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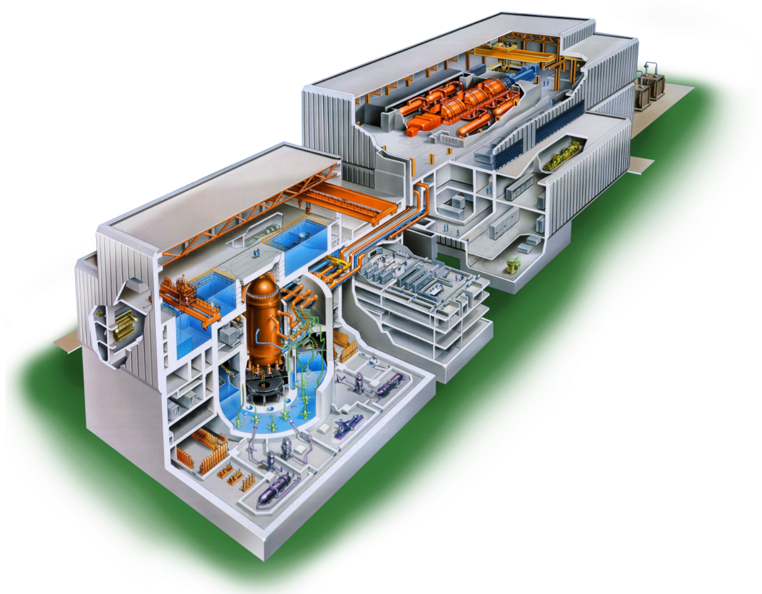

Chapter 15

Accident and Analysis

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15.0 Accident and Analysis

General

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events. The system response analysis is based upon the core loading shown in Figure 4.3-1 and is used to identify the limiting events for the ABWR. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-2, will not affect the sensitivities demonstrated by this study. Evaluation of these limiting events for each plant cycle will assure that the criteria in Appendix 4B are met.

GEH has developed a unique systematic approach to plant safety consistent with the GEH boiling water reactor technology base. The key to the GEH approach to plant safety is the Nuclear Safety Operational Analysis (NSOA). A generic NSOA has been developed for each of the recent GEH BWR product lines. It has then been modified to be compatible with the specific plant configuration being evaluated. Key inputs into the NSOA are derived from the applicable regulations and through industry codes and standards. The generic NSOA for ABWR is presented in Appendix 15A.

GEH has evaluated the entire spectrum of events in the NSOA to establish the most limiting or design basis events in a meaningful manner. It is the design basis events that are quantified in this chapter.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (AOOs) (e.g., loss of electrical load), off-design abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire Control Rod Drive System).

15.0.1 Nuclear Safety Operational Analysis

In Appendix 15A, Nuclear Safety Operational Analyses, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for all plant designs to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on event probability, which means that events more likely to occur are tested against more restrictive limits. This is consistent with industry practice and the applicable regulatory requirements.

The starting point for the NSOA is the establishment of unacceptable safety results. This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components which are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry codes and standards.

The focal point of the NSOA is the event analysis, in which all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required front-line safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation for AOOs, accidents, and plant capability demonstrations are evaluated. All events are analyzed until a stable condition is obtained. This assures that the event being evaluated does not have a characteristic for long-term consideration which is important.

In the event analysis all essential systems, operator actions, and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA, a complete and consistent set of safety actions (i.e., those required to prevent unacceptable results) has been developed. For transients and accidents, a single-failure-proof path to plant shutdown must be shown. The application of a single-failure criterion to these events is imposed as an additional measure of conservatism in the NSOA process.

15.0.2 Event Analytical Objective

The spectrum of postulated initiating events developed from the NSOA was divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence. The limiting events in each combination of category and frequency were evaluated using the core loading in Figure 4.3-1 to determine the limiting events. The plant safety analysis evaluates the ability of the plant to operate without unacceptable safety results within regulatory guidelines. This objective is met by satisfying the criteria in Appendix 4B.

15.0.3 Analytical Categories

Each event analyzed is assigned to one of eight categories listed in Chapter 15 of Regulatory Guide 1.70.

15.0.4 Event Evaluation

15.0.4.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed in the sensitivity study are described within the categories designated in Subsection 15.0.3. The frequency of occurrence of each event is summarized based upon the NSOA and currently available operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of three frequency groups defined in Regulatory Guide 1.70.

15.0.4.2 Identified Results

Events analyzed for each plant must meet the criteria in Appendix 4B.

15.0.4.3 Sequence of Events and Systems Operations

Each transient or accident evaluated in the sensitivity study is discussed and evaluated in terms of:

(1) A step-by-step sequence of events from initiation to final stabilized condition

(2) The extent to which normally operating plant instrumentation controls are assumed to function

(3) The extent to which the plant and reactor protection systems are required to function

(4) The credit taken for the functioning of normally operating plant systems

(5) The operation of engineered safety systems that is required

This sequence of events is supported by the NSOA for the transient or accident. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA.

15.0.4.4 Analysis Basis

The sensitivity study results given in this chapter are based upon the core loading given in Figure 4.3-1. These sensitivities are valid for other fuel designs and core loadings.

15.0.4.4.1 Evaluation Models

The computer codes used in the analysis of the transients and accidents in this chapter are shown in Table 15.0-1-A. These models have been approved by the USNRC.

15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed for the sensitivity analysis documented within this section have values for input parameters and initial conditions as specified in Table 15.0-1. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

The normal maximum allowable reactor operating condition is the 100%-power/100%-flow condition. The maximum power measurement uncertainty is usually ~2%. Therefore, the sensitivity analyses are based on 102% power level. The transient results at this condition are more severe than that at rated condition.

The analytical values for some system characteristics, like SRV delay/stroke time, reactor internal pump coastdown time constant, etc., bound the design specification for that system. These values will be checked during startup tests.

All setpoints for the protection system assumed in the analyses are conservative, which includes instrument uncertainty, calibration error and instrument drift. The nominal and allowable values for these setpoints, (see Technical Specifications) assume that the setpoints will not exceed what are assumed in the analyses.

In conclusion, the input parameters and initial conditions (including uncertainties) used in the sensitivity study are conservative values and bound the operating band.

15.0.4.4.3 Initial Power/Flow Operating Constraints

The power/flow map used for the system response analysis is shown in Figure 15.0-1. The analyses basis for most of the sensitivity analyses is 102% thermal power at rated core flow (100%). Rated core flow can be achieved with either nine or ten pumps in operation. This operating point is the apex of the operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins. Referring to Figure 15.0-1, the apex of the bounded power/flow map is point A, the upper bound is the design flow control line (102% rod line A-D), the lower bound is the zero power line H'-J, the right bound is the maximum flow line A'-H', and the left bound is the natural circulation line D-J.

The power/flow map (A-D-J-H'-A') represents the operational region covered by abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map (e.g., the moisture carryover protection region, the licensed power limit and other restrictions based on pressure and thermal margin criteria) must be observed. See Subsection 4.4.3.3 for power/flow map operating instructions. The upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the GETAB operating limit.

Certain localized events are evaluated at other than the above-mentioned conditions. These conditions are discussed pertinent to the appropriate event.

The power/flow operating map for a plant may differ from that used in the system response analysis given in this chapter. Differences in the map will not change the designation of limiting events. The operating map used at a plant will be provided by the COL applicant to the USNRC for information (Subsection 4.4.2.1).

15.0.4.5 Evaluation of Results

The results of the system response analyses are presented in Table 15.0-2. Based on these results, the limiting events have been identified. Reasons why the other events are not limiting are given in the event documentation. The limiting events which establish CPR operating limit include:

(1) **Limiting Pressurization Events:** Inadvertent closure of one turbine control valve and generator load rejection with all bypass valve failure.

(2) **Limiting Decrease in Core Coolant Temperature Events:** Feedwater Controller Failure—Maximum Demand

For the core loading in Figure 4.3-1, the resulting initial core MCPR operating limit is 1.17. The operating limit based on the plant loading pattern will be provided by the COL applicant to the USNRC for information (Subsection 15.0.5.2 for COL license information requirement).

Results of the transient analyses for individual plant reference core loading patterns will differ from the results shown in this chapter. However, the relative results between core associated events do not change. Therefore, only the results of the identified limiting events given in Table 15.0-4 will be provided by the COL applicant to the USNRC for information (Subsection 15.0.5.1).

15.0.4.5.1 Effect of Single Failures and Operator Errors

The effect of a single equipment failure or malfunction or operator error is provided in Appendix 15A.

15.0.4.5.2 Analysis Uncertainties

The analysis uncertainties meet the criteria in Appendix 4B.

A summary of applicable accidents is provided in Table 15.0-5, which compares GEH calculated amount of failed fuel to that used in worst-case radiological calculations for the core shown in Figure 4.3-1. Radiological calculations for a plant initial core will be provided by the utility to the USNRC for information (see Subsection 15.0.5 for COL license information requirements).

15.0.4.5.3 Barrier Performance

The significant areas of interest for internal pressure damage are the high-pressure portions of the reactor coolant pressure boundary (i.e., the reactor vessel and the high pressure pipelines attached to the reactor vessel). The plant shall meet the criteria in Appendix 4B.

15.0.4.5.4 Radiological Consequences

This chapter describes the consequences of radioactivity release for the core loading in Figure 4.3-1 during three types of events: (1) incidents of moderate frequency (anticipated operational occurrences); (2) infrequent incidents (abnormal operational occurrences); and (3) limiting faults (design basis accidents). For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

15.0.5 COL License Information

15.0.5.1 Anticipated Operational Occurrences (AOO)

The results of the events identified in Subsection 15.0.4.5 for plant core loading will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

15.0.5.2 Operating Limits

The operating limit resulting from the analyses normally provided in this subsection will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

15.0.5.3 Design Basis Accidents

Results of the design basis accidents, including radiological consequences, will be provided by the COL applicant referencing the ABWR design to the USNRC for information.

Table 15.0-1 Input Parameters and Initial Conditions for System Response Analysis Transients

1.	Thermal Power Level (MWt)	
	Warranted Value	3926
	Analysis Value	4005
2.	Steam Flow (kg/h)	
	Warranted Value	7.64×10^6
	Analysis Value	7.84×10^6
3.	Core Flow (kg/h)	
	Rated	52.2×10^6
	Maximum	58.0×10^6
4.	Feedwater Flow Rate (kg/s)	
	Warranted Value	2122
	Analysis Value	2179
5.	Feedwater Temperature (°C)	217
6.	Vessel Dome Pressure (MPaG)	7.17
7.	Vessel Core Pressure (MPaG)	7.23
8.	Turbine Bypass Capacity (% NBR)	33
9.	Core Coolant Inlet Enthalpy (kJ/g)	1.23
10.	Turbine Inlet Pressure (MPaA)	6.85
11.	Fuel Lattice	N
12.	Core Leakage Flow (%)	11.67
13.	Required MCPR Operating Limit	1.17
14.	MCPR Safety Limit	1.07
15.	Doppler Coefficient (–)¢/°C	
	Analysis Data for Power Increase Events (REDY only)*	0.429
	Analysis Data for Power Decrease Events (REDY only)*	0.180
16.	Void Coefficient (–)¢/% Rated Voids	
	Analysis Data for Power Increase Events (REDY only)*	11.6
	Analysis Data for Power Decrease Events (REDY only)*	2.5
17.	Core Average Rated Void Fraction (%) (REDY only)*	43.4
18.	Scram Reactivity, $\$\Delta k$ Analysis Data (REDY only)*	Table 15.0-5
19.	Control Rod Drive Position versus time	Table 15.0-6

Table 15.0-1 Input Parameters and Initial Conditions for System Response Analysis Transients (Continued)

20.	Nuclear characteristics used in ODYN simulations	EOEC†
21.	Number of Reactor Internal Pumps	10
22.	Safety/Relief Valve Capacity (%NBR) at 7.89 MPaG	91.3
	Quantity Installed	18
23.	Relief Function Delay (s)	0.45
24.	Relief Function Opening Time (s)	0.15
25.	Safety Function Delay (s)	0.0 ‡
26.	Safety Function Opening Time (s)	0.3
27.	Setpoints for Safety/Relief Valves	
	Safety Function (MPaG)	8.12, 8.19, 8.26, 8.33, 8.39
	Relief Function (MPaG)	7.89, 7.96, 8.03, 8.10, 8.17, 8.24
28.	Number of Valve Groupings Simulated	
	Safety Function (No.)	5
	Relief Function (No.)	6
29.	S/R Valve Reclosure Setpoint — Both Modes (% of setpoint)	
	— Maximum Safety Limit (used in analysis)	98
	— Minimum Operational Limit	93
30.	High Flux Trip (% NBR)	
	Analysis Setpoint (125 x 1.02)	127.5
31.	High Pressure Scram Setpoint (MPaG)	7.62
32.	Vessel level Trips (m above bottom of separator skirt bottom)	
	Level 8—(L8) (m)	1.73
	Level 4—(L4) (m)	1.08
	Level 3—(L3) (m)	0.57
	Level 2—(L2) (m)	–0.75
33.	APRM Simulated Thermal Power Trip Scram % NBR	
	Analysis Setpoint (115 x 1.02)	117.3
	Time Constant (s)	7
34.	Reactor Internal Pump Trip Delay (s)	0.16
35.	Recirculation Pump Inertia for Analysis (MPa) ƒ	
	— Trip of RIPs for mitigation	≤ 2.60
	— Accident	≥ 1.72
36.	Total Steamline Volume (m^3)	113.2

Table 15.0-1 Input Parameters and Initial Conditions for System Response Analysis Transients (Continued)

37.	Set pressure of Recirculation pump trip (MPaG)	7.76

* For transients simulated on the ODYN model, this input is calculated by ODYN.

† EOEC = End of Equilibrium Cycle

‡ This is a programming convenience number.

ƒ The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_o n}{g T_o}$$

where t = Inertia time constant (s)
J_o = Pump motor inertia (kg•m^2)
n = Pump speed (rps)
g = Gravitational constant (m/s^2)
T_o = Pump shaft torque (kg•m)

Table 15.0-1a Computer Codes Used in the Analysis of Transients and Accidents

Sub Section I.D.	Figure I.D.	Event	Analysis Code
15.1		Decrease in core coolant temperature	
15.1.1	15.1-1	Loss of Feedwater heating	PANACEA
15.1.2	15.1-2	Runout of one feedwater pump	ODYNA
15.1.2	15.1-3	Runout of two feedwater pumps	ODYNA
15.1.3	15.1-4	Opening of one Bypass Valve	REDYA
15.1.3	15.1-5	Opening of all Control and Bypass Valves	REDYA
15.2		Increase in Reactor Pressure	
15.2.1	15.2-1	Closure of One Turbine Control Valve	ODYNA
15.2.1	15.2-2	Pres. Regulator Downscale Fail.	ODYNA
15.2.2	15.2-3	Generator Load Rejection, Bypass on	ODYNA
15.2.2	15.2-4	Generator Load Rejection, Failure of One Bypass Valve	ODYNA
15.2.2	15.2-5	Generator Load Rejection with Bypass Off	ODYNA
15.2.3	15.2-6	Turbine Trip Bypass On	ODYNA
15.2.3	15.2-7	Turbine Trip w/Failure of One Bypass Valve	ODYNA
15.2.3	15.2-8	Turbine Trip Bypass Off	ODYNA
15.2.4	15.2-9	Inadvertent MSIV Closure	ODYNA
15.2.5	15.2-10	Loss of Condenser Vacuum	ODYNA
15.2.6	15.2-11	Loss of Aux. Power Transformer	ODYNA
15.2.6	15.2-12	Loss of Aux. Power Transformer and One Startup Transformer	ODYNA
15.2.7	15.2-13	Loss of All Feedwater Flow	REDYA
15.3		Decrease in Reactor Coolant System Flow Rate	
15.3.1	15.3-1	Trip of Three Reactor Internal Pumps	REDYA
15.3.1	15.3-2	Trip of All Reactor Internal Pumps	REDYA
15.3.2	15.3-3	Fast Runback of One Reactor Internal Pump	REDYA
15.3.2	15.3-4	Fast Runback of All Reactor Internal Pumps	REDYA
15.3.3	15.3-5	Seizure of One Reactor Internal Pump	REDYA
15.4		Reactivity and Power Distribution Anomalies	
15.4.5	15.4-1	Fast Runout of One Reactor Internal Pump	REDYA

Table 15.0-1a Computer Codes Used in the Analysis of Transients and Accidents

Sub Section I.D.	Figure I.D.	Event	Analysis Code
15.4.5	15.4-2	Fast Runout of All Reactor Internal Pumps	REDYA
15.5		Increase in Reactor Coolant Inventory	
15.5.1	15.5-1	Inadvertent HPCF Startup	REDYA
15.6		Decrease in Reactor Coolant Inventory	
15.6.5		Steam System Pipe Break Outside Containment	See Response to Question 440.92
15.6.5		LOCA Within RCPB	See Response to Question 440.92
15.6.6		Feedwater Line Break	See Response to Question 440.92

Table 15.0-2 Results Summary of System Response Analysis Transient Events

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max. Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blow-down	Duration of Blow-down (s)
15.1		Decrease in core coolant temperature									
15.1.1		Loss of Feedwater Heating	112.8	7.17	7.44	7.02	112.8	0.07	†	0	0
15.1.2	15.1-2	Runout of One Feedwater Pump	104.5	7.18	7.43	7.03	101.8	0.06	†	0	0
15.1.2	15.1-3	Feedwater Controller Failure—Maximum Demand	139.0	8.17	8.33	8.12	105.9	0.10	†‡	10	6
15.1.3	15.1-4	Opening of One Bypass Valve	102.1	7.17	7.41	7.02	100.0	*f*	†	0	0
15.1.3	15.1-5	Opening of All Control and Bypass Valves	102.0	7.88	7.95	7.86	100.0	*f*	†‡	0	0
15.1.4		Inadvertent Opening of One SRV				SEE	TEXT				
15.1.6		Inadvertent RHR Shutdown Cooling				SEE	TEXT				
15.2		Increase in Reactor Pressure									
15.2.1	15.2-1a	Fast Closure of One Turbine Control Valve	129.4	7.36	7.61	7.23	103.6	0.10	†	0	0

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max. Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blow-down	Duration of Blow-down (s)
15.2.1	15.2-1b	Slow Closure of One Turbine Control Valve	110.3	7.33	7.58	7.19	103.3	0.09			
15.2.1	15.2-2	Pressure Regulator Downscale Fails	154.8	8.41	8.57	8.35	103.0	**	†	18	6
15.2.2	15.2-3	Generator Load Rejection, Bypass On	148.1	8.16	8.30	8.11	100.2	0.06	†	10	5
15.2.2	15.2-4	Generator Load Rejection, Failure of One Bypass Valve	155.3	8.26	8.41	8.20	100.5	0.07	†‡	14	5
15.2.2	15.2-5	Generator Load Rejection with Failure of All Bypass Valves	184.6	8.44	8.60	8.39	102.3	0.10	†‡	18	6
15.2.3	15.2-6	Turbine Trip Bypass—On	122.1	8.14	8.30	8.10	100.0	0.05	†	10	5
15.2.3	15.2-7	Turbine Trip w/Failure of One Bypass Valve	131.9	8.25	8.39	8.18	100.0	0.05	†‡	14	5
15.2.3	15.2-8	Turbine Trip with Failure of All Bypass Valves	158.6	8.44	8.60	8.37	100.6	0.08	†‡	18	6

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max. Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blow-down	Duration of Blow-down (s)
15.2.4	15.2-9	Inadvertent MSIV Closure	102.1	8.30	8.47	8.25	100.1	*f*	†	18	5
15.2.5	15.2-10	Loss of Condenser Vacuum	122.3	8.14	8.30	8.10	100.0	*f*	†	10	5
15.2.6	15.2-11	Loss of AC Power	113.2	8.13	8.28	8.11	100.0	0.05	†	10	5
15.2.7	15.2-12	Loss of All Feedwater Flow	102.0	7.17	7.42	7.02	100.1	*f*	†	0	0
15.2.8		Feedwater Piping Break			SEE	TEXT					
15.2.9		Failure of RHR Shutdown Cooling			SEE	TEXT					
15.3		Decrease in Reactor Coolant System Flow Rate									
15.3.1	15.3-1	Trip of Three Reactor Internal Pumps	102.0	7.19	7.45	7.03	100.1	0.04	†	0	0
15.3.1	15.3-2	Trip of All Reactor Internal Pumps	102.0	8.16	8.25	8.11	100.2	**	*†		
15.3.2	15.3-3	Fast Runback of One Reactor Internal Pump	102.0	7.16	7.44	7.02	100.0	*f*	†	0	0
15.3.2	15.3-4	Fast Runback of All Reactor Internal Pumps	102.0	7.17	7.45	7.02	100.0	*f*	†‡	0	0

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max. Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blow-down	Duration of Blow-down (s)
15.3.3	15.3-5	Seizure of One Reactor Internal Pump	102.0	7.17	7.44	7.02	100.0	ƒ	††	0	0
15.3.4		One Pump Shaft Break			SEE	TEXT					
15.4		Reactivity and Power Distribution Anomalies									
15.4.1.1		RWE-Refueling			SEE	TEXT					
15.4.1.2		RWE-Startup			SEE	TEXT					
15.4.2		RWE at Power			SEE	TEXT					
15.4.3		Control Rod Misoperation			SEE	TEXT					
15.4.4		Abnormal Startup of One Reactor Internal Pump			SEE	TEXT					
15.4.5	15.4-2	Fast Runout of One Reactor Internal Pump	89.8	6.97	7.09	6.92	116.1	‡‡	†	0	0
15.4.5	15.4-3	Fast Runout of All Reactor Internal Pumps	135.0	7.11	7.33	7.01	168.5	ƒƒ	‡	0	0
15.4.7		Mislocated Bundle Accident				SEE	TEXT				

Table 15.0-2 Results Summary of System Response Analysis Transient Events (Continued)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (MPaG)	Max. Vessel Bottom Pressure (MPaG)	Max. Steamline Pressure (MPaG)	Max. Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blow-down	Duration of Blow-down (s)
15.4.8		Misoriented Fuel Bundle Accident						0.09	a+		
15.4.9		Rod Ejection Accident				SEE	TEXT	0.09	a+		
15.4.10		Control Rod Drop Accident				SEE	TEXT				
15.5		Increase in Reactor Coolant Inventory									
15.5.1	15.5-1	Inadvertent HPCF Startup	102.0	7.17	7.41	7.02	100.0	*ƒ*	†	0	0

* Frequency definition is discussed in Subsection 15.0.4.1.

† Moderate Frequency

‡ This event should be classified as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

ƒ Not limiting (see Subsection 15.0.4.5).

** CPR Criterion does not apply. PCT <593.3°C

†† Limiting Fault

‡‡ Transients initiated from low power.

a+ Moderate Frequency. This event should be classified as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

Table 15.0-3 Summary of Accidents

		Failed Fuel Rods	
Subsection I.D.	Title	GEH Calculated Value	NRC Worst-Case Assumption
15.2.1	Pressure Regulator Downscale Failure	None	<0.2%
15.3.1	Trip of All Reactor Internal Pumps	None	<0.2%
15.3.3	Seizure of one Reactor Internal Pump	None	None
15.3.4	Reactor Internal Pump Shaft Break	None	None
15.6.2	Instrument Line Break	None	None
15.6.4	Steam System Pipe Break Outside Containment	None	None
15.6.5	LOCA Within RCPB	None	100%
15.6.6	Feedwater Line Break	None	None
15.7.1.1	Main Condenser Gas Treatment System Failure	N/A	N/A
15.7.3	Liquid Radwaste Tank Failure	N/A	N/A
15.7.4	Fuel-Handling Accident	<125	125
15.7.5	Cask Drop Accident	None	All Rods in Cask

Table 15.0-4 Core-Wide Transient Analysis Results To Be Provided for Different Core Design

Transient	Max. Neutron Flux (%NBR)	Max. Core Average Surface Heat Flux (%NBR)	Δ CPR	Figure
Closure of One Turbine Control Valve	X	X	X	X
Load Rejection with all Bypass Valves Failure	X	X	X	X
Feedwater Controller Failure—Maximum Demand	X	X	X	X

Table 15.0-5 Scram Reactivity Curves

Control Fraction	Scram Reactivity BOC 1*	EOEC*
0.0	0.0	0.0
0.05	– 0.235	– 0.082
0.10	– 0.473	– 0.170
0.20	– 0.945	– 0.360
0.30	– 1.611	– 0.656
0.40	– 2.576	– 1.122
0.50	– 4.295	– 1.875
0.60	– 7.160	– 3.366
0.70	–13.60	– 6.728
0.80	–25.44	–14.08
0.90	–33.44	–27.05
1.00	–34.56	–31.20

* BOC = Beginning of Cycle 1
EOEC = End of Equilibrium Cycle

Table 15.0-6 ABWR FMCRD Scram Time

Rod Insertion (%)	Scram Time (seconds) (Including Solenoid De-energization) Used in Analysis
10	0.46
40	1.208
60	1.727
100	3.719

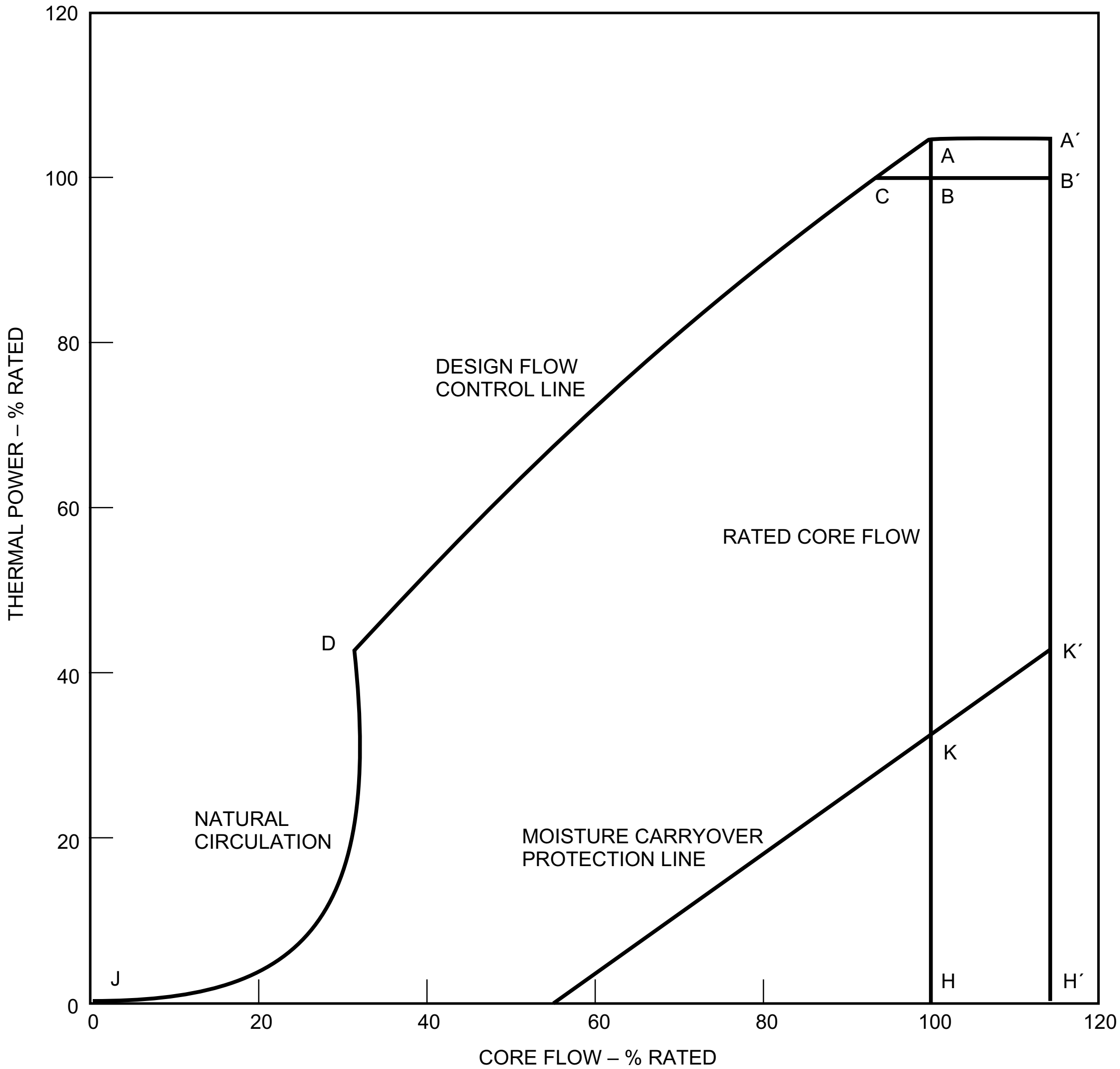


Figure 15.0-1 System Response Analysis Power/Flow Map

15.1 Decrease in Reactor Coolant Temperature

15.1.1 Loss of Feedwater Heating

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in at least two ways:

(1) Steam extraction line to heater is closed.

(2) Feedwater is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the feedwater bypasses the heater and no heating of that feedwater occurs. In either case, the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations.

The ABWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C feedwater heating. The reference steam and power conversion system shown in Figures 10.1-1 to 10.1-3 meets this requirement. In fact, the feedwater temperature drop based on the reference heat balance (Figure 10.1-1) is less than 30°C . Therefore, the use of 55.6°C temperature drop in the transient analysis is conservative.

This event has been conservatively estimated to incur a loss of up to 55.6°C of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) includes a logic intended to mitigate the consequences of a loss of feedwater heating capability. The system will be constantly monitoring the actual feedwater temperature and comparing it with a reference temperature. When a loss of feedwater heating is detected (i.e., when the difference between the actual and reference temperatures exceeds a ý T setpoint, which is currently set at 16.7°C), the FWCS sends an alarm to the operator. The operator can then take actions to mitigate the event. This will avoid a scram and reduce the ý CPR during the event. The same signal is also sent to the RCIS to initiate the SCRRI (selected control rods run-in) to automatically reduce the reactor power and avoid a scram. This will prevent the reactor from violating any thermal limits.

Because this event is very slow, the operator action or automatic SCRRI will terminate this event. Therefore, the worst event is the loss of feedwater heating resulting in a temperature difference just below the ýT setpoint. However, a loss of 55.6°C feedwater temperature is analyzed to bound this event.

15.1.1.1.2 Frequency Classification

The probability of this event is considered low enough to warrant it being categorized as an infrequent incident. However, because of the lack of a sufficient frequency database, this transient disturbance is analyzed as an incident of moderate frequency.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Table 15.1-1 lists the sequence of events for this transient.

15.1.1.2.1.1 Identification of Operator Actions

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The high simulated thermal power trip (STPT) scram is the primary protection system trip in mitigating the consequences of this event. However, the power increase in this event is not high enough to initiate this scram. Operation of engineered safety features (ESF) is not expected for this transient.

15.1.1.3 Core and System Performance

15.1.1.3.1 Input Parameters and Initial Conditions

The transient is simulated by programming a change in feedwater enthalpy corresponding to a 55.6°C loss in feedwater heating. Another case with the ý T setpoint in FWCS of 16.7°C is also analyzed.

15.1.1.3.2 Results

Because the power increase during this event is relatively slow, it can be treated as a quasi steady-state transient. The 3-D core simulator, PANACEA, has been used to evaluate this event for the equilibrium cycle. The results are summarized in Tables 15.1-2 and 15.1-2a.

The MCPR response of this event is small due to the mild thermal power increase with shifting axial shape. The worst ý CPR response is 0.07.

No scram is initiated in this event. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not change significantly (less than 0 .039 MPaD) and, consequently, the RCPB is not threatened.

15.1.1.4 Barrier Performance

As noted previously, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.1.2 Feedwater Controller Failure—Maximum Demand

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow.

The ABWR FWCS uses a triplicated digital control system, instead of a single-channel analog system, as used in current BWR designs (BWR 2-6). The digital systems consist of a triplicated fault-tolerant digital controller, the operator control stations and displays. The digital controller contains three parallel processing channels, each containing the microprocessor-based hardware and associated software necessary to perform all the control calculations. The operator interface provides information regarding system status and the required control functions.

Redundant transmitters are provided for key process inputs, and input voting and validation are provided such that faults can be identified and isolated. Each system input is triplicated internally and sent to the three processing channels (Figure 15.1-1). The channels will produce the same output during normal operation. Interprocessor communication provides self-diagnostic capability. A two-out-of-three voter compares the processor outputs to generate a validated output to the control actuator. A separate voter is provided for each actuator. A “ringback” feature feeds back the final voter output to the processors. A voter failure will thereby be detected and alarmed. In some cases, a protection circuit will lock the actuator into its existing position promptly after the failure is detected.

Table 15.1-3 lists the failure modes of a triplicated digital control system and outlines the effects of each failure. Because of the triplicated architecture, it is possible to take one channel out of service for maintenance or repair while the system is online. Modes 2 and 5 of Table

15.1-3 address a failure of a component while an associated redundant component is out of service. This type of failure could potentially cause a system failure. However, the probability of a component failure during servicing of a counterpart component is considered to be so low that these failure modes will not be considered incidents of moderate frequency, but, rather, limiting faults.

Adverse effects minimization is mentioned in the effects of Mode 2. This feature stems from the additional intelligence of the system provided by the microprocessor. When possible, the system will be programmed to take action in the event of some failure which will reduce the severity of the transient. For example, if the total steam flow or total feedwater flow signals fail, the FWCS will detect this by the input reasonability checks and automatically switch to one-element mode (i.e., control by level feedback only). The level control would essentially be unaffected by this failure.

The only credible single failures which would lead to some adverse effect on the plant are Modes 6 (failure of the output voter) and 7 (control actuator failure). Both of these failures would lead to a loss of control of only one actuator (i.e., only one feedwater pump with increasing flow). A voter failure is detected by the ringback feature. The FWCS will initiate a lockup of the actuator upon detection of the failure. The probabilities of failure of the variety of control actuators are very low based on operating experience. In the event of one pump run-out, the FWCS would then reduce the demand to the remaining pump, thereby automatically compensating for the excessive flow from the failed pump. Therefore, the worst single failure in the FWCS causes a runout of one feedwater pump to its maximum capacity. However, the demand to the remaining feedwater pump will decrease to offset the increased flow of the failed pump. The effect on total flow to the vessel will not be significant. The worst additional single failure would cause all feedwater pumps to run out to their maximum capacity. However, the probability of this to occur is extremely low.

15.1.2.1.2 Frequency Classification

15.1.2.1.2.1 Runout of One Feedwater Pump

Although the frequency of occurrence for this event is very low, this event is conservatively evaluated as an incident of moderate frequency.

15.1.2.1.2.2 Feedwater Controller Failure—Maximum Demand

The frequency of occurrence for this event is estimated to be so low that it should be classified as a limiting fault as specified in Chapter 15 of Regulatory Guide 1.70. Nonetheless, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

15.1.2.2.1.1 Runout of One Feedwater Pump

With momentary increase in feedwater flow, the water level rises and then settles back to its normal level. Table 15.1-4 lists the sequencing of events for Figure 15.1-2.

15.1.2.2.1.2 Feedwater Controller Failure—Maximum Demand

With excess feedwater flow, the water level rises to the high-level reference point, at which time the feedwater pumps and the main turbine are tripped and a scram is initiated. Table 15.1-5 lists the sequence of events for Figure 15.1-3. The figure shows the changes in important variables during this transient.

15.1.2.2.1.3 Identification of Operator Actions

15.1.2.2.1.3.1 Runout of One Feedwater Pump

Because no scram occurs for runout of one feedwater pump, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.2.2.1.3.2 Feedwater Controller Failure—Maximum Demand

The operator should:

(1) Observe that high feedwater pump trip has terminated the failure event

(2) Switch the feedwater controller from auto to manual control to try to regain a correct output signal

(3) Identify causes of the failure and report all key plant parameters during the event

15.1.2.2.2 Systems Operation

15.1.2.2.2.1 Runout of One Feedwater Pump

Runout of a single feedwater pump requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.1.2.2.2.2 Feedwater Controller Failure—Maximum Demand

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems. Important system operational actions for this event include (1) high level tripping of the main turbine and feedwater pumps, (2) scram and recirculation pump trip (RPT) due to turbine trip, and (3) low water level initiation of the RCIC System to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.3 Core and System Performance

15.1.2.3.1 Input Parameters and Initial Conditions

The runout capacity of one feedwater pump is assumed to be 75% of rated flow at the design pressure of 7.35 MPaG. The total feedwater flow for all pumps runout is assumed to be 130% of rated at the design pressure of 7.35 MPaG.

15.1.2.3.2 Results

15.1.2.3.2.1 Runout of One Feedwater Pump

The simulated runout of one feedwater pump event is presented in Figure 15.1-2. When the increase of feedwater flow is sensed, the feedwater controller starts to command the remaining feedwater pump to reduce its flow immediately. The vessel water level increases slightly (about 15 cm) and then settles back to its normal level. The vessel pressures only increase about 0.01 MPaD. MCPR remains above the safety limit.

15.1.2.3.2.2 Feedwater Controller Failure—Maximum Demand

The simulated runout of all feedwater pumps is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 18 seconds. Scram occurs and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. It is calculated that the MCPR is right at the safety limit. Therefore, the design limit for the moderate frequent incident is met. The Turbine Bypass System opens to limit peak pressure in the steamline near the SRVs to 8.12 MPaG and the pressure at the bottom of the vessel to about 8.33 MPaG.

The level will gradually drop to the Low Level reference point (Level 2), activating the RCIC System for long-term level control.

The COL applicant will provide reanalysis of this event for the specific core configuration.

15.1.2.4 Barrier Performance

As previously noted, the consequence of this event does not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.3 Pressure Regulator Failure—Open

15.1.3.1 Identification of Causes and Frequency Classifications

15.1.3.1.1 Identification of Causes

The ABWR Steam Bypass and Pressure Control System (SB&PCS) uses a triplicated digital control system instead of an analog system as used in current BWR designs (BWR 2-6). The SB&PCS controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system will result in a maximum demand to all actuators for all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent opening of one turbine control valve or one turbine bypass valve. In this case, the SB&PCS will sense the pressure change and command the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Because the effect of a sudden opening of one bypass valve, which bypasses about 11% of rated steam flow when full opened, is more severe than the sudden opening of one turbine control valve (which is almost wide open at rated power), it is assumed for purposes of this transient analysis that a single failure causes a single bypass valve to fail open.

As presented in Subsection 15.1.2.1.2, multiple failures might cause the SB&PCS to erroneously issue a maximum demand to all turbine control valves and bypass valves. Should this occur, all turbine control valves and bypass valves could be fully opened. However, the probability of this event is extremely low, and, hence, the event is considered as a limiting fault. However, the criteria of moderate frequency incidents are conservatively applied to this event.

15.1.3.1.2 Frequency Classification

15.1.3.1.2.1 Inadvertent Opening of One Turbine Bypass Valve

This transient disturbance, estimated to occur with very low frequency, is conservatively categorized as one of moderate frequency.

15.1.3.1.2.2 Inadvertent Opening of all Turbine Control Valves and Bypass Valves

The frequency of occurrences for this event is estimated to be extremely low. The event should thus be classified as a limiting fault as specified in Chapter 15 of Regulatory Guide 1.70. Nonetheless, since the consequence of this event has no significant impact on the operating CPR limit, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

15.1.3.2.1.1 Inadvertent Opening of One Turbine Bypass Valve

Table 15.1-6 lists the sequence of events for Figure 15.1-4.

15.1.3.2.1.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

Table 15.1-7 lists the sequence of events for Figure 15.1-5.

15.1.3.2.1.3 Identification of Operator Actions

15.1.3.2.1.3.1 Inadvertent Opening of One Turbine Bypass Valves

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

15.1.3.2.1.3.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (5.69 MPaG) in the run mode, the following sequence of operator actions is expected during the course of the event. Once isolation occurs, the pressure will increase to a point where the SRVs open. The operator should:

(1) Monitor that all rods are in

(2) Monitor reactor water level and pressure

(3) Observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries

(4) Observe that the reactor pressure relief valves open at their setpoint

(5) Observe that RCIC initiated on low-water level

(6) Secure RCIC when reactor pressure and level are under control

(7) Monitor reactor water level and continue cooldown per the normal procedure

(8) Complete the scram report and initiate a maintenance survey of the SB&PCS before reactor restart

15.1.3.2.2 Systems Operation

15.1.3.2.2.1 Inadvertent Opening of One Turbine Bypass Valve

This event does not require any protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.1.3.2.2.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems, except as otherwise noted.

Initiation of RCIC System functions occurs when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized.

If these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and are expected to be less severe than those already experienced by the system.

15.1.3.3 Core and System Performance

15.1.3.3.1 Input Parameters and Initial Conditions

A five-second isolation valve closure (maximum isolation valve closing time plus instrument delay) instead of a three second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure setpoint for main steamline isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

15.1.3.3.2 Results

15.1.3.3.2.1 Inadvertent Opening of One Turbine Bypass Valve

The simulated inadvertent opening of one turbine bypass valve is presented in Figure 15.1-4. When the decrease in reactor pressure is sensed, the pressure control system starts immediately to command turbine control valves to close to maintain the reactor pressure. The vessel water level increases slightly (about 10 cm) and then settles back to its normal level. Reactor pressure decreases by about 0.069 MPaD. MCPR remains above the safety limit.

15.1.3.3.2.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

Figure 15.1-5 presents graphically how the high water level turbine trip and the isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

Depressurization results in formation of voids in the reactor coolant and causes a decrease in reactor power almost immediately. The depressurization rate is large enough such that water level swells to the sensed level trip setpoint (L8), initiating main turbine and feedwater pump trips. Position switches on the turbine stop valves initiate reactor scram and a trip of four RIPs.

After a pressurization resulting from the turbine stop valve closure, pressure again drops and continues to drop until turbine inlet pressure is below the low turbine pressure isolation setpoint when main steamline isolation finally terminates the depressurization. The turbine trip and isolation limit the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. No significant reduction in fuel thermal margins occur; therefore, this event does not have to be analyzed for specific core configurations.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required because the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which fuel, pressure vessel or containment are designed. During the event of inadvertent opening of all turbine control and bypass valves, peak pressure in the bottom of the vessel reaches 8.02 MPaG, which is below the ASME code limit of 9.48 MPaGfor the reactor coolant pressure boundary. Vessel dome pressure reaches 7.88 MPaG, below the setpoint of the second pressure relief group. Minimum vessel dome pressure of 4.96 MPaG occurs at about 40 seconds.

15.1.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.4 Inadvertent Safety/Relief Valve Opening

15.1.4.1 Identification of Causes and Frequency Classification

15.1.4.1.1 Identification of Causes

Cause of inadvertent opening is attributed to malfunction of the valve or an operator initiated opening. It is therefore simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-8 lists the sequence of events for this event.

