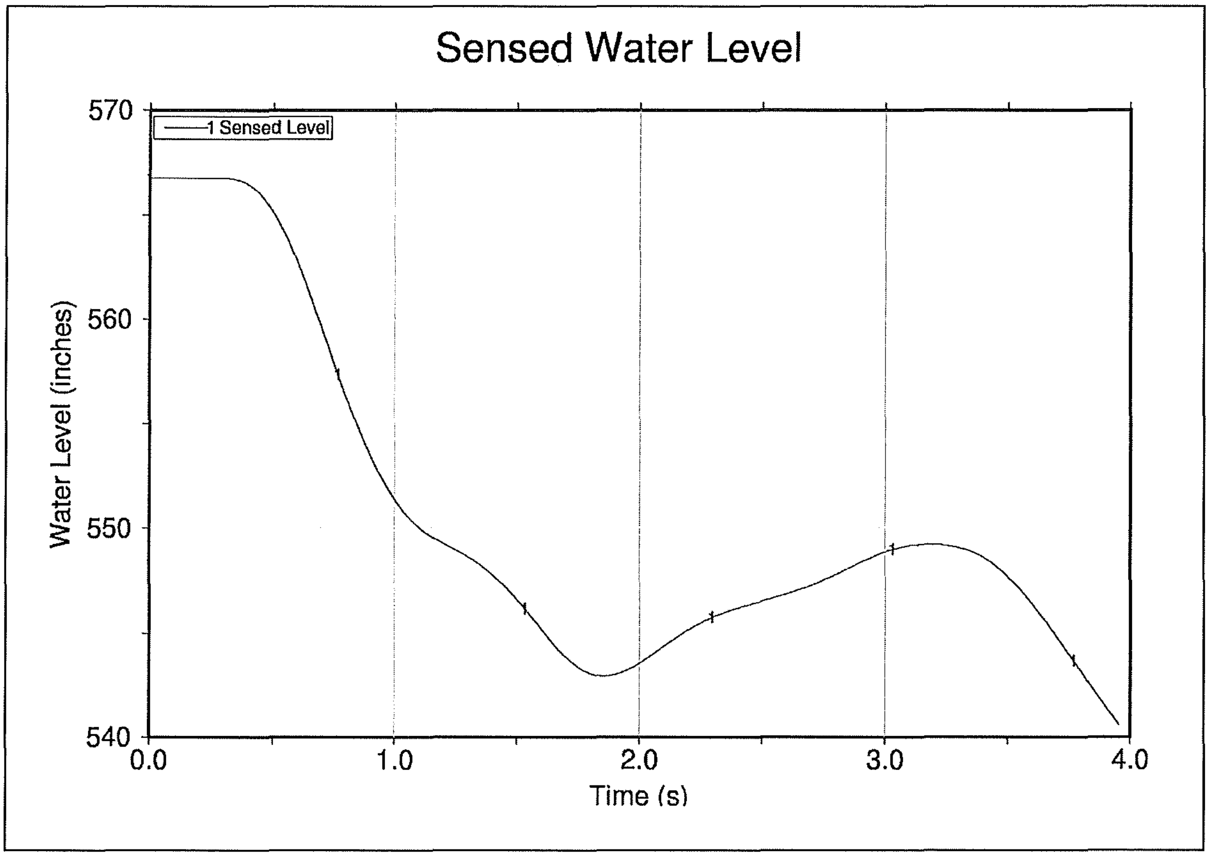


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS
AND TURBINE TRIP WITHOUT BYPASS
(TYPICAL)

Figure 15D.2.2-1-1, Rev. 67

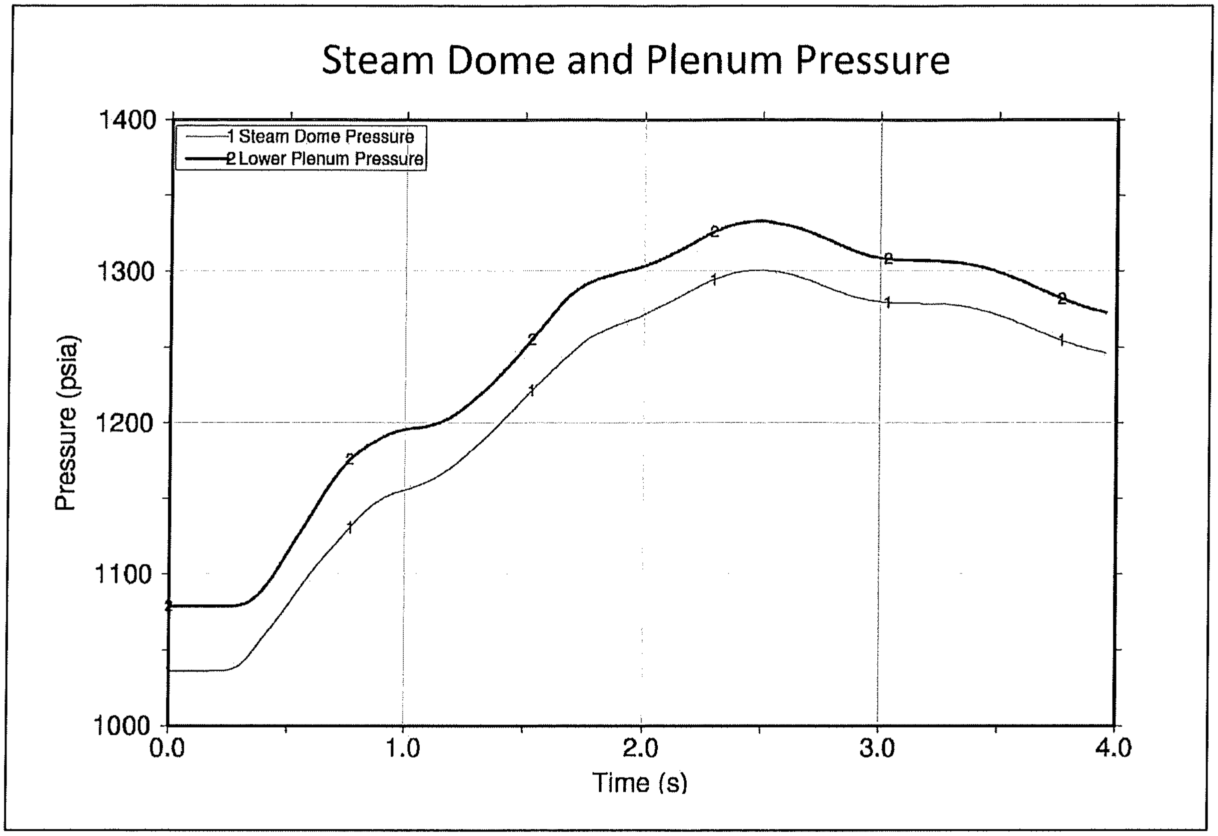


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS
AND TURBINE TRIP WITHOUT BYPASS
(TYPICAL)