15.1.4.2.1.1 Identification of Operator Actions

The plant operator must reclose the valve as soon as possible and check that reactor and T-G output return to normal. If the valve cannot be closed, plant shutdown should be initiated.

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.3 Core and System Performance

The opening of one SRV allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The SB&PCS senses the nuclear system pressure decrease and within a few seconds closes the turbine control valves far enough to stabilize the reactor vessel pressure at a slightly lower value and the reactor settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected and this event does not have to be reanalyzed for specific core configurations.

The discharge of steam to the suppression pool causes the temperature of the suppression pool to increase. When the pool temperature reaches the high temperature setpoint, the suppression pool cooling function of the RHR System is automatically initiated. The pool temperature continues to increase due to the mismatch of cooling capacity and steam discharged into the pool. When the pool temperature reaches the next setpoint of 43.3°C, a reactor scram is automatically initiated. In this analysis a conservative scram set point of 48.9°C was assumed.

15.1.4.4 Barrier Performance

As presented previously, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequence of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Because this activity is contained in the primary containment, there will be no exposures to operating personnel. Because this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside Containment in a PWR

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions, no conceivable malfunction in the shutdown cooling system could cause a temperature reduction.

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderate temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Because no single failure could cause this event, it should be categorized as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram occurs before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-9.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation because the nuclear system pressure is too high to permit operation of the Shutdown Cooling Mode (SDC) of the RHRs.

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature is controlled by the operator in the same manner normally used to control power in the startup range.

15.1.6.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR SDC mode could result in a slow power increase due to the reactivity insertion. This power rise is terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here and this event does not have to be analyzed for specific core configuration.

15.1.6.4 Barrier Performance

As previously presented, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Because this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

Table 15.1-1 Sequence of Events for Loss of Feedwater Heating

Time (s)	Event
0	Initiate a 55.6°C (or 16.7°C) temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level.
100 (est.)	Reactor variables settle into new steady state.

Table 15.1-2 Loss of 55.6°C Feedwater Heating

	BOC* to EOC*
Change in Core Power (%)	12.8
Change in MCPR	0.07

* BOC = Beginning of Cycle
EOC = End of Cycle

Table 15.1-2a Loss of 16.7°C Feedwater Heating

	BOC * to EOC*
Change in Core Power (%)	3.9
Change in MCPR	0.02

* BOC = Beginning of Cycle
EOC = End of Cycle

Table 15.1-3 Single Failure Modes for Digital Controls

Modes	Description	Effects
1.	Critical input failure	None—Redundant transmitter takes over—Operator informed of failure
2.	Input failure while one sensor out of service	Possible system failure. Adverse effects minimized when possible
3.	Operator switch single contact failure	None—Triplicated contacts
4.	Processor channel failure	None—Redundant processors maintain control; Operator informed of failure
5.	Processor failure while one channel out of service	System failure
6.	Voter failure	Loss of control of one actuator (i.e., one feedwater pump only). FWCS will lock up actuators.
7.	Actuator failure	Loss of one actuator (i.e., one feedwater pump only)

Table 15.1-4 Sequence of Events for Figure 15.1-2

Time (s)	Events
0	Initiate simulated runout of one feedwater pump (at system design pressure of 7.35 MPaG the pump runout flow is 75% of rated feedwater flow).
~0.1	Feedwater controller starts to reduce the feedwater flow from the other feedwater pump.
16.6	Vessel water level reaches its peak value and starts to return to its normal value.
~60 (est.)	Vessel water level returns to its normal value.

Table 15.1-5 Sequence of Events for Figure 15.1-3

Time (s)	Event
0	Initiate simulated runout of all feedwater pumps (130% at system design pressure of 7.35 MPaG on feedwater flow).
18.35	L8 vessel level setpoint initiates trip of main turbine and feedwater pumps.
18.36	Reactor scram and trip of 4 RIPs are actuated by stop valve position switches.
18.5	Main turbine bypass valves opened due to turbine trip.
20.1	SRVs open due to high pressure.
>25	SRVs close.
>40 (est.)	Water level dropped to low water level setpoint (Level 2).
>70 (est.)	RCIC flow into vessel (not simulated).

Table 15.1-6 Sequence of Events for Figure 15.1-4

Time (s)	Events
0	Simulate one bypass valve to open.
~0.5	Pressure control system senses the decrease of reactor pressure and commands control valves to close.
5.0	Reactor settles at another steady state.

Table 15.1-7 Sequence of Events for Figure 15.1-5

Time (s)	Events
0	Simulate all turbine control valves and bypass valves to open.
2.8	Turbine control valves wide open.
2.87	Vessel water level (L8) trip initiates main turbine and feedwater pump trips.
2.9	Main turbine stop valves reach 85% open position and initiates reactor scram and trip of 4 RIPs.
2.97	Turbine stop valves closed.
17.2	Vessel water level reaches L2 setpoint. The remaining 6 RIPs are tripped. RCIC is initiated.
36.2	Low turbine inlet pressure trip initiates main steamline isolation.
41.2	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
47.2 (est.)	RCIC flow enters vessel (not simulated).

Table 15.1-8 Sequence of Events for Inadvertent Safety/Relief Valve Opening

Time (s)	Event
0	Initiated opening of one SRV.
0.5 (est.)	Relief flow reaches full flow.
15 (est.)	System establishes new steady-state operation.
750 (est.)	Suppression pool temperature reaches setpoint; suppression pool cooling function is initiated.
1200 (est.)	Suppression pool temperature reaches setpoint; reactor scram is automatically initiated.

Table 15.1-9
Sequence of Events for Inadvertent RHR Shutdown Cooling Operation

Approximate Elapsed Time	Event
0	Reactor at states B or D (of Appendix 15A) when RHR shutdown cooling inadvertently activated.
0–10 min.	Slow rise in reactor power.
+10 min.	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

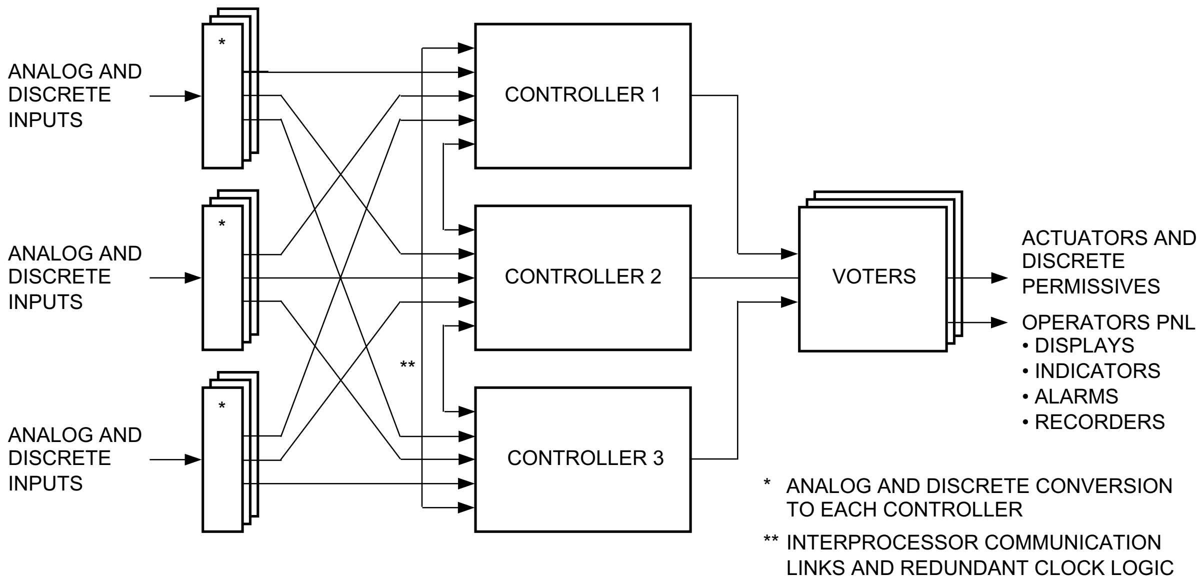


Figure 15.1-1 Simplified Block Diagram of Fault-Tolerant Digital Controller System

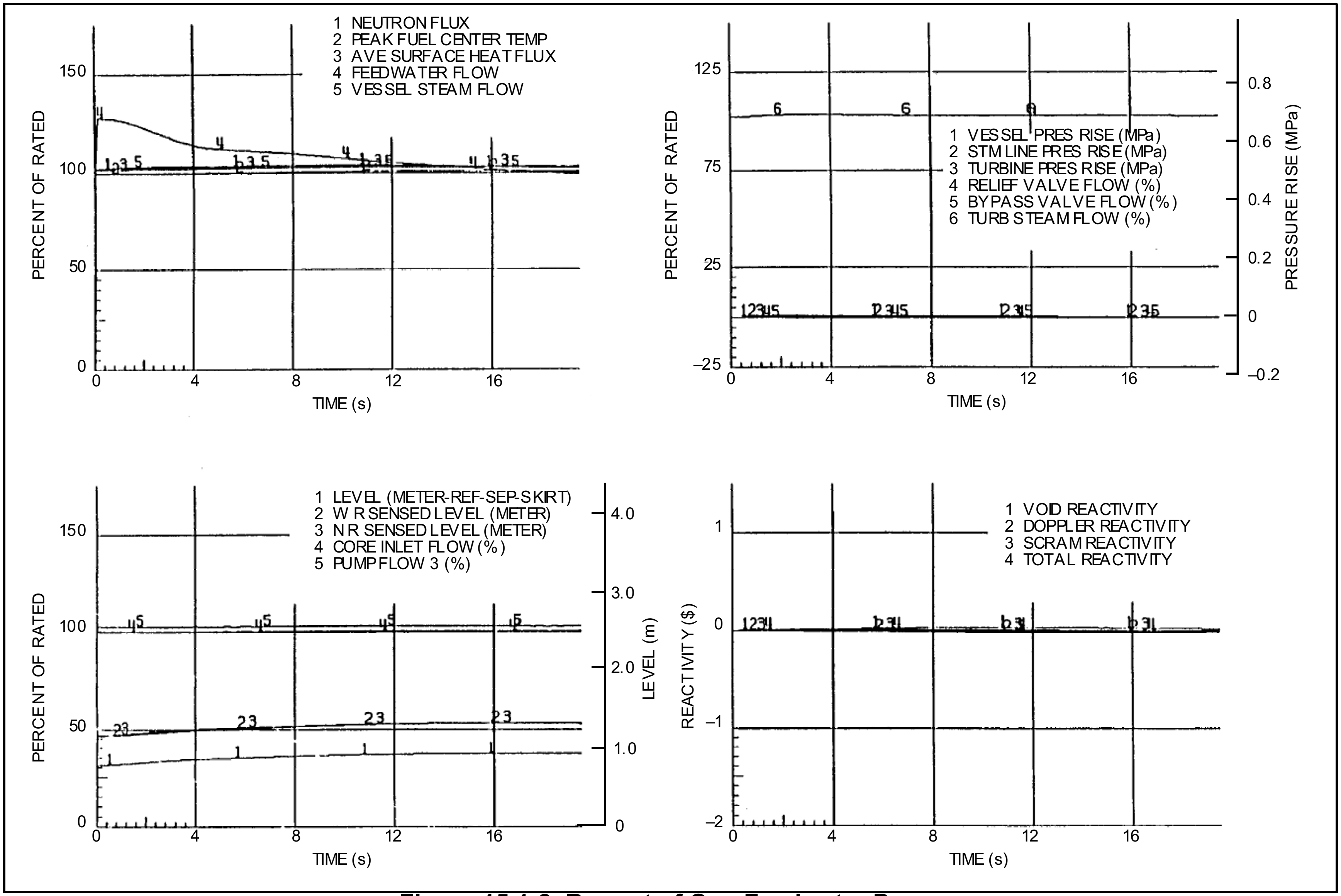


Figure 15.1-2 Runout of One Feedwater Pump

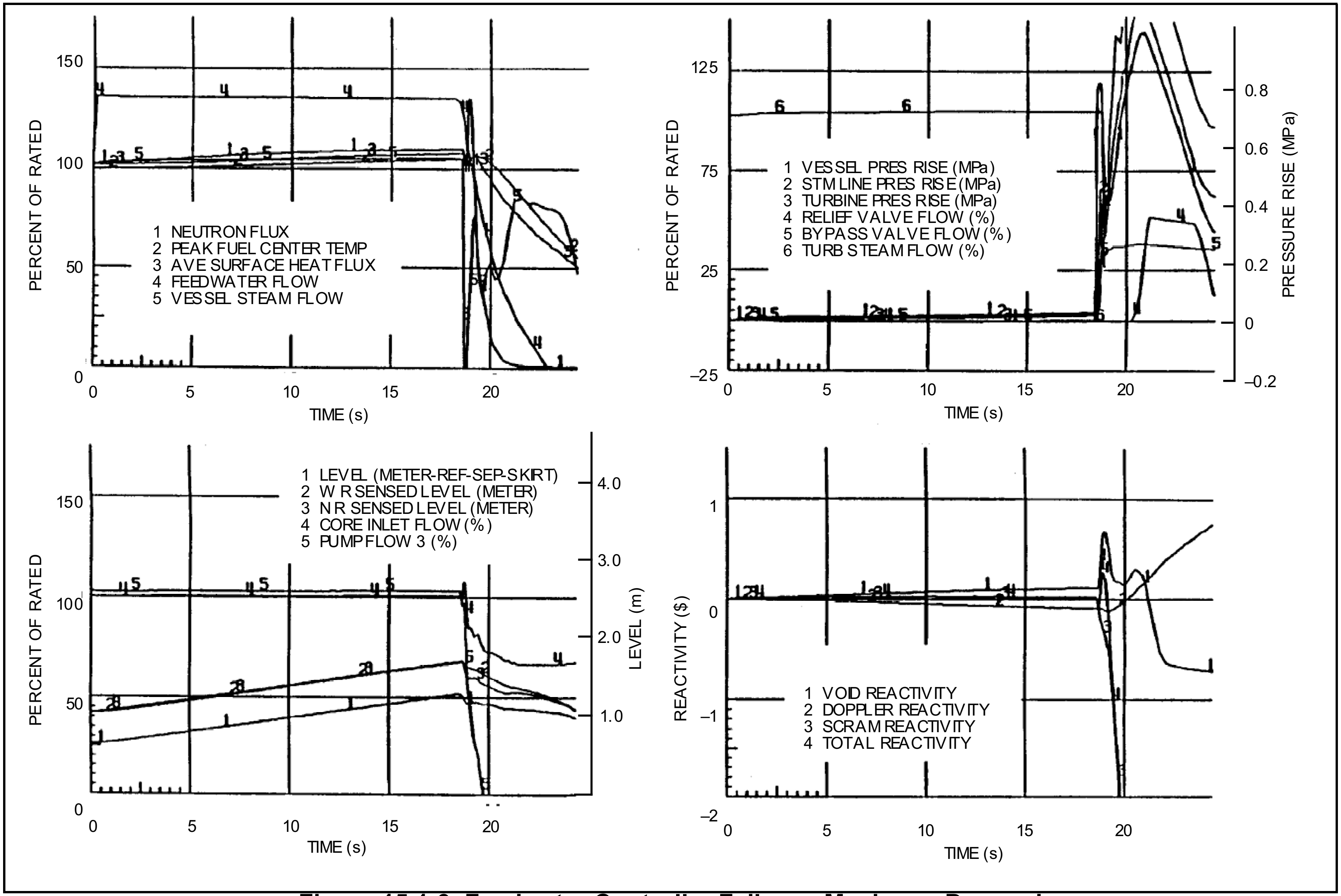


Figure 15.1-3 Feedwater Controller Failure—Maximum Demand

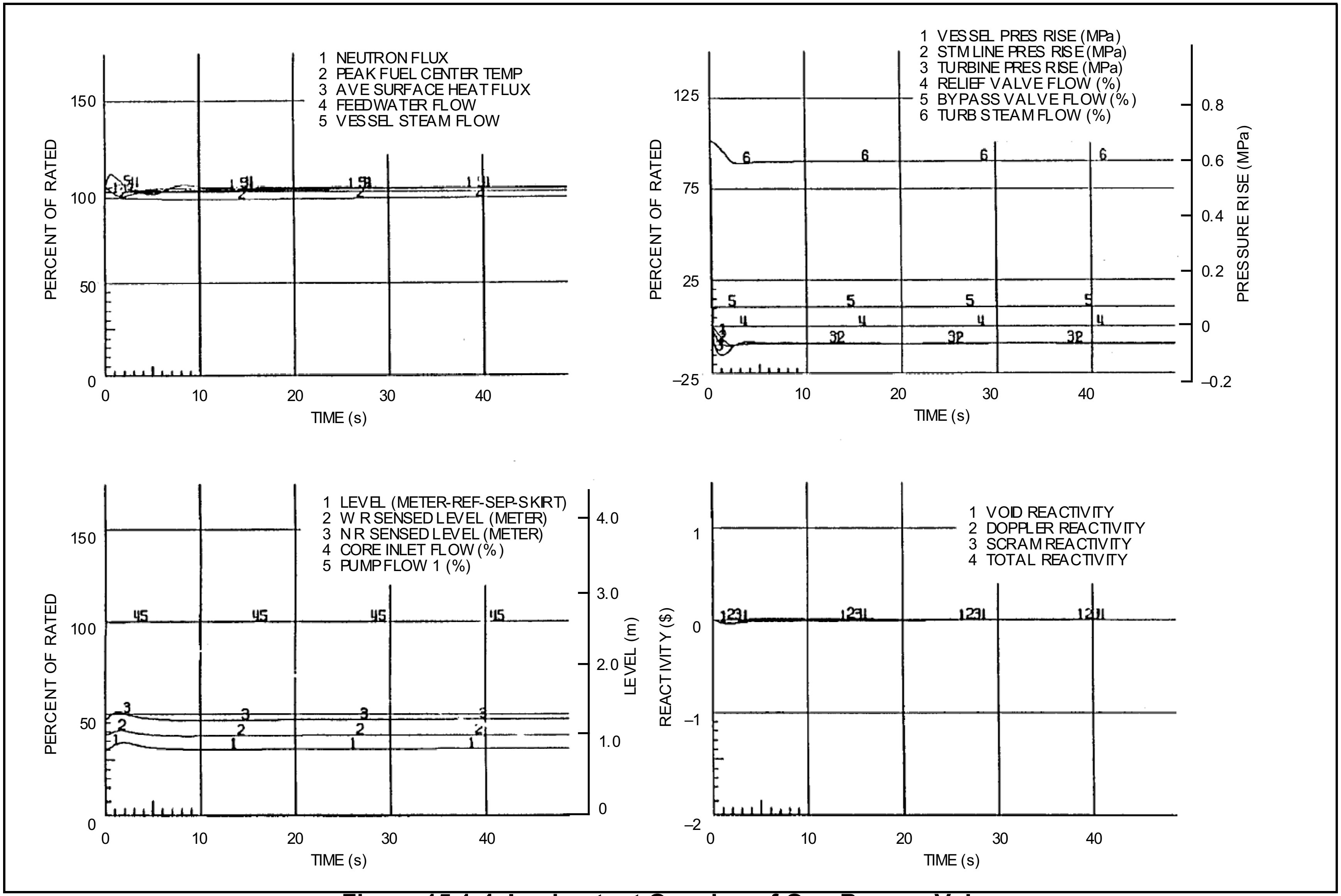


Figure 15.1-4 Inadvertent Opening of One Bypass Valve

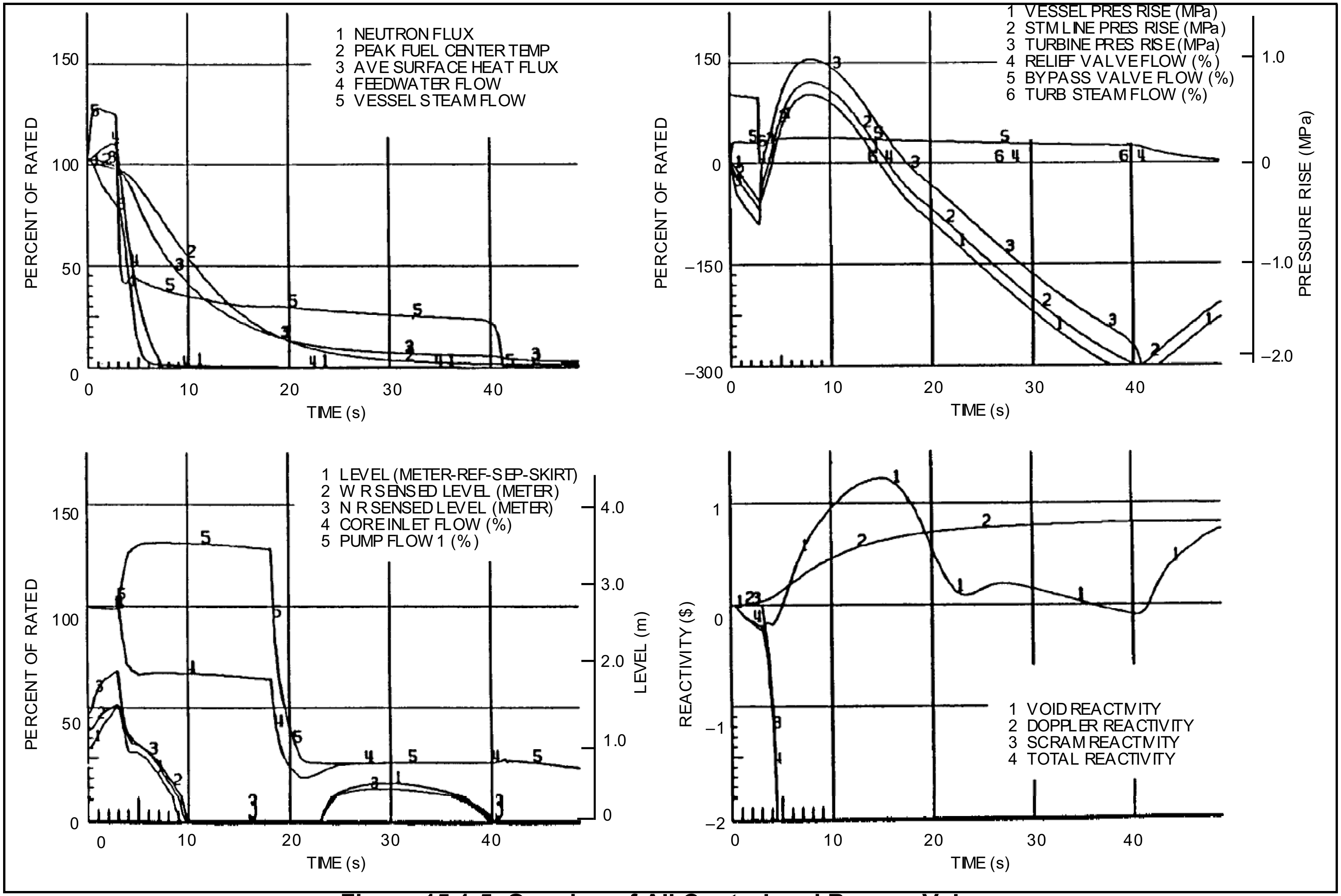


Figure 15.1-5 Opening of All Control and Bypass Valves

15.2 Increase in Reactor Pressure

15.2.1 Pressure Regulator Failure—Closed

15.2.1.1 Identification of Causes and Frequency Classification

15.2.1.1.1 Identification of Causes

The ABWR Steam Bypass and Pressure Control System (SB&PCS) uses a triplicated digital control system instead of an analog system as used in BWR/2 through BWR/6. The SB&PCS controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system will result in a minimum demand to all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent closure of one turbine control valve or one turbine bypass valve if it is open at the time of failure. In this case, the SB&PCS will sense the pressure change and command the remaining control valves or bypass valves, if needed, to open, and thereby automatically mitigate the transient and try to maintain reactor power and pressure.

Because turbine bypass valves are normally closed during normal full power operation, it is assumed for purposes of this transient analysis that a single failure causes a single turbine control valve to fail closed. Should this event occur at full power, the opening of remaining control valves may not be sufficient to maintain the reactor pressure, depending on the turbine design. Neutron flux will increase due to void collapse resulting from the pressure increase. A reactor scram will be initiated when the high flux scram setpoint is exceeded.

No single failure will cause the SB&PCS to issue erroneously a minimum demand to all turbine control valves and bypass valves. However, as discussed in Subsection 15.1.2.1.1, multiple failures might cause the SB&PCS to fail and erroneously issue a minimum demand. Should this occur, it would cause full closure of turbine control valves as well as inhibit steam bypass flow and thereby increase reactor power and pressure. When this occurs, reactor scram will be initiated when the high reactor flux scram setpoint is reached. This event is analyzed here as the simultaneous failure of two control processors, called “pressure regulator downscale failure.” However, the probability of this event occurring is extremely low, and hence the event is considered as a limiting fault.

15.2.1.1.2 Frequency Classification

15.2.1.1.2.1 Inadvertent Closure of One Turbine Control Valve

This event is conservatively treated as a moderate frequency event, although the voter/ actuator failure rate is very low.

15.2.1.1.2.2 Pressure Regulator Downscale Failure

The probability of occurence of this event is calculated to be extremely low, as shown in Appendix 15D. This event is treated as a limiting fault.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Inadvertent Closure of One Turbine Control Valve

Postulating an actuator failure of the SB&PCS (Subsection 15.2.1.1.1) will cause one turbine control valve to close. The pressure will increase because the reactor is still generating the initial steam flow. The SB&PCS will open the remaining control valves and some bypass valves. This sequence of events is listed in Table 15.2-1a for Figure 15.2-1, for a fast closure, and in Table 15.2-1b for Figure 15.2-1, for a slow closure.

15.2.1.2.1.1 Pressure Regulator Downscale Failure

Table 15.2-2 lists the sequence of events for Figure 15.2-2.

15.2.1.2.1.2 Identification of Operator Actions

The operator should:

(1) Monitor that all rods are in

(2) Monitor reactor water level and pressure

(3) Observe turbine coastdown and break vacuum before the loss of steam seals (check turbine auxiliaries)

(4) Observe that the reactor pressure relief valves open at their setpoint

(5) Monitor reactor water level and continue cooldown per the normal procedure

(6) Complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart

15.2.1.2.2 Systems Operation

15.2.1.2.2.1 Inadvertent Closure of One Turbine Control Valve

Normal plant instrumentation and control are assumed to function. This event takes credit for high neutron flux scram to shut down the reactor.

After a closure of one turbine control valve, the steam flow rate that can be transmitted through the remaining three turbine control valves depends upon the turbine configuration. For plants with full-arc turbine admission, the steam flow through the remaining three turbine control valves is at least 95% of rated steam flow. On the other hand, this capacity drops to about 85% of rated steam flow for plants with partial-arc turbine admission. Therefore, this transient is less severe for plants with full-arc turbine admission. In this analysis, cases with full-arc and partial-arc turbine admission are analyzed to cover all potential operating conditions.

This event is sensitive to the closure time of the turbine control valve, and the bypass capacity available during this event. A wide range of closure time, including very slow closure, has been assumed in the analysis. A fast closure causes the reactor to be scrammed on high neutron flux trip, while a slow closure allows the reactor to settle in another steady state.

The turbine bypass capacity during this event is controlled by the setpoint of the maximum combined steam flow limits in the pressure control system. A nominal 115% setpoint will allow for about 12% bypass capacity, while a nominal 125% setpoint for about 22%, assuming a 3% bypass bias. It is concluded from analysis that the nominal setpoint for the maximum combined flow limiter should be set at 115% for plants with full-arc turbine admission, and at 125% for plants with partial-arc turbine admission.

15.2.1.2.2.2 Pressure Regulator Downscale Failure

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems. Specifically, this event takes credit for high neutron flux scram to shut down the reactor. High system pressure is limited by the pressure relief valve system operation.

15.2.1.3 Core and System Performance

15.2.1.3.1 Inadvertent Closure of One Turbine Control Valve

A simulated fast closure of one turbine control valve (2.5 sec) is presented in Figure 15.2-1. The analysis assumes that about 85% of rated steam flow can pass through the remaining three turbine control valves.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux increase is limited to 124% NBR by the reactor scram. Peak fuel surface heat flux does not exceed 103.6% of its initial value. MCPR for this transient is still above the safety MCPR limit (ýCPR = 0.10). Therefore, the design basis is satisfied.

A slow closure of one turbine control valve is also analyzed as shown in Figure 15.2-1a. In this case, the neutron flux increase does not reach the high neutron flux scram setpoint. Since the available turbine bypass capacity is high enough to bypass all steam flow not passing through the remaining three turbine control valves, the reactor power settles back to its steady state. During the transient, the peak fuel surface heat flux does not exceed 103.6% of its initial value. MCPR is still above the safety limit (ýCPR = 0.09). Therefore, the design basis is satisfied.

The applicant will provide reanalysis of this event for the specific core configuration.

15.2.1.3.2 Pressure Regulator Downscale Failure

A pressure regulator downscale failure is simulated at 102% NBR power as shown in Figure 15.2-2.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. When the sensed neutron flux reaches the high neutron flux scram setpoint, a reactor scram is initiated. The neutron flux increase is limited to 155% NBR by the reactor scram. Peak fuel surface heat flux does not exceed 103% of its initial value. It is estimated that less than 0.2% of rods will get into transition boiling. Therefore, the design limit for the limiting fault event is met.

15.2.1.4 Barrier Performance

15.2.1.4.1 Inadvertent Closure of One Turbine Control Valve

Peak pressure at the SR valves reaches 7.31 MPaG. The peak vessel bottom pressure reaches 8.57 MPaG, which is below the transient pressure limit of 9.48 MPaG.

15.2.1.4.2 Pressure Regulator Downscale Failure

Peak pressure at the SRVs reaches 8.35 MPaG. The peak nuclear system pressure reaches 8.57 MPaG at the bottom of the vessel, which is below the nuclear barrier pressure limit.

15.2.1.5 Radiological Consequences

15.2.1.5.1 Inadvertent Closure of One Turbine Control Valve

The consequences of this event do not result in any fuel failures, nor any discharge to the suppression pool. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.2.1.5.2 Pressure Regulator Downscale Failure

During this event, less than 0.2% of fuel rods get into transition boiling. No fuel failures are expected. However, it is conservatively assumed that 0.2% of fuel rods fail in the radiological dose calculation. The results show that both the whole body dose and thyroid dose are well within 10% of 10CFR100 requirements. Therefore, the acceptance criteria are met.

15.2.2 Generator Load Rejection

15.2.2.1 Identification of Causes and Frequency Classification

15.2.2.1.1 Identification of Causes

Fast closure of the turbine control valves (TCVs) is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive overspeed of the turbine-generator (T-G) rotor. Closure of the main TCVs will cause a sudden reduction in steam flow, which results in an increase in system pressure and reactor shutdown.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the “fast” opening mode by the SB&PCS, which uses a triplicated digital controller. As presented in Subsection 15.1.2.1.1, no single failure can cause all turbine bypass valves (TBVs) to fail to open on demand. The worst single failure can only cause one TBV fail to open on demand. Therefore, the probability of this to occur is very low. Therefore, generator load rejection with failure of one TBV is considered an infrequent event, while generator load rejection with failure of all TBVs is a limiting fault.

15.2.2.1.2 Frequency Classification

15.2.2.1.2.1 Generator Load Rejection

This event is categorized as an incident of moderate frequency.

15.2.2.1.2.2 Generator Load Rejection with Failure of One Bypass Valve

This event should be categorized as an infrequent event. However, criteria for moderate frequent incidents are conservatively applied.

15.2.2.1.2.3 Generator Load Rejection with Failure of All Bypass Valves

Frequency Basis: Thorough search of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. Combining the actual frequency of a generator load rejection with the failure rate of bypass yields a frequency of a generator load rejection with bypass failure. With the triplicated fault-tolerant design used in ABWR, this failure frequency is lowered much more. Therefore, this event should be classified as a limiting fault; however, criteria for moderate frequent incidents are conservetively applied.

15.2.2.2 Sequence of Events and System Operation

15.2.2.2.1 Sequence of Events

15.2.2.2.1.1 Generator Load Rejection—Turbine Control Valve Fast Closure

A loss of generator electrical load from high power conditions produces the sequence of events listed in Table 15.2-3.

15.2.2.2.1.2 Generator Load Rejection with Failure of One Bypass Valve

A loss of generator electrical load from high power conditions with failure of one bypass valve produces the sequence of events listed in Table 15.2-4.

15.2.2.2.1.3 Generator Load Rejection with Failure of All Bypass Valves

A loss of generator electrical load at high power with failure of all bypass valves produces the sequence of events listed in Table 15.2-5.

15.2.2.2.1.4 Identification of Operator Actions

The operator should:

(1) Verify proper bypass valve performance

(2) Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value

(3) Observe that the pressure regulator is controlling reactor pressure at the desired value

(4) Observe reactor peak power and pressure

(5) Verify relief valve operation

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection and reactor protection systems unless stated otherwise.

Turbine control valve (TCV) fast closure initiates a scram trip signal for power levels greater than 40% NB rated. In addition, a trip of four of ten RIPs is initiated. Both of these trip signals satisfy the single-failure criterion and credit is taken for these protection features.

The pressure relief system, which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.2.2.2.2 Generator Load Rejection with Failure of One Bypass Valve

Same as Subsection 15.2.2.2.2.1, except that failure of one main TBV is assumed for the entire event.

15.2.2.2.2.3 Generator Load Rejection with Failure of All Bypass Valves

Same as Subsection 15.2.2.2.2.1, except that failure of all TBVs is assumed for the entire event.

15.2.2.3 Core and System Performance

15.2.2.3.1 Input Parameters and Initial Conditions

The turbine electrohydraulic control system (EHC) detects load rejection before a measurable turbine speed change takes place.

The TCV trip closure time is 0.08 seconds or greater.

The reactor is operating in the manual flow-control mode when load rejection occurs. Results do not significantly differ if the plant had been operating in the automatic flow-control mode.

The bypass valve opening characteristics are simulated using the specified delay together with the specified opening characteristic required for bypass system operation.

Events caused by low water level trips such as an initiation of the RCIC System function is not required. Should this event occur, it will follow sometime after the primary concerns of fuel thermal margin and overpressure effects have occurred, and is expected to be less severe than those already experienced by the system.

15.2.2.3.2 Results

15.2.2.3.2.1 Generator Load Rejection with Bypass

Figure 15.2-3 shows the results of the generator trip from the 102% rated power conditions. Peak neutron flux rises 48% above NB rated conditions.

The average fuel surface heat flux shows no increase from its initial value, and MCPR does not significantly decrease below its initial value. Therefore, this event does not have to be reanalyzed for a specific core configuration.

15.2.2.3.2.2 Generator Load Rejection with Failure of One Bypass Valve

Figure 15.2-4 shows that, for the case of one bypass valve failure, peak neutron flux reaches about 155% of rated, and the average fuel surface heat flux still shows no increase from its initial value.

The MCPR for this event is above the safety limit. Therefore, this event does not have to be analyzed for a specific core configuration.

15.2.2.3.2.3 Generator Load Rejection with Failure of All Bypass Valves

Figure 15.2-5 shows that, for the case of all bypass valves failure, peak neutron flux reaches about 185% of rated, and average surface heat flux reaches 102.3% of its initial value. The MCPR for this event is right at the safety limit and meets the criteria for moderate frequent incidents. The event should be analyzed for a specific core configuration.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure at the SRVs reaches 8.11 MPaG. The peak vessel bottom pressure reaches 8.31 MPaG, below the transient pressure limit of 9.48 MPaG.

15.2.2.4.2 Generator Load Rejection with Failure of One Bypass Valve

Peak pressure at the SRVs reaches 8.19 MPaG. The peak vessel pressure at the bottom of the vessel reaches 8.41 MPaG, below the pressure limit.

15.2.2.4.3 Generator Load Rejection with Failure of All Bypass Valves

Peak pressure at the SRVs reaches 8.39 MPaG. The peak nuclear system pressure reaches 8.60 MPaG at the bottom of the vessel, below the pressure limit.

15.2.2.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5. Therefore, the radiological exposures noted in Subsection 15.2.4.5 for Type 2 exposure cover these consequences of this event.

15.2.3 Turbine Trip

15.2.3.1 Identification of Causes and Frequency Classification

15.2.3.1.1 Identification of Causes

A variety of turbine or nuclear system malfunctions will initiate a turbine trip. Some examples are moisture separator and heater drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PCS. As presented in Subsection 15.2.2.1.1, any single failure can only cause one bypass valve fail to open on demand. Only multiple failures can cause all bypass valves fail to open on demand.

15.2.3.1.2 Frequency Classification

15.2.3.1.2.1 Turbine Trip

This transient is categorized as an incident of moderate frequency. In defining the frequency of this event, turbine trips which occur as a byproduct of other transients such as loss of condenser vacuum or reactor high level trip events are not included. However, spurious low vacuum or high level trip signals which cause an unnecessary turbine trip are included in defining the frequency. To get an accurate event-by-event frequency breakdown, this type of division of initiating causes is required.

15.2.3.1.2.2 Turbine Trip with Failure of One Bypass Valve

This event is conservatively considered as an incident of moderate frequency.

15.2.3.1.2.3 Turbine Trip with Failure of All Bypass Valves

This disturbance should be categorized as a limiting fault. Frequency is as follows:

Frequency Basis: The failure rate of the bypass is presented in Subsection 15.2.2.1.2.3. Combining this with the turbine trip frequency yields the frequency in general. The ABWR design reduces this frequency much more to classify as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.2.3.2 Sequence of Events and Systems Operation

15.2.3.2.1 Sequence of Events

15.2.3.2.1.1 Turbine Trip

Turbine trip at high power produces the sequence of events listed in Table 15.2-6.

15.2.3.2.1.2 Turbine Trip with Failure of One Bypass Valve

Turbine trip at high power with failure of one bypass valve produces the sequence of events listed in Table 15.2-7.

15.2.3.2.1.3 Turbine Trip with Failure of All Bypass Valves

Turbine trip at high power with failure of all bypass valves produces the sequence of events listed in Table 15.2-8.

15.2.3.2.1.4 Identification of Operator Actions

The operator should:

(1) Verify auto-transfer of buses supplied by generator to incoming power (if automatic transfer does not occur, manual transfer must be made)

(2) Monitor and maintain reactor water level at required level

(3) Check turbine for proper operation of all auxiliaries during coastdown

(4) Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes

(5) Put the mode switch in the startup position before the reactor pressure decays to <5.86 MPaG

(6) Secure the RCIC operation if auto initiation occurred due to low water lever

(7) Monitor control rod drive positions and the SRNMS

(8) Investigate the cause of the trip, make repairs as necessary, and complete the scram report

(9) Cool down the reactor per standard procedure if a restart is not intended

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip

All plant control systems maintain normal operation unless specifically designated to the contrary.

Turbine stop valve closure initiates a reactor scram trip via position signals to the protection system. Credit is taken for successful operation of the Reactor Protection System (RPS).

Turbine stop valves closure initiates a trip of four RIPs, thereby reducing the core flow.

The pressure relief system, which operates the relief valves independently when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed.

15.2.3.2.2.2 Turbine Trip with Failure of One Bypass Valve

Same as Subsection 15.2.3.2.2.1, except that a failure of one bypass valve is assumed.

15.2.3.2.2.3 Turbine Trip with Failure of All Bypass Valves

Same as Subsection 15.2.3.2.2.1, except that failure of all main turbine bypass valves is assumed for the entire transient time period analyzed.

15.2.3.3 Core and System Performance

15.2.3.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is 0.1 seconds.

A reactor scram is initiated by position switches on the stop valves when the valves are less that 85% open.

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips four of the reactor internal pumps.

15.2.3.3.2 Results

15.2.3.3.2.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 102% NBR power conditions as shown in Figure 15.2-6.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 122% of rated by the stop valve scram and the trip of four RIPs. Peak fuel surface heat flux does not exceed its initial value. Therefore, this event does not have to be reanalyzed for a specific core configuration.

15.2.3.3.2.2 Turbine Trip with Failure of One Bypass Valve

Same as Subsection 15.2.3.3.2.1, except the peak neutron flux is 132% of rated. This event is shown in Figure 15.2-7.

15.2.3.3.2.3 Turbine Trip with Failure of All Bypass Valves

A turbine trip with failure of the bypass system is simulated at 102% NBR power conditions in Figure 15.2-8.

Peak neutron flux reaches 159% of its rated value, and average surface heat flux reaches 100.6% of its initial value. Therefore, this transient is less severe than the generator load rejection with failure of bypass transient presented in Subsection 15.2.2.1.2.3.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 8.30 MPaG, which is below the ASME Code limit of 9.48 MPaG for the reactor coolant pressure boundary. Vessel dome pressure does not exceed 8.14 MPaG. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of One Bypass Valve

Peak pressure at the bottom of the vessel reaches 8.39 MPaG, while vessel dome pressure does not exceed 8.25 MPaG. Both are below the pressure limit of 9.48 MPaG.

15.2.3.4.3 Turbine Trip with Failure of All Bypass Valves

The SRVs open and close sequentially as the stored energy is dissipated and the pressure falls below the setpoints of the valves. Peak nuclear system pressure reaches 8.6 MPaG at the vessel bottom; therefore, the overpressure event is below the RCPB pressure limit. Peak dome pressure does not exceed 8.44 MPaG.

15.2.3.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is less than those consequences identified in

Subsection 15.2.4.5 for a Type 2 event. Therefore, the radiological exposures noted in Section 15.2.4.5 cover the consequences of this event.

15.2.4 MSIV Closures

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

Various steamline and nuclear system malfunctions, or operator actions, can initiate main steamline isolation valve (MSIV) closure. Examples are low steamline pressure, high steamline flow, high steamline radiation, low water level or manual action.

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All Main Steamline Isolation Valves

This event is categorized as an incident of moderate frequency. To define the frequency of this event as an initiating event and not the byproduct of another transient, only the following contribute to the frequency: (1) manual action (purposely or inadvertent); (2) spurious signals such as low pressure, low reactor water level, low condenser vacuum and, (3) equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSIV may cause an immediate closure of all other MSIVs, depending on reactor conditions. If this occurs, it is also included in this category. During the MSIV closure, position switches on the valves provide a reactor scram if the valves in two or more main steamlines are less than 85% open (except for interlocks which permit proper plant startup). Protection system logic, however, permits the test closure of one valve without initiating scram from the position switches.

15.2.4.1.2.2 Closure of One Main Steamline Isolation Valve

This event is categorized as an incident of moderate frequency. One MSIV may be closed at a time for testing purposes; this is done manually. Operator error or equipment malfunction may cause a single MSIV to be closed inadvertently. If reactor power is greater than about 80% when this occurs, a high flux scram may result (if all MSIVs close as a result of the single closure, the event is considered as a closure of all MSIVs).

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-9 lists the sequence of events for Figure 15.2-9.

15.2.4.2.1.1 Identification of Operator Actions

The following is the sequence of operator actions expected during the course of the event, assuming no restart of the reactor. The operator should:

(1) Observe that all rods have inserted

(2) Observe that the relief valves have opened for reactor pressure control

(3) Check that RCIC auto starts on the impending low reactor water level

(4) Switch the feedwater controller to the manual position

(5) Secure RCIC when the reactor vessel level has recovered to a satisfactory level

(6) Initiate RHR operation when the reactor pressure has decayed sufficiently

(7) Determine the cause of valve closure before resetting the MSIV isolation

(8) Observe turbine coastdown and break vacuum before the loss of sealing steam (check T-G auxiliaries for proper operation)

(9) Check that conditions are satisfactory prior to opening and resetting MSIVs

(10) Survey maintenance requirements and complete the scram report

15.2.4.2.2 Systems Operation

15.2.4.2.2.1 Closure of All Main Steamline Isolation Valves

MSIV closure initiates a reactor scram trip via position signals to the protection system.

Credit is taken for successful operation of the protection system.

The pressure relief system, which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation setpoints, is assumed to function normally during the time period analyzed.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.2.2.2 Closure of One Main Steamline Isolation Valve

A closure of a single MSIV at any given time will not initiate a reactor scram directly. This is because the valve position scram trip logic is designed to accommodate single valve operation and testability during normal reactor operation at limited power levels. Credit is taken for the operation of the pressure and flux signals to initiate a reactor scram.

All plant control systems maintain normal operation unless specifically designated to the contrary.

15.2.4.3 Core and System Performance

15.2.4.3.1 Input Parameters and Initial Conditions

The main steam isolation valves close in 3 to 4.5 seconds. The worst case (the 3 second closure time) is assumed in this analysis. No credit was taken for instrument delay.

Position switches on the valves initiate a reactor scram when the valves are less than 85% open. Closure of these valves causes the dome pressure to increase. Four RIPs are tripped when the high pressure setpoint is reached.

ABWR has motor-driven feedwater pumps. However, a conservative feedwater flow coastdown model was used in order to bound both the motor-driven and steam turbine driven feedwater pump designs.

15.2.4.3.2 Results

15.2.4.3.2.1 Closure of All Main Steamline Isolation Valves

Figure 15.2-9 shows the changes in important nuclear system variations for the simultaneous isolation of all main steamlines while the reactor is operating at 102% of NBR power. Neutron flux increases slightly, and fuel surface heat flux shows no increase.

Four RIPs are tripped due to high pressure. Water level decreases sufficiently to cause a trip of remaining 6 RIPs and the initiation of the RCIC system on the Level 2 (L2) trip at some time greater than 10 seconds. However, there is a delay up to 30 seconds before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.4.3.2.2 Closure of One Main Steamline Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to approximately 75 to 80% of design conditions in order to avoid high flux scram, high pressure scram, or full isolation from high steam flow in the “live” lines. With a 3 second closure of one MSIV during 102% rated power conditions, the steam flow disturbance may raise vessel pressure and reactor power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIVs at full power. No quantitative analysis is furnished for this event. However, no significant change in thermal margins is experienced and no fuel damage occurs. Peak pressure remains below SRV setpoints. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steamline Isolation Valves

The nuclear system relief valves begin to open at approximately 2.9 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 8.47 MPaG, below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline is 8.25 MPaG.

15.2.4.4.2 Closure of One Main Steamline Isolation Valve

No significant effect is imposed on the RCPB, since, if closure of the valve occurs at an unacceptable high operating power level, a flux or pressure scram may result. The main turbine bypass system continues to regulate system pressure via the other three open steamlines.

15.2.4.5 Radiological Consequences

15.2.4.5.1 General Observations

The radiological impact of transients involves consequences which do not lead to fuel rod damage as a direct result of the event itself. Additionally, many events do not lead to the depressurization of the primary system but only the venting of sensible heat and energy via fluids at coolant loop activity through relief valves to the suppression pool. In the case of previously defective fuel rods, a depressurization transient will result in considerably more fission product carryover to the suppression pool than hot-standby transients. The time duration of the transient varies from several minutes to more than four hours.

These observations lead to the realization that radiological aspects can involve a broad spectrum of results. For example:

(1) Transients where appropriate operator action (seconds) results in quick return (minutes) to planned operation, little radiological impact results.

(2) Where major RCPB equipment failure requires immediate plant shutdown and its attendant depressurization under controlled shutdown timetables (4 hours), the radiological impact is greater.

To envelope the potential radiological impact, a worst case like example No. 2 is described below. However, it should be noted that most transients are like example (1) and the radiological envelope conservatively overpredicts the actual radiological impact by a factor greater than 100.

15.2.4.5.2 Depressurization—Shutdown Evaluation

15.2.4.5.2.1 Fission Product Release from Fuel

While no fuel rods are damaged as a consequence of this event, fission product activity associated with normal coolant activity levels as well as that released from previously defective rods will be released to the suppression pool as a consequence of SRV actuation and vessel depressurization. The release of activity from previously defective rods is based in part upon measurements obtained from operating BWR plants (Reference 15.2-1).

Because each of those transients identified previously (which cause SRV actuation) will result in various vessel depressurization and steam blowdown rates, the transient evaluated in this section is that one which maximizes the radiological consequences for all transients of this nature. This transient is the closure of all main steamline isolation valves. The activity airborne in the containment is based on the analysis presented in Reference 15.2-1. The results of these analyses are presented in Table 15.2-10, which was used in evaluating the radiological dose consequences in this section.

15.2.4.5.2.2 Fission Product Release to Environment

Because this event does not result in the immediate need to purge the containment, it is assumed that purging of the containment through the SGTS occurs under average annual meteorological conditions and commences 8 hours after initiation of the event. The SGTS efficiency for iodine is 99% for organic forms and 99.9% for other forms. Reference 15.2-2 contains a description of the containment purge release model used. The integrated release to the environment is presented in Table 15.2-11.

15.2.4.5.3 Radiological Exposures

The offsite radiological doses for this event are presented in Table 15.2-12. COL applicants need to update the calculations to conform to the as-designed plant and site specific parameters (see Subsection 15.2.10 for COL license information). It should be noted that the radiological doses in the table are exposures per event. For the isolation transient, this event is not expected to occur more than 2.5 times per year; therefore, it is conservative to assume the yearly commitments for these transients will be ~2.5 times the individual values.

15.2.5 Loss of Condenser Vacuum

15.2.5.1 Identification of Causes and Frequency Classification

15.2.5.1.1 Identification of Causes

Various system malfunctions which can cause a loss of condenser vacuum due to some single equipment failure are designated in Table 15.2-13.

15.2.5.1.2 Frequency Classification

Although the frequency of occurrence of this event is expected to be infrequent, this event is categorized as an incident of moderate frequency.

15.2.5.2 Sequence of Events and Systems Operation

15.2.5.2.1 Sequence of Events

Table 15.2-14 lists the sequence of events for Figure 15.2-10.

15.2.5.2.1.1 Identification of Operator Actions

The operator should:

(1) Verify auto transfer of buses supplied by generator to incoming power—if automatic transfer has not occurred, manual transfer must be made

(2) Monitor and maintain reactor water level at required level

(3) Check turbine for proper operation of all auxiliaries during coastdown

(4) Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes

(5) Put the mode switch in the STARTUP position before the reactor pressure decays to <5.86 MPaG

(6) Secure the RCIC operation if the auto-initiation occurred due to low water level

(7) Monitor control rod drive positions and the SRNM

(8) Investigate the cause of the trip, make repairs as necessary, and complete the scram report

(9) Cooldown the reactor per standard procedure if a restart is not intended

15.2.5.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

Tripping functions incurred by sensing main turbine condenser vacuum are presented in Table 15.2-15.

15.2.5.3 Core and System Performance

15.2.5.3.1 Input Parameters and Initial Conditions

Turbine stop valves full stroke closure time is 0.1 seconds.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 85% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40% NBR power level.

The analysis presented here is a hypothetical case with a conservative 6.78 kPa/s vacuum decay rate. Thus, the bypass system is available for several seconds, because the bypass is signaled to close at a vacuum level of about 3.38 kPa less than the stop valve closure.

15.2.5.3.2 Results

Under this hypothetical 6.78 kPa/s vacuum decay condition, the turbine bypass valves and MSIV closure would follow main turbine trip about 5 seconds after it initiates the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of MSIV closure tends to be minimal, because the closure of main turbine stop valves, and subsequently the bypass valves, has already shut off the main steamline flow. Figure 15.2-10 shows the transient expected for this event. It is assumed that the plant is initially operating at 102% of NBR power conditions. Peak neutron flux reaches 122% of NBR power, while average fuel surface heat flux shows no increase. SRVs open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 8.30 MPaG at the vessel bottom. The overpressure transient is below the RCPB transient pressure limit of 9.48 MPaG. Vessel dome pressure does not exceed 8.14 MPaG. A comparison of these values to those for turbine trip at high power shows the similarities between these two transients. The prime difference is the subsequent main steamline isolation.

15.2.5.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5; therefore, the radiological exposures noted in Subsection 15.2.4.5 for Type 2 events cover the consequences of this event.

15.2.6 Loss of Non-Emergency AC Power to Station Auxiliaries

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

The non-emergency AC power to the station auxiliaries is provided by three unit auxiliary transformers. The unit auxiliary transformers are powered by the unit turbine/generator via a medium voltage generator breaker. Each unit auxiliary transformer (UAT) provides power to three electrical buses which provide the unit's auxiliary loads, including the reactor internal pumps (RIPs), as follows: UAT-A provides power to a RIP MG with 3 RIPs and a separate bus powers 2 RIPs directly (i.e. no MG), UAT-B powers 2 RIPs directly (i.e., no MG), and UAT-C provides power to a RIP MG with 3 RIPs. Following a generator trip and during plant startup, the medium voltage generator breaker is open but the high voltage breaker at the switchyard remains closed to backfeed power from the normal preferred power grid to the unit auxiliary transformers.

15.2.6.1.1.1 Loss of Unit Auxiliary Transformer

Causes for interruption or loss of power from the unit auxiliary transformers can arise from transformer (main or unit auxiliary) malfunction or isolated phase bus failures.

A loss of a unit auxiliary or main transformer is assumed to result in a generator trip and the opening of the generator and high voltage breakers. The generator trip will cause a reactor scram and an immediate trip of four RIPs not connected to M/G sets immediately. The opening of generator and high voltage breakers will result in a loss of power to all unit auxiliary transformers. However, the remaining six RIPs are powered by M/G sets. The M/G sets are capable of holding the RIPs at their original speeds for one second, then the RIPs will coastdown at a speed of 10%/s for two seconds, and trip at three seconds after the start of the event.

15.2.6.1.1.2 Loss of Grid Connections

Loss of grid connection can result from major shifts in electrical loads, loss of loads, lightning, storms, wind, etc., which contribute to electrical grid instabilities. These instabilities could cause equipment damage if unchecked. Protective relay schemes automatically disconnect electrical sources and loads to mitigate damage and regain electrical grid stability.

Should this occur, it would result in the same sequence of events as described above in Subsection 15.2.6.1.1.1.

15.2.6.1.2 Frequency Classification

15.2.6.1.2.1 Loss of Unit Auxiliary Transformer

Although the frequency of this event is low enough to be an infrequent event, this transient disturbance is analyzed as an incident of moderate frequency.

15.2.6.1.2.2 Loss of Grid Connections

Although the frequency of this event is low enough to be an infrequent event, this transient disturbance is analyzed as an incident of moderate frequency.

15.2.6.2 Sequence of Events and Systems Operation

15.2.6.2.1 Sequence of Events

15.2.6.2.1.1 Loss of Unit Auxiliary Power Transformer

Table 15.2-16 lists the sequence of events for Figure 15.2-11.

15.2.6.2.1.2 Loss of Grid Connections

This event is similar to a loss of unit auxiliary transformer as discussed in Subsection 15.2.6.2.1.1.

15.2.6.2.1.3 Identification of Operator Actions

The operator should (1) maintain the reactor water level by use of the RCIC System and control reactor pressure by use of the safety/relief valves, (2) verify that the turbine DC oil pump is operating satisfactorily to prevent turbine bearing damage, and (3) verify proper switching and loading of the emergency diesel generators.

The following is the sequence of operator actions expected during the course of the events when no immediate restart is assumed. The operator should:

(1) Verify all rods are in

(2) Check that diesel generators start and carry the vital loads

(3) Check that the RCIC System starts when reactor vessel level drops to the initiation point after the relief valves open

(4) Break vacuum before the loss of sealing steam occurs

(5) Check T-G auxiliaries during coastdown

(6) Secure the RCIC System when both reactor pressure and level are under control

(7) Continue cooldown per the normal procedure

(8) Complete the scram report and survey the maintenance requirements

15.2.6.2.2 Systems Operation

15.2.6.2.2.1 Loss of Unit Auxiliary Transformer

This event, unless otherwise stated, assumes and takes credit for normal functioning of plant instrumentation and controls, plant protection and reactor protection systems.

The reactor is subjected to a complex sequence of events when the plant loses all auxiliary power. Estimates of the responses of the various reactor systems (assuming loss of the unit auxiliary transformer) provide the following simulation sequence:

(1) A generator trip occurs at time = 0, which initiates a scram and trip of four RIPs (already tripped at time = 0, see (2) below).

(2) All electrical pumps, including feedwater pumps, including 4 RIPs not connected to the M/G sets, are tripped at a reference time, t = 0, with normal coastdown times for the reactor internal pumps.

(3) The remaining six RIPs powered by M/G sets are capable of maintaining their original speeds for one second, then coast down at a speed of 10%/s for two seconds, and trip at three seconds after the start of the event.

(4) The loss of the main condenser circulating water pumps, occurs at the same time normal power is lost to the system (t=0) which is conservatively assumed to cause the condenser vacuum to drop to the main turbine trip setting, within 8 seconds causing stop valve closure, assuming 1.69 kPa/s vacuum decay rate.

(5) At approximately 28 seconds, the loss of condenser vacuum is expected to reach the MSIV and bypass valves closure setpoint and initiate steamline isolation.

Operation of the RCIC System function is not simulated in this analysis as its operation occurs at some time beyond the primary concerns of fuel thermal margin and overpressure effects of this analysis.

15.2.6.2.2.2 Loss of Grid Connections

Same as Subsection 15.2.6.2.2.1.

15.2.6.3 Core and System Performance

15.2.6.3.1 Loss of Unit Auxiliary Power Transformer

Figure 15.2-11 shows graphically the simulated transient. The initial portion of the transient is similar to the load rejection transient. At eight seconds, the turbine trips, on low condenser vacuum. Main steamline isolation valves and turbine bypass valves close at 28 seconds on their condenser vacuum setpoint.

Sensed level drops to the RCIC initiation setpoint at approximately 21 seconds after loss of auxiliary power.

There is no significant increase in fuel temperature or decrease in the MCPR value, fuel thermal margins are not threatened and the design basis is satisfied. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.6.3.2 Loss of Grid Connections

Same as Subsection 15.2.6.3.1. This event does not have to be reanalyzed for specific core configurations.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Unit Auxiliary Transformer

The consequences of this event do not result in any significant temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints. The pressure at the bottom of the vessel is limited to a maximum value of 8.28 MPaG, which is below the vessel pressure limit of 9.48 MPaG.

15.2.6.4.2 Loss of Grid Connections

Same as Subsection 15.2.6.4.1.

15.2.6.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5; therefore, the radiological exposures noted in Subsection 15.2.4.5 for Type 2 events cover the consequences of this event.

15.2.7 Loss of Feedwater Flow

15.2.7.1 Identification of Causes and Frequency Classification

15.2.7.1.1 Identification of Causes

A loss of feedwater flow could occur from pump failures, loss of electrical power, operator errors, or reactor system variables such as a high vessel water level (L8) trip signal.

15.2.7.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.2.7.2 Sequence of Events and Systems Operation

15.2.7.2.1 Sequence of Events

Table 15.2-17 lists the sequence of events for Figure 15.2-12.

15.2.7.2.1.1 Identification of Operator Actions

The operator should ensure RCIC actuation so that water inventory is maintained in the reactor vessel. Additionally, the operator should monitor reactor water level and pressure control and T-G auxiliaries during shutdown.

The following is the sequence of operator actions expected during the course of the event when no immediate restart is assumed. The operator should:

(1) Verify all rods in, following the scram

(2) Verify trip of four RIPs

(3) Verify RCIC initiation

(4) Verify that the remaining recirculation pumps trip on reactor low level (L2)

(5) Continue operation of the RCIC System until decay heat diminishes to a point where the RHR System can be put into service

(6) Monitor turbine coastdown, break vacuum as necessary

(7) Complete scram report and survey maintenance requirements

15.2.7.2.2 Systems Operation

Loss of feedwater flow results in a reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. The Reactor Protection System responds within one second after this trip to scram the reactor. The low level (L3) scram trip function meets the single-failure criterion. Four of the RIPs are tripped at Level 3.

15.2.7.3 Core and System Performance

The results of this transient simulation are presented in Figure 15.2-12. Feedwater flow terminates at approximately 5 seconds. Subcooling decreases, causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 10 seconds. Water level continues to drop until, first, the recirculation flow is runback at Level 4 (L4) and then the vessel level (L3) scram trip setpoint is reached, whereupon the reactor is shut down and the four RIPs are tripped. Vessel water level continues to drop to the L2 trip. At this time, the remaining six RIPs are tripped and the RCIC operation is initiated. MCPR remains considerably above the safety limit, because increases in heat flux are not experienced. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.2.7.4 Barrier Performance

The consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.2.7.5 Radiological Consequences

The consequences of this event do not result in any fuel failure. Therefore, no analysis of the radiological consequences is required.

15.2.8 Feedwater Line Break

Refer to Subsection 15.6.6.

15.2.9 Failure of RHR Shutdown Cooling

The RHR System performs low pressure core cooling, containment heat removal, containment spray and shutdown cooling functions. The RHR System has three independent divisions, each of which contains the necessary piping, pumps, valves, heat exchangers, instrumentation and electrical power for operation. Each division also has its own cooling water supply, diesel generator and room cooling system. For the shutdown cooling function, each division has its own suction line from and return line to the RPV. Thus, each of the three RHR divisions is completely independent of the other divisions in its shutdown cooling function. The RHR System reduces the primary system temperature to 51.7°C within 24 hours of plant shutdown.

Normally, in evaluating component failure considerations associated with RHR System shutdown cooling mode operation, active pumps, valves or instrumentation would be assumed to fail. If the single active failure criterion is applied to the RHR System, one of the three RHR divisions would be inoperable. However, the two operable RHR divisions could achieve cold shutdown to 100°C within 36 hours after reactor shutdown.

Failure of offsite power is another case which could affect the shutdown cooling function. The plant will have two independent offsite power supplies. If either or both offsite power supplies are lost, each RHR division has its own diesel generator which will permit operating that division at its rated capacity. Application of the single active failure criterion would still leave two RHR divisions operational.

The RHR System description and performance evaluation in Subsection 5.4.7 describes the models, assumptions and results for shutdown cooling with two RHR divisions operational.

15.2.10 COL License Information

15.2.10.1 Radiological Effects of MSIV Closures

COL applicants will evaluate the radiological effects of the inadvertent closure of MSIVs for the final plant design and the site parameters (Subsection 15.2.4.5.3).

15.2.11 References

15.2-1 F. G. Brutshscy, et al., “Behavior of Iodine in Reactor Water During Plant Shutdown and Startup”, August 1972 (NEDO-10585).

15.2-2 H. Careway, V. Nguyen, and P. Stancavage, “Radiological Accident—The CONAC03 CODE”, December 1981 (NEDO-21143-1).

Table 15.2-1a Sequence of Events for Figure 15.2-1

Time (s)	Event
0	Simulate one main turbine control valve to fast close.
0	Failed turbine control valve starts to close.
3.0	Neutron flux reaches high flux scram setpoint and initiates a reactor scram.
2.8	Turbine bypass valves start to open.
8.1	Water level reaches Level 3 setpoint. Four RIPs are tripped.

Table 15.2-1b Sequence of Events for Figure 15.2-1a

TIme (s)	Event
0	Simulate one main turbine control valve to slow close.
0	Failed turbine control valve starts to close.
16.0	Neutron flux reaches its peak. No scram is initiated.
15.6	Turbine bypass valves start to open.
~30	Reactor power settles back to steady state.

Table 15.2-2 Sequence of Events for Figure 15.2-2

Time (s)	Event
0	Simulate zero steam flow demand to main turbine and bypass valves.
0	Turbine control valves start to close.
1.0	Neutron flux reaches high flux scram setpoint and initiates a reactor scram.
2.4	Four RIPs are tripped due to high dome pressure.
2.6	Safety/relief valves open due to high pressure.
8.9	Safety/relief valves close.
9.4	Group 1 safety/relief valves open again to relieve decay heat
9.8	Group 2 safety/relief valves open again to relieve decay heat.
15 (est.)	Safety/relief valves close.

Table 15.2-3 Sequence of Events for Figure 15.2-3

Time (s)	Event
–0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure and main turbine bypass system operation.
0.0	Fast control valve closure (FCV) initiates reactor scram and a trip of four RIPs.
0.07	Turbine control valves closed.
0.1	Turbine bypass valves start to open.
1.9	Safety/relief valves open due to high pressure.
7.0	Safety/relief valves close.

Table 15.2-4 Sequence of Events for Figure 15.2-4

Time (s)	Event
–0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure and main turbine bypass system operation.
0.0	One turbine bypass valve fails to operate on demand.
0.0	Fast control valve closure (FCV) initiates reactor scram and a trip of four RIPs.
0.07	Turbine control valves closed.
0.1	Remaining bypass valves start to open.
1.6	Safety/relief valves open due to high pressure.
6.9	Safety/relief valves close.

Table 15.2-5 Sequence of Events for Figure 15.2-5

TIme (s)	Event
–0.015	Turbine-generator detection of loss of electrical load.
0.0	Turbine-generator load rejection sensing devices trip to initiate turbine control valves fast closure.
0.0	Turbine bypass valves fail to operate.
0.0	Fast control valve closure (FCV) initiates reactor scram and trip of four RIPs.
0.07	Turbine control valves closed.
1.3	Safety/relief valves open due to high pressure.
7.6	Safety/relief valves close.
8.3	Safety/relief valves open again to relieve decay heat.
>15.0 (est.)	Safety/relief valves close again.

Table 15.2-6 Sequence of Events for Figure 15.2-6

Time (s)	Event
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 85% open position and initiate reactor scram and trip of four RIPs.
0.1	Turbine stop valves close.
0.1	Turbine bypass valves start to open to regulate pressure.
2.0	Safety/relief valves open due to high pressure.
6.9	Safety/relief valves close.

Table 15.2-7 Sequence of Events for Figure 15.2-7

Time (s)	Event
0.0	Turbine trip initiates closure of main stop valves.
0.0	One turbine bypass valve fails to operate.
0.01	Main turbine stop valves reach 85% open position and initiate reactor scram and trip of four RIPs.
0.1	Turbine stop valves close.
0.1	Remaining bypass valves start to open.
1.7	Safety/relief valves open due to high pressure.
6.8	Safety/relief valves close.

Table 15.2-8 Sequence of Events for Figure 15.2-8

Time (s)	Event
0.0	Turbine trip initiates closure of main stop valves.
0.0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 85% open position and initiate reactor scram and a trip of four RIPs.
0.1	Turbine stop valves close.
1.4	Safety/relief valves open due to high pressure.
7.4	Safety/relief valves close.
8.2	Safety/relief valves open again to relieve decay heat.
>14.0 (est.)	Safety/relief valves close again.

Table 15.2-9 Sequence of Events for Figure 15.2-9

Time (s)	Event
0.0	Closure of all main steamline isolation valves (MSIV).
0.45	MSIVs reach 85% open.
0.45	MSIVs position trip scram initiated.
2.6	Four RIPs are tripped due to high reactor pressure.
2.9	Safety/relief valves open due to high pressure.
7.8	Safety/relief valves close.
9.0	Safety/relief valves open again to relieve decay heat.
>10.0 (est.)	Vessel water level reaches L2 setpoint. RCIC is initiated. The remaining six RIPs are tripped.
>13.0 (est.)	Group 1 safety/relief valves close again.
>40.0 (est.)	RCIC flow into vessel (not included in simulation).

Table 15.2-10 Post-Transient Primary Containment Inventory (Air Plus Water) (megabecqueral)

Isotope	1 Min	10 Min	1 Hour	2 Hour	4 Hour	8 Hour*	12 Hour	1 Day	4 Days	30 Days
I-131	1.1E+07	1.E+07	1.1E+07	1.0E+07	8.5E+06	6.3E+06	4.8E+06	1.9E+06	9.3E+03	
I-132	1.7E+07	1.6E+07	1.1E+07	8.1E+06	3.7E+06	8.5E+05	1.9E+05	2.2E+03	4.8E+09	
I-133	2.4E+07	2.3E+07	2.1E+07	1.9E+07	1.6E+07	1.0E+07	7.0E+06	2.0E+06	1.1E+03	
I-134	2.6E+07	2.3E+07	1.1E+07	4.8E+06	8.5E+05	2.7E+04	8.5E+02	2.7E+02		
I-135	2.3E+07	2.2E+07	1.9E+07	1.6E+07	1.1E+07	5.6E+06	2.8E+06	3.3E+05	1.0E+00	
Totals	1.0E+08	9.5E+07	7.4E+07	5.8E+07	4.0E+07	2.3E+07	1.5E+07	4.3E+06	1.0E+04	
KR-83M	7.4E+06	7.0W+06	7.0E+06	5.2E+06	3.6E+06	1.7E+06	3.7E+05	4.4E+04	7.8E+01	
KR-85M	1.6E+07	1.6E+07	1.4E+07	1.2E+07	8.9E+06	4.8E+06	1.4E+06	3.6E+04	1.0E+05	
KR-85	7.4E+05	7.4E+05	7.4E+05	7.4E+05	7.4E+05	7.4E+05	4.1E+05	6.7E+04	1.3E+00	
KR-87	3.1E+07	2.8E+07	1.8E+07	1.0E+07	3.5E+06	4.1E+05	2.5E+04	5.9E+00		
KR-88	4.4E+07	4.4E+07	3.5E+07	2.7E+07	1.7E+07	6.3E+06	1.3E+06	1.1E+04	5.2E-09	
KR-89	4.4E+07	6.3E+06	1.1E+02	2.2E-04						
XE131M	3.7E+05	3.7E+05	3.7E+05	3.7E+05	3.7E+05	3.7E+05	2.0E+05	3.3E+04	5.6E-01	
XE133M	5.6E+06	5.6E+06	5.6E+06	5.6E+06	5.2E+06	5.2E+06	2.6E+06	3.7E+05	2.8E+00	
XE-133	1.3E+08	1.3E+08	1.3E+08	1.3E+08	1.3E+08	1.3E+08	6.7E+07	1.1E+07	1.4E+02	
XE135M	2.4E+07	1.6E+07	1.8E+06	1.2E+05	5.9E+02	1.5E-02	2.0E-07			
XE-135	1.7E+07	1.7E+07	1.6E+07	1.5E+07	1.4E+07	9.3E+06	3.7E+06	2.5E+05	2.0E-02	
XE-137	1.0E+08	1.9E+07	2.3E+03	4.4E-02	1.6E+11					
XE-138	1.1E+08	6.7E+07	5.9E+06	3.1E+05	8.9E+02	7.0E-03	3.1E-08			
Totals	5.3E+08	3.6E+08	2.4E+08	2.1E+08	1.8E+08	1.6E+08	7.6E+07	1.1E+07	1.5E+02	

* Beginning of Containment Purge

Table 15.2-11 Activity Released to the Environment (megabecqueral)

Isotope	12 Hours	1 Day	4 Days	30 Days
I-131	1.0E+02	2.4E+02	3.2E+02	3.2E+02
I-132	8.5E+00	1.1E+01	1.1E+01	1.1E+01
I-133	1.6E+02	3.4E+02	4.1E+02	4.1E+02
I-134	1.5E-01	1.5E-01	1.5E-01	1.5E-01
I-135	7.8E+01	1.3E+02	1.4E+02	1.4E+02
TOTALS	3.5E+02	7.2E+02	8.7E+02	8.7E+02
KR-83M	9.3E+04	1.0E+05	1.0E+05	1.0E+05
KR-85M	1.6E+06	2.3E+06	2.3E+06	2.3E+06
KR-85	3.3E+05	6.7E+05	7.4E+05	7.4E+05
KR-87	8.1E+04	8.5E+04	8.5E+04	8.5E+04
KR-88	1.9E+06	2.4E+06	2.4E+06	2.4E+06
KR-89	Insignificant			
XE131M	1.7E+05	3.4E+05	3.7E+05	3.7E+05
XE133M	2.2E+06	4.4E+06	4.4E+06	4.4E+06
XE-133	5.9E+07	1.1E+08	1.2E+08	1.2E+08
XE135M	7.8E-04	7.8E-04	7.8E-04	7.8E-04
XE-135	3.7E+06	5.9E+06	6.3E+06	6.3E+06
XE-137	Insignificant			
XE-138	3.4E-04	3.4E-04	3.4E-04	3.4E-04
TOTALS	6.9E+07	1.3E+08	1.4E+08	1.4E+08

Table 15.2-12 Dose Evaluation and Meteorology

Dispersion sec/m^3	Thyroid mGy	W Body mGy	Beta mGy	Skin mGy
1.0E-5	3.0E-4	8.5E-3	1.3E-2	2.2E-2
5.0E-6	1.5E-4	4.3E-3	6.6E-3	1.1E-2
1.0E-6	3.0E-5	8.5E-4	1.3E-3	2.2E-3
5.0E-7	1.5E-5	4.3E-4	6.6E-4	1.1E-3
1.0E-7	3.0E-6	8.5E-5	1.3E-4	2.2E-4

Table 15.2-13 Typical Rates of Decay for Condenser Vacuum

	Cause	Estimated Vacuum Decay Rate
(1)	Failure of Isolation of Steam Jet Air Ejectors	<3.32 kPa/min
(2)	Loss of Sealing Steam to Shaft Gland Seals	Approximately 3.32 to 6.79 kPa/min
(3)	Opening of Vacuum Breaker Valves	Approximately 6.79 to 40.66 kPa/min
(4)	Loss of One or More Circulating Water Pumps	Approximately 13.56 to 81.10 kPa/min

Table 15.2-14 Sequence of Events for Figure 15.2-10

Time (s)	Event
–3.0	Initiate simulated loss of condenser vacuum at 6.79 kPa/s.
0.0	Low condenser vacuum main turbine trip actuated.
0.01	Main turbine trip initiates scram and a trip of four RIPS.
2.0	Safety/relief valves open due to high pressure.
5.0	Low condenser vacuum initiates main steamline isolation valve closure.
5.0	Low condenser vacuum initiates main bypass valve closure.
7.0	Safety/relief valves close.
8.9	Safety/relief valves open again to relieve decay heat.
13.7	Vessel water level reaches L2 setpoint and initiates RCIC.
>16.0 (est.)	Safety/relief valves close again.
>40.0 (est.)	RCIC flow enters vessel (not included in simulation).

Table 15.2-15 Trip Signals Associated with Loss of Condenser Vacuum

Vacuum (cm of Hg)	Protective Action Initiated
69 to 71	Normal Vacuum Range
51 to 58	Main Turbine Trip (Stop Valve Closures)
18 to 25	Mainsteam Line Isolation Valve (MSIV) Closure and Bypass Valve Closure

Table 15.2-16 Sequence of Events for Figure 15.2-11

Time (s)	Event
0.0	Loss of unit auxiliary power transformer, which initiates a generator trip.
0.0	Turbine control valve fast closure is initiated.
0.0	Turbine control valve fast closure initiates main turbine bypass system operation.
0.0	Four RIPs which are not connected to the M/G sets are tripped.
0.0	Fast control valve closure (FCV) initiates a reactor scram.
0.0	Feedwater pumps are tripped
0.07	Turbine control valves closed.
0.1	Turbine bypass valves start to open
1.9	Safety/relief valves open due to high pressure.
3.0	Remaining six RIPs powered by M/G sets are tripped.
6.3	Turbine bypass valves close, after operating for 6 seconds.
7.1	Safety/relief valves close.
7.8	Safety/relief valves open again to relieve decay heat.
8.0	Turbine trips due to low condenser vacuum.
15.6	Safety/relief valves close again.
18.3	Safety/relief valves open again to relieve decay heat.
21.0 (est.)	Vessel water level reaches L2 setpoint.
24.0 (est.)	Safety/relief valves close again.
28.0 (est.)	Closure of MSIV and turbine bypass valves is initiated via low condenser vacuum (not simulated).
51.0 (est.)	RCIC flow enters vessel (not simulated).

Table 15.2-17 Sequence of Events for Figure 15.2-12

Time (s)	Event
0	Trip of all feedwater pumps initiated.
2.8	Vessel water level reaches Level 4 and initiates recirculation flow runback.
5	Feedwater flow decays to zero.
7.5	Vessel water level (L3) trip initiates reactor scram and trip of four RIPs.
19.5	Vessel water level reaches Level 2. The remaining six RIPs are tripped. RCIC is initiated.
49.5	RCIC flow enters vessel.

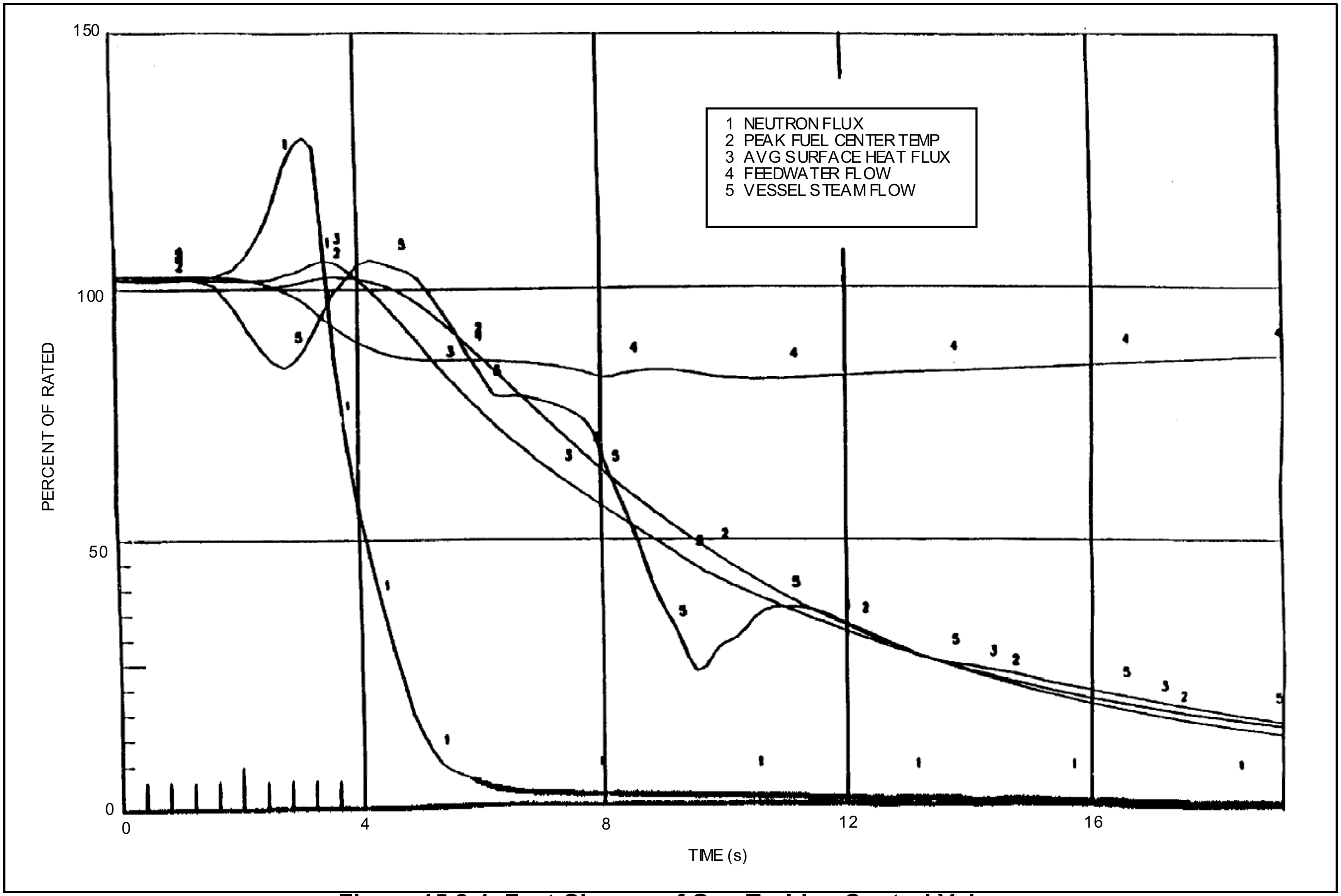


Figure 15.2-1 Fast Closure of One Turbine Control Valve

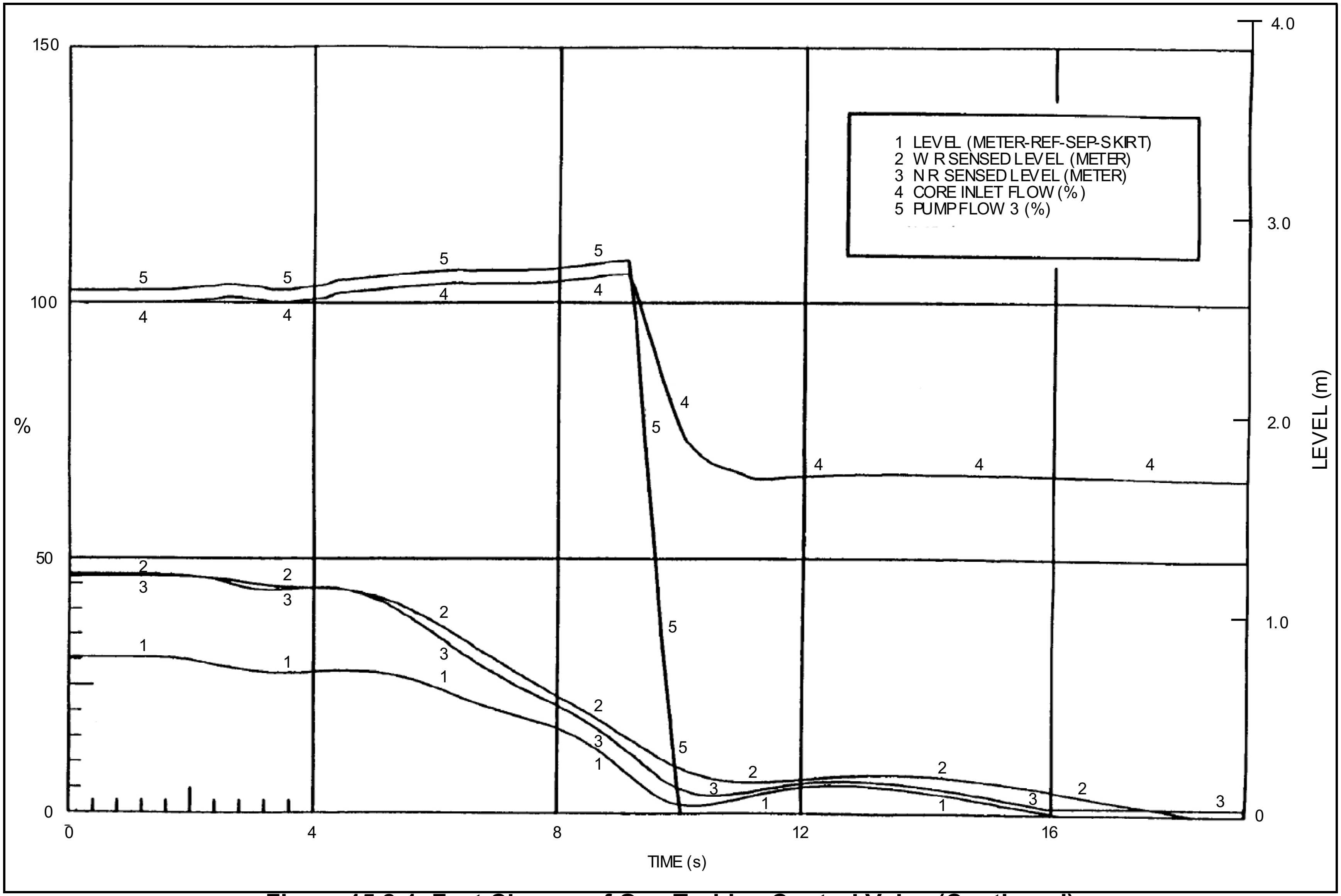


Figure 15.2-1 Fast Closure of One Turbine Control Valve (Continued)

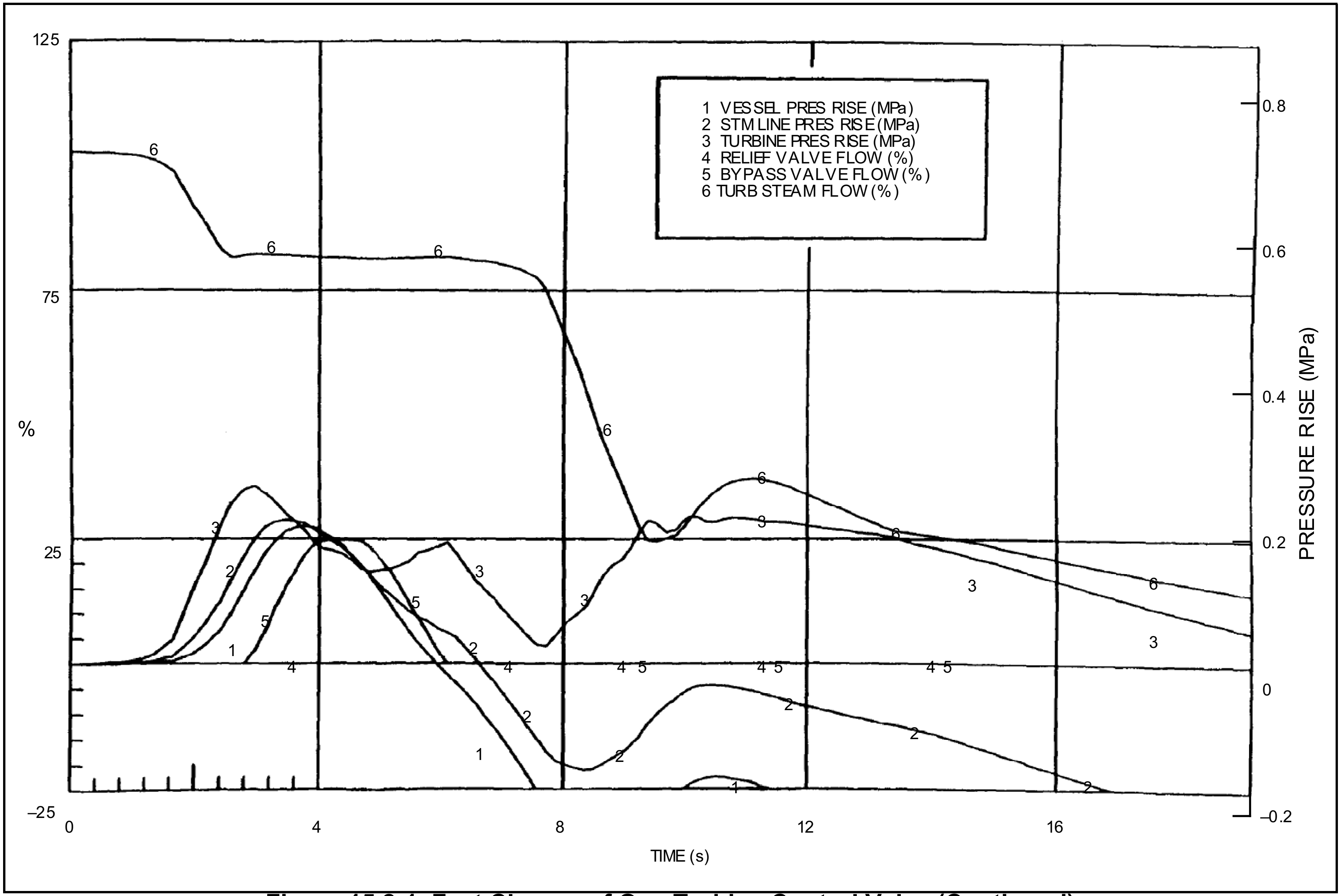


Figure 15.2-1 Fast Closure of One Turbine Control Valve (Continued)