Figure 15D.2.2-1-2, Rev. 66

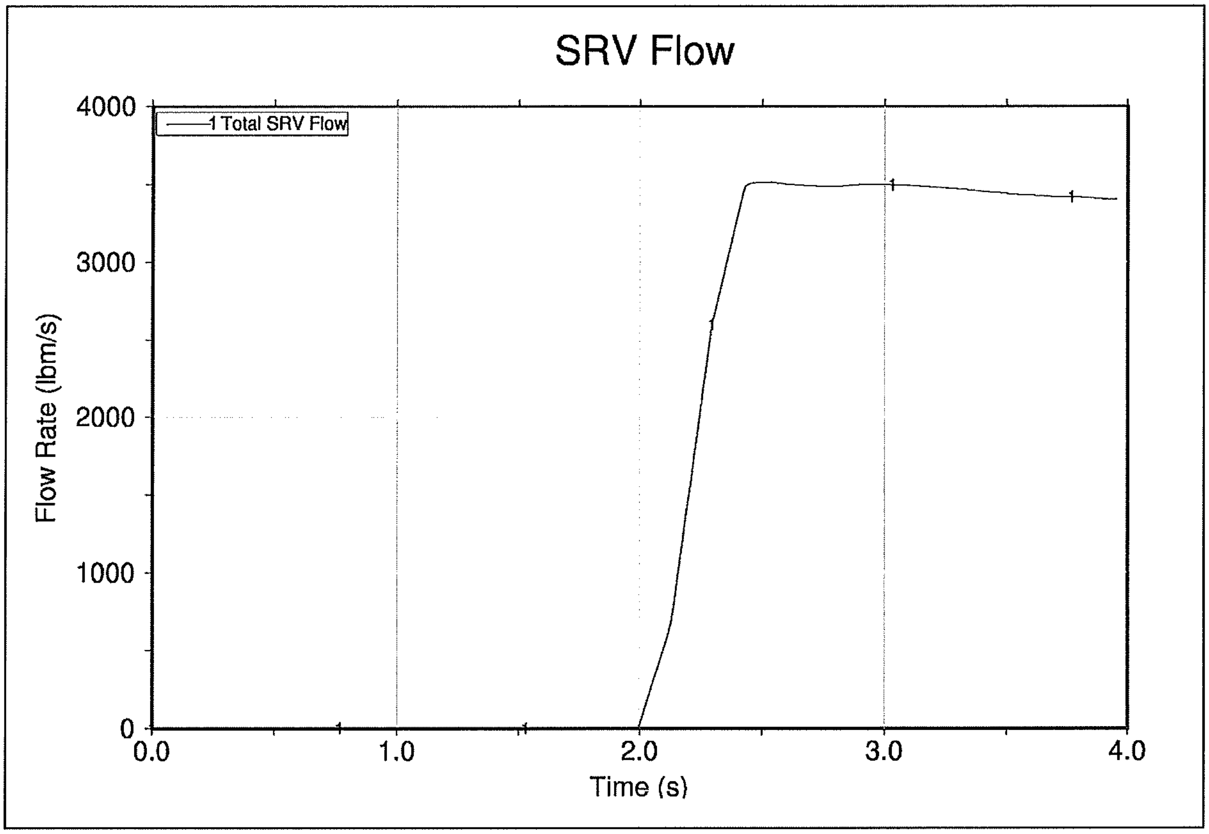


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS AND TURBINE TRIP WITHOUT BYPASS (TYPICAL)

Figure 15D.2.2-1-4, Rev. 66

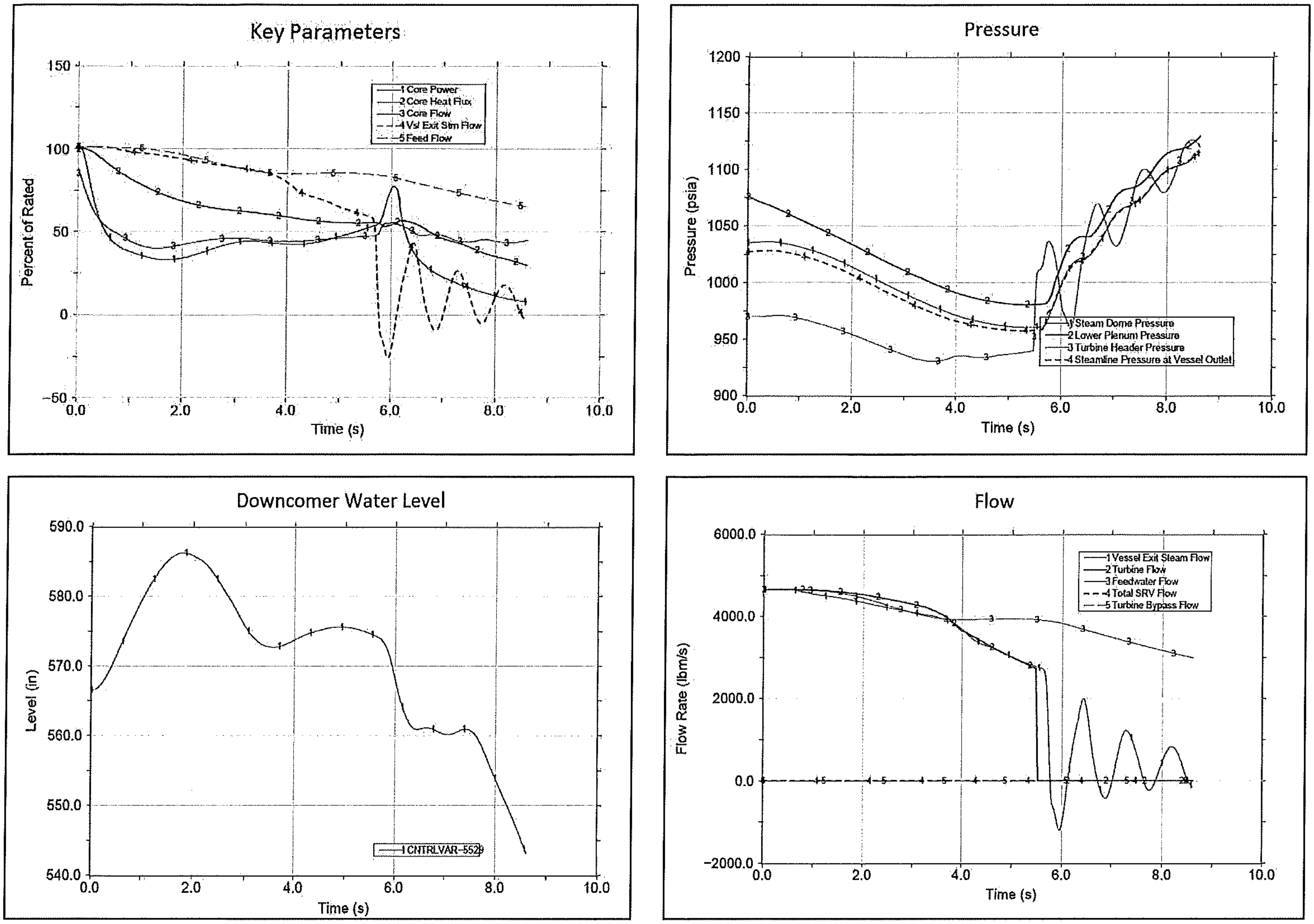


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNIT 2
FINAL SAFETY ANALYSIS REPORT