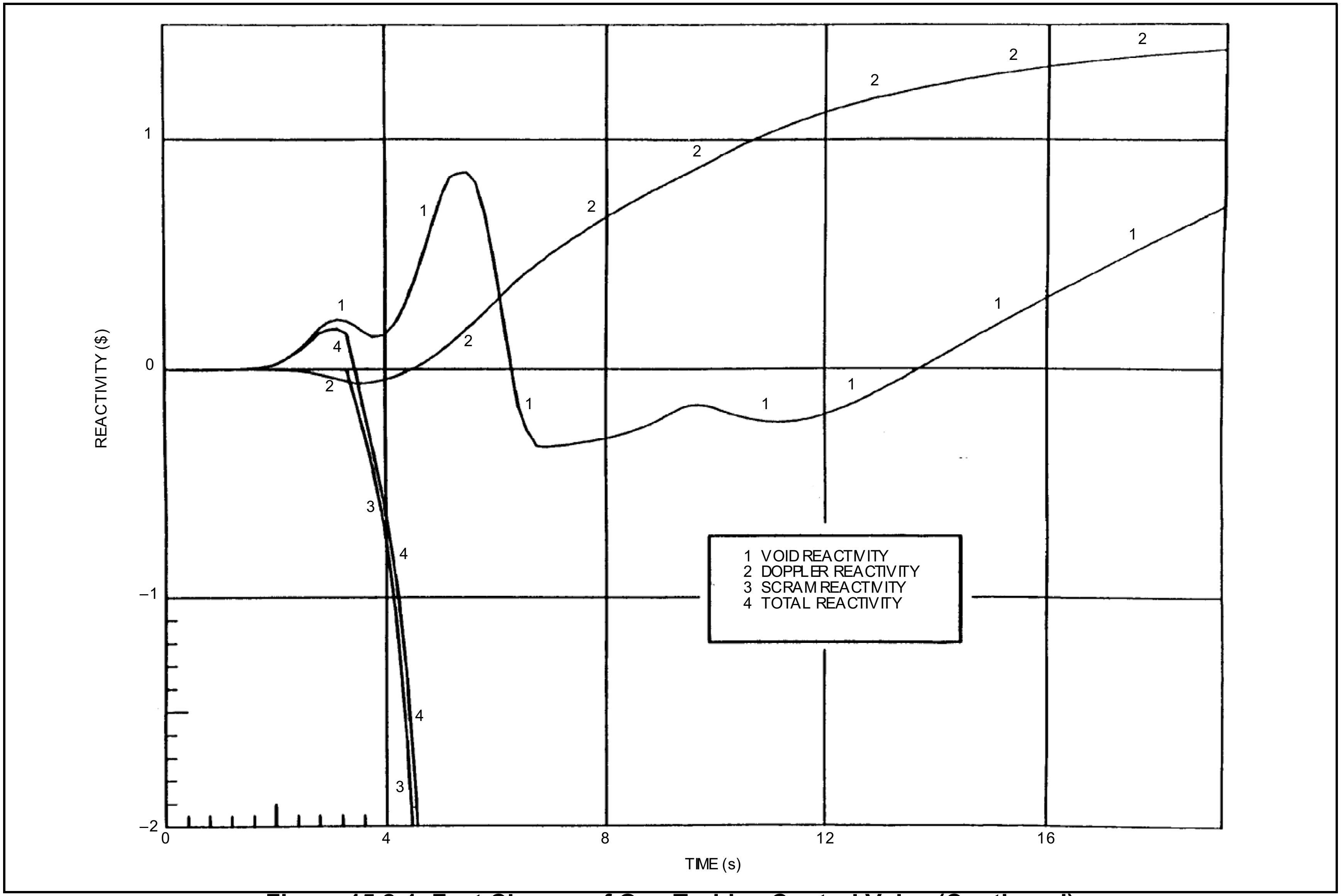


Figure 15.2-1 Fast Closure of One Turbine Control Valve (Continued)

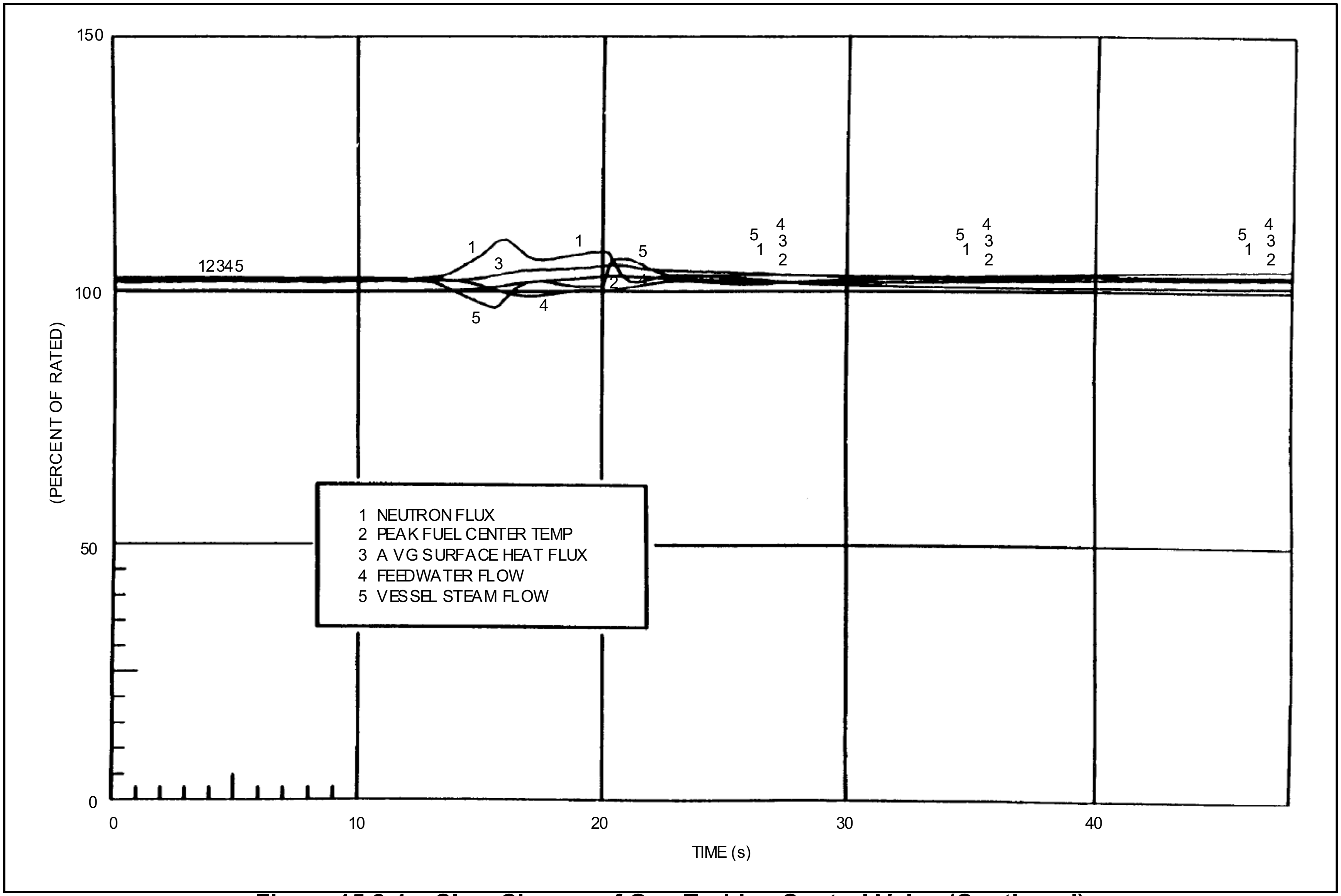


Figure 15.2-1a Slow Closure of One Turbine Control Valve (Continued)

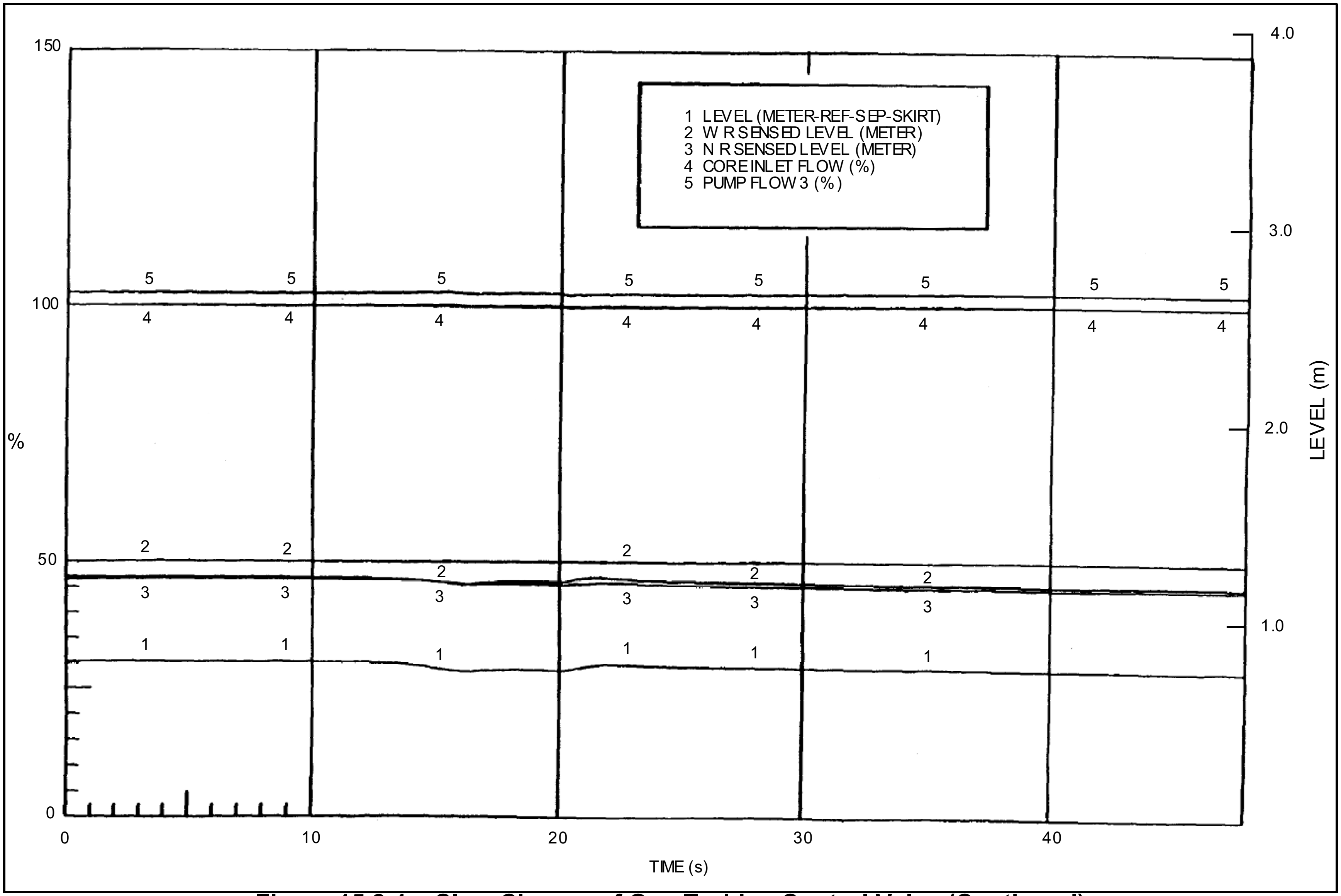


Figure 15.2-1a Slow Closure of One Turbine Control Valve (Continued)

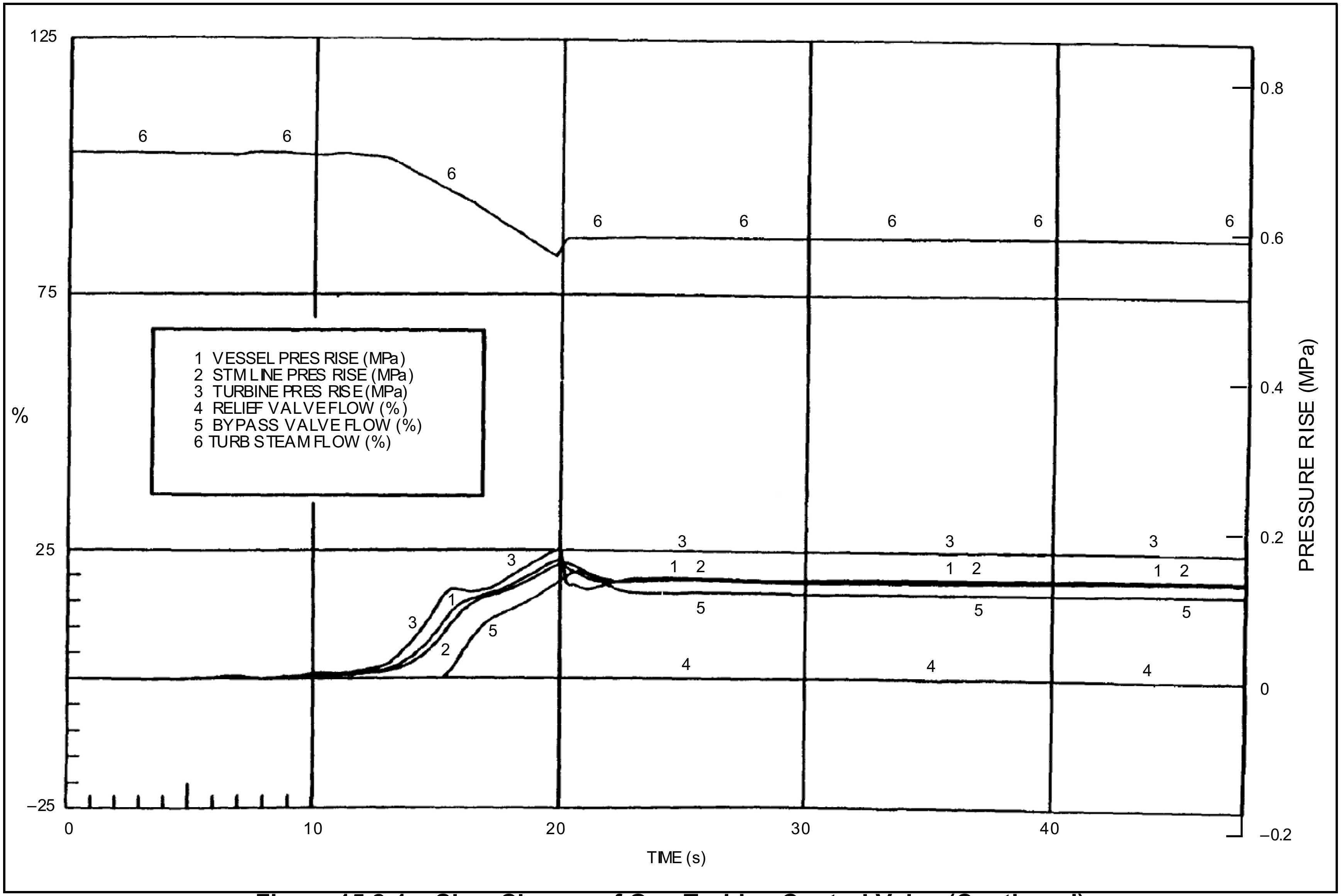


Figure 15.2-1a Slow Closure of One Turbine Control Valve (Continued)

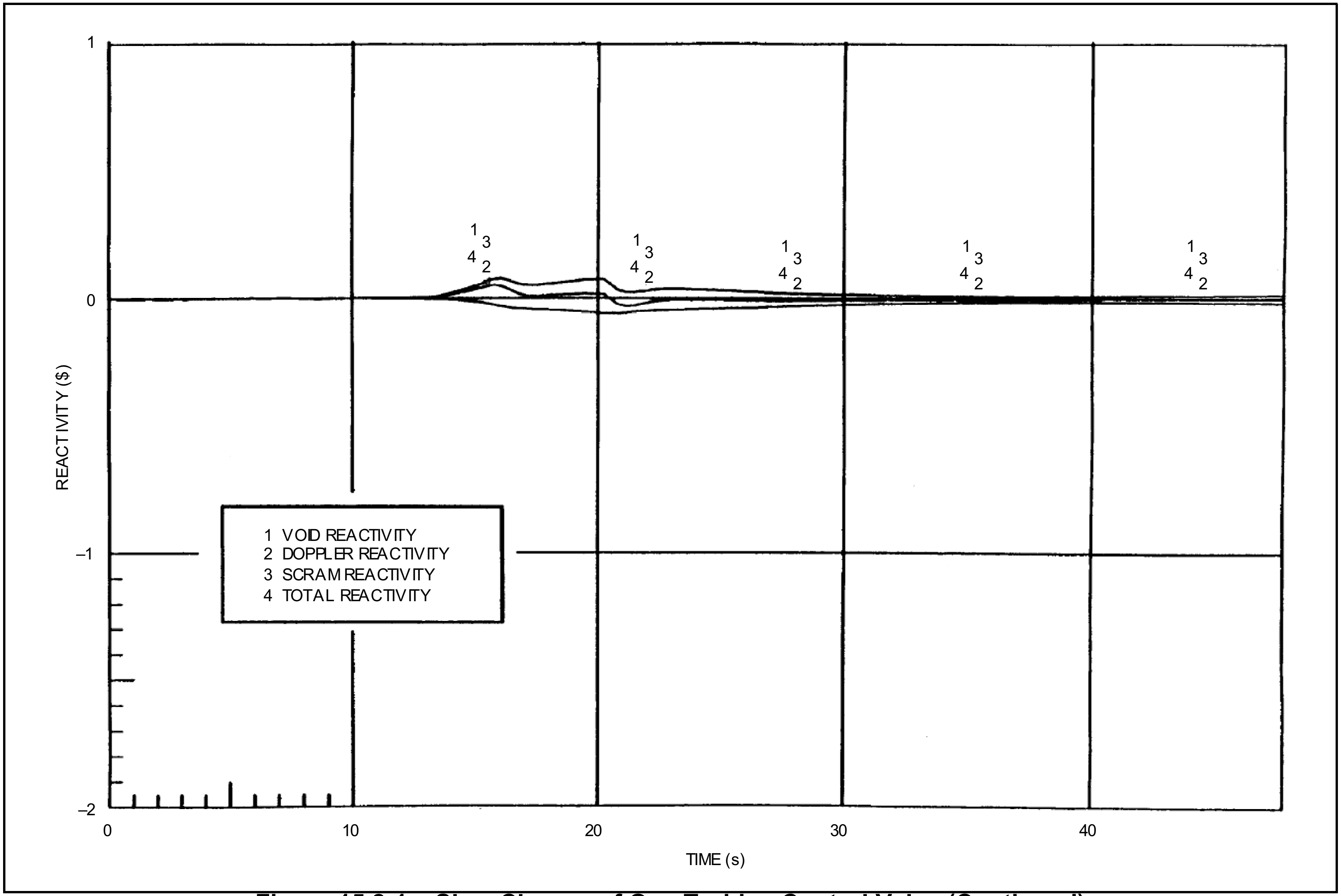


Figure 15.2-1a Slow Closure of One Turbine Control Valve (Continued)

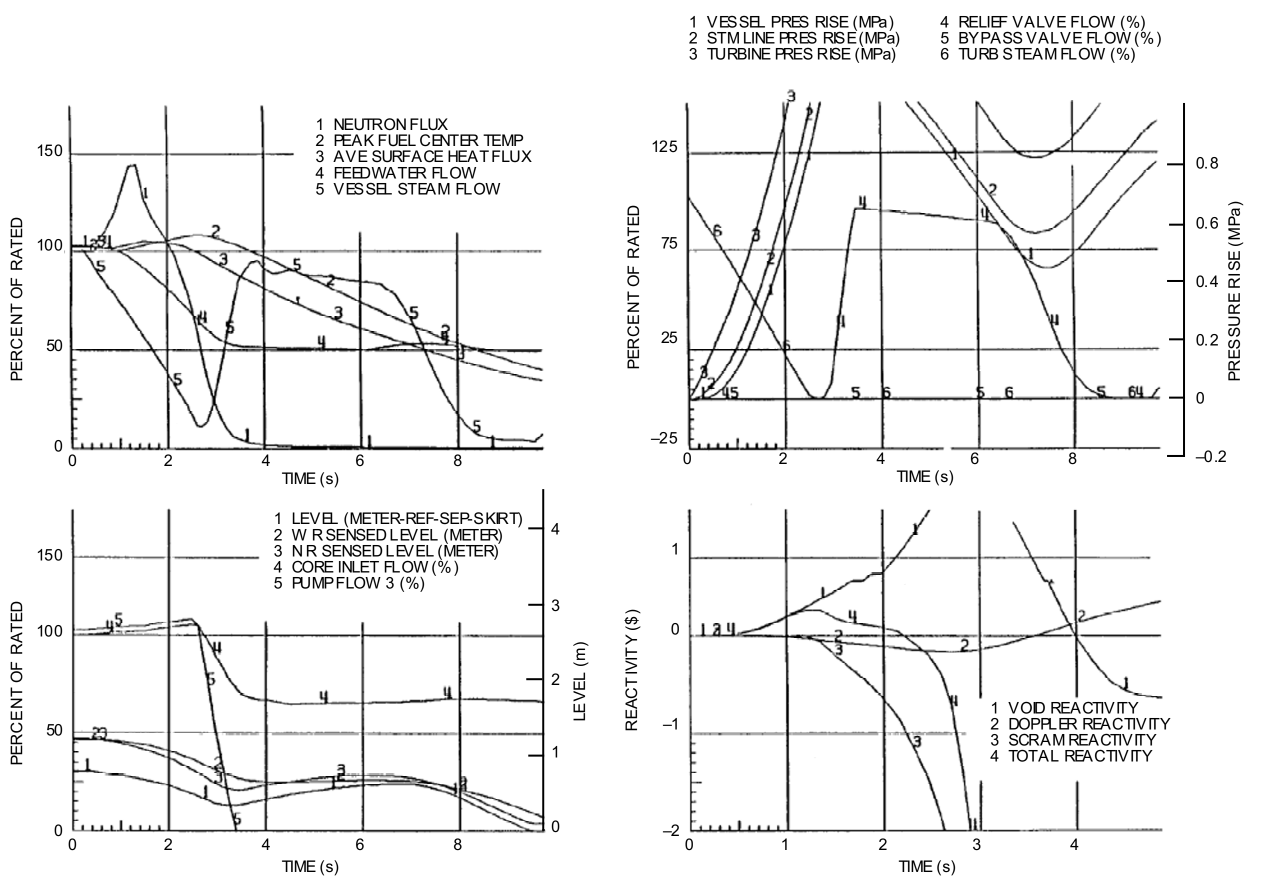


Figure 15.2-2 Pressure Regulator Downscale Failure

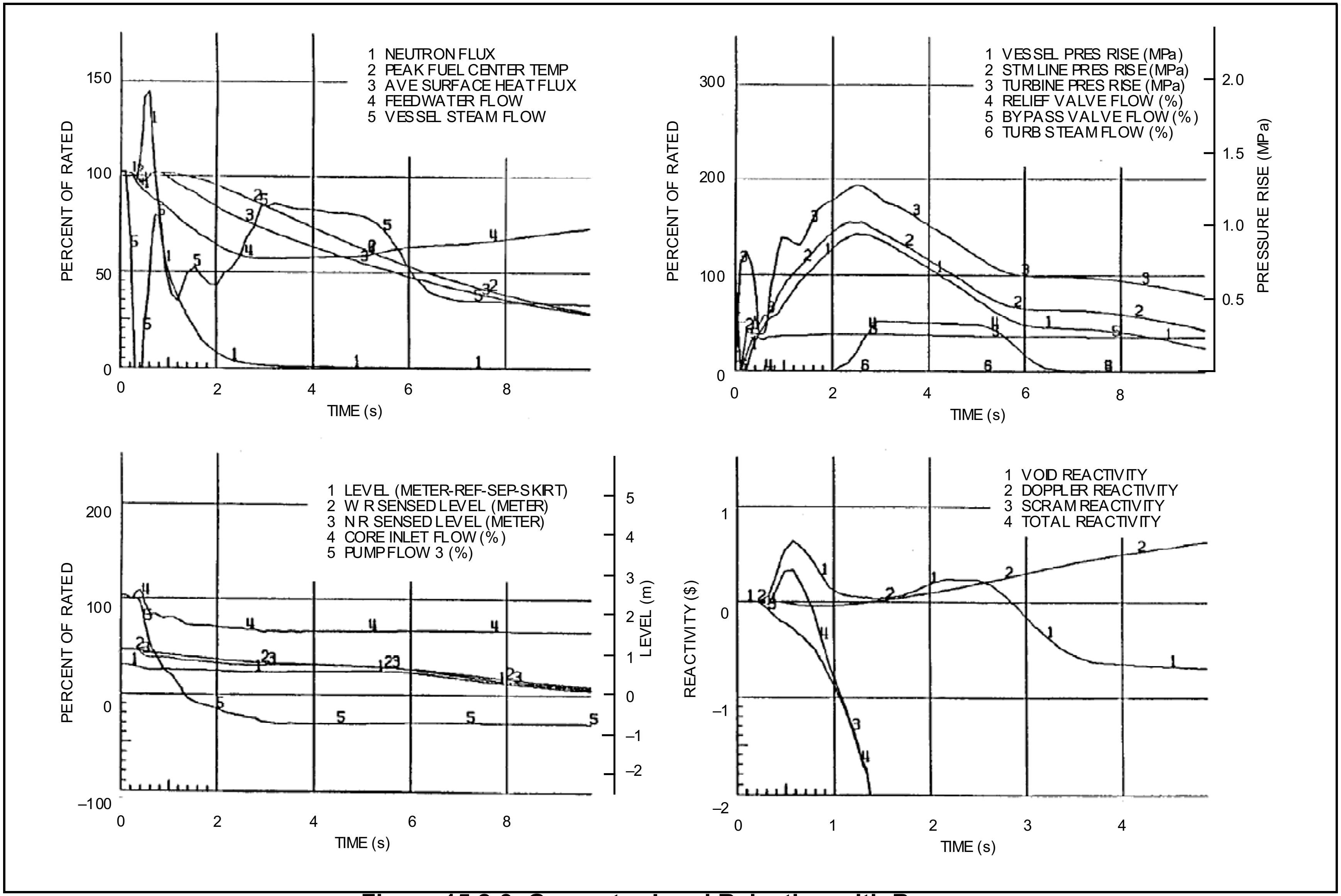


Figure 15.2-3 Generator Load Rejection with Bypass

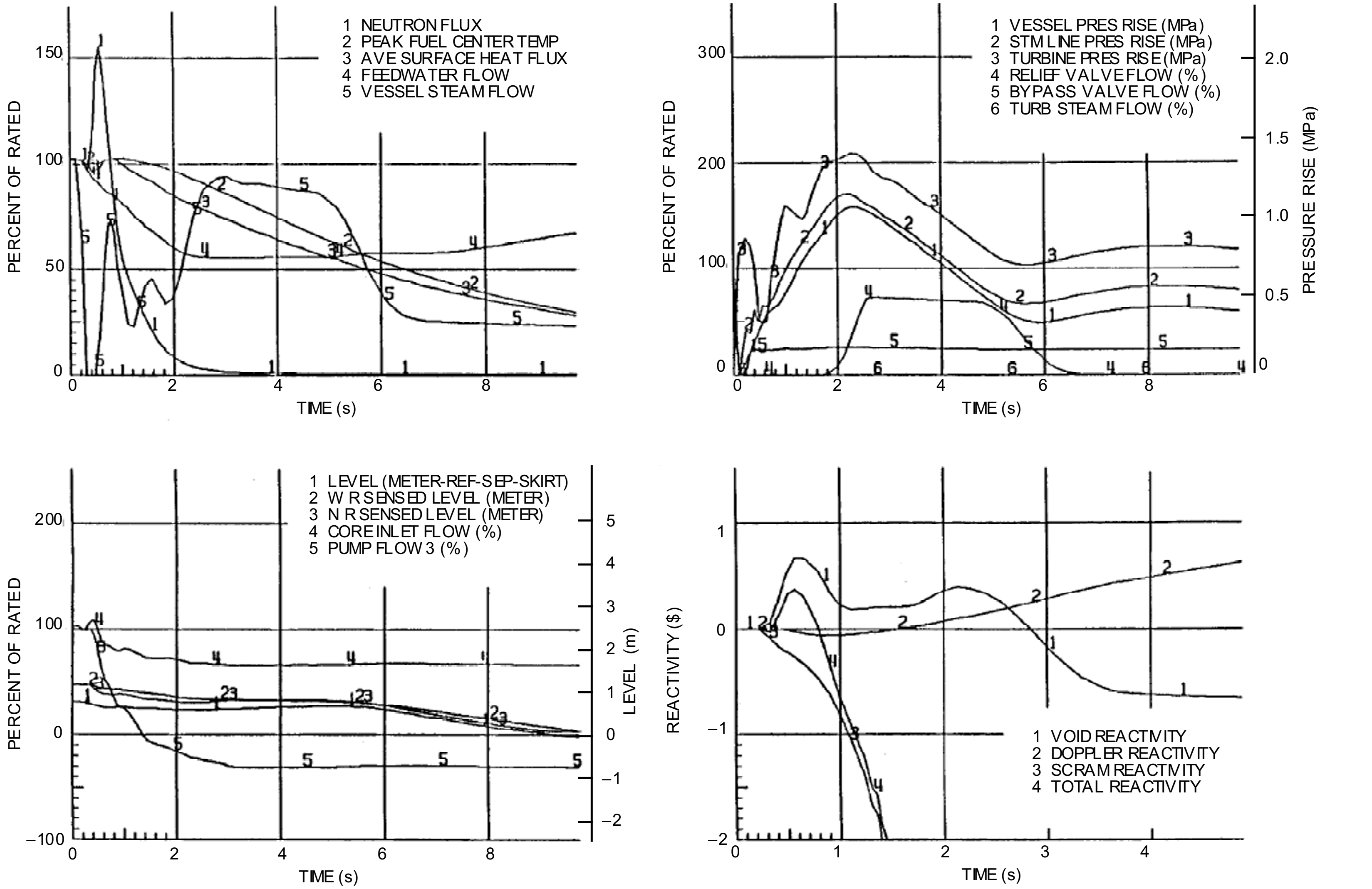


Figure 15.2-4 Load Rejection with One Bypass Valve Failure

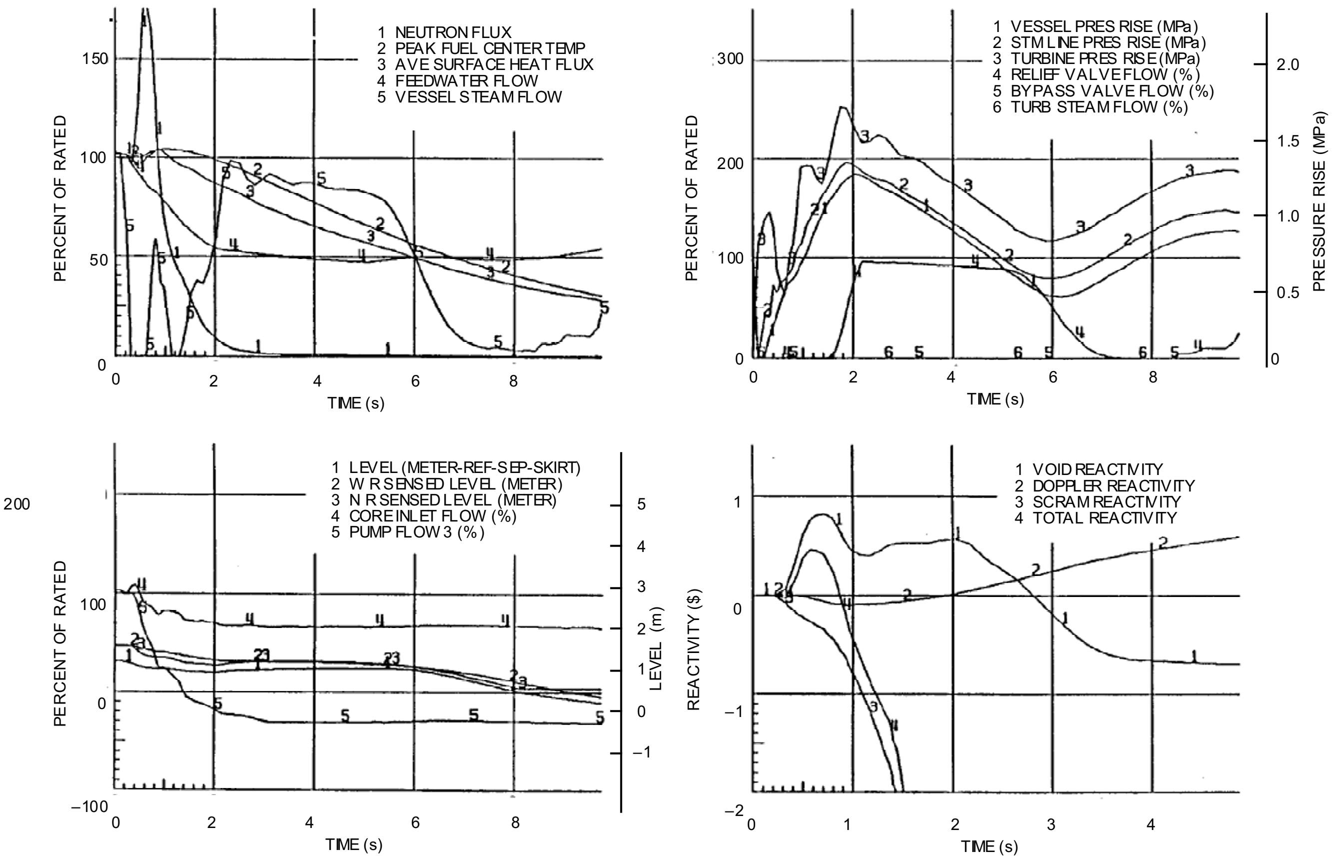


Figure 15.2-5 Load Rejection with All Bypass Valves Failure

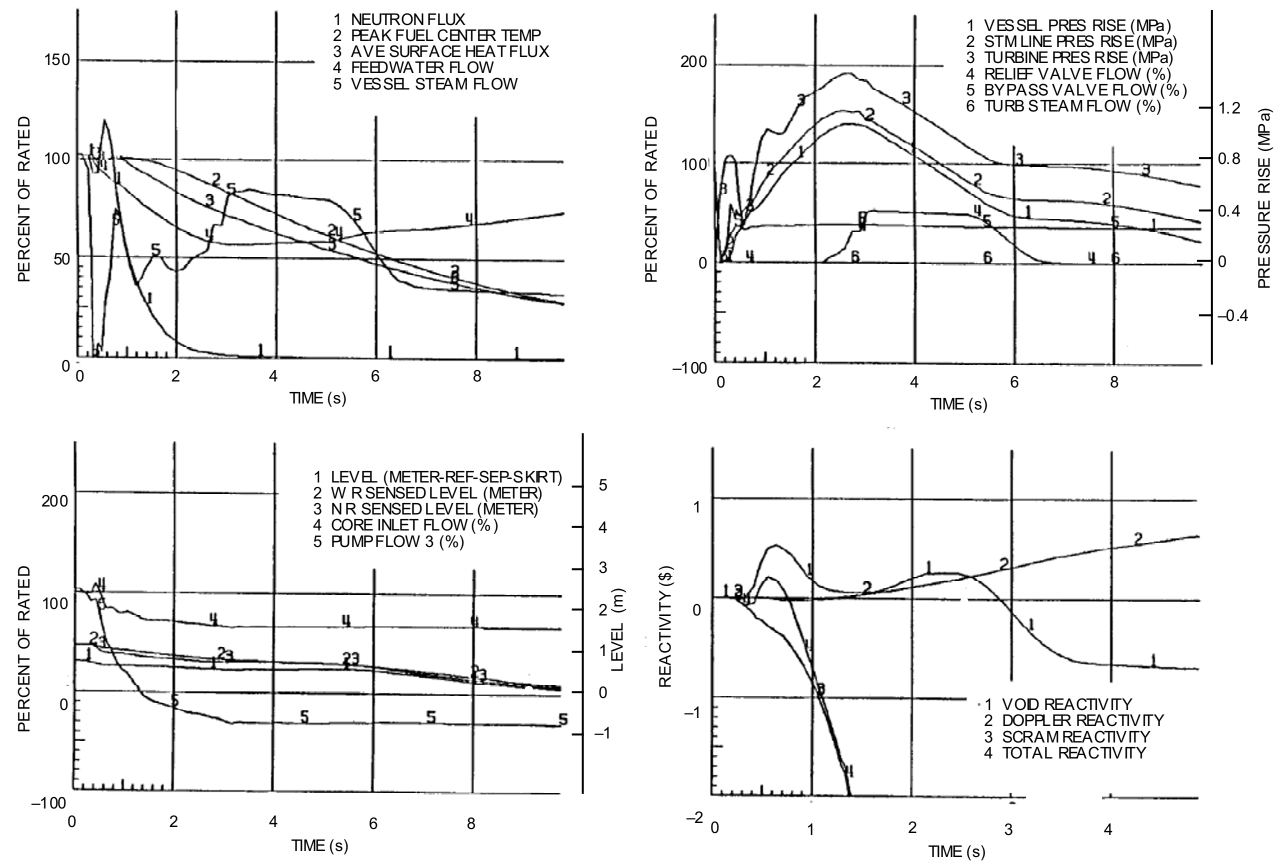


Figure 15.2-6 Turbine Trip with Bypass

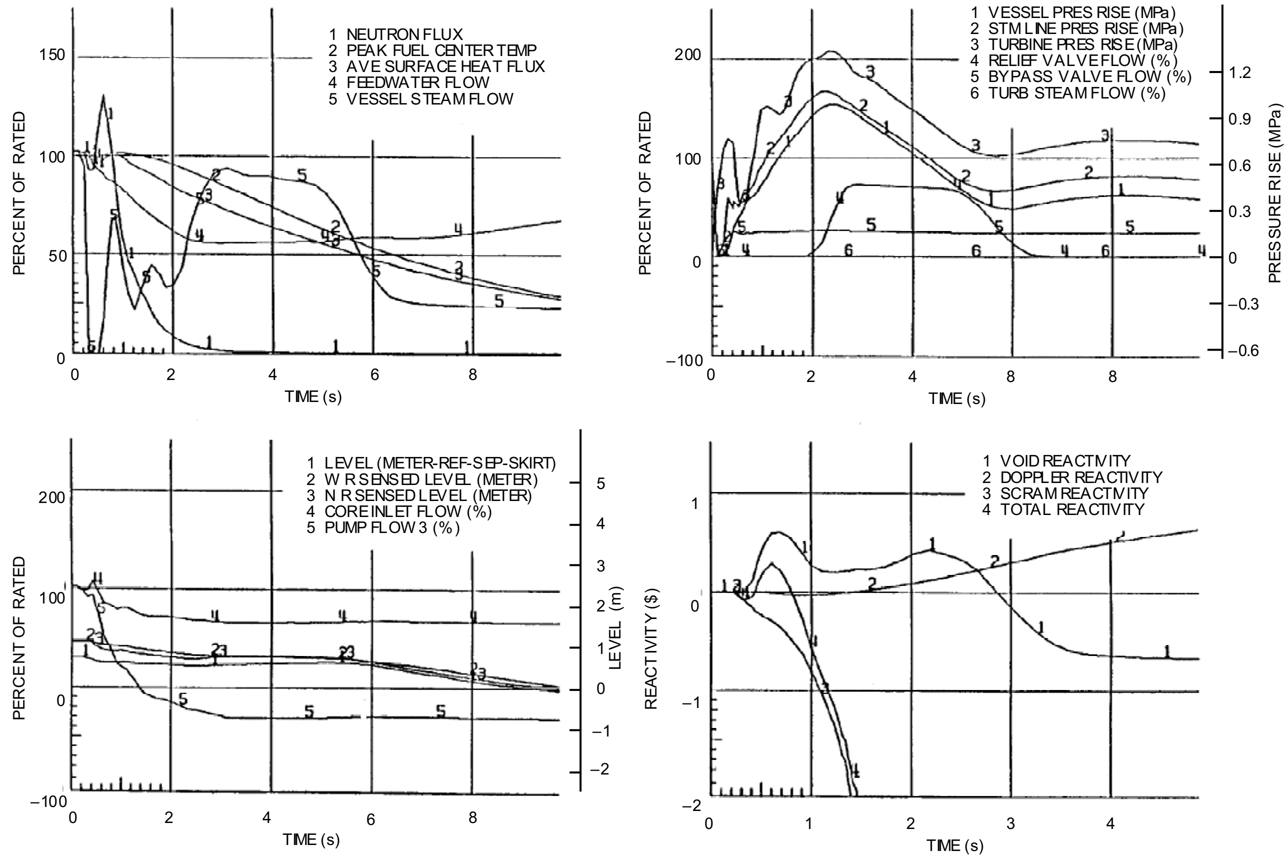


Figure 15.2-7 Turbine Trip with One Bypass Valve Failure

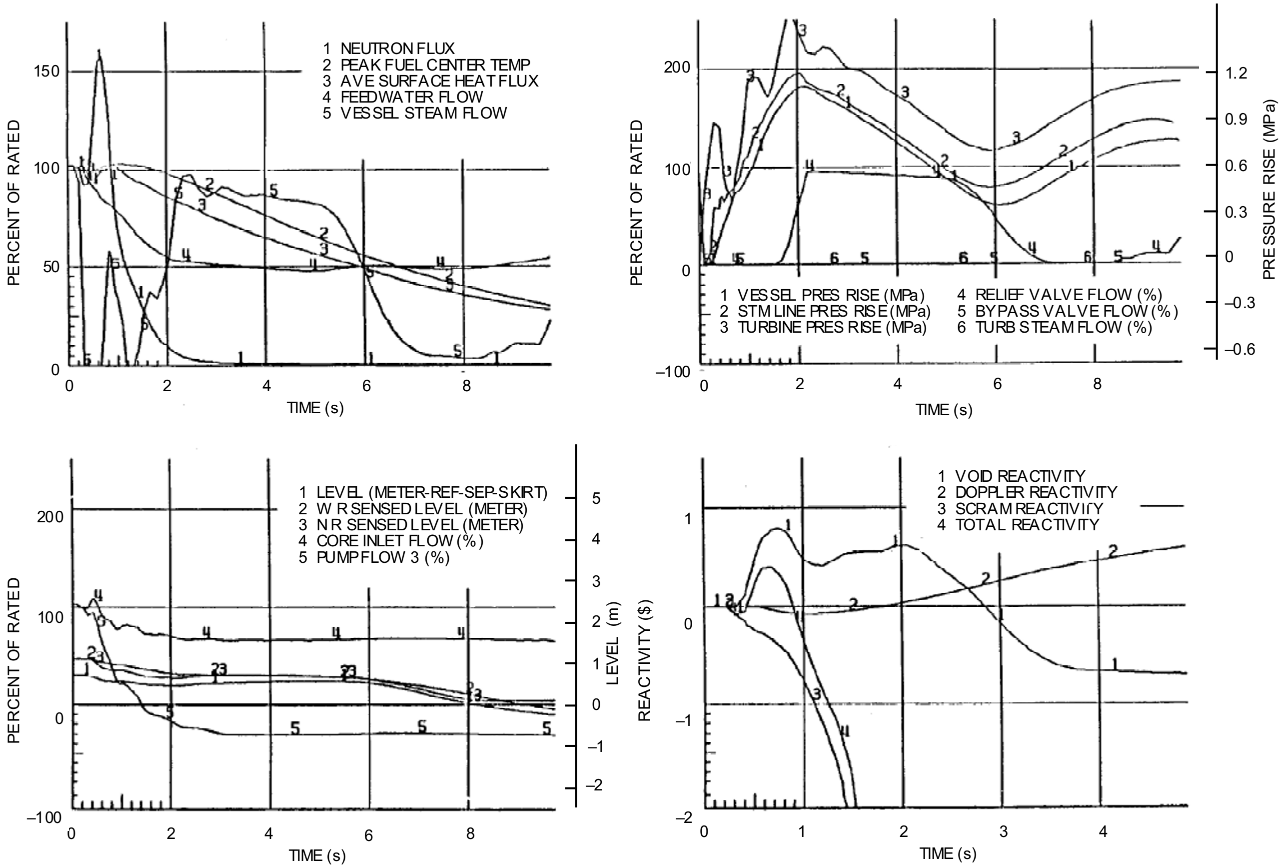


Figure 15.2-8 Turbine Trip with All Bypass Valves Failure

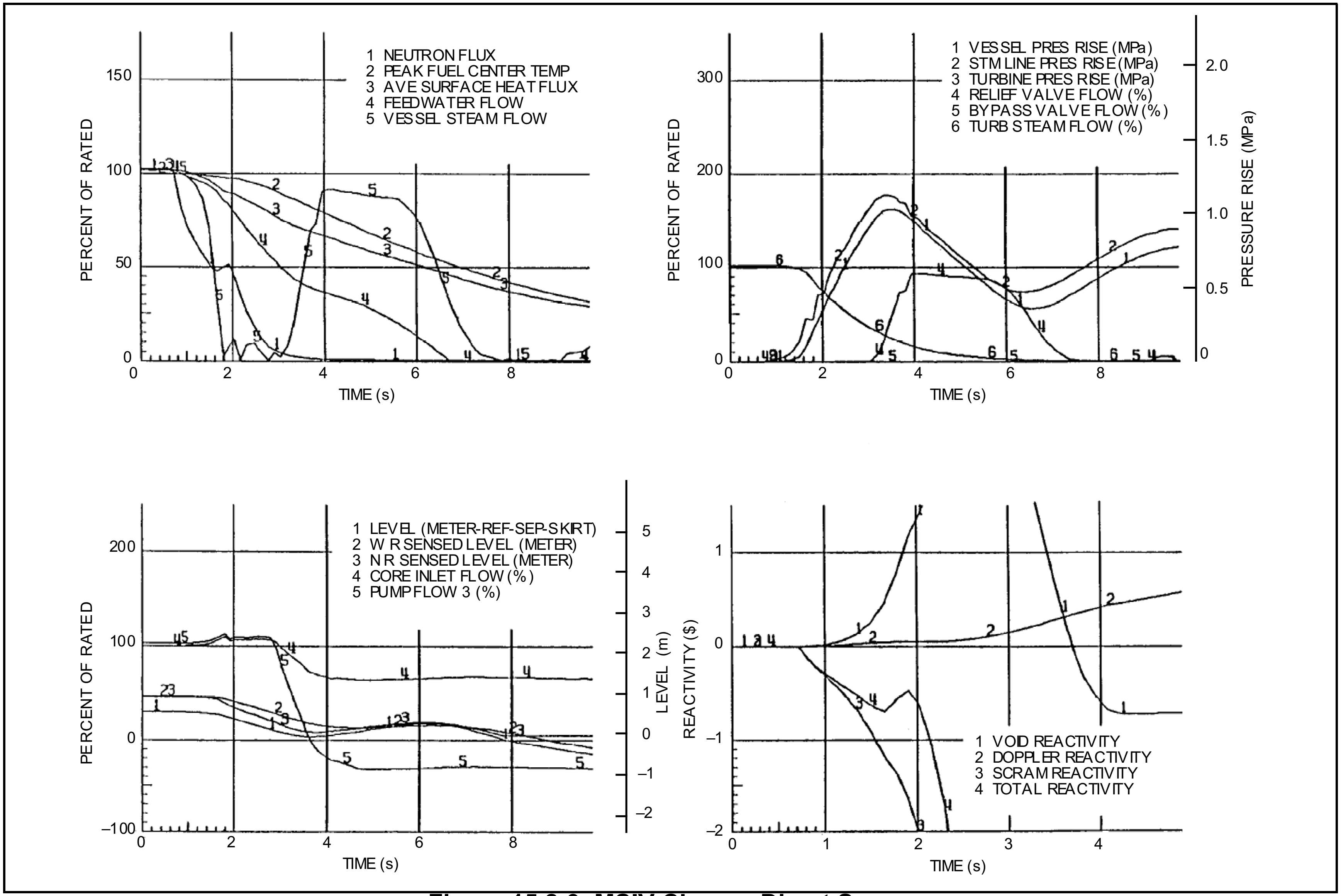


Figure 15.2-9 MSIV Closure Direct Scram

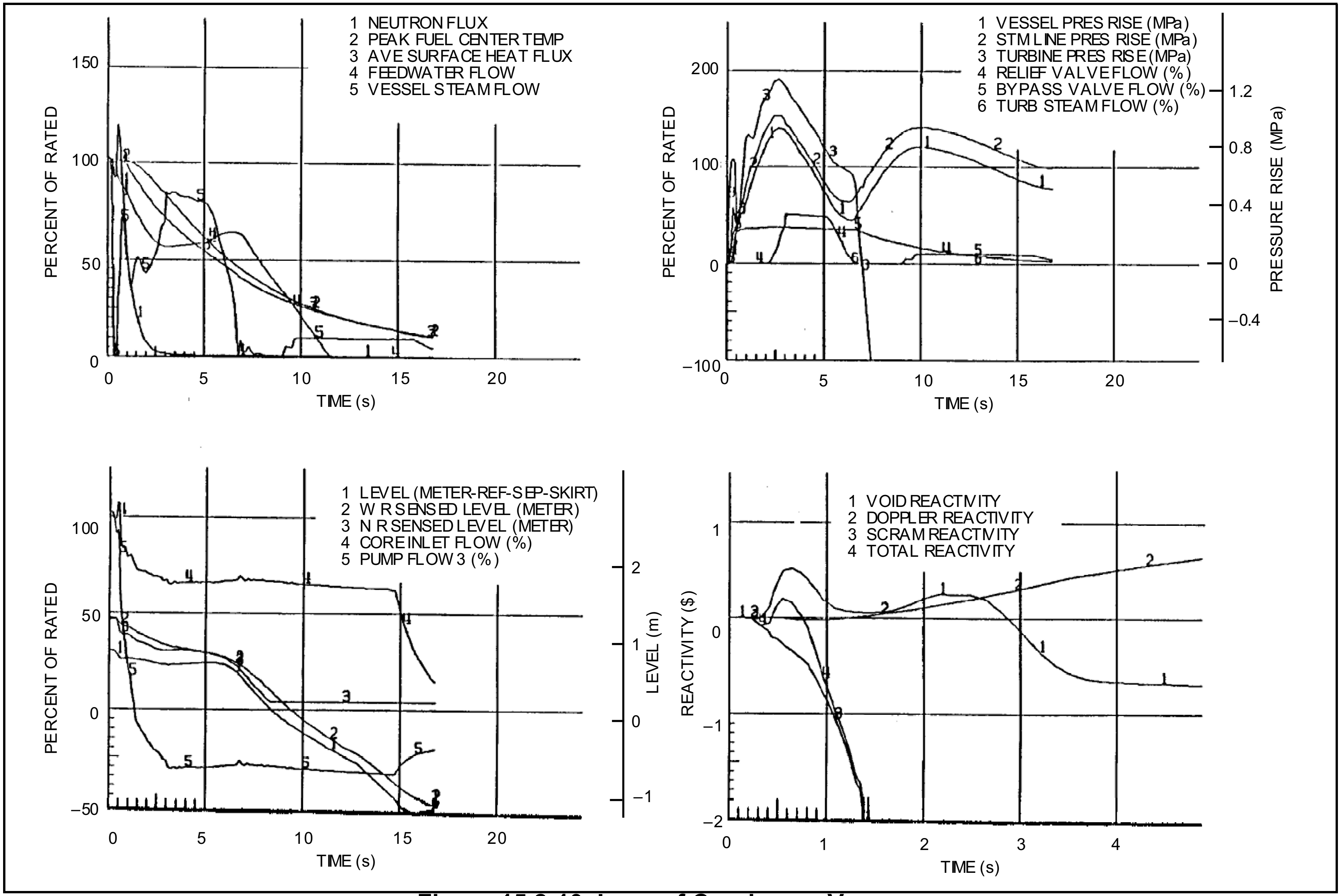


Figure 15.2-10 Loss of Condenser Vacuum

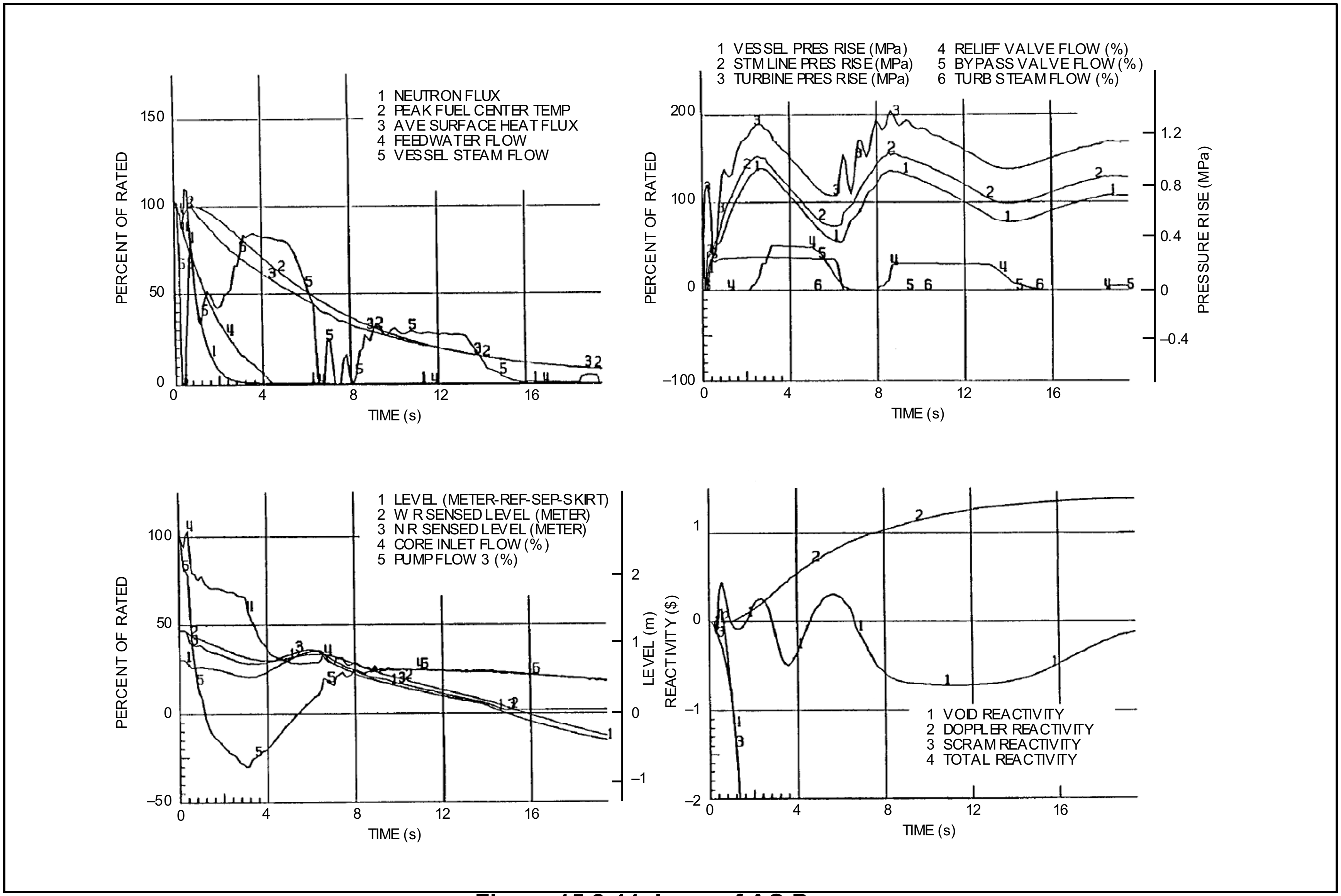


Figure 15.2-11 Loss of AC Power

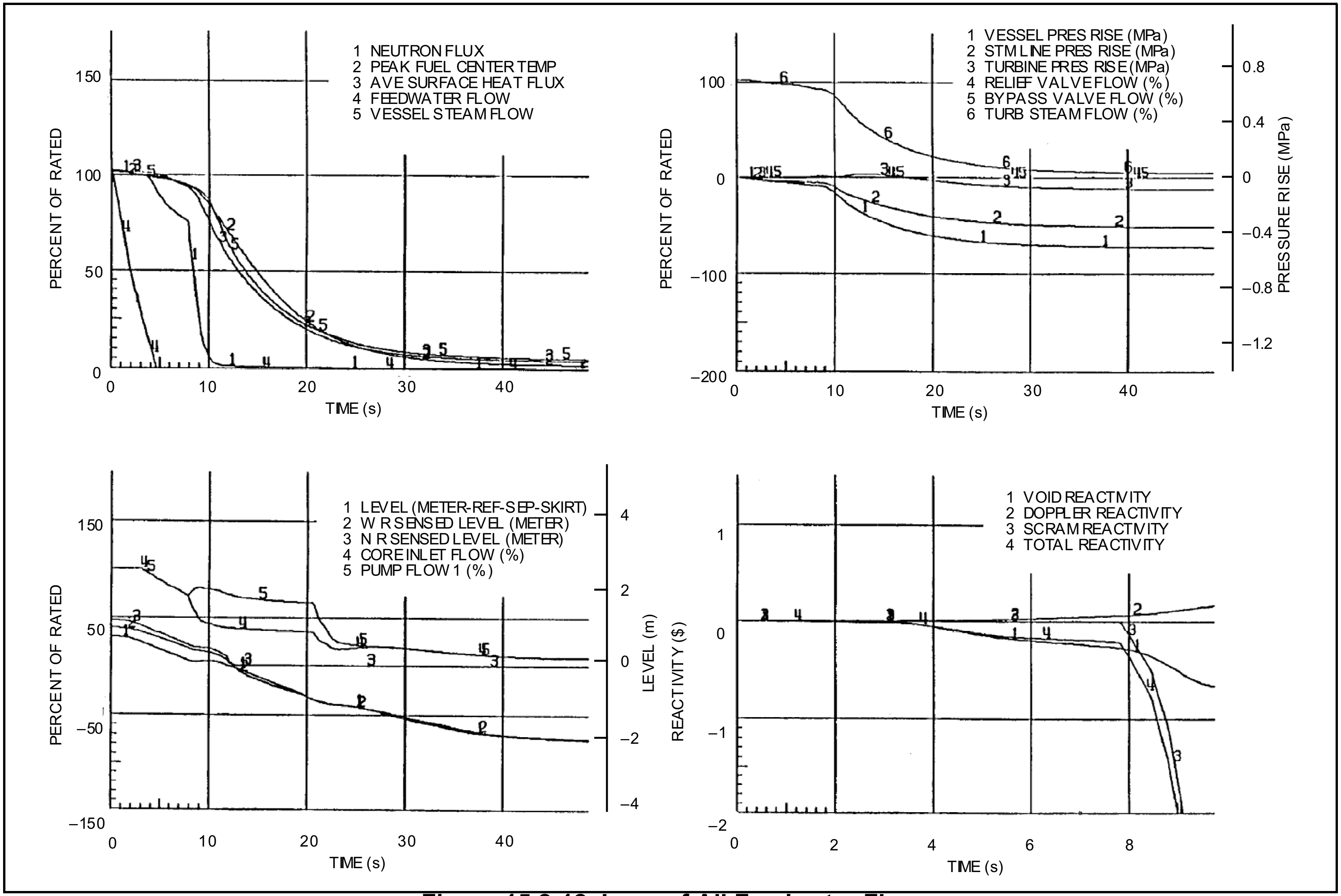


Figure 15.2-12 Loss of All Feedwater Flow

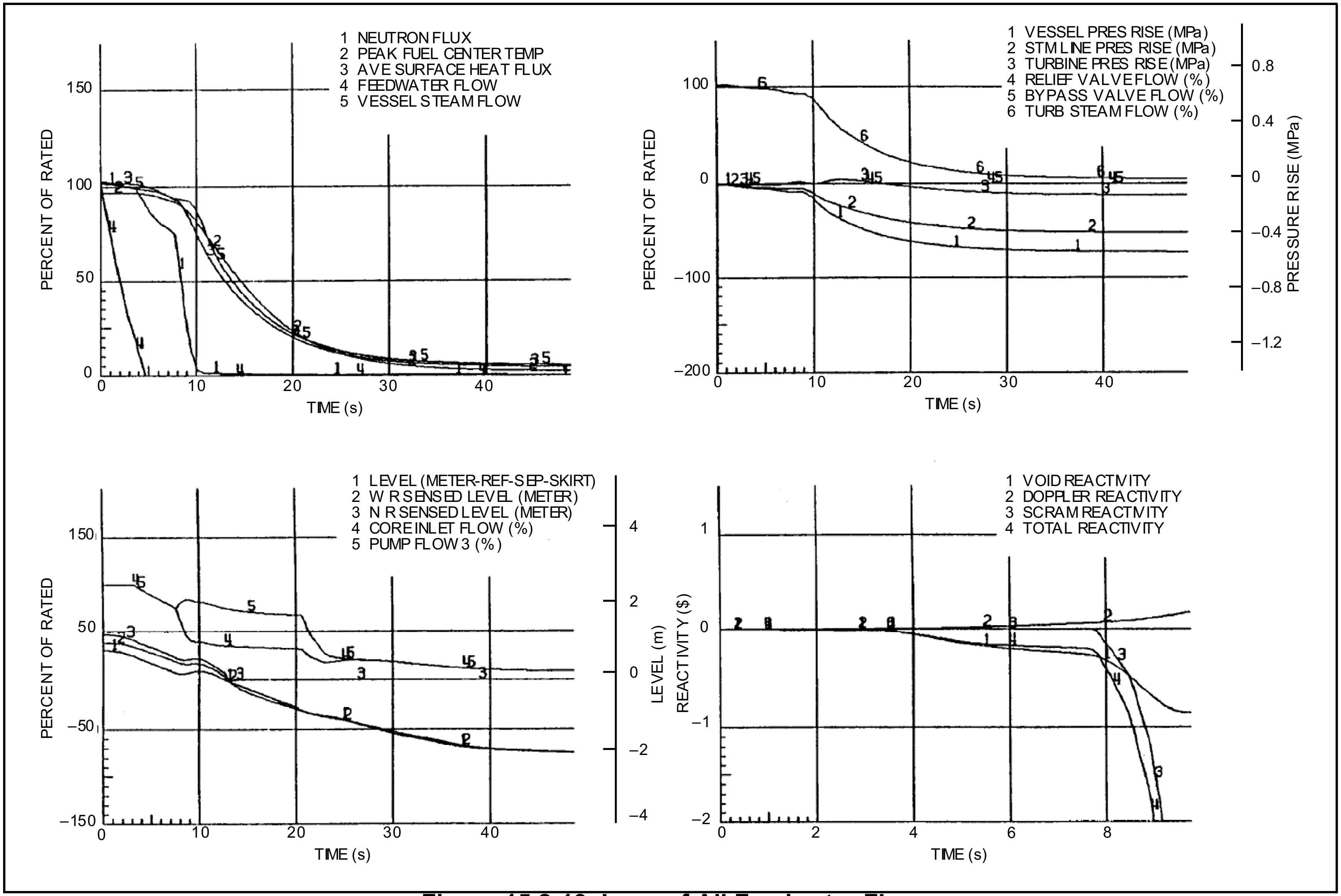


Figure 15.2-13 Loss of All Feedwater Flow

15.3 Decrease in Reactor Coolant System Flow Rate

15.3.1 Reactor Internal Pump Trip

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Reactor internal pump (RIP) motor operation can be tripped off by design for intended reduction of other transient core and RCPB effects, as well as randomly by unpredictable operational failures. Intentional tripping will occur in response to:

(1) Reactor vessel water level L3 setpoint trip (4 RIPs)

(2) Reactor vessel water level L2 setpoint trip (the other 6 RIPs)

(3) TCV fast closure or stop valve closure (the same 4 RIPs as L3 trip)

(4) High pressure setpoint trip (the same 4 RIPs as L3 trip)

(5) Motor overcurrent protection (single pump)

(6) Motor overload and short circuit protection (single pump)

Random tripping will occur in response to:

(1) Operator error.

(2) Loss of electrical power source to the pumps.

(3) Equipment or sensor failures and malfunctions which initiate the above intended trip response. However, all trip logics use redundant digital designs. Single failures in the UAT or MPT and/or their protection circuits can result in loss of preferred power source to the plant.

Thus, the worst single-failure event is a loss of electrical power bus, which supplies power to RIPs. Since four buses are used to supply power to the RIPs, the worst single failure can only cause three RIPs to trip.

A loss of AC power to station auxiliaries may cause RIPs to trip. However, not all RIPs would be tripped at the same time due to the M-G sets. Transients caused by a loss of AC power are discussed in Subsection 15.2.6.

The effect of an additional single failure on this event (i.e., trip of three RIPs) is the tripping of additional RIPs. For example, if an additional power bus fails at the same time, the number of RIPs tripped are five or six, instead of three. However, the probability of this occurring is low.

This event should be classified as a limiting fault. In this analysis, the trip of all RIPs is provided to bound the events of low probability.

When a rapid core flow reduction caused by a trip of all RIPs is sensed, a reactor scram is initiated to terminate the power generation. The core flow reduces rapidly due to the relatively small inertia of the RIPs. However, natural circulation is still available to keep the reactor core covered and cooled.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of Three Reactor Internal Pumps

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of All Reactor Internal Pumps

This event is categorized as an infrequent low probability event with special acceptance for fuel failure (see Subsection 15.3.1.5.2).

15.3.1.2 Sequence of Events and Systems Operation

15.3.1.2.1 Sequence of Events

15.3.1.2.1.1 Trip of Three Reactor Internal Pumps

Table 15.3-1 lists the sequence of events for Figure 15.3-1.

15.3.1.2.1.2 Trip of All Reactor Internal Pumps

Table 15.3-2 lists the sequence of events for Figure 15.3-2.

15.3.1.2.1.3 Identification of Operator Actions

15.3.1.2.1.3.1 Trip of Three Reactor Internal Pumps

Because no scram occurs for trip of three RIPs, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. The operator should also determine the cause of failure prior to returning the system to normal operation.

15.3.1.2.1.3.2 Trip of All Reactor Internal Pumps

The operator should ascertain that the reactor scram is initiated. If the main turbine and feedwater pumps are tripped resulting from reactor water level swell, the operator should regain control of reactor water level through RCIC operation, monitoring reactor water level and pressure after shutdown. When both reactor pressure and level are under control, the operator should secure RCIC as necessary. The operator should also determine the cause of the trip prior to returning the system to normal operation.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of Three Reactor Internal Pumps

Tripping of three RIPs requires no protection system or safeguard system operation. This analysis assumes normal functioning of plant instrumentation and controls.

15.3.1.2.2.2 Trip of All Reactor Internal Pumps

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems.

If a trip of all RIPs is caused by an electrical power supply to the RIPs, a reactor scram will be initiated at time 0 due to load rejection or turbine trip at time 0. For other causes, a reactor scram will be initiated upon the condition of high simulated thermal power scram, turbine trip due to high water level, or rapid core flow coastdown. High system pressure is limited by the pressure relief valve system operation.

Since the event becomes more severe when the reactor scram is delayed, the analysis conservatively assumes that the reactor scram is initiated by the last signal (i.e., core flow rapid coastdown scram). It is also conservatively assumed that the event is caused by a common mode failure in all ASDs, which results in a trip of all RIPs.

15.3.1.3 Core and System Performance

15.3.1.3.1 Input Parameters and Initial Conditions

Pump motors and pump rotors are simulated with minimum specified rotating inertias. The nuclear conditions for the beginning of cycle (BOC) are used to provide conservative bounding analysis.

15.3.1.3.2 Results

15.3.1.3.2.1 Trip of Three Reactor Internal Pumps

Figure 15.3-1 shows the results of losing three RIPs. MCPR remains above the safety limit; thus, the fuel thermal limits are not violated. During this transient, level swell is not sufficient to cause turbine trip and scram.

Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.1.3.2.2 Trip of all Reactor Internal Pumps

Figure 15.3-2 graphically shows this event with the minimum specified rotating inertia for the RIPs. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip, thereby tripping the main turbine and feed pumps. Subsequent events, such as initiation of the RCIC System occurring late in this event, have no significant effect on the results. The peaking cladding temperature (PCT) during this event is calculated to be less than

600°C, which is below the applicable limit of 1200°C. The cladding temperature during this event is shown in Figure 15.3-2a. Since the time that the cladding temperature is above the coolant saturated temperature is less than 60 seconds, and the peak cladding temperature is less than 600°C, no fuel failure is expected.

This event is very sensitive to the core condition. It is expected that about 60% of the rods will be in transition boiling at the beginning of the core life, and about 6% at the end of the first fuel cycle. This value drops to about 4% at the end of the equilibrium cycle. However, no fuel failures are expected.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of Three Reactor Internal Pumps

The results shown in Figure 15.3-1 indicate that peak pressures stay well below the 9.48 MPa limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.4.2 Trip of All Reactor Internal Pumps

The results shown in Figure 15.3-2 indicate that peak pressures stay well below the limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

15.3.1.5 Radiological Consequences

15.3.1.5.1 Trip of Three Reactor Internal Pumps

This event does not result in any fuel failures, nor any discharge to the suppression pool. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.1.5.2 Trip of All Reactor Internal Pumps

The approved procedures for radiological dose calculation for this event are as follows:

(1) For fuel rods with less than or equal to 20 GWd/T exposure, fuel failures are assumed if the PCT stays above 600°C for more than 60 seconds.

(2) For fuel rods with greater than 20 GWd/T exposure, rods that are in transition boiling shall be assumed to fail radiological dose calculations.

(3) The radiological doses shall be less than 10% of 10CFR100 requirements.

As discussed in Subsection 15.3.1.3.2.2, the PCT during this event is less than 600°C and the time at high temperature is less than 60 seconds. Therefore, no fuel failures need to be assumed for fuel rods with less than or equal to 20 GWd/T exposure.

In general, fuel rods with more than 20 GWd/T exposure are those remaining in the core for more than two fuel cycles. In the equilibrium cycle, these fuel bundles only account for about

45% of the total bundles. The power generated by these bundles is usually 20% less than that of the hottest bundles. Less than 0.2% of these rods get into transition boiling. Therefore, the requirements of 10% of 10CFR100 are met.

15.3.2 Recirculation Flow Control Failure—Decreasing Flow

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

The Recirculation Flow Control System (RFCS) uses a triplicated, fault-tolerant digital control system, instead of an analog system, as used in BWR/2 through BWR/6. The RFCS controls all 10 reactor internal pumps (RIPs) at the same speed. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system will result in a minimum demand to all RIPs. A voter or actuator failure may result in an inadvertent runback of one RIP at its maximum drive speed (~40%/s). In this case, the RFCS will sense the core flow change and command the remaining RIPs to increase speeds and thereby automatically mitigate the transient and maintain the core flow.

As presented in Subsection 15.1.2.1.1, multiple failures in the control system might cause the RFCS to erroneously issue a minimum demand to all RIPs. Should this occur, all RIPs could reduce speed simultaneously. Each RIP drive has a speed limiter which limits the maximum speed change rate to 5%/s. However, the probability of this event occurring is very low, and, hence, the event should be considered as a limiting fault. However, criteria for moderately frequent incidents are conservatively applied.

15.3.2.1.2 Frequency Classification

15.3.2.1.2.1 Fast Runback of One Reactor Internal Pump

The failure rate of a voter or an actuator is very low. However, it is analyzed as an incident of moderate frequency.

15.3.2.1.2.2 Fast Runback of All Reactor Internal Pumps

This event should be classified as a limiting fault event. However, criteria for moderate frequent incidents are conservatively applied.

15.3.2.2 Sequence of Events and Systems Operation

15.3.2.2.1 Sequence of Events

15.3.2.2.1.1 Fast Runback of One Reactor Internal Pump

Table 15.3-3 lists the sequence of events for Figure 15.3-3.

15.3.2.2.1.2 Fast Runback of All Reactor Internal Pumps

Table 15.3-4 lists the sequence of events for Figure 15.3-4.

15.3.2.2.1.3 Identification of Operator Actions

15.3.2.2.1.3.1 Fast Runback of One Reactor Internal Pump

As soon as possible, the operator verifies that no operating limits are being exceeded. The operator determines the cause of failure prior to returning the system to normal.

15.3.2.2.1.3.2 Fast Runback of All Reactor Internal Pumps

As soon as possible, the operator verifies that no operating limits are being exceeded. If they are, corrective actions must be initiated. Also, the operator determines the cause of the failures prior to returning the system to normal.

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Fast Runback of One Reactor Internal Pump

Normal plant instrumentation and control is assumed to function.

15.3.2.2.2.2 Fast Runback of All Reactor Internal Pumps

Normal plant instrumentation and control is assumed to function.

15.3.2.3 Core and System Performance

15.3.2.3.1 Input Parameters and Initial Conditions

15.3.2.3.1.1 Fast Runback of One Reactor Internal Pump

Failure can result in the maximum speed of the RIP decreasing at a rate of 40%/s as limited by the pump drive.

15.3.2.3.1.2 Fast Runback of All Reactor Internal Pumps

A downscale failure of the master controller will generate a zero flow demand signal to all RIPs. Each individual RIP drive has a speed limiter which limits the maximum speed decrease to a rate of 5%/s. Core flow decreases to approximately 40% of rated. This is the flow expected when the RIPs are maintained at their minimum speeds.

15.3.2.3.2 Results

15.3.2.3.2.1 Fast Runback on One Reactor Internal Pump

Figure 15.3-3 illustrates the fast runback of one RIP event with the maximum rate which is limited by hydraulic means. The MCPR remains above the safety limit. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.2.3.2.2 Fast Runback of All Reactor Internal Pumps

Figure 15.3-4 illustrates the expected event. Design of limiter operation is intended to render this event to be less severe than the trip of all RIPs. No fuel damage is expected to occur. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.3.2.4 Barrier Performance

15.3.2.4.1 Fast Runback of One Reactor Internal Pump

Peak pressures are less than those for the “Fast Runback of All RIPs” presented in Subsection 15.3.2.4.2.

15.3.2.4.2 Fast Runback of All Reactor Internal Pumps

Pressure in the vessel bottom is not higher than its initial value and below the ASME code limit.

15.3.2.5 Radiological Consequences

While the consequences of the events identified previously do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences presented in Subsection 15.2.4.5 for a Type 2 event. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.3 Reactor Internal Pump Seizure

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a reactor internal pump (RIP) is considered a design basis accident (DBA) event. It has been evaluated as being a very mild accident in relation to others DBAs such as the LOCA. (Refer to Section 5.1 for special mechanical considerations and Chapter 7 for electrical aspects.)

The seizure event postulated is not expected to be the mode failure of such a device. Safe shutdown components (e.g., electrical breakers, protective circuits) preclude an instantaneous seizure event.

15.3.3.1.1 Identification of Causes

The cause of RIP seizure represents the unlikely event of instantaneous stoppage of the pump motor shaft of one reactor internal pump. This event produces a very rapid decrease of pump flow as a result of the large hydraulic resistance introduced by the stopped rotor. Consequently, a decrease in core inlet flow and core cooling capability occurs. However, with only one out of ten RIPs seized, the core flow decrease is small (~10%), so the event is very mild.

15.3.3.1.2 Frequency Classification

This event is considered to be a limiting fault but results in effects which can satisfy an event of greater probability (i.e., infrequent incident classification).

15.3.3.2 Sequence of Events and Systems Operations

15.3.3.2.1 Sequence of Events

Table 15.3-5 lists the sequence of events for Figure 15.3-5.

15.3.3.2.1.1 Identification of Operator Actions

Because no scram occurs for one RIP seizure, no immediate operator action is required. As soon as possible, the operator verifies that no operating limits are being exceeded. Also, the operator determines the cause of failure and proceeds to shutdown the plant for repair.

15.3.3.2.2 Systems Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls, plant protection, and reactor protection systems. Acceptance Criterion II.8 of SRP Section 15.3.3 provides that only safety grade equipment is be used to mitigate the consequences of this event. It also provides that safety functions be accomplished assuming the worst single failure of a safety system active component. Acceptance Criterion II.10 of SRP Section 15.3.3 also provides that the analysis assume turbine trip and coincident loss of offsite power. Should a coincident loss of offsite power occur, the consequences would be similar to the consequences of the loss of offsite power (LOPP) transient described in Subsection 15.2.6.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the plant instrumentation and controls, plant protection, and reactor protection systems will not cause this accident to be more severe than analyzed.

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The REDYA transient model (References 4.4-13, 4.4-14, and 4.4-15) is used to simulate this event.

15.3.3.3.2 Input Parameters and Initial Conditions

For the purpose of evaluating consequences to the fuel thermal limits, this event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 102% NBR power. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value (i.e., the least negative value in Table 15.0-1).

15.3.3.3.3 Results

Figure 15.3-5 shows the analysis results of this event. Table 15.3-5 lists the sequence of events for Figure 15.3-5.

15.3.3.4 Barrier Performance

As shown in Figure 15.3-5, system pressure during this event is not higher than the original values. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 Radiological Consequences

The consequences of the events identified do not result in any fuel failures or SRV actuation.

15.3.4 Reactor Internal Pump Shaft Break

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a RIP is considered a DBA event. It has been evaluated as a very mild accident in relation to other DBAs such as the LOCA (Refer to Chapter 5 for specific mechancial considerations and Chapter 7 for electrical aspects.).

This postulated event is bounded by the more limiting case of RIP seizure. Quantitative results for this more limiting case are presented in Subsection 15.3.3.

15.3.4.1.1 Identification of Causes

The case of RIP shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one reactor internal pump. This event produces a very rapid decrease of pump flow as a result of the break of the pump shaft. Consequently, it results in a small decrease in core inlet flow and core cooling capability.

15.3.4.1.2 Frequency Classification

This event is considered a limiting fault but results in effects which can easily satisfy an event of greater probability (i.e., infrequent incident classification).

15.3.4.2 Sequence of Events and Systems Operations

15.3.4.2.1 Sequence of Events

A postulated instantaneous break of the shaft of one RIP (Subsection 15.3.4.1.1) causes the core flow to decrease rapidly. The sequence of events is the same as that presented in Subsection 15.3.3.2.1.

15.3.4.2.1.1 Identification of Operator Actions

Same as Subsection 15.3.3.2.1.1.

15.3.4.2.2 Systems Operation

Same as Subsection 15.3.3.2.2.

15.3.4.3 Core and System Performance

The severity of this pump shaft break event is bounded by the pump seizure event (Subsection 15.3.3). This can be demonstrated easily by consideration of these two events. In either of these two events, the recirculation drive flow of the affected pump decreases rapidly. In the case of the pump seizure event, the pump flow decreases faster than the normal flow coastdown as a result of the large hydraulic resistance introduced by the stopped rotor. For the pump shaft break event, the hydraulic resistance caused by the broken pump shaft is less than that of the stopped rotor for the pump seizure event. Therefore, the core flow decrease following a pump shaft break effect is slower than the pump seizure event. Thus, it can be concluded that the potential effects of the hypothetical pump shaft break accident are bounded by the effects of the pump seizure event and this event does not have to be reanalyzed for specific core configurations.

15.3.4.4 Barrier Performance

The reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 Radiological Consequences

The consequences of this event do not result in any fuel failures or SRV actuation.

15.3.5 References

None.

Table 15.3-1 Sequence of Events for Figure 15.3-1

Time (s)	Event
0	Trip of three RIPs initiated.
0.6	Pump flow reverses in the three tripped pumps.
10.0	Core flow reaches its steady state.
>12.0 (est.)	Core power reaches its steady state.

Table 15.3-2 Sequence of Events for Figure 15.3-2

Time (s)	Event
0	Trip of all RIPs initiated.
1.22	The rate of change of the reactor vessel core flow reaches the rapid core flow coastdown setpoint and initiates reactor scram.
1.85	High reactor vessel water level (L8) initiates main turbine trip and feedwater pump trip.
1.97	Turbine trip initiates bypass operation.
3.7	Safety/relief valves open due to high pressure.
9.0	Safety/relief valves close.
28 (est.)	Vessel water level (L2) setpoint reached (not simulated).
58 (est.)	RCIC flow enters vessel (not simulated).

Table 15.3-3 Sequence of Events for Figure 15.3-3

Time (s)	Event
0	Initiates fast runback of one RIP.
~0.1	Core flow starts to decrease.
1.0	RIPs other than the failed one start to increase their speeds.
1.1	Pump flow reverses in the affected RIP.
2.5	Pump speed reaches its minimum speed in the affected RIP.
5.0	Core flow reaches its steady state.

Table 15.3-4 Sequence of Events for Figure 15.3-4

Time (s)	Event
0.0	Initiates fast runback of all RIPs.
14.0	RIPs reach their minimum speed.
20.0	Core flow settles at its steady state.
30.0	Core power settles at its steady state.

Table 15.3-5 Sequence of Events for Figure 15.3-5

Time (s)	Event
0.0	Single pump seizure is initiated.
0.001	Seized RIP stops.
0.1	Pump flow reverses through the seized RIP.
1.0	Other RIPs start to increase their pump speed.
2.0	Core flow reaches its steady state.
10.0	Core power reaches its steady state.