SUSQUEHANNA GENERATOR LOAD REJECT WITHOUT BYPASS
AND TURBINE TRIP WITHOUT BYPASS
(TYPICAL)

Figure 15D.2.2-1-5, Rev. 66

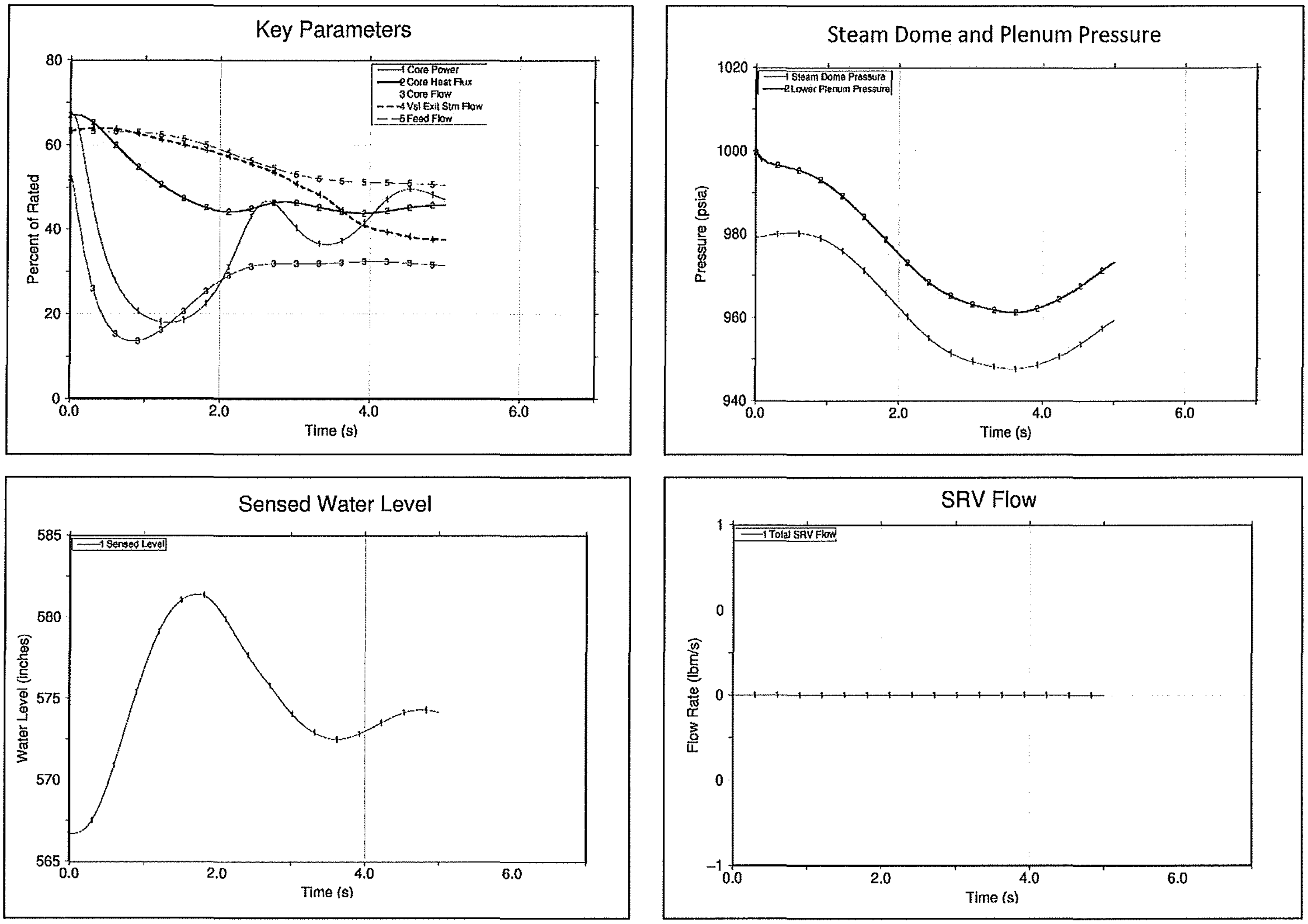


FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT TWO LOOP OPERATION
TYPICAL OF UNIT 2

FIGURE 15D.3.3-5, Rev 0



FSAR REV. 70

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PUMP SEIZURE ACCIDENT SINGLE LOOP OPERATION
TYPICAL OF UNIT 2

FIGURE 15D.3.3-6, Rev 0