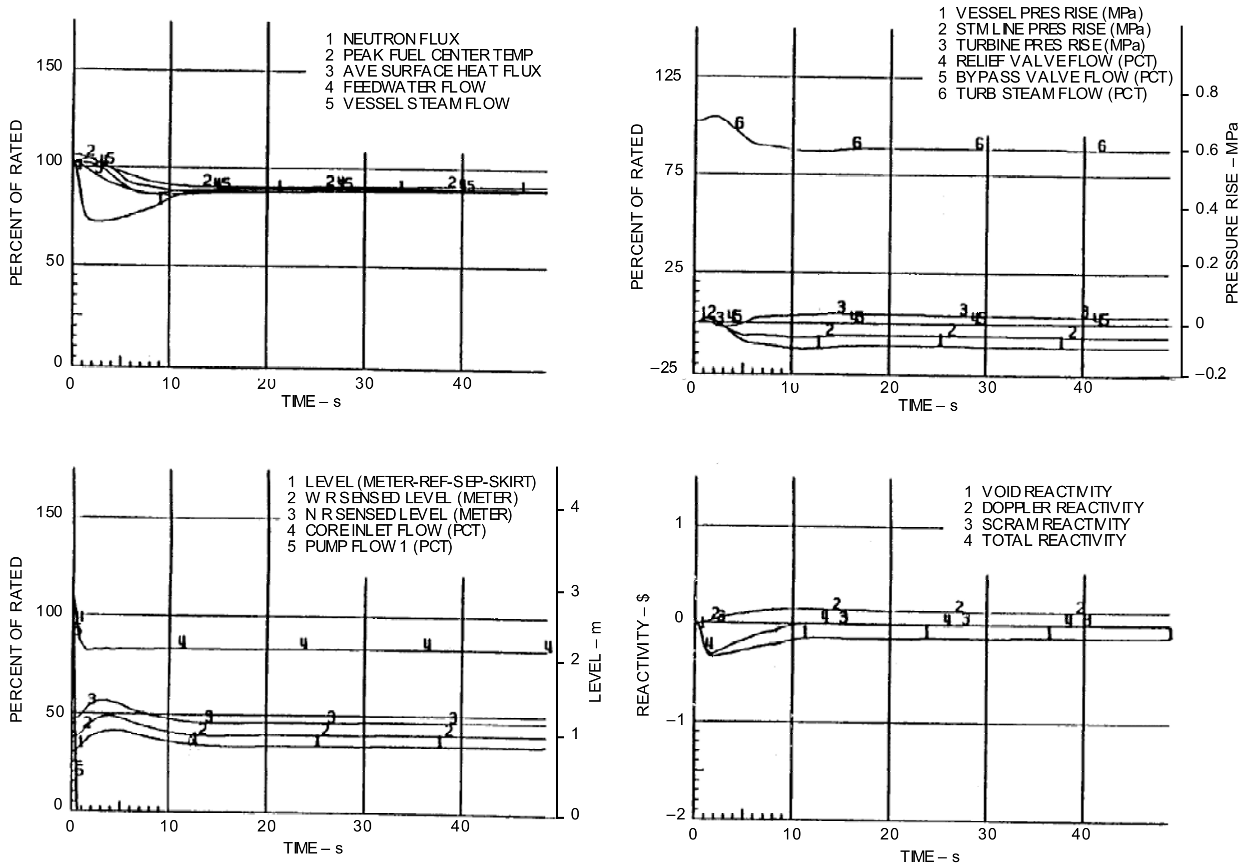


Figure 15.3-1 Three Pump Trip

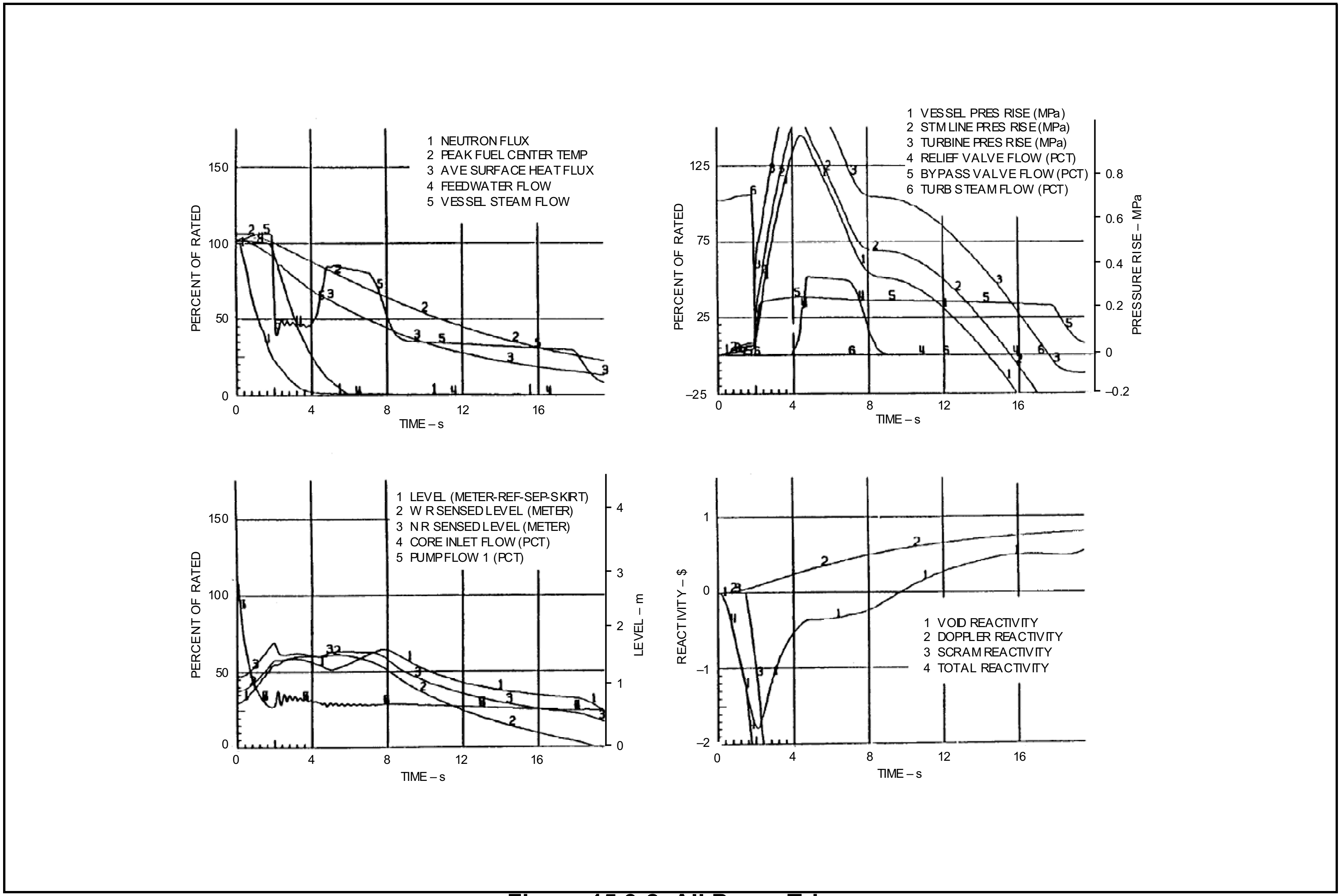


Figure 15.3-2 All Pump Trip

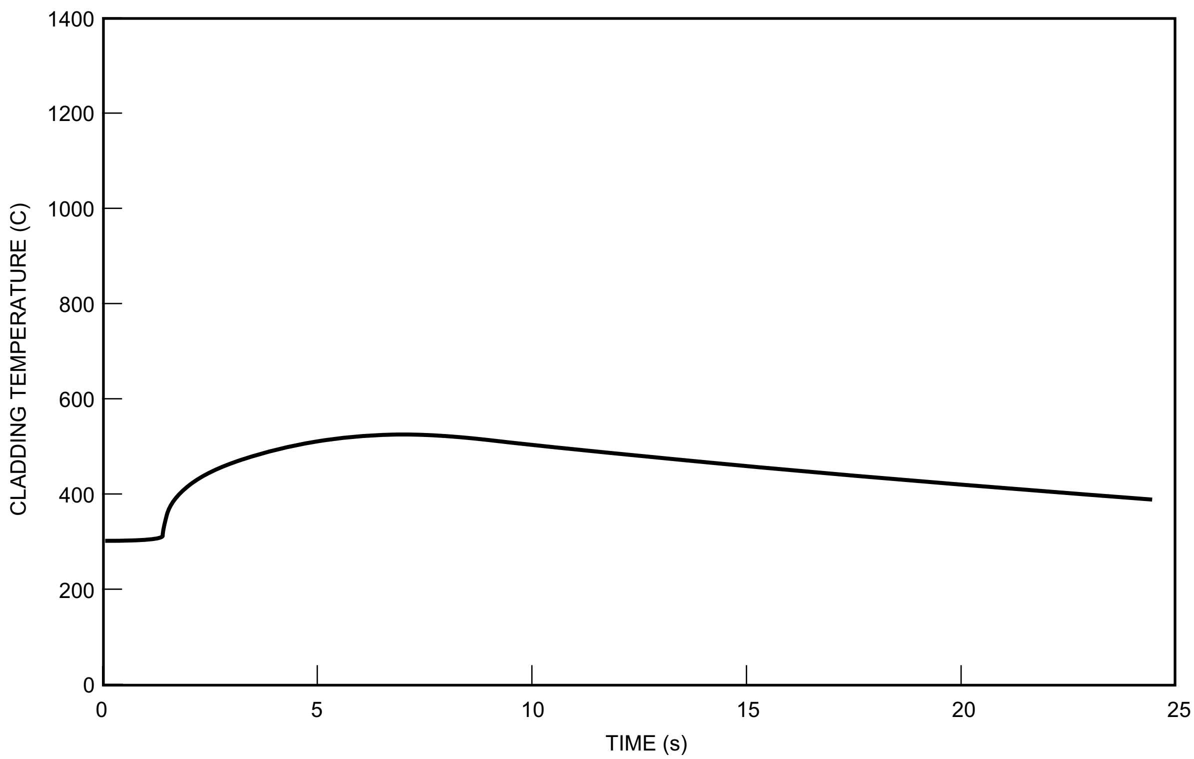


Figure 15.3-2a Cladding Temperature During All Pump Trip

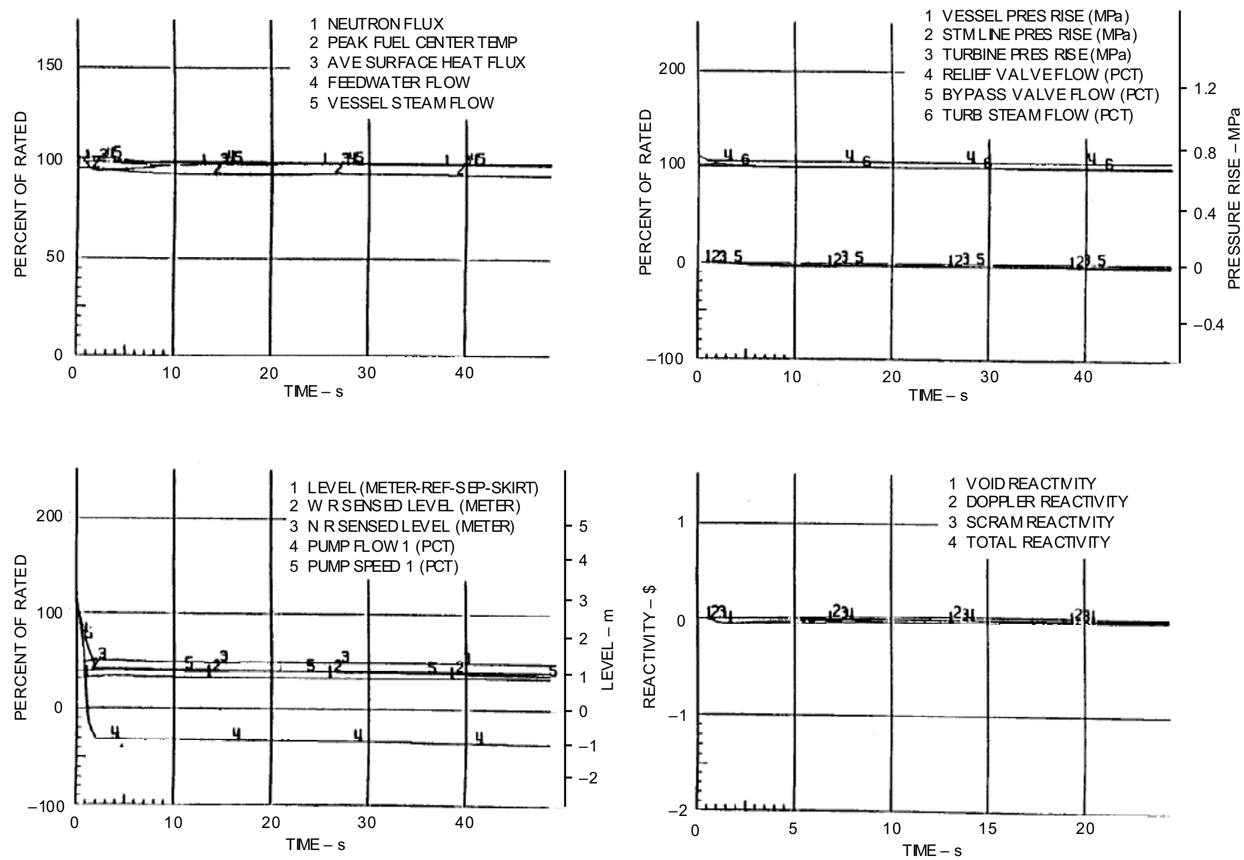


Figure 15.3-3 Fast Runback of One RIP

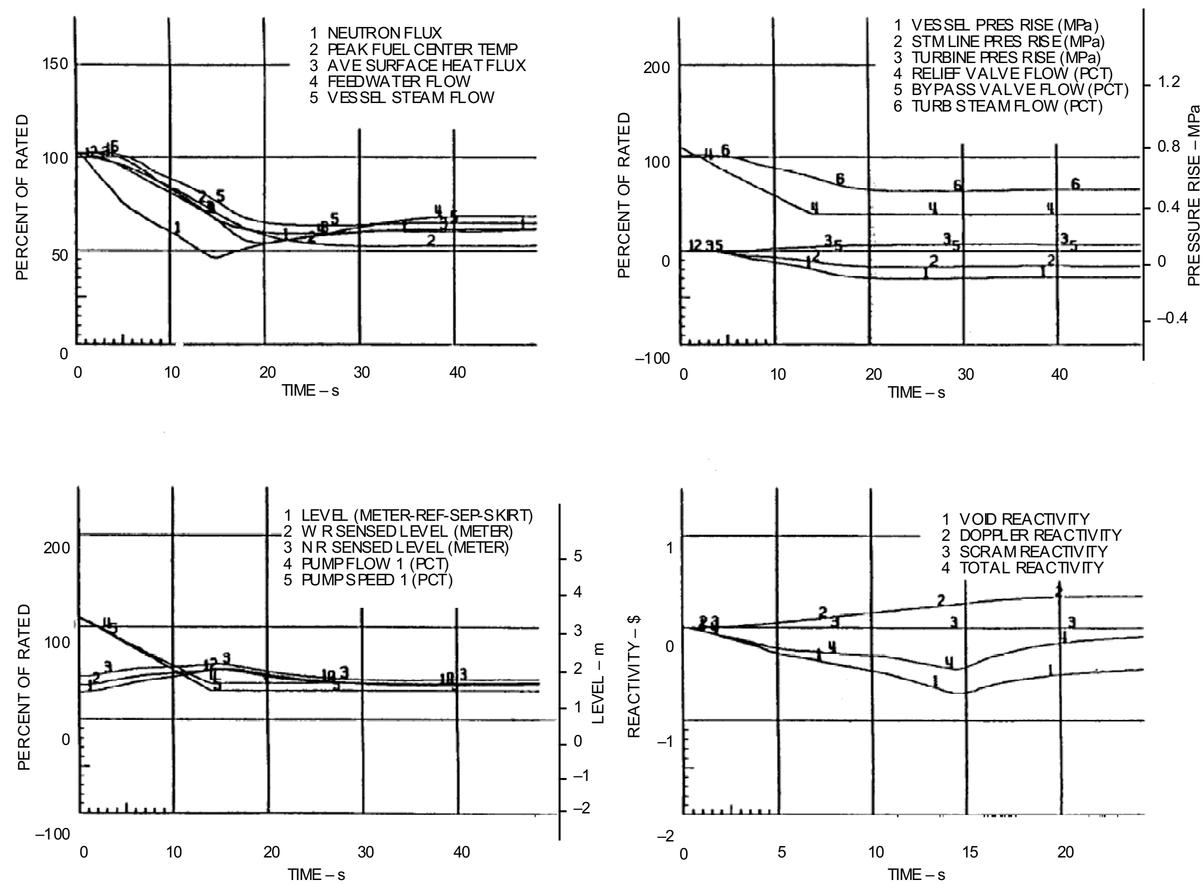


Figure 15.3-4 Fast Runback of All RIPs

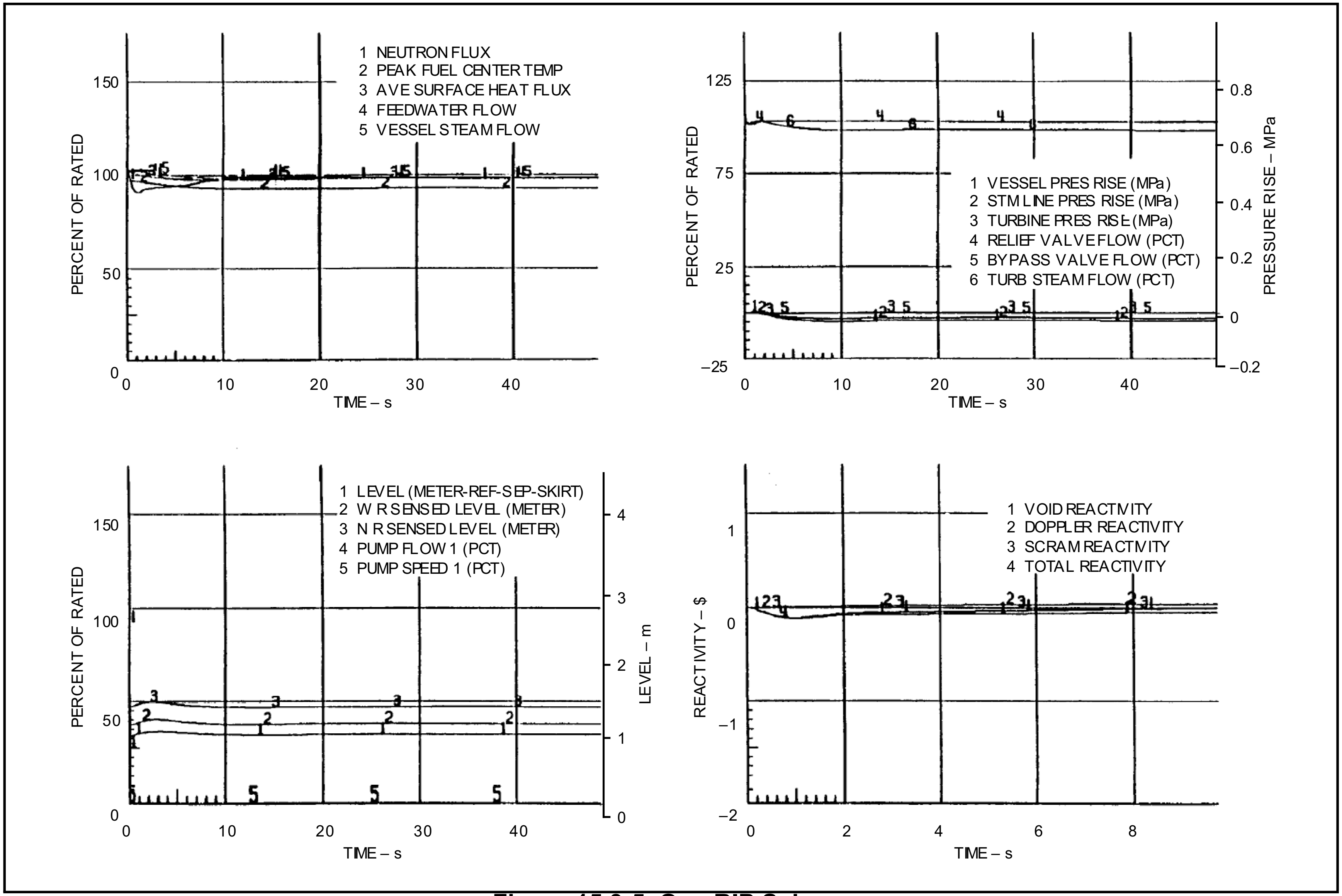


Figure 15.3-5 One RIP Seizure

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Rod Withdrawal Error—Low Power

15.4.1.1 Control Rod Removal Error During Refueling

15.4.1.1.1 Identification of Causes and Frequency Classification

The event considered here is inadvertent criticality due to the complete withdrawal or removal of the most reactive rod during refueling. The probability of the initial causes, alone, is considered low enough to warrant its being categorized as an infrequent incident, because there is no postulated set of circumstances which results in an inadvertent rod withdrawal error (RWE) while in the REFUEL mode.

15.4.1.1.2 Sequence of Events and Systems Operation

15.4.1.1.2.1 Initial Control Rod Removal or Withdrawal

During refueling operation, safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

15.4.1.1.2.2 Fuel Insertion with Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell with a withdrawn control rod, it is required that all control rods are fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

15.4.1.1.2.3 Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RCIS SINGLE/GANG switch is in the SINGLE position. When the RCIS switch is in the GANG position, only one control rod pair with the same HCU may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RCIS interlock. Because the core is designed to meet shutdown requirements with any one control rod pair (with the same HCU) withdrawn, the core remains subcritical.

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the bayonet coupling system, does not physically permit the upward removal of the control rod without decoupling by rotation and the simultaneous or prior removal of the four adjacent fuel bundles.

15.4.1.1.2.5 Identification of Operator Actions

No operator actions are required to preclude this event, because the protection system design as previously presented will prevent its occurrence.

15.4.1.1.3 Core and System Performance

Because the possibility of inadvertent criticality during refueling is precluded, the core and system performances are not analyzed. The withdrawal of the highest worth control rod during refueling does not result in criticality. This is verified experimentally by performing shutdown margin checks (see Subsection 4.3 for a description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Because no fuel damage can occur, no radioactive material will be released from the fuel. Therefore, this event is not reanalyzed for specific core configurations.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance is not made for this event because there is no postulated set of circumstances for which this event could occur.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences is not made for this event because no radioactive material is released from the fuel.

15.4.1.2 Continuous Control Rod Withdrawal Error During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

It is postulated that during a reactor startup, a gang of control rods or a single control rod is inadvertently withdrawn continuously due to a procedural error by the operator or a malfunction of the automated rod movement control system.

The Rod Control and Information System (RCIS) has a dual channel rod pattern control function that prevents withdrawal of any out-of-sequence rods from 100% control rod density (CRD) to 50% CRD (i.e., for Group 1 to Group 4 rods). It also has bank position withdraw sequence constraints such that, if the withdraw sequence constraints are violated, the rod pattern control function of the RCIS will initiate a rod block. The bank position constraints are in effect from 50% CRD to the low power setpoint.

The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block if the flux excursion, caused by rod withdrawal, generates a period shorter than 20 seconds. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds. Any single SRNM rod block trip initiates a rod block. Any two divisional scram trips out of four divisions initiates a scram. The SRNM also has upscale rod block and upscale scram functions as a double protection for flux excursion. A detailed description of the period-based trip function is presented in Chapter 7.

For this transient to happen, a large reactivity addition must be introduced. The reactor must be critical, with control rod density greater than 50%. Additionally, the BPWS logic must fail such that a gang of rods can be continuously withdrawn. The causes of the event are summarized in Table 15.4-1. The probability for this event to occur is considered low enough to warrant its being categorized as an infrequent incident.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

The sequence of events of a typical continuous control rod withdrawal error during reactor startup is shown in Table 15.4-2.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to terminate this event, since the SRNM period-based trip functions will initiate and terminate this event.

15.4.1.2.3 Core and System Performance

15.4.1.2.3.1 Analysis Method and Analysis Assumptions

The analysis uses the reactivity insertion analysis code described in Reference 15.4-2. It is a two-dimensional adiabatic code assuming no heat transfer to the coolant. The analysis consists of three steps. In Step 1, with the error rods being continuously withdrawn from full-in, the model is used to calculate the average power and period change as a function of time with a continuous reactivity insertion simulating the RWE event. In Step 2, the power versus time data are used as input to a calculation of the SRNM rod block and scram trip times. Both the rod block trip and scram trip times are then determined. In Step 3, the reactivity insertion input to the adiabatic model is adjusted such that after the period reaches the rod block setpoint (20 s), there is no further reactivity insertion. The RWE transient is then recalculated by the model with the adjusted reactivity input. The reactor scram time is also adjusted based on the time determined in Step 2. The calculated fuel enthalpy does not consider local peaking effect. In Step 4, the peak fuel enthalpy that includes the local peaking effect is calculated.

Other assumptions used in the analysis are:

(1) The standard BWR data of the adiabatic model is used.

(2) The scram reactivity shape is derived from the design core, assuming no failing rods and same scram speed for all rods.

(3) Six delayed neutron groups are assumed.

15.4.1.2.3.2 Analysis Conditions and Results

(1) Analysis Conditions

(a) The reactor is assumed to be in the critical condition before the control rod withdrawal, with an initial power of 0.001% rated, and a temperature of 286°C at the fuel cladding surface.

(b) The worth of the withdrawn rods (gang) is 3%ý k from full-in to full-out. Gang rod withdrawal is used as during a normal startup.

(c) The control rod withdrawal speed is 30 mm/s, the nominal FMCRD withdrawal speed.

(d) With the gang rod withdrawal, the reactor period monitored by any SRNM is relatively the same. Any single channel bypass of the SRNM does not affect the result.

(2) Analysis Result

With this 3%ýk reactivity insertion, the flux excursion generates a period of approximately 4 seconds. The rod block trip is initiated at 14 seconds after the start of the transient. The scram is initiated at about 25 seconds. The event is terminated by the scram. The peak fuel enthalpy reached is approximately 69.5 J/g, which is 0.63 J/g higher than the initial fuel enthalpy. The result is illustrated in Figure 15.4-1.

15.4.1.2.3.3 Evaluation Based On Criteria

Due to the effective protection function of the period-based trip function, the fuel enthalpy increase is small. The criterion of 170 cal/gm for fuel enthalpy increase under RWE event is satisfied.

An additional analysis was performed with the same assumptions and conditions as stated in Subsections 15.4.1.2.3.1 and 15.4.1.2.3.2, but assuming no protection function from the SRNM. Under this condition, the APRM setdown scram trip at 15% power provides the protection function. Flux and power excursion caused by continuous rod withdrawal error reaches the 15% power scram level and the reactor scrams. The result showed that the final peak fuel enthalpy was approximately 146.5 J/g, lower than the RWE criteria for fuel integrity.

15.4.1.2.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no fuel damage in this event and only with mild change in gross core characteristics.

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Features of the ABWR Automatic Thermal Limit Monitoring System (ATLM)

In the ABWR, the Automatic Thermal Limit Monitoring (ATLM) System performs the rod block monitoring function. The ATLM System is a dual channel subsystem of the Rod Control and Information System (RCIS). In each ATLM channel there are two independent thermal limit monitoring devices. One device monitors the MCPR limit and protects the operating limit of the MCPR, and the other device monitors the APLHGR limit and protects the operating limit of the APLHGR. The rod block algorithm and setpoint of the ATLM System are based on actual online core thermal limit information. If any one of the two limits is reached, either due to control rod withdrawal or recirculation flow increase, control rod withdrawal permissive is removed. Detailed description of the ATLM System is presented in Reference 15.4-1 and Chapter 7.

15.4.2.2 Identification of Causes and Frequency Classification

The causes of a potential RWE transient are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function will block any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods will be stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no RWE transient event.

15.4.2.3 Sequence of Event and System Operation

Due to an operator error or a malfunction of the automated rod withdrawal sequence control logic, a single control rod or a gang of control rods is withdrawn continuously. The ATLM operating thermal limit protection function of either MCPR or MLHGR protection algorithm stops further control rod withdrawal when either operating limit is reached. There is no basis for occurrence of the continuous control rod withdrawal error event in the power range.

No operator action is required to preclude this event, because the plant design as described above prevents its occurrence.

15.4.2.4 Core and System Performance

The performance of the ATLM System of the RCIS prevents the RWE event from occurring. The core and system performance are not affected by such an operator error or control logic malfunction. There is no need to analyze this event.

15.4.2.5 Barrier Performance

An evaluation of the barrier performance is not made for this event, because there is no postulated set of circumstances for which this event could occur.

15.4.2.6 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluations presented in Subsections 15.4.1 and 15.4.2 and does not have to be reanalyzed for specific core configurations.

15.4.4 Abnormal Startup of Idle Reactor Internal Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from manual action by the operator to initiate pump operation. It is assumed that the remaining nine RIPs are already operating.

The normal restart procedure requires the operator to reduce the pump speeds of running RIPs to , at or near, their minimum speeds (i.e., 30% of rated speed) before the restart of the idle RIP.Plant operating procedures specify the maximum allowable speed for the nine operating RIPs, for a normal restart of one RIP. Therefore, an abnormal restart occurs only when an operator error (i.e., operator ignoring the procedure) occurs. Should an abnormal restart occur, the much higher reverse flow at the idle RIP requires the inverter to provide electrical current much higher than the normal. This overcurrent requirement activates the overcurrent protection logic of the adjusstable speed dirve (ASD) which supplies the power to the idle RIP. This ASD is tripped by the protection logic. Therefore, an abnormal restart of the idle RIP becomes a trip of one RIP, which is presented in Subsection 15.3.1.

15.4.4.1.1.1 Normal Restart of Reactor Internal Pump

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Reactor Internal Pump at High Power

This transient should be considered as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.4.4.2 Sequence of Events and Systems Operation

15.4.4.2.1 Sequence of Events

Table 15.4.3 lists the sequence of events for an abnormal startup of an idle RIP.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) Adjust rod pattern, as necessary, for new power level following idle RIP start

(2) Reduce the speed of the running RIPs to, at or near, their minimum speeds

(3) Start the idle loop pump and adjust speed to match the running RIPs (monitor reactor power)

(4) Readjust power, as necessary, to satisfy plant requirements per standard procedure

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF action occurs as a result of the event.

15.4.4.3 Core and System Performance

An abnormal restart of an idle RIP becomes a trip of one RIP event, which is presented in Subsection 15.3.1.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event because no significant pressure increases are incurred during this transient (Subsection 15.3.1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, because no radioactive material is released from the fuel.

15.4.5 Recirculation Flow Control Failure with Increasing Flow

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

The ABWR Recirculation Flow Control System (RFCS) uses a triplicated, fault-tolerant digital control system. The RFCS controls all ten reactor internal pumps (RIPs) at the same speed. As presented in Subsection 15.1.2.1.1, no credible single failure in the control system results in a maximum demand to all RIPs. A voter or actuator failure may result in an inadvertent runout of one RIP at its maximum drive speed (~40%/s). In this case, the RFCS senses the core flow change and commands the remaining RIPs to decrease speed and thereby automatically mitigate the transient and maintains the core flow.

As presented in Subsection 15.1.2.1.1, multiple failures in the control system might cause the RFCS to erroneously issue a maximum demand to all RIPs. Should this occur, all RIPs could increase speed simultaneously. Each RFCS processing channel has a speed demand limiter which limits the maximum speed change rate to 5%/s. However, the probability of this event occurring is very low, and, hence, the event should be considered as a limiting fault.

15.4.5.1.2 Frequency Classification

15.4.5.1.2.1 Fast Runout of One Reactor Internal Pump

The failure rate of a voter or an actuator is very low. However, it is analyzed as an incident of moderate frequency.

15.4.5.1.2.2 Fast Runout of All Reactor Internal Pumps

This event should be considered as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

15.4.5.2.1.1 Fast Runout of One Reactor Internal Pump

Table 15.4.4 lists the sequence of events for Figure 15.4-2.

15.4.5.2.1.2 Fast Runout of All Reactor Internal Pumps

Table 15.4.5 lists the sequence of events for Figure 15.4-3

15.4.5.2.1.3 Identification of Operator Actions

The operator should:

(1) Transfer flow control to manual and reduce the flow to minimum

(2) Identify cause of the failure

Reactor pressure is controlled as required, depending on whether scram occurs and, if scram occurs, whether a restart or cooldown is planned. In general, following a scram, the corrective action is to hold reactor pressure and condenser vacuum for restart after the malfunction has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

(1) Observe that all rods are in

(2) Check the reactor water level and maintain above low level (L2) trip to prevent RCIC initiation

(3) Switch the reactor mode switch to the STARTUP position

(4) Maintain vacuum and turbine seals

(5) Transfer the recirculation flow controller to the manual position and reduce setpoint to zero

(6) Survey maintenance requirements and complete the scram report

(7) Monitor the turbine coastdown and auxiliary systems

(8) Establish a restart of the reactor per the normal procedure

15.4.5.2.2 Systems Operation

The analysis of this transient assumes and takes credit for normal functioning of plant instrumentation and controls and the reactor protection system. Operation of engineered safeguards is not expected.

15.4.5.3 Core and System Performance

15.4.5.3.1 Input Parameters and Initial Conditions

In each of these events, the most severe consequences result when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is 59% NBR power and 42% core flow. The maximum speed increasing rate of 40%/s is assumed for one RIP runout.

For all RIPs runout, 5%/s is assumed for the speed limit. The maximum core flow achieved by all RIPs runout is conservatively assumed to be 120% of rated.

15.4.5.3.2 Results

15.4.5.3.2.1 Fast Runout of One Reactor Internal Pump

Figure 15.4-2 presents the analysis of a fast runout of one RIP with its maximum speed increase rate of 40%/s. Table 15.4.4 provides the sequence of events of this failure.

The increase in core flow causes a rise in neutron flux. The peak neutron flux reached is 90% of NBR value, which is below the high neutron flux scram setpoint. The accompanying average fuel surface heat flux reaches 68% of NBR (116.1% of initial) at approximately 5.0 s and average fuel temperature increases 35°C. Acceptance Criterion II.2(b) of SRP Section 15.4.4 provides that fuel clad integrity shall be maintained by ensuring that the CPR remains above the MCPR safety limit. Because this event does not result in a significant increase in pressure and it is initiated from a low power condition, no MCPR calculation was performed.

Reactor pressure is presented in Subsection 15.4.5.4.

15.4.5.3.2.2 Fast Runout of All Reactor Internal Pumps

Figure 15.4-2 illustrates the fast runout of all RIPs with a maximum speed increase rate of 5%/s. Table 15.4.5 shows the sequence of events for this failure. Flux scram occurs at approximately 8.6 s, peaking at 135% of NB rated, while the average surface heat flux reaches 99% of NB rated (168.1% of initial) at approximately 9.2 s. No fuel failure is expected. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Runout of One Reactor Internal Pump

This transient results in a slight increase in reactor vessel pressure (Figure 15.4-2) and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Runout of All Reactor Internal Pumps

This transient results in a slight increase in reactor vessel pressure (Figure 15.4-3) and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event because no radioactive material is released from the fuel.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs. This is a PWR event.

15.4.7 Mislocated Bundle Accident

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the loading of a fuel bundle in an improper location and subsequent operation of the core. Three errors must occur for this event to take place in the equilibrium core loading. First, a bundle must be placed into a wrong location in the core. Second, the bundle which was supposed to be loaded where the error occurred is also put in an incorrect location or discharged. Third, the mislocated bundles are overlooked during the core verification process performed following core loading.

Provisions to prevent potential fuel loading errors are included in the plant Operating Procedures/Technical Specification.

15.4.7.1.2 Frequency Classification

This unlikely event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed that the bundle is misplaced in the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as a limiting fault, because the expected frequency is very low based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the mislocated bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A mislocated bundle error, undetected by incore instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.7.3 Core and System Performance

Mislocated bundle analyses are not performed for reload cores because, based on analysis of data available from past reloads, the probability that a mislocated fuel bundle loading error will result in a CPR less than the safety limit is sufficiently small.

For ABWR initial core, the mismatch of exposures and integrated bundle power between misloaded bundles are less severe than the equilibrium cycle. Therefore, the consequence of a postulated MBA for the initial core is less severe than that for the equilibrium cycle. Consequently, the conclusion drawn from the reload core analysis as previously presented is applicable to the ABWR initial core. Hence, no specific analysis is required.

The COL applicant will provide an analysis to confirm that the consequences of a fuel bundle mislocated event meet all requirements approved by the NRC. See Subsection 15.4.11.1 for COL license information.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance is not made for this event, because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.7.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event because no radioactive material is released for the fuel.

15.4.8 Misoriented Fuel Bundle Accident

15.4.8.1 Identification of Causes and Frequency Classification

15.4.8.1.1 Identification of Causes

The misoriented fuel bundle (MOFB) event discussed in this section is the situation in which a bundle has been loaded in the correct location but is rotated by 90 or 180 degrees. The rotation could result in non-uniform water gaps which could cause an increase in local rod power through increased moderation. The initiator for a reactor with a MOFB is an operator placing the bundle into the core in a misoriented position. The next step in the accident progression is failure to detect the MOFB. A verification procedure is recommended to detect a MOFB. This verification procedure requires two core scans. One scan is with an underwater TV camera positioned close enough to read the bundle serial numbers on top of the lifting bail (first attribute) and to check the orientation of the bosses on the bail (second attribute). The other scan is with a TV camera positioned sufficiently above the core to allow viewing one complete 4 bundle cell for the following four attributes: boss on lifting bail, channel fasteners, channel buttons, and “cell look alike”. Two independent reviewers (checkers A and B) are recommended to verify tapes from the above procedure.

15.4.8.1.2 Frequency Classification

A generic model was developed based on the recommended verification procedure to quantify the probability of operating a reactor with a MOFB. An event tree was constructed to find this probability using human error rates from NUREG/CR-1278. The results show that the probability of operating the reactor with a MOFB is lower than the probability of a large break LOCA. However, since at the time of this submittal the NRC has not approved this classification, the MOFB has been treated as a moderately frequent event, and analyzed accordingly.

15.4.8.2 Sequence of Events and Systems Operation

15.4.8.2.1 Sequence of Events

The postulated sequence of events for the misoriented fuel bundle accident (MOFB) is presented in Table 15.4-7.

15.4.8.2.2 Systems Operation

A misoriented fuel bundle accident, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

15.4.8.3 Core and System Performance

The MOFB event was analyzed for a reference core loading utilizing a bundle which is very similar to the reference fuel bundle design. This bundle design is defined in Tab AY of Reference 15.4-3. The only difference in the MOFB bundle design slight modifications to the radial enrichment distribution which were made to reduce the ΔR-factor. The maximum ΔR-factor under rotated conditions was determined to be 0.035. The bundle used in this analysis exhibited energy capabilities equivalent to the reference bundle design and the 15% thermal margin requirement was maintained. The infinite lattice void coefficients for both designs were compared and there was no change. The methods for analyzing the misoriented fuel bundle are described in detail in Reference 15.4-4 and approved in Reference 15.4-5. The ΔCPR calculated for this event is reported in Table 15.0-2.

The COL applicant will provide an analysis to confirm that the consequences of a fuel bundle misoriented event meet all requirements approved by the NRC. See Subsection 15.4.11.2 for COL license information.

15.4.8.4 Barrier Performance

An evaluation of the barrier performance is not made for this event because it is a mild and highly localized event. No perceptible change in the core pressure is observed.

15.4.8.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event because no radioactive material is released from the fuel.

15.4.9 Rod Ejection Accident

15.4.9.1 Identification of Causes and Frequency Classification

The rod ejection accident is caused by a major break on the FMCRD housing, outer tube or associated CRD pipe lines. Due to a break of this type, the reactor pressure exerted on the CRD spud pushes down the hollow piston and the ballnut with a large force. The shaft screw and the motor are forced to unwind. A passive brake mechanism is installed in the FMCRD system to prevent the control rod from moving. The design of the brake is presented in Section 4.6.1. The probability of the initial causes (i.e., a CRD pipe line break or housing break) is considered low enough to warrant its being categorized as a limiting fault. Even if this accident does happen, the brake prevents the control rod from ejection. Should the brake fail, the check valve will serve as a backup brake to prevent the rod ejection.

15.4.9.2 Sequence of Events and Systems Operation

If a major break occurs on the FMCRD housing, the reactor pressure will provide forces that could cause the shaft screw to unwind. The FMCRD brake mechanism prevents the rod from moving. Therefore, no rod ejection can occur.

15.4.9.3 Core and System Performance

The FMCRD brake mechanism prevents this event from occurring. There is no need to analyze this event.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance is not made for this accident since there is no circumstance for which this event would occur.

15.4.9.5 Radiological Consequences

The radiological analysis is not required.

15.4.10 Control Rod Drop Accident

15.4.10.1 Features of the ABWR Fine Motion Control Rod Drives

As presented in Subsection 4.6.1, the Fine Motion Control Rod Drive(FMCRD) System has several new features that are unique compared with locking piston control rod drives.

In each FMCRD, there are dual Class 1E separation-detection devices that will detect the separation of the control rod from the CRD if the control rod is stuck and separated from the ballnut of the CRD. The control rods are normally inserted into the core and withdrawn with the hollow piston, which is connected with the control rod, resting on the ballnut. The separation-detection device is used at all times to ascertain that the hollow piston and control rod are resting on the ballnut of the FMCRD. The separation-detection devices sense motion of a spring-loaded support for the ball screw and, in turn, the hollow piston and the control rod. Separation of either the control rod from the hollow piston or the hollow piston from the ballnut will be detected immediately. When separation has been detected, the interlocks preventing rod withdrawal will operate to prevent further control rod withdrawal. Also, an alarm signal will be initiated in the control room to warn the operator.

There is also the unique highly reliable bayonet type coupling between the control rod blade and the control rod drive. With this coupling, the connection between the blade and the drive cannot be separated unless they are rotated 45 degrees. This rotation is not possible during reactor operation. There are procedural coupling checks to assure proper coupling. There is also the automated overtravel check in the RCIS logic during automated operation. Finally, there is the latch mechanism on the hollow piston part of the drive. If the hollow piston is separated from the ballnut and rest of the drive due to a stuck rod, the latch will limit any subsequent rod drop to a distance of 8 inches. More detailed descriptions of the FMCRD System are presented in Subsection 4.6.1.

15.4.10.2 Identification of Causes and Frequency Classification

For the rod drop accident to occur, it is necessary for such highly unlikely events as failure of both Class 1E separation-detection devices, or the failure of the rod block interlock, and the failure of the latch mechanism to occur simultaneously with the occurrence of a stuck rod on the same FMCRD. This would permit hollow piston separation from the ballnut.

Alternatively, separation of the blade from the hollow piston would require either that the control rod was installed without coupling and the coupling checks failed, or there is structural failure of this coupling. Under such circumstances of this coupling failure, the rod drop accident can only occur with the simultaneous failure of both separation-detection devices (or the failure of the rod block interlock), together with the occurrence of a stuck rod on the same FMCRD.

In either case, because of the low probability of such simultaneous occurrence of these multiple independent events, there is no basis to postulate this event to occur.

15.4.10.3 Sequence of Events and System Operation

15.4.10.3.1 Sequence of Events

The bayonet coupling and procedural coupling checks will preclude the uncoupling of the control rod from the hollow piston of the FMCRD. If the control rod is stuck, the separation-detection devices will detect the separation of the control rod and hollow piston from the ballnut of the FMCRD, and rod block interlock will prevent further rod withdrawal. The operator will be alarmed for this separation.

There is no basis for the control rod drop event to occur.

15.4.10.3.2 Identification of Operator Actions

No operator actions are required to preclude this event. However, the operator will be notified by the separation-detection alarm if separation is detected.

15.4.10.4 Core and System Performance

The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.

15.4.10.5 Barrier Performance

An evaluation of the barrier performance is not made for this accident, since there is no circumstance for which this event could occur.

15.4.10.6 Radiological Consequences

The radiological analysis is not required.

15.4.11 COL License Information

15.4.11.1 Mislocated Fuel Bundle Accident

COL applicants will provide an analysis to confirm that the consequences of a fuel bundle mislocated event meet all requirements approved by the NRC (Subsection 15.4.7.3).

15.4.11.2 Misoriented Fuel Bundle Accident

COL applicants will provide an analysis to confirm that the consequences of a fuel bundle misoriented event meets all requirements approved by the NRC (Subsection 15.4.8.3).

15.4.12 References

15.4-1 Not Used.

15.4-2 C. J. Paone and J. A. Woolley, “Rod Drop Accident Analysis for Large Boiling Water Reactors”, Licensing Topical Report, March 1972 (NEDO-10527, Supplements 1 and 2).

15.4-3 “GE Fuel Bundle Design”, NEDE-31152P, December 1988.

15.4-4 R.E. Engel (GE) to D.G. Eisenhut (NRC), “Fuel Assembly Loading Error”, MFN-219-77, November 30, 1977.

15.4-5 D.G. Eisenhut (NRC) to R.E. Engel (GE), MFN-200-78, May 8, 1978.

Table 15.4-1 Causes of Control Rod Withdrawal Error

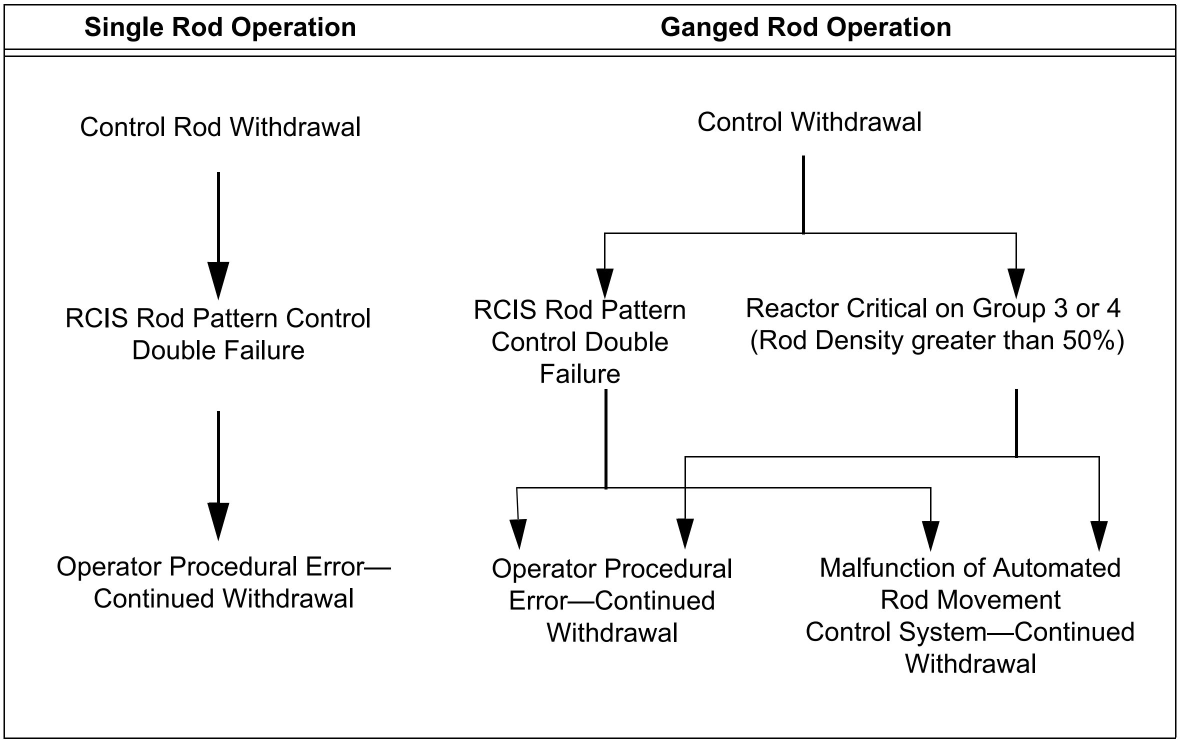


Table 15.4-2 Sequence of Events for Continuous Control Rod Withdrawal Error During Reactor Startup

Time (s)	Events
—	Rod Control & Information System (RCIS) logics to prevent continuous control rod withdrawal fail (from both channels).
0	Operator withdraws a gang of rods (or a single rod) continuously; or a gang of rods (or single rod) is withdrawn continuously due to a malfunction of the Automated Rod Movement Control System.
~6	Neutron flux increases rapidly due to the continuous reactivity addition, with a very short period.
14	The SRNM Period-Based Rod Block Trip initiates rod block due to short period (less than the 20-second setpoint).
24.8	The SRNM Period-Based Scram Trip initiates reactor scram due to short period (less than the 10-second setpoint).
~27	Reactor is scrammed and the event is terminated.

Table 15.4-3 Sequence of Events for Abnormal Startup of Idle RIP

Time (s)	Events
0	Operator starts idle RIP with running RIPs at higher than minimum speeds.
0	Interlock fails to prevent restart.
0.1 (estimated)	Overcurrent protection logic trips the electrical bus.
0.1 (estimated)	One or two RIPs are tripped due to the bus trip.
For Other Sequences of Events, see Table 15.3-1.	

Table 15.4-4 Sequence of Events for Figure 15.4-2

Time (s)	Events
0	Simulate fast runout of one RIP.
3.6	Neutron flux reaches its peak value.
10.0 (estimated)	Reactor variables settle into new steady-state.

Table 15.4-5 Sequence of Events for Figure 15.4-3

Time (s)	Event
0	Initiate fast runout of all RIPs.
8.6	Reactor APRM high flux scram trip initiated.
12.0 (estimated)	Turbine control valves start to close upon falling turbine pressure.
12.2	Four RIPs trip at vessel water level L3.
47.0 (estimated)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints.
>100 (estimated)	Reactor variables settle into new steady-state.

Table 15.4-6 Sequence of Events of the Mislocated Bundle Accident

(1)	During the core loading operation, a bundle is loaded into the wrong core location.
(2)	Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.
(3)	During the core verification procedure, the two errors are not observed.
(4)	The plant is brought to full power operation without detecting misplaced bundles.
(5)	The plant continues to operate throughout the cycle.

Table 15.4-7 Sequence of Events of the Misoriented Fuel Bundle Accident

(1)	During the core loading operation, a bundle is rotated 90 or 180 degrees.
(2)	During the core verification procedure, this error goes undetected.
(3)	The plant is brought to full power operation without detecting the misoriented bundle.
(4)	The plant continues to operate throughout the cycle.

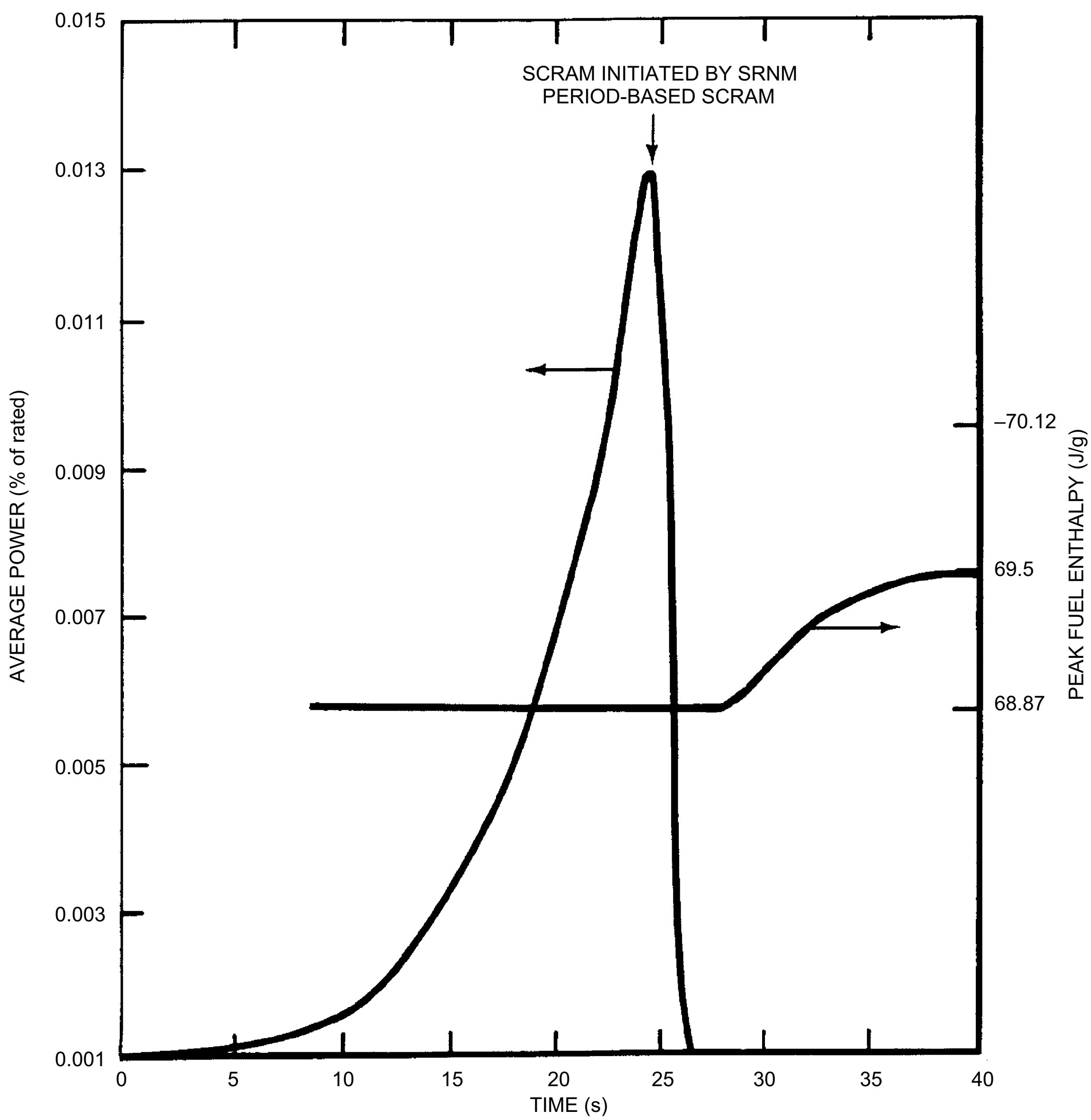


Figure 15.4-1 Transient Changes for Control Rod Withdrawal Error During Startup

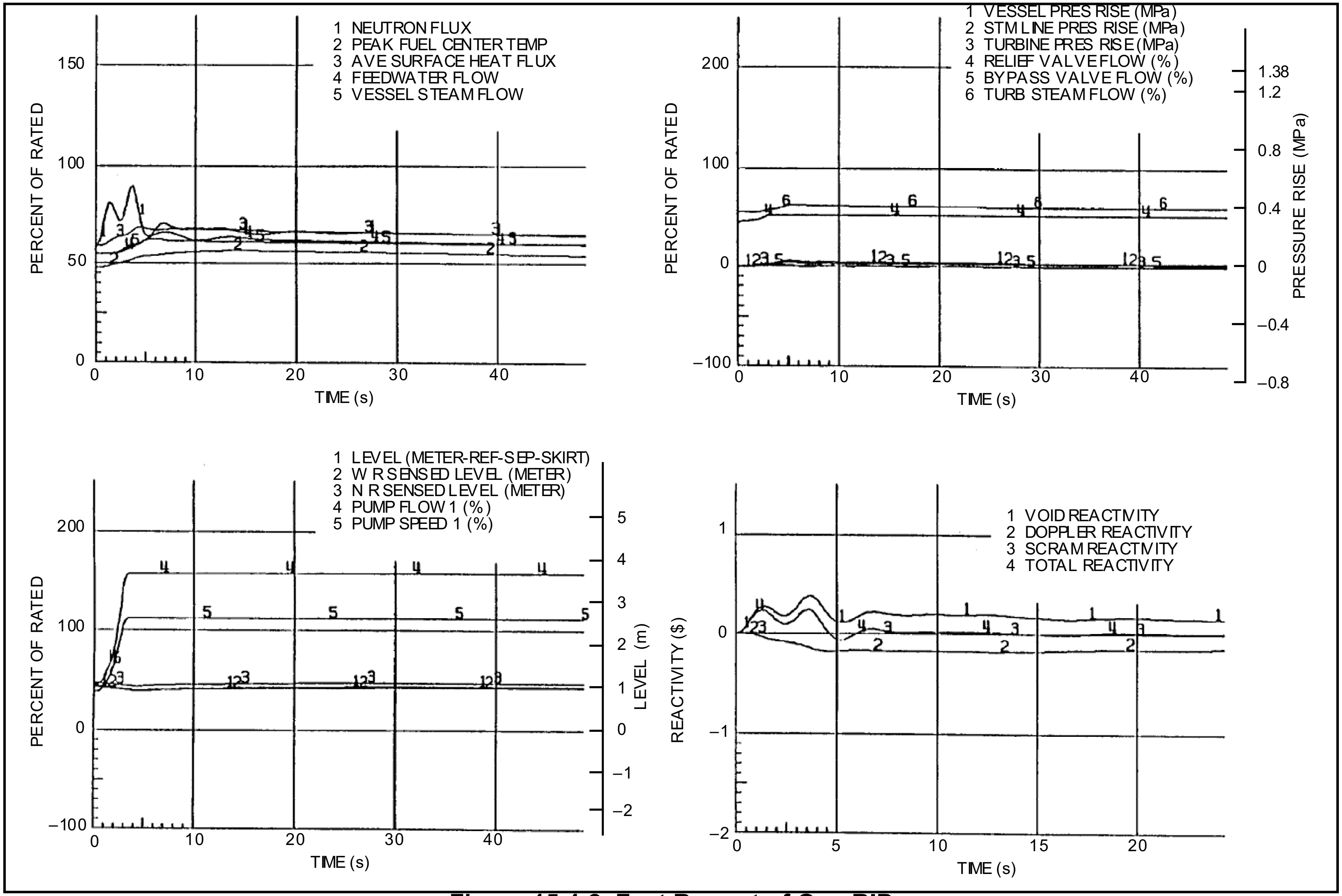


Figure 15.4-2 Fast Runout of One RIP

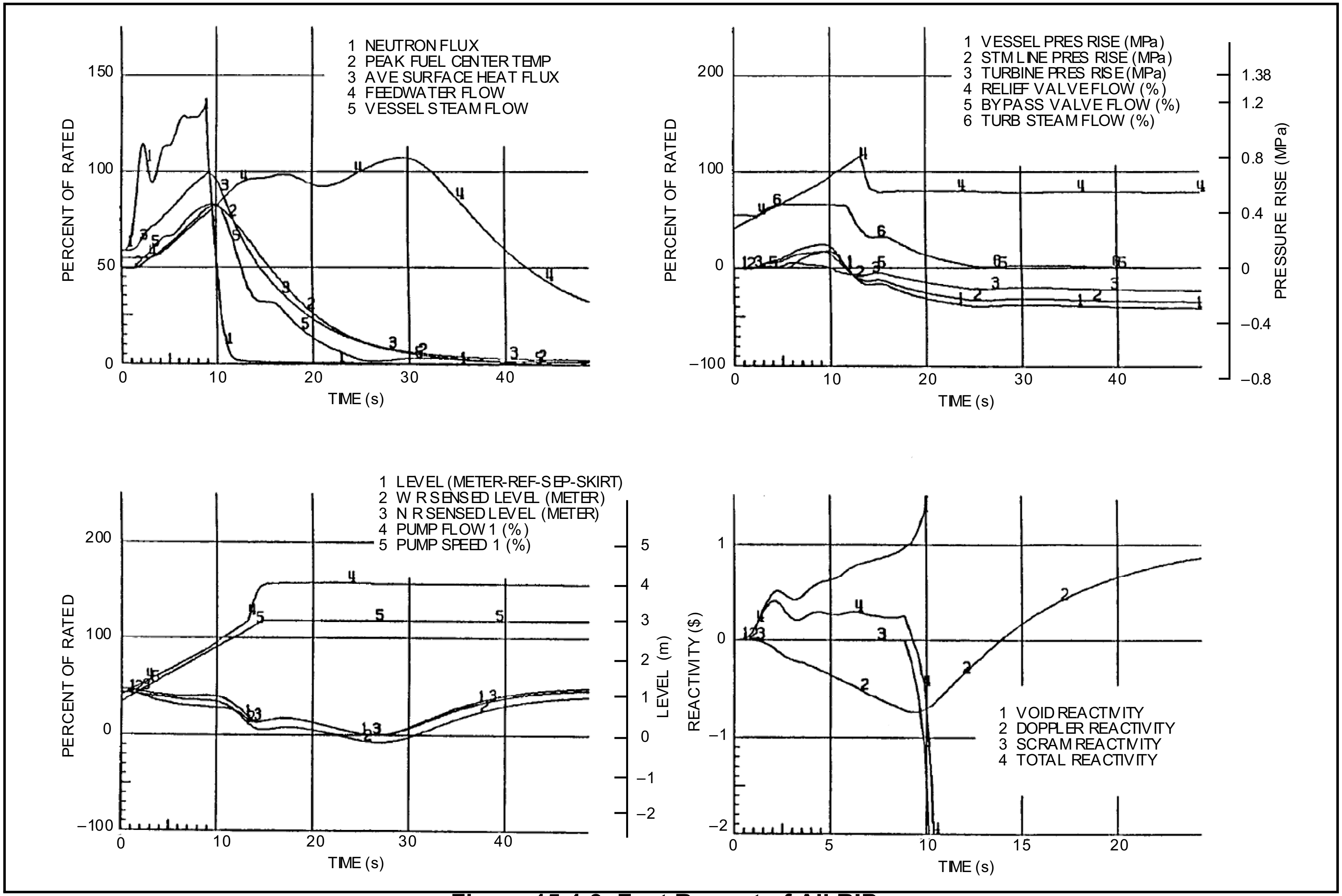


Figure 15.4-3 Fast Runout of All RIPs

15.5 Increase in Reactor Coolant Inventory

15.5.1 Inadvertent HPCF Startup

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCF System is postulated for this analysis (i.e., operator error).

15.5.1.1.2 Frequency Classification

This transient disturbance is categorized as an incident of moderate frequency.

15.5.1.2 Sequence of Events and System Operation

15.5.1.2.1 Sequence of Events

Table 15.5-1 lists the sequence of events for Figure 15.5-1.

15.5.1.2.1.1 Identification of Operator Actions

Small changes in plant conditions are experienced. The operator should, after hearing the alarm that the HPCF System has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator shuts down the system.

15.5.1.2.2 System Operation

To properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant instrumentation and controls—specifically, the pressure regulation and the vessel level control which respond directly to this event.

Required operation of engineered safeguards other than what is described is not expected for this transient event.

The system is assumed to be in the manual flow control mode of operation.

15.5.1.3 Core and System Performance

15.5.1.3.1 Input Parameter and Initial Conditions

The water temperature of the HPCF System is assumed to be 4.4°C with an enthalpy of 25.6 J/g.

Inadvertent startup of the HPCF System is chosen to be analyzed, because it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.3.2 Results

Figure 15.5-1 shows the simulated transient event for the manual flow control mode. It begins with the introduction of cold water into the upper core plenum. Within 1 s, the full HPCF flow is established at approximately 3.2% of rated feedwater flow rate. This flow is nearly 138% of the HPCF flow at rated pressure. No delays are considered because they are not relevant to the analysis.

Addition of cooler water to the upper plenum causes a reduction in steam flow, which results in some depressurization as the pressure regulator responds to the event. The flux level settles out slightly below operating level. Pressure and thermal variations are relatively small and no significant consequences are experienced. MCPR remains above the safety limit and, therefore, fuel thermal margins are maintained. Therefore, this event does not have to be reanalyzed for specific core configurations.

15.5.1.3.3 Consideration of Uncertainties

Important analytical factors, including reactivity coefficient and feedwater temperature change, are assumed to be at the worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 Barrier Performance

Figure 15.5-1 shows a slight pressure reduction from initial conditions; therefore, no further evaluation is required, as RCPB pressure margins are maintained.

15.5.1.5 Radiological Consequences

Because no activity is released during this event, a detailed evaluation is not required.

15.5.2 Chemical Volume Control System Malfunction (or Operator Error)

This section is not applicable to the BWR.

15.5.3 BWR Transients Which Increase Reactor Coolant Inventory

These events are presented and considered in Sections 15.1 and 15.2.

Table 15.5-1 Sequence of Events for Figure 15.5-1

Time (s)	Event
0	Simulate HPCF cold water injection.
1	Full flow established for HPCF.
2	Depressurization effect stabilized.

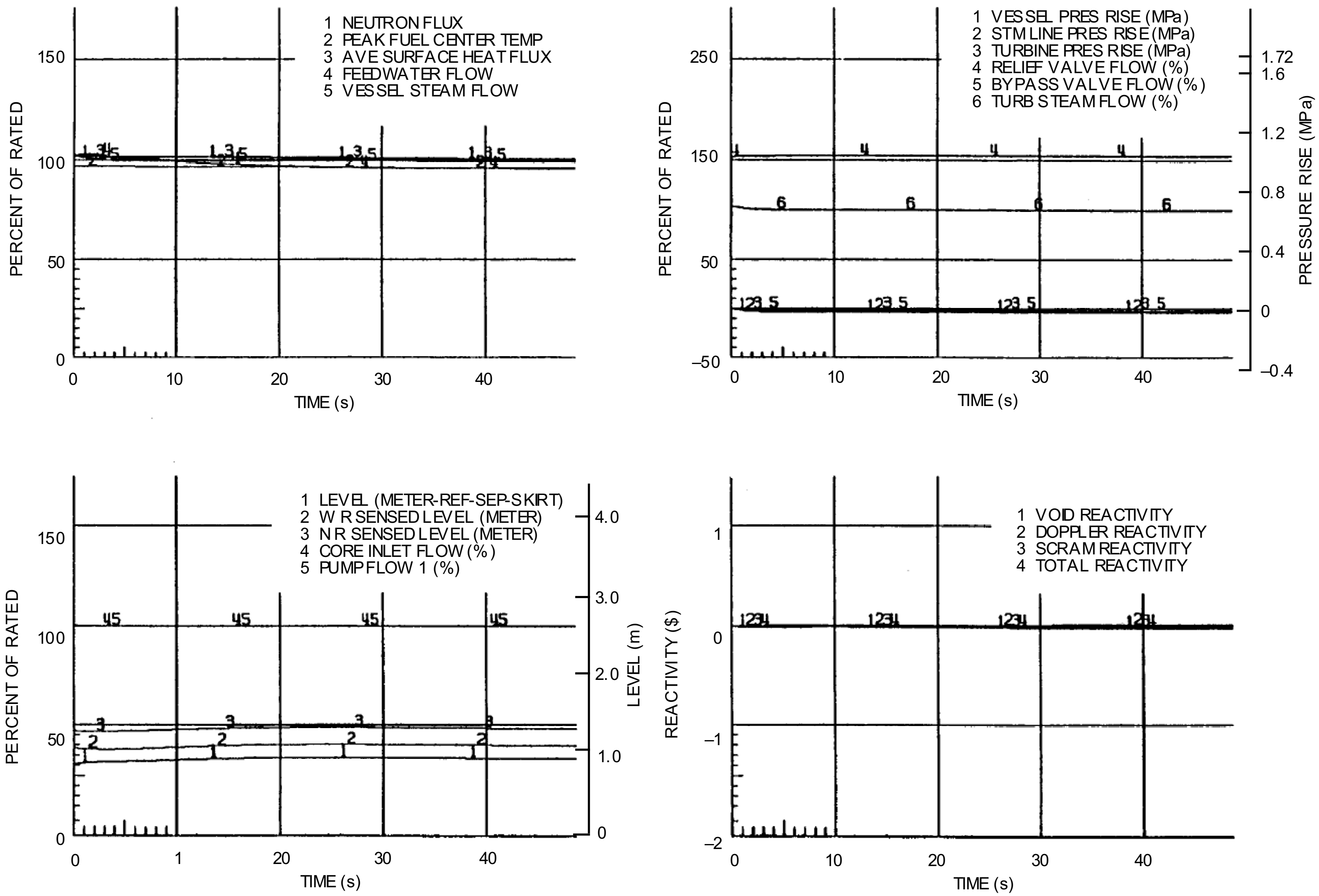


Figure 15.5-1 Inadvertent Startup of HPCF

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Safety/Relief Valve Opening

This event is presented and analyzed in Subsection 15.1.4.

15.6.2 Failure of Small Line Carrying Primary Coolant Outside Containment

This event postulates a small steam or liquid line pipe break inside or outside the primary containment, but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent. This event is less limiting than the postulated events presented in Subsections 15.6.4, 15.6.5, and 15.6.6.

This postulated event represents the envelope evaluation for small line failure inside and outside the primary containment relative to sensitivity for detection.

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. These lines are designed to high quality engineering standards and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

A circumferential rupture of an instrument line which is connected to the Primary Coolant System is postulated to occur outside the drywell, but inside the reactor building. This event could conceivably occur also in the drywell. However, the associated effects would not be as significant as those from the failure in the reactor building.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Systems Operations

15.6.2.2.1 Sequence of Events

The leak may result in noticeable increases in radiation, temperature, humidity, or noise levels in the secondary containment or abnormal indications of actuations caused by the affected instrument.

Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of the orderly shutdown and depressurization of the reactor vessel.

15.6.2.2.2 Systems Operation

A presentation of plant and reactor protection system action and ESF action is given in Sections 6.3, 7.3, and 7.6.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

There is no single failure or operator error that will significantly affect the system response to this event.

15.6.2.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.6.4, 15.6.5, and 15.6.6. Consequently, instrument line breaks are considered to be bounded specifically by the steamline break (Subsection 15.6.4). Details of this calculation, including those pertinent to core and system performance, are presented in Subsection 15.6.4.3.

15.6.2.3.1 Input Parameters and Initial Conditions

All information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steamline break) event are presented in Section 6.3.

15.6.2.3.2 Results

No fuel damage or core uncovering occurs as a result of this accident. Similarly, instrument line breaks are within the spectrum considered in ECCS performance calculations presented in Subsection 6.3.3.

15.6.2.4 Barrier Performance

The following assumptions and conditions are the basis for the mass loss during the release period of this event:

(1) The instrument line releases coolant into the Reactor Building for a period of ten minutes at normal operating temperature and pressure. Following this 10-minute period, the operator is assumed to have isolated the event and taken steps to SCRAM the reactor to reduce reactor pressure over a period of 5.4 hours.

(2) The flow from the instrument line is limited by reactor pressure and a 0.64 cm diameter flow restricting orifice inside the drywell. The Moody critical blowdown model is applicable, and the flow is critical at the orifice (Reference 15.6-1).

The total integrated mass of fluid released into the Reactor Building is 5442 kg, with approximately 2270 kg being flashed to steam.

15.6.2.5 Radiological Analysis

15.6.2.5.1 General

The radiological analysis is based upon conservative assumptions considered acceptable to the NRC. Though the Standard Review Plan does not provide detailed guidance, the assumptions found in Table 15.6-1 assume that all of the iodine available in the flashed water is transported via the HVAC System or blowout panels to the environment without prior treatment by the Standby Gas Treatment System. Other isotopes in the water contribute only negligibly to the total dose.

15.6.2.5.2 Fission Product Release

The iodine activity in the coolant is assumed to be at the maximum equilibrium Technical Specification limit (see Subsection 15.6.4.5.1.1, Case 1) for continuous operation. The iodine released to the Reactor Building atmosphere and to the environment are presented in Table 15.6-2.

15.6.2.5.3 Results

Results of the analysis (Table 15.6-3) are within the 10% of 10CFR100 specified in the Standard Review Plan. COL applicants need to update the analysis to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information).

15.6.3 Steam Generator Tube Failure

This section is not applicable to the direct cycle BWR.

15.6.4 Steam System Piping Break Outside Containment

This event involves postulating a large steamline pipe break outside containment. It is assumed that the largest steamline, instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steamline failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steamline break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steamline rupture, the failure of a main steamline is assumed to occur.

15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.4.2 Sequence of Events and Systems Operation

15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the result of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting event for breaks outside the containment is a complete severance of one of the four main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.6-4.

The reactor operator maintains reactor vessel water inventory and core cooling with the RCIC System or with one of the HPCF Systems. Without operator action, the RCIC and the HPCF Systems would initiate automatically on low water level following isolation of the mainsteam supply system (i.e., MSIV closure). The core remains covered throughout the accident and there is no fuel damage.

15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle. Flow from the downstream side is initially limited by the total area of the flow restrictors within the reactor vessel steam outlet nozzles for the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

Discussions of plant and reactor protection system action and ESF action are presented in Sections 6.3, 7.3 and 7.6.

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The steamline break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in Subsection 6.3.3.3. For the steamline break outside the containment, because the break is isolatable, either the RCIC System or one of the HPCF systems can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either one of the HPCF systems or the RCIC System would still allow sufficient flow to keep the core covered with water (see Section 6.3 and Appendix 15A for analysis details).

15.6.4.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are presented in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

15.6.4.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 6.3-1.

15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to Section 6.3 for ECCS analysis.

15.6.4.4 Barrier Performance

Because this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs:

(1) The reactor is operating at the power level associated with maximum mass release.

(2) Nuclear system pressure is initially 7.17 MPa.

(3) An instantaneous circumferential break of the main steamline occurs.

(4) Isolation valves start to close at 0.5 s on high flow signal and are fully closed at 5.5 s.

(5) The Moody critical flow model (Reference 15.6-1) is applicable.

Initially, only steam will issue from the broken end of the steamline. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steamline break is 34,817 kg (21,949 kg of liquid and 12,868 kg of steam).

15.6.4.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the “design basis analysis.”

A schematic of the release path is shown in Figure 15.6-1.

15.6.4.5.1 Design Basis Analysis

The specific models, assumptions and the program used for computer evaluation are described in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-5.

General Compliance or Alternate Approach Statement (RG 1.5):

This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a steamline break accident for a BWR.

The key implementation assumptions used by General Electric in the analyses are as follows:

(1) All regulatory position requirements implemented.

(2) Site boundary and LPZ χ/Q are in conformance with NRC Regulatory Guide 1.145.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The impact of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ABWR standard design. The resultant dose is within regulatory limits.

15.6.4.5.1.1 Fission Product Release from Fuel

There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break. This level of activity is consistent with an offgas release rate of 3.7 GBq/s for Case 1 and 14.8 GBq/s for Case 2 referenced to a 30 minute decay. The iodine concentration in the reactor coolant is:

	MBq/g	
	Case 1	**Case 2**
I-131	0.001739	0.03515
I-132	0.01536	0.30747
I-133	0.01206	0.24161
I-134	0.02634	0.52688
I-135	0.01647	0.3293

Other isotopes of high intrinsic activity such as N-16 have been precluded due to their extremely short half lives.

15.6.4.5.1.2 Fission Product Transport to the Environment

The transport pathway is a direct unfiltered release to the environment. The MSIV detection and closure time of 5.0 s (maximum MSIV closing time and instrument delay) results in a discharge of 12,870 kg of steam and 21,953 kg of liquid from the break. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in Table 15.6-6.

15.6.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-7 and are less than the guidelines of 10CFR100. COL applicants need to update the calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information.).

15.6.5 Loss-of-Coolant Accident (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary)—Inside Containment

This event postulates a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also assumed to be coincident with a safe shutdown earthquake (SSE) for the mechanical design of components.

The event has been analyzed quantitatively in Sections 6.3 (Emergency Core Cooling Systems), 6.2 (Containment Systems), 7.3 and 7.1 (Instrumentation and Controls), and 8.3 (Onsite Power Systems). Therefore, the following discussion provides only information not presented in the subject sections. All other information is cross-referenced.

The postulated event represents the envelope evaluation for liquid or steamline failures inside containment.

15.6.5.1 Identification of Causes and Frequency Classification

15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a loss-of-coolant accident coincident with an SSE. The subject piping is of high quality, designed to construction industry codes and standards, and for seismic and environmental conditions. However, because such an accident provides an upper limit estimate for the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.5.2 Sequence of Events and Systems Operation

15.6.5.2.1 Sequence of Events

The sequence of events associated with this accident is presented in Table 6.3-2 for core system performance.

Following the pipe break and scram, the MSIV begins closing on the low level 1.5 signal. The low water level or high drywell pressure signal initiates RCIC, HPCF and RHR flooding systems.

15.6.5.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator, after assuring that all rods have been inserted, should perform the following:

(1) Determine plant conditions by observing the annunciators.

(2) After observing that the ECCS flows are initiated, check that the diesel generators have started and are on standby condition and confirm that the Service Water System is operating in the LOCA mode.

(3) After the RHR System and other auxiliary systems are in proper operation, the operator should periodically monitor the oxygen concentration in the drywell and wetwell.

15.6.5.2.3 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe breaks, sizes and locations are presented in Sections 6.2 and 6.3, including the severance of main steamlines, ECCS lines, feedwater lines, or other process system lines. The most severe nuclear system effects and the greatest potential release of radioactive material to the containment result from a complete circumferential break of one of the two HPCF injection lines. The minimum required functions of any reactor and plant protection system are presented in Sections 6.2, 6.3, 7.3, 7.6 and 8.3, and Appendix 15A.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Models

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide conservative assessment of the expected consequences of this improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.3, 7.3, 7.6, 8.3 and Appendix 15A.

15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 6.3-1.

15.6.5.3.3 Results

Results of this event are presented in detail in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured.

Continued long-term core cooling is demonstrated. Radiological impact is minimized and within limits. Continued operator control and surveillance is examined and provided.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of any primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Sections 3.8.2.3, 3.6, and 6.2 for details and results of the analyses).

15.6.5.5 Radiological Consequences

Two specific analyses are provided for the evaluation of the radiological consequences of a design basis LOCA, one for offsite dose evaluations and the second for control room dose evaluations. Both analyses are based upon assumptions provided in Regulatory Guide 1.3 except where noted. The analysis is based upon a process flow diagram shown in Figure 15.6-2 and accident parameters specified in Table 15.6-8.

15.6.5.5.1 Fission Product Release and Pathways to the Environment

Fission product releases are based upon Regulatory Guide 1.3, in that it is assumed that of the fission products found in the core, 100% of the noble gases and 50% of the iodines are released from the core. Of these iodines, 50% are assumed to plate out, leaving 25% of the total core inventory of iodine airborne and available for release. The chemical species differentiation for the iodine isotopes released to the containment atmosphere is assumed as specified in Regulatory Guide 1.3 as 91% elemental form, 4% organic form, and the remaining 5% as particulate form. Following the release of fission products to the containment atmosphere from the reactor pressure vessel, the fission products are subject to holdup and radioactive decay, removal processes, and leakage to other plant areas and to the environment.

Two specific pathways are analyzed in releasing fission products to the environment. The first pathway is leakage to the Reactor Building (secondary containment) via penetrations and engineered safety feature (ESF) components. This leakage pathway is assumed as not greater than an equivalent release of 0.5% by weight per day of the primary containment free air weight per plant Technical Specifications. The secondary containment is a multi-compartment self-contained structure maintained at negative pressure with respect to the environment, thereby providing a significant holdup volume for fission product releases. All leakage pathways from the primary containment, except the main steamlines and the feedwater lines, terminate in the Reactor Building. Leakage through the steamlines is treated separately below, and leakage through the feedwater lines is assumed negligible assuming the proper isolation and filling of the feedwater lines upstream of the primary containment through the feedwater system. Flow through the Reactor Building/secondary containment is directed via the Standby Gas Treatment System to the plant stack through HEPA and charcoal filters. Credit is taken for holdup, assuming 50% mixing in the secondary containment without plateout and other removal processes except filtration in the SGTS (Table 15.6-8). It is assumed that for the first 20 minutes after an isolation signal, the SGTS is drawing the Reactor Building down to negative pressures, and therefore all leakage during this time period is assumed without effective filtration. Following this 20 minute period, full filtration is assumed for the remainder of the period.

The removal process in the primary containment and for leakages from the primary containment is described in the following sections. Subsection 15.6.5.5.1.1 discusses reductions in airborne iodine due to water attrition, while Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3 discuss removal processes for leakages downstream of the MSIVs.

15.6.5.5.1.1 Suppression Pool Scrubbing

The BWR suppression pool, though designed primarily as a pressure suppression mechanism for vessel blowdown, serves also as an excellent medium for the intrainment and capturing of all fission products except the noble gases. The design and operational characteristics of the BWR provide for a release pathway from the vessel and drywell into the suppression pool for all cases involving vessel depressurization and, therefore, for removal of fission products by scrubbing in the suppression pool. The NRC has accepted the fact that the suppression pool is capable of removing fission products and provides for credit to incorporate this phenomenon in design basis analysis by recourse to the requirements of Standard Review Plan 6.5.5. The requirements of SRP 6.5.5 state that any flow directed through the pool can be credited with a decontamination factor (DF) of 10 providing the requirements of Subsection II are met and that the total decontamination is a combination of the decontamination applied to flow through the pool to that fraction of the release which bypasses the pool. The following paragraphs describe the determination of the bypass fraction for the calculation of overall pool decontamination.

The requirements of Regulatory Guide 1.3 stipulate an instantaneous release of fission products from the vessel to the containment atmosphere. Coincident with an instantaneous release, under LOCA conditions, the BWR pressure vessel will be depressurized, resulting in the purging of the primary containment atmosphere to the suppression pool. This situation is shown in Figure

15.6-3, which shows the fractions of airborne particulate as a function of time in the drywell and wetwell airspaces, assuming a decontamination factor of 10 for that flow which is purged either through the horizontal vents or the safety/relief valves. The figure shows that the airborne inventory is reduced by almost a factor of ten within two minutes of the initiation of the blowdown event.

However, the application of the precepts of Regulatory Guide 1.3 do not indicate the most likely train of events in a core damage event, which is what is implied in the design basis release assumptions. Both Regulatory Guide 1.3 and its predecessor, TID-14844, are based upon non-mechanistic assumptions and devices and are in the process of being replaced. Therefore, consideration of a range of accident progressions beyond the rigidly narrow scope of Regulatory Guide 1.3 is given below to evaluate potential suppression pool bypass under more realistic conditions.

The basic assumption of this evaluation of suppression pool bypass conditions is that an event occurs which challenges the reactor core causing sufficient damage to release approximately half the fission product volatile iodines. Damage to the core is limited to this extent, implying the ability to recover core cooling and limit in-vessel damage. Such an assumption complies with the intent of design basis licensing, in that the exact means by which the core is challenged is not specified; but given the challenge, the response and adequacy of the plant design is tested. In addition, the assumption of resumption of core cooling and recovery with limited release is fully justifiable, since the ABWR incorporates multiple cooling modes with redundant safety grade cooling systems. Events leading to more significant core damage are not considered as design basis, since they assume massive damage with “multiple failures to the design safety systems.” Such events are of exceedingly low probability and are described and evaluated in Chapter 19. Therefore, broadly speaking, events which lead to the assumed damage can be divided into two categories, break and non-break. Break events are those through which primary coolant are released directly to the primary containment atmosphere, and non-break events are those in which the primary coolant boundary is not breached. Both types of events will be considered below to provide a bounding analysis for suppression pool bypass.

In considering the non-break events, core damage is primarily the result of failure to maintain proper core water level, resulting in uncovering the core with subsequent release of fission products upon overheating of the fuel rods. To consider the train of events in such a case, the MAAP code (see Appendix 19E for a description of the MAAP code) was used to model vessel response. Based upon the MAAP analysis, releases would begin shortly after core water level reaches the bottom of the core and would proceed rapidly. During this period, isolation of the Primary Coolant System and containment would have been automatically tripped on low water level and the MSIVs, as well as all the other isolation valves, would have tripped, effectively isolating all flow from the primary containment. Therefore, the released fission products would be exposed to three primary influences: (1) plateout and removal in the dryers and separators, (2) leakage from the MSIVs into the main steamlines, and (3) flow through the SRVs into the suppression pool.

The release of volatile fission products would occur over a period of 10–20 minutes, during which steam or hydrogen flow from the core region would be very small. Using an upper bound estimate of 2 kg/s of steam generation during this period, the vessel flushing rate would be once every ten minutes. Therefore, during this period, 0.13% of the flow would bypass the pool through MSIV leakage. The remaining fraction would be transported through the SRVs. Without recovery of cooling water after this period, significant damage would occur to the core beyond that of a design basis event. With the recovery of water, the energy generated from decay heat which would be evident in overall core temperature rise and core degradation would cause a rapid pulse of steam, resulting in purging of the pressure vessel of all airborne materials. Based upon the MAAP analysis, it is conservatively estimated that 9×10^3 kg of steam would be generated in a short period of time (on the order of minutes), resulting in a vessel purge rate of seven to eight complete exchanges. Therefore, effectively all fission products remaining airborne in the vessel or lines would be purged to the suppression pool. The effective pool bypass fraction would then be 0.13% for an integrated overall DF of 9.8 without credit for plateout or 4.9 with a factor of two plateouts.

The break case follows a similar logic. Initially, following a break, massive depressurization of the pressure vessel would occur, causing all non-condensables in the drywell to be purged into the wetwell air space through both the horizontal vents and the safety/relief valves. Isolation of the containment and associated lines would be automatically initiated on depressurization. Following this rapid depressurization, there would follow a period during which the water level in the vessel would drop to the bottom of the core, resulting in the eventual release of fission products from the core. Since in a break case the path of least resistance would be through the break, the fission products would be effectively purged to the drywell airspace. In this case, the temperatures and surface areas involved would provide adequate plateout areas to validate the Regulatory Guide 1.3 plateout factor of 2. Like the non-break case, the total release is limited, implying resumption of cooling and a massive release of steam upon resumption of cooling. In the case of reflood with a break, because of the large volume of the drywell, conservatively 80% of the drywell volume is purged during the reflood period. If complete mixing is assumed, which is reasonable because of the dynamic flows involved, it is then found that 55.6% of the airborne fission products are purged to the suppression pool in the few minutes needed to reflood the core. Therefore, in this case an integrated pool DF of 2 is calculated.

In summary, it is found that for DBA conditions, the suppression pool is capable of reducing the elemental and particulate airborne iodine inventory by a factor of 2. Credit is taken for the proper operation of redundant safety grade systems subject to the single-failure criteria.

15.6.5.5.1.2 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exists which permits the primary containment atmosphere, or in the non-break case pressure vessel air space, direct access to the main steamlines and that the MSIVs leak at the maximum technical specification. Furthermore, it is assumed that the most critical MSIV fails in the open position. Therefore, the total leakage through the steamlines is equal to the maximum technical specification for the plant.

The main steamlines are graded (Table 3.2-1) as Seismic Classification I Quality Group B from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV, thereby providing a qualified safety grade mitigation system for fission product leakage, which in this case is limited by the leakage criteria specific in the technical specifications for the MSIV. The primary purpose of this system is to stop any potential flow through the main steamlines. Downstream of the seismic restraint referred to above, the steamlines pass through the Reactor Building-Control Building interface into the steam tunnel located in the Control Building upper floor. This steam tunnel is a heavily-shielded Seismic Category I structure designed primarily to shield the Control Building complex. From the Control Building the steamlines pass through the Control Building-Turbine Building interface into the Turbine Building steam tunnel, which is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines and their associated branch lines outboard of the last Reactor Building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Subsection 3.2.5.3) which determine the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the primary containment through the main steamlines involves the determination of (1) probable and alternate flow pathways, (2) physical conditions in the pathways, and (3) physical phenomena which affect the flow and concentration of fission products in the pathways. The most probable pathway for fission product transport from the main steamlines is found to be from the outboard MSIVs into the drain lines coming off the outboard MSIV and then into the Turbine Building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines. Consideration of the main steamlines and drain line complex downstream of the Reactor Building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination.

(1) The main steamlines and drain lines are high quality lines inspected on a regular schedule.

(2) The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.

(3) The main steamlines and drain lines are enclosed in a shielded corridor which protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as-built inspection is required to verify that no damage could be expected from other components and structures in a SSE.

(4) The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to assure their survival under SSE conditions. A large database of experience in the survival of these types of components under actual earthquake conditions exists which proves this contention (Reference 15.6-4). In the case of the ABWR, further margin for survival can be expected, since the ABWR lines are designed through dynamic analysis to survive such events, whereas in the case of the actual experience database, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Therefore, based upon the facts above, the main steamlines and drain lines in the ABWR are used as mitigative components in the analysis of leakage from the MSIVs.

The analysis of leakage from the MSIVs follows the procedures and conditions specified in Reference 15.6-4. Two flow paths are analyzed for dose contributions. The first pathway through the drain lines is expected to dominate because of the incorporation of a safety grade isolation valve on the outboard drain line which will open the line for flow down the drain line under LOCA conditions. The second pathway through the main steamlines into the turbine is expected to carry less than 0.3% of the flow based upon a determination that the maximum leakage past the turbine stop valves with an open drain line would permit only 0.3% flow for the valves to operate within specification. Specific values used and results of the main steamline leakage analysis are given in Table 15.6-8.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

15.6.5.5.1.3 Condenser and Turbine Modeling

The condenser and turbine are modeled as detailed in Reference 15.6-4 with specific values used given in Table 15.6-8. Both volumes are modeled primarily as stagnant volumes, assuming the shutdown of all active components. Both turbine and condenser are used as mitigative volumes based upon the determination that such components designed to standard engineering practice are sufficiently strong to withstand SSE conditions due wholly to their design (Reference 15.6-4). The only requirement in the design of the condenser is that it be bolted to the building basemat to prevent walking during an earthquake. The turbine has no such restriction and may possibly move. The requirement on these components for purposes of mitigation is only that they survive as a volume and not that they provide functionality or leaktightness following an earthquake.

Release from the condenser/Turbine Building pathway is assumed via diffuse sources in the Turbine Building. The two major points of release in the Turbine Building are expected to be the truck doors at the far end of the Turbine Building and the maintenance panels located midway on the Turbine Building on the side opposite the service building. Releases are assumed to be ground level releases. See Subsection 15.6.5.5.3 for applicable meteorology.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

15.6.5.5.2 Control Room

The ABWR control room is physically integrated with the Reactor Building and Turbine Buildings and is located between these structures (Figure 15.6-4). During a LOCA, exposure to the operators will consist of contributions from airborne fission products entrained into the control room ventilation system and gamma shine from the Reactor Building and airborne fission products external to the Control Building. Of these contributions, the last two involving gamma shine are negligible, since the inhabited portions of the ABWR control room are physically located underground with sufficient shielding overhead (a minimum of 1.6 meters of concrete) and in the side walls (1.2 meters) to protect the operators from the normal steamline gamma shine. Such shielding is more than sufficient to protect the operators given any amount of airborne fission products.

Therefore, exposure to the operators will consist almost entirely of fission products entrained into the control room environment from the atmosphere. The ABWR control room uses a redundant safety grade HVAC System with 100 mm (four-inch) charcoal filters for removal of iodines and two wall-mounted automatically controlled intake vents. The locations of the vents are given in Figure 15.6-4. Because of the location of these vents, it cannot be assumed that at least one vent will be uncontaminated, given most conditions of meteorology. Therefore, no credit for dual intakes was taken. In addition, the location of these vents with respect to the potential release points shows that, given any wind flow condition, the vents may be contaminated only by a release from the Reactor Building or Turbine Building but not both. Nevertheless, for purposes of conservative calculations, it was arbitrarily assumed that for 30% of the time stagnant meteorological conditions were assumed such that the primary intake vent was contaminated by both sources. For the remaining 70% of the time, only the more significant source was assumed to contaminate the primary intake vent.

Infiltration of airborne contamination to the control room was considered negligible, owing to the pathway for access to the control room complex. Entry into the control room is via the Service Building and a labyrinth doorway entry system through double doors into the clean portions of the Service Building. From the Service Building, additional controlled access through double doors provides entry into the control room. In each of these entry/access door systems, positive pressure is maintained to vent infiltrated air to the outside and away from the control room complex. As such, no contamination is anticipated beyond the initial access entry way from which infiltrating air is purged to the environment.

Control room dose is based upon fission product releases modeled as described in Subsection 15.6.5.5.1 and the values presented in Table 15.6-8. Operator exposure was based upon those conditions given in Table 15.6-8 and occupancy factors as shown below derived from SRP 6.4. Meteorology was derived as is specified in Subsection 15.6.5.5.3.2.

Time	Occupancy Factor
0–1 day	1.0
1–4 days	0.6
>4 days	0.4

15.6.5.5.3 Meteorology

15.6.5.5.3.1 Offsite Meteorology

Tier 2 involves the use of a generic U.S. site which does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore, for the evaluation of offsite accident conditions, recourse was made to Regulatory Guides 1.145 and 1.3 for meteorological definitions. Specifically, the table found in Section C.2.g(4) of Regulatory Guide 1.3 was used to define the meteorological parameters for use with the models found in Regulatory Guide 1.145. All releases were defined as ground level incorporating building wake conditions using the minimum ABWR building cross section.

Unlike the other design basis accidents found in Chapter 15, the LOCA accident analysis requires the development of meteorological conditions over a 30 day period. To develop a bounding 30 day set of four χ/Q dispersion parameters, recourse was made to Regulatory Guide 1.3 and the metrological prescription found under Subsection 2.g. From this prescription, the χ/Q values for 30 days were “walked” in from a 4828 m LPZ to approximately 1140 meters where the 30 day thyroid dose became 3 Sv. By plotting these resulting four χ/Q values on log-log paper a straight line curve was established from which a 2-hour 95% LPZ χ/Q and annual average χ/Q value were back fitted with a small factor of conservatism in the derivation so that the resultant integrated dose was less than 300 Rem. The resultant straight line plot and χ/Q values are shown in Figure 15.6-6. The end points are the 95% 2-hour LPZ χ/Q of 4.11E-04 and annual average (8760 hour) χ/Q of 1.17E-06 from which the intermediate values given in Table 15.6-13 (shown as Chp 2 values) were derived as specified in Regulatory Guide 1.145.

15.6.5.5.3.2 Control Room Meteorology

No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. Unlike the offsite dose methodology, which is a relatively straight forward application of Regulatory Guides 1.3 and 1.145, the parameters and methods by which the control room intake concentrations can be calculated are poorly characterized and currently not codified in a usable form. Therefore, for application to the ABWR, a back-calculation was used to provide an estimate of the meteorological χ/Q dispersion parameters which would provide for the maximum acceptable dose under SRP 6.4. Since the calculation covers a period of 30 days, a variation in meteorological χ/Q was assumed for variations in wind direction and wind speed. The variation factors chosen were taken from Table 1 of Reference 15.6-3 and are shown below.

Time Period	Murphy-Campe χ/Q Improvement Factor
0–8 hours	1.0
8–24 hours	0.59
1–4 days	0.375
> 4 days	0.165

Also, since the control room may be contaminated from two physically separated sources, the Reactor Building stack base or the Turbine Building truck doors, reference was made to the most recently published work of Ramsdell to evaluate the differences in χ/Q for releases from each source to the control building. Using the methodology given in References 15.6-5 and 15.6-6, it was determined that releases from the Turbine Building at 108 meters from the control room intake would be a factor of six lower in concentration for an equal release than releases from the Reactor Building stack base at 41 meters from the nearest Control Building intake. Therefore, a factor of six improvement in χ/Q was assumed for releases from the Turbine Building.

For application to specific site analysis, two methods exist for determination of control room dose. The first method is a one-on-one comparison of the χ/Q values in Table 15.6-14 to the site χ/Qs. If the site χ/Qs are for all values less than the values in Table 15.6-14, then the control room doses are less than regulatory requirements. If this is not true, then a site specific calculation needs to be performed for the site. For this purpose, an isotope-by-isotope release rate table is given in Tables 15.6-10 and 15.6-12, from which actual calculations can be made.

15.6.5.5.4 Results

The results of this analysis are presented in Tables 15.6-13 and 15.6-14 for both offsite and control room dose evaluations and are within current regulatory guidelines. COL applicants need to update the analysis to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information).

15.6.6 Cleanup Water Line Break—Outside Containment

To evaluate liquid process line pipe breaks outside containment, the failure of a cleanup water line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the cleanup water line, representing the most significant liquid line outside containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break—Inside Containment) has been quantitatively analyzed and is presented in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is cross-referenced to appropriate Chapter 6 subsections.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A cleanup water line break is assumed without the cause being identified. The subject piping is designed to high quality, to strict engineering codes and standards, and to seismic environmental requirements.

15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault (liquid line break).

15.6.6.2 Sequence of Events and System Operation

15.6.6.2.1 Sequence of Events

The sequence of events is presented in Table 15.6-15.

15.6.6.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator should perform the following (shown for informational purposes only):

(1) determine that a line break has occurred

(2) ensure that if vessel water level is below level 3 that reactor has scrammed,

(3) monitor vessel water level and ensure actuation of ECCS as needed, and

(4) implement site radiation incident procedures.

These actions occur over an elapsed time of 3–4 hours.

15.6.6.2.3 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS. The reactor protection system (safety/relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF Systems and HPCF System are assumed to operate normally. RCIC will automatically isolate to high RCIC room temperature caused by steam escaping from the break prior to closing the isolation valves.

15.6.6.2.4 The Effect of Single Failures and Operator Errors

The cleanup water line outside the containment is a special case of the general LOCA break spectrum presented in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in detail in Subsection 6.3.3.3. For the cleanup water line break outside the containment, because the break is isolatable, either the RCIC System or one of the HPCF Systems provides adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either one of the HPCF Systems or the RCIC System still provides sufficient flow to keep the core covered with water (see Section 6.3 and Appendix 15A for analysis details).

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative and envelope assessment of the consequences of the postulated failure (i.e., severance) of one the feedwater piping lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions presented in Table 6.3-1.

15.6.6.3.2 Qualitative Results

The cleanup water line break outside the containment is less limiting than either of the steamline breaks outside the containment (analysis presented in Sections 6.3 and/or 15.6.4), or the feedwater line break inside the containment (analysis presented in Subsections 6.3.3 and 15.6.5).

The reactor vessel is isolated on water level L1.5, and the RCIC and the HPCF Systems together restore the reactor water level to the normal elevation if needed. The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (see Section 6.3 for details).

15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.6.4. The cleanup water system piping break is less severe than the main steamline break. Results of analysis of this event can be found in Subsections 6.2.3 or 6.2.4.

15.6.6.5 Radiological Consequences

15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC.

15.6.6.5.2 Analysis

The analysis is based on a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are presented in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-16. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-5.

15.6.6.5.2.1 Fission Product Release

There is no fuel damage as a consequence of this accident.

At the initiation of this accident it is assumed that the total non-filtered inventory in both the regenerative and non-regenerative heat exchangers is released through the break. Inventory in the demineralizer is prevented from being released by back flow check valves from exiting that component. A break on the downstream side of the demineralizer would be bounded due to the demineralizer action compared to a break on the upstream side of the demineralizer.

Isolation of the CUW line is conservatively analyzed based upon actuation of the flow differential pressure instrumentation. This instrumentation has a built in 45 second time delay so that for the initial 45 seconds of the accident full flow through the CUW line subject to flow restriction by a 140 cm^2 flow restrictor located in the primary containment. After the initial 45 second flow, motor operated isolation valves will close over a period of 30 seconds. During this period of 75 seconds, flow of reactor water is assumed at the maximum equilibrium reactor water concentration given in Subsection 15.6.4.5.1.1, case 1, with flashing to steam at reactor temperature and pressure. In addition, iodine spiking based upon a differential reactor depressurization from 7.24 MPa to 6.69 MPa in 20 seconds and using the spiking source terms given in Table 15.6-16 is assumed. Noble gas activity in the reactor coolant is negligible and is therefore ignored in this analysis.

15.6.6.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of a tortuous path from the lowest levels of the reactor building through designed rupture disks to the pipe chases terminating with flow directed into the main steam tunnel. The main steam tunnel incorporates an over pressurization flow chimney which will route the flow finally to turbine building upper deck. Flow ejected into this area will most probably be entrained into the turbine building HVAC and directed to the plant stack. However credit for this pathway to the stack is not assumed and releases to the turbine building are considered environmental releases out turbine building doors. Because the release pathway experiences significant surface areas in the reactor building, steam tunnel and turbine building, a credit for iodine plateout of 0.5 is assumed.

15.6.6.5.2.3 Results

The calculated exposures for the analysis are presented in Table 15.6-18 and are a small fraction of 10CFR100 guidelines. COL applicants need to update the calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information.).

15.6.7 COL License Information

15.6.7.1 Iodine Removal Credit

The COL applicant will recalculate iodine removal credit as outlined in Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3.

15.6.8 References

15.6-1 F.J. Moody, “Maximum Two-Phase Vessel Blowdown from Pipes”, ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-2 H.A. Careway, V.D. Nguyen, and P.P. Stancavage, “Radiological Accident Evaluation - The CONAC03 Code”, December 1981 (NEDO-21143-1).

15.6-3 K.G. Murphy, and K.M. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19”, 13th ASC Air Cleaning Conference, June 1974.

15.6-4 L.S. Lee, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems”, February 1991 (NEDC-31858P).

15.6-5 J.V. Ramsdell, “Atmospheric Diffusion for Control Room Habitability Assessments”, May 1988 (NUREG/CR-5055).

15.6-6 Ramsdell, J.V., “Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes”, 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

Table 15.6-1 Instrument Line Break Accident Parameters

I	Data and assumptions used to estimate source terms	
	A. Power level	4005 MWt
	B. Mass of fluid released	13610 kg
	C. Mass of fluid flashed to steam	2270 kg
	D. Duration of accident	8 h
	E. Number of bundles in core	872
II	Data and assumptions used to estimate activity released	
	A. Iodine water concentration	15.6.4.5.1.1, case 1
	B. Iodine Spiking (MBq/bundle)	
	I-131	7.77E+04
	I-132	1.18E+05
	I-133	1.85E+05
	I-134	2.00E+05
	I-135	1.78E+05
	C. Iodine plateout fraction	50%
	D. Reactor Building Flow rate	200%/h
	E. SGTS Filter Efficiency	None assumed
III	Dispersion and Dose Data	
	A. Meteorology	Table 15.6-3
	B. Boundary and LPZ distances	Table 15.6-3
	C. Method of Dose Calculation	Reference 15.6-2
	D. Dose conversion assumptions	Reference 15.6-2, RG 1.109, and ICRP 30
	E. Activity Inventory/releases	Table 15.6-2
	F. Dose Evaluations	Table 15.6-3

Table 15.6-2 Instrument Line Break Accident Isotopic Inventory

Reactor Building Inventory (Megabecquerel)						
Isotope	**1- min**	**10-min**	**1-hour**	**2-hour**	**4-hour**	**8-hour**
I-131	3.77E+01	3.27E+02	2.60E+04	1.73E+04	1.38E+04	4.59E+00
I-132	3.68E+02	3.11E+03	2.31E+05	1.44E+05	1.17E+05	1.17E+01
I-133	2.59E+02	2.24E+03	1.75E+05	1.16E+05	9.29E+04	2.72E+01
I-134	7.22E+02	5.92E+03	3.89E+05	2.26E+05	1.86E+05	2.65E+00
I-135	3.77E+02	3.25E+03	2.52E+05	1.64E+05	1.32E+05	2.90E+01
Total	1.76E+03	1.48E+04	1.07E+06	6.68E+05	5.41E+05	7.52E+01
Isotopic Release to Environment (Megabecquerel)						
Isotope	**1- min**	**10-min**	**1-hour**	**2-hour**	**4-hour**	**8-hour**
I-131	6.36E–01	5.77E+01	2.77E+04	6.81E+04	1.27E+05	1.41E+05
I-132	6.18E+00	5.51E+02	2.52E+05	5.96E+05	1.09E+06	1.19E+06
I-133	4.37E+00	3.96E+02	1.87E+05	4.59E+05	8.51E+05	9.44E+05
I-134	1.21E+01	1.06E+03	4.44E+05	9.92E+05	1.76E+06	1.90E+06
I-135	6.36E+00	5.74E+02	2.71E+05	6.59E+05	1.21E+06	1.34E+06
Total	2.97E+01	2.64E+03	1.18E+06	2.77E+06	5.04E+06	5.51E+06

Table 15.6-3 Instrument Line Break Accident Results

Meteorology* and Dose Results			
Meteorology (s/m^3)	**Distance (m)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
8.59E–03	max	3.0E–01	6.0E–03
1.37E-03	Chp 2	4.8E–02	9.4E–04
2.19E–04	800	7.6E–03	1.5E–04
1.11E–04	1600	3.9E–03	7.9E–05
5.61E–05	3200	2.0E–03	4.0E–05
3.73E–05	4800	1.3E–03	2.6E–05

* Meteorology calculated using Regulatory Guide 1.145 for a ground level 1.0 m/s, F stability release. "Max" = maximum meteorology to meet 10% of 10CFR100 limits.

Table 15.6-4 Sequence of Events for Steamline Break Outside Containment

Time (s)	Event
0	Guillotine break of one main steamline outside primary containment.
~0.5	High steamline flow signal initiates closure of main steamline isolation valve
<1.0	Reactor begins scram.
≤5.0	Main steamline isolation valves fully closed.
38	Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 7.58 MPa.
30	RCIC initiates on vessel low-water Level 2.
50	RCIC begins injection.
199	HPCF initiates on low water level.
236	One HPCF begins injection (the other HPCF is unavailable due to the single failure assumption).
1–2 hours	Normal reactor cooldown procedure established.

Table 15.6-5 Steamline Break Accident Parameters

I	**Data and assumptions used to estimate source terms.**		
	A.	Power Level	4005 MWt
	B.	Fuel damage	none
	C.	Reactor coolant activity	Subsection 15.6.4.5
	D.	Steam mass released	12,870 kg
	E.	Water mass released	21,953 kg
II	**Data and assumptions used to estimate activity released**		
	A.	MSIV closure time (break time until fully closed)	5.0 s
	B.	Maximum release time	2 h
III	**Dispersion and Dose Data**		
	A.	Meteorology	Table 15.6-7
	B.	Boundary and LPZ distances	Table 15.6-7
	C.	Method of Dose Calculation	Reference 15.6-2
	D.	Dose conversion Assumptions	Reference 15.6-2, RG 1.109, and ICRP 30
	E.	Activity Inventory/release	Table 15.6-6
	F.	Dose Evaluations	Table 15.6-7

Table 15.6-6 Main Steamline Break Accident Activity Released to Environment (megabecquerel)

Isotope	Case 1	Case 2
I-131	7.29E+04	1.46E+06
I-132	7.10E+05	1.42E+07
I-133	5.00E+05	9.99E+06
I-134	1.40E+06	2.79E+07
I-135	7.29E+05	1.46E+07
Total Halogens	3.41E+06	6.81E+07
KR-83M	4.07E+02	2.44E+03
KR-85M	7.18E+02	4.29E+03
KR-85	2.26E+00	1.36E+01
KR-87	2.44E+03	1.47E+04
KR-88	2.46E+03	1.48E+04
KR-89	9.88E+03	5.92E+04
KR-90	2.55E+03	1.55E+04
XE-131M	1.76E+00	1.06E+01
XE-133M	3.39E+01	2.04E+02
XE-133	9.47E+02	5.70E+03
XE-135M	2.89E+03	1.74E+04
XE-135	2.70E+03	1.62E+04
XE-137	1.23E+04	7.40E+04
XE-138	9.44E+03	5.66E+04
XE-139	4.33E+03	2.59E+04
Total Noble Gases	5.11E+04	3.07E+05

Table 15.6-7 Main Steamline Break Meteorology* Parameters and Radiological Effects

Meteorology (s/m^3)	**Distance (m)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
Case 1			
8.12E–03	max	3.0E–01	7.4E–03
1.37E–03	Chp 2	2.6E–02	6.2E–04
1.18E–03	300	2.2E–02	5.4E–04
2.19E–04	800	4.1E–03	1.0E–04
Case 2			
8.12E–03	max	3.0E+00	7.4E–02
1.37E–03	Chp 2	5.1E–01	1.3E–02
1.18E–03	300	4.4E–01	1.1E–02
2.19E–04	800	8.1E–02	2.0E–03

* Meteorology calculated using Regulatory Guide 1.145 for a ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 10% of 10CFR100 limits.

Table 15.6-8 Loss of Coolant Accident Parameters

I	**Data and assumptions used to estimate source terms.**	
	A. Power Level	4005 MWt
	B. Fraction of Core Inventory Released	
	Noble Gases	100%
	Iodines	50%
	C. Iodine Initial Plateout Fraction	50%
	D. Iodine Chemical Species	
	Elemental	91%
	Particulate	5%
	Organic	4%
	E. Suppression Pool Decontamination Factor–Section 15.6.5.5.1.1	
	Noble Gas	1
	Organic Iodine	1
	Elemental Iodine	2
	Particulate	2
	Pool Bypass Area	46.5 cm^2
II	**Data and Assumptions used to estimate activity released.**	
	A. Primary Containment Leakage	
	(1) Penetration and ESF Equipment	0.5%/day
	(2) MSIV Leakage (Total all lines)	66.1 L/min
	B. Reactor Building Leakage	150%/h
	(1) 0–20 min	150%/h
	(2) >20 min	50%/d
	(3) Mixing Efficiency	50%
	C. SGTS	
	Filter Efficiency (15.2 cm)	97%
	Drawdown Time	20 min
	D. MSIV Leakage—see Reference 15.6-4 for standard parameters	
	Main Steamline Length	47.9 m
	Drain Line length	71.6 m
	Main Steamline IR/OR	31.98/35.55 cm
	Drain Line IR/OR	3.33/4.45 cm
	Main Steamline Insulation	12.0 cm
	Drain Line Insulation	6.5 cm
	Plateout and Resuspension Factors	Ref. 15.6-5

Table 15.6-8 Loss of Coolant Accident Parameters (Continued)

	E. Condenser data	
	Free Air Volume	6230 m^3
	Fraction of Volume involved	20%
	Leakage Rate	11.6%/d
	Iodine Removal Factor	
	Elemental	0.993
	Particulate	0.993
	Organic	0
III	**Control Room Data**	
	A. Control Room Volumes	
	Total Free Air Volume	5,509 m^3
	Gamma Room Volume (room size)	1,400 m^3
	B. Recirculation Rates	
	Filtered Intake	0.944 m^3/s
	Unfiltered Intake	0.0
	Filtered Recirculation	0.47 m^3/s
	Filter Efficiency (100 mm)	99%
IV	**Dispersion and Dose Data**	
	A. Meteorology	Sec 15.6.5.5.3 Tbls 15.6-13, 15.6-14
	B. Dose Calculation Method (semi-infinite)	Ref 15.6-2 & 15.6-3, RG 1.109
	C. Dose Conversion Assumptions	Ref 15.6-2, 15.6-3
	D. Activity/Releases	Tbls 15.6-9, 15.6-10, 15.6-11, 15.6-12 Appendix 15F
	E. Dose Evaluation	Tbls 15.6-13,15.6-14

Table 15.6-9 Iodine Activities

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
A. Primary Containment Inventory (megabecquerel)										
I-131	5.2E+11	5.2E+11	5.2E+11	5.2E+11	4.8E+11	4.8E+11	4.8E+11	4.4E+11	3.4E+11	2.7E+10
I-132	7.4E+11	7.0E+11	5.6E+11	4.1E+11	2.2E+12	6.7E+10	1.9E+10	5.2E+8	1.6E+11	0
I-133	1.1E+12	1.0E+12	1.0E+12	1.0E+12	9.3E+11	8.1E+11	7.0E+11	4.8E+11	4.1E+10	2.8E+1
I-134	1.1E+12	1.0E+12	5.2E+11	2.4E+11	4.8E+10	2.1E+9	8.9E+7	6.7E+3	0	0
I-135	1.0E+12	1.0E+12	8.9E+11	8.1E+11	6.7E+11	4.4E+11	2.8E+11	7.8E+10	4.1E+7	0
Total	4.5E+12	4.3E+12	3.5E+12	3.0E+12	4.3E+12	1.8E+12	1.5E+12	1.0E+12	5.4E+11	2.7E+10
B. Reactor Building Inventory (megabecquerel)										
I-131	1.7E+6	1.5E+7	9.3E+7	1.9E+8	3.7E+8	7.0E+8	9.6E+8	1.4E+9	1.7E+9	1.3E+8
I-132	2.5E+6	2.1E+7	1.0E+8	1.6E+8	1.7E+8	9.3E+7	3.7E+7	1.6E+6	7.8E–4	0
I-133	3.6E+6	3.1E+7	1.9E+8	3.7E+8	7.0E+8	1.1E+9	1.4E+9	1.5E+9	2.0E+8	1.4E–1
I-134	4.1E+6	3.0E+7	1.0E+8	9.3E+7	3.7E+7	2.9E+6	1.7E+5	2.1E+1	0	0
I-135	3.4E+6	2.9E+7	1.7E+8	3.1E+8	4.8E+8	5.9E+8	5.6E+8	2.5E+8	1.9E+5	0
Total	1.5E+7	1.3E+8	6.5E+8	1.1E+9	1.8E+9	2.5E+9	2.9E+9	3.2E+9	1.9E+9	1.3E+8
C.1 MSIV Pathway—Condenser Inventory (megabecquerel)—Elemental Iodine										
I-131	0	0	7.8E+6	4.8E+7	2.0E+8	6.3E+8	1.1E+9	2.4E+9	4.1E+9	3.1E+7
I-132	0	0	8.5E+6	4.1E+7	8.9E+7	8.5E+7	4.4E+7	2.6E+6	1.9E–3	0
I-133	0	0	1.6E+7	9.6E+7	3.7E+8	1.0E+0	1.6E+9	2.4E+9	4.8E+8	3.3E–2
I-134	0	0	8.1E+6	2.4E+7	2.0E+7	2.7E+6	2.0E+5	3.5E+1	0	0
I-135	0	0	1.4E+7	8.1E+7	2.7E+8	5.6E+8	6.3E+8	4.1E+8	4.8E+5	0
Total	0	0	5.4E+7	2.9E+8	9.5E+8	2.3E+9	3.4E+9	5.2E+9	4.6E+9	3.1E+7
C.2 MSIV Pathway—Condenser Inventory (megabecquerel)—Organic Iodine (Primary Containment)										
I-131	0	0	6.7E+5	4.1E+6	1.7E+7	5.6E+7	9.3E+7	2.1E+8	5.9E+8	1.3E+8
I-132	0	0	7.0E+5	3.4E+6	7.8E+6	7.0E+6	3.7E+6	2.3E+5	2.8E–4	0
I-133	0	0	1.4E+6	8.1E+6	3.2E+7	8.9E+7	1.4E+8	2.1E+10	7.0E+7	1.4E–1
I-134	0	0	7.0E+5	2.0E+6	1.7E+6	2.3E+5	1.7E+4	3.0E+0	0	0
I-135	0	0	1.2E+6	6.7E+6	2.3E+7	4.8E+7	5.6E+7	3.6E+8	6.7E+4	0
Total	0	0	4.6E+6	2.4E+7	8.1E+7	2.0E+8	2.9E+8	2.2E+10	6.6E+8	1.3E+8

Table 15.6-9 Iodine Activities (Continued)

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
C.3 MSIV Pathway—Condenser Inventory in Curies—Resuspended Organic										
I-131	0	0	2.8E+3	5.6E+3	3.4E+4	8.9E+4	2.7E+5	9.3E+5	4.8E+7	9.6E+7
I-132	0	0	2.2E+3	3.7E+3	8.5E+3	1.2E+4	5.9E+3	1.7E+3	0	0
I-133	0	0	5.6E+3	1.1E+4	5.9E+4	1.5E+5	3.6E+5	9.3E+5	5.9E+6	5.2E–1
I-134	0	0	1.4E+3	2.1E+3	1.3E+3	1.0E+3	5.6E+1	1.2E+0	0	0
I-135	0	0	4.4E+3	8.5E+3	3.6E+4	7.4E+4	1.1E+5	1.6E+5	9.6E+3	0
Total	0	0	1.6E+4	3.1E+4	1.4E+5	3.2E+5	7.5E+5	2.0E+6	5.4E+7	9.6E+7
C.4 Condenser Inventory (megabecquerel)—Combined										
I-131	0	0	8.5E+6	5.6E+7	2.2E+8	7.0E+8	1.2E+9	2.6E+9	4.8E+9	2.6E+8
I-132	0	0	9.3E+6	4.4E+7	9.6E+7	9.3E+7	4.8E+7	2.8E+6	2.2E–3	0
I-133	0	0	1.7E+7	1.1E+8	4.1E+8	1.1E+9	1.7E+9	2.7E+9	5.6E+8	7.0E–1
I-134	0	0	8.9E+6	2.6E+7	2.2E+7	2.9E+6	2.2E+5	4.1E+1	0	0
I-135	0	0	1.5E+7	8.5E+7	2.9E+8	5.9E+8	7.0E+8	4.4E+8	5.6E+5	0
Total	0	0	5.9E+7	3.2E+8	1.0E+9	2.5E+9	3.7E+9	5.7E+9	5.4E+9	2.6E+8
D.1 Control Room Inventory (megabecquerel)										
I-131	8.4E-1	7.3E+1	1.4E+2	5.8E+1	2.0E+1	2.6E+1	2.2E+1	3.6E+1	3.2E+1	1.9E+0
I-132	1.2E+0	1.0E+2	1.5E+2	4.6E+1	8.7E+0	3.4E+0	8.9E–1	4.0E–2	0	0
I-133	1.8E+0	1.5E+2	2.8E+2	1.1E+2	3.7E+1	4.2E+1	3.2E+1	3.7E+1	3.8E+0	3.8E-9
I-134	1.9E+0	1.5E+2	1.4E+2	2.8E+1	2.0E+0	1.1E–1	4.0E–3	5.4E–7	0	0
I-135	1.7E+0	1.4E+2	2.4E+2	9.3E+1	2.6E+1	2.2E+1	1.3E+1	6.3E+0	3.6E–3	0
Total	7.4E+0	6.2E+2	9.5E+2	3.4E+2	9.3E+1	9.3E+1	6.8E+1	8.0E+1	3.5E+1	1.9E+0
D.2 Control Room Integrated Activity (megabecquerel–seconds)										
I-131	1.7E+1	1.5E+4	5.4E+5	3.3E+5	2.3E+5	3.0E+5	3.1E+5	1.3E+6	8.0E+6	1.5E+7
I-132	2.5E+1	2.1E+4	6.7E+5	3.1E+5	1.5E+5	7.4E+4	2.4E+4	1.2E+4	0	0
I-133	3.5E+1	3.2E+4	1.1E+6	6.5E+5	4.4E+5	5.2E+5	4.8E+5	1.6E+6	3.2E+6	2.0E+5
I-134	3.9E+1	3.2E+4	8.1E+5	2.5E+5	6.4E+4	8.4E+3	4.1E+2	2.0E+1	0	0
I-135	3.3E+1	3.0E+4	1.0E+6	5.6E+5	3.4E+5	3.2E+5	2.2E+5	4.1E+5	1.7E+5	0
Total	1.5E+2	1.3E+5	4.1E+6	2.1E+6	1.2E+6	1.2E+6	1.0E+6	3.3E+6	1.1E+7	1.5E+7

Table 15.6-10 Iodine Activity Release to the Environment

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
A. Release from Reactor Building to Environment (megabecquerel)										
I-131	2.9E+4	2.6E+6	9.6E+6	9.6E+6	1.0E+7	1.3E+7	1.7E+7	3.6E+7	1.9E+8	6.7E+8
I-132	4.1E+4	3.7E+6	1.3E+7	1.3E+7	1.4E+7	1.4E+7	1.5E+7	1.5E+7	1.5E+7	1.5E+7
I-133	5.9E+4	5.6E+6	2.0E+7	2.0E+7	2.1E+7	2.6E+7	3.3E+7	5.6E+7	1.2E+8	1.3E+8
I-134	6.7E+4	5.6E+6	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7
I-135	5.6E+4	5.2E+6	1.9E+7	1.9E+7	2.0E+7	2.3E+7	2.6E+7	3.1E+7	3.5E+7	3.5E+7
Total	2.5E+5	2.3E+7	8.0E+7	8.1E+7	8.4E+7	9.5E+7	1.1E+8	1.6E+8	7.0E+8	8.6E+8
B.1 MSIV Pathway Release to Environment—Elemental (megabecquerel)										
I-131	0	0	5.6E+1	9.3E+2	9.3E+3	6.3E+4	1.8E+5	8.9E+5	1.0E+7	3.0E+7
I-132	0	0	6.3E+1	8.5E+2	5.6E+3	1.8E+4	2.7E+4	3.3E+4	3.4E+4	3.4E+4
I-133	0	0	1.1E+2	1.9E+3	1.7E+4	1.1E+5	3.0E+5	1.1E+6	4.8E+6	5.6E+6
I-134	0	0	6.3E+1	6.3E+2	2.3E+3	3.6E+3	3.7E+3	3.7E+3	1.3E+2	3.7E+3
I-135	0	0	1.0E+2	1.6E+3	1.3E+4	7.0E+4	1.6E+5	3.7E+5	5.6E+5	5.6E+5
Total	0	0	3.9E+2	5.9E+3	4.8E+4	2.7E+5	6.6E+5	2.4E+6	1.6E+7	3.6E+7
B.2 MSIV Pathway Release to Environment—Organic (megabecquerel)										
I-131	0	0	6.7E+2	1.1E+4	1.1E+5	7.8E+5	2.2E+6	1.1E+7	1.6E+8	1.4E+9
I-132	0	0	7.4E+2	1.0E+4	6.7E+4	2.2E+5	3.3E+5	4.1E+5	4.1E+5	4.1E+5
I-133	0	0	1.3E+3	2.3E+4	2.1E+5	1.4E+6	3.6E+6	1.4E+7	7.0E+7	8.1E+7
I-134	0	0	7.8E+2	7.8E+3	2.8E+4	4.4E+4	4.4E+4	4.4E+4	4.4E+4	4.4E+4
I-135	0	0	1.2E+3	1.9E+4	1.6E+5	8.5E+5	1.9E+6	4.4E+6	7.0E+6	7.0E+6
Total	0	0	4.7E+3	7.1E+4	5.8E+5	3.3E+6	8.0E+6	3.0E+7	2.4E+8	1.5E+9
B.3 MSIV Pathway Release to Environment—Resuspended Organic (megabecquerel)										
I-131	0	0	6.7E+0	2.7E+1	2.2E+2	1.4E+3	4.8E+3	3.7E+4	6.3E+6	5.2E+8
I-132	0	0	5.6E+0	2.1E+1	8.1E+1	2.9E+2	4.4E+2	6.7E+2	7.0E+2	7.0E+2
I-133	0	0	1.3E+1	5.2E+1	4.1E+2	2.4E+3	7.4E+3	4.4E+4	1.4E+6	3.3E+6
I-134	0	0	4.1E+0	1.3E+1	2.8E+1	4.8E+1	5.6E+1	5.6E+1	5.6E+1	5.6E+1
I-135	0	0	1.1E+1	4.1E+1	2.6E+2	1.4E+3	3.2E+3	1.1E+4	3.4E+4	3.5E+4
Total	0	0	4.1E+1	1.5E+2	1.0E+3	5.5E+3	1.6E+4	9.4E+4	7.7E+6	5.2E+8

Table 15.6-10 Iodine Activity Release to the Environment (Continued)

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
B.4 Release from Condenser to Environment—Sum of B.1+B.2+B.3 (megabecquerel)										
I-131	0	0	7.0E+2	1.2E+4	1.2E+5	8.5E+5	2.4E+6	1.2E+7	1.8E+8	2.0E+9
I-132	0	0	8.1E+2	1.1E+4	7.0E+4	2.4E+5	3.5E+5	4.4E+5	4.4E+5	4.4E+5
I-133	0	0	1.5E+3	2.4E+4	2.3E+5	1.5E+6	3.7E+6	1.5E+7	7.4E+7	8.9E+7
I-134	0	0	8.5E+2	8.5E+3	3.0E+4	4.8E+4	4.8E+4	4.8E+4	4.8E+4	4.8E+4
I-135	0	0	1.3E+3	2.1E+4	1.7E+5	9.3E+5	2.0E+6	5.2E+6	7.4E+6	7.4E+6
Total	0	0	5.1E+3	7.7E+4	6.2E+5	3.5E+6	8.5E+6	3.3E+7	2.6E+8	2.1E+9

Table 15.6-11 Noble Gas Activities

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
A. Primary Containment Inventory (megabecquerel)										
Kr-83m	4.4E+11	4.4E+11	3.2E+11	2.2E+11	1.0E+11	2.3E+10	5.2E+9	5.9E+7	1.3E–4	0
Kr-85	4.4E+10	4.4E+10	4.4E+10	4.4E+10	4.4E+10	4.4E+10	4.4E+10	4.4E+10	4.1E+10	3.1E+10
Kr-85m	1.0E+12	9.6E+11	8.5E+11	7.4E+11	5.2E+11	2.8E+11	1.5E+11	2.2E+10	2.4E+5	0
Kr-87	1.9E+12	1.7E+12	1.1E+12	6.3E+11	2.1E+11	2.4E+10	2.7E+9	3.7E+6	0	0
Kr-88	2.7E+12	2.6E+12	2.1E+12	1.7E+12	1.0E+12	3.7E+11	1.4E+11	7.0E+9	1.2E+2	0
Kr-89	2.7E+12	3.7E+11	7.0E+6	1.4E+1	0	0	0	0	0	0
Xe-131m	2.3E+10	2.3E+10	2.3E+10	2.3E+10	2.3E+10	2.3E+10	2.3E+10	2.2E+10	1.8E+10	2.9E+9
Xe-133	8.1E+12	8.1E+12	8.1E+12	8.1E+12	8.1E+12	7.8E+12	7.8E+12	7.0E+12	4.4E+12	1.1E+11
Xe-133m	3.4E+11	3.4E+11	3.4E+11	3.3E+11	3.3E+11	3.1E+11	2.9E+11	2.5E+11	9.6E+10	2.4E+7
Xe-135	1.1E+12	1.0E+12	1.0E+12	9.3E+11	7.8E+11	5.9E+11	4.1E+11	1.7E+11	7.0E+8	0
Xe-135m	1.5E+12	1.0E+12	1.1E+11	7.8E+9	3.7E+7	9.6E+2	2.4E–2	0	0	0
Xe-137	5.9E+12	1.2E+12	1.3E+8	2.4E+3	8.1E–7	0	0	0	0	0
Xe-138	6.7E+12	4.1E+12	3.6E+11	2.0E+10	5.6E+7	4.4E+2	3.6E–3	0	0	0
Total	3.2E+13	2.2E+13	1.4E+13	1.3E+13	1.1E+13	9.4E+12	8.8E+12	7.5E+12	4.6E+12	1.4E+11
B. Reactor Building Inventory (megabecquerel)										
Kr-83m	1.6E+6	1.3E+7	5.9E+7	8.5E+7	7.8E+7	3.3E+7	1.0E+7	1.9E+5	0	0
Kr-85	1.5E+5	1.3E+6	8.1E+6	1.7E+7	3.4E+7	6.3E+7	8.9E+7	1.4E+8	2.1E+8	1.6E+8
Kr-85m	3.4E+6	2.9E+7	1.6E+8	2.8E+8	4.1E+8	4.1E+8	2.9E+8	7.0E+7	1.2E+3	0
Kr-87	6.7E+6	5.2E+7	2.1E+8	2.4E+8	1.6E+8	3.4E+7	5.2E+6	1.2E+4	0	0
Kr-88	9.3E+6	7.8E+7	4.1E+8	6.3E+8	7.4E+8	5.2E+8	2.7E+8	2.2E+7	5.9E–1	0
Kr-89	9.3E+6	1.1E+7	1.3E+3	5.6E–3	0	0	0	0	0	0
Xe-131m	8.1E+4	7.0E+5	4.4E+6	8.9E+6	1.7E+7	3.2E+7	4.4E+7	7.0E+7	8.9E+7	1.4E+7
Xe-133	2.8E+7	2.4E+8	1.5E+9	3.1E+9	5.9E+9	1.1E+10	1.5E+10	2.3E+10	2.3E+10	5.6E+8
Xe-133m	1.2E+6	1.0E+7	6.3E+7	1.3E+8	2.4E+8	4.4E+8	5.6E+8	7.8E+8	4.8E+8	1.2E+5
Xe-135	3.6E+6	3.1E+7	1.8E+8	3.5E+8	5.9E+8	8.1E+8	8.5E+8	5.6E+8	3.5E+6	0
Xe-135m	5.2E+6	2.9E+7	2.0E+7	3.0E+6	2.9E+4	1.3E+0	4.8E–5	0	0	0
Xe-137	2.0E+7	3.4E+7	2.5E+4	9.3E–1	0	0	0	0	0	0
Xe-138	2.2E+7	1.2E+8	6.7E+7	7.4E+6	4.1E+4	6.3E–1	0	0	0	0
Total	1.1E+8	6.5E+8	2.7E+9	4.9E+9	8.2E+9	1.3E+10	1.7E+10	2.4E+10	2.4E+10	7.2E+8

Table 15.6-11 Noble Gas Activities (Continued)

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
C. Condenser Inventory (megabecquerel)										
Kr-83m	0	0	5.6E+6	2.4E+7	4.8E+7	3.3E+7	1.3E+7	3.5E+5	0	0
Kr-85	0	0	7.8E+5	4.8E+6	2.0E+7	6.3E+7	1.1E+8	2.6E+8	9.6E+8	2.0E+9
Kr-85m	0	0	1.4E+7	7.8E+7	2.4E+8	4.1E+8	3.7E+8	1.3E+8	5.6E+3	0
Kr-87	0	0	1.9E+7	7.0E+7	9.6E+7	3.4E+7	6.7E+6	2.2E+4	0	0
Kr-88	0	0	3.6E+7	1.8E+8	4.4E+8	5.2E+8	3.5E+8	4.1E+7	2.7E+0	0
Kr-89	0	0	1.2E+2	1.5E–3	0	0	0	0	0	0
Xe-131m	0	0	4.1E+5	2.6E+6	1.0E+7	3.3E+7	5.6E+7	1.3E+8	4.1E+8	1.9E+8
Xe-133	0	0	1.4E+8	8.9E+8	3.6E+9	1.1E+10	1.9E+10	4.1E+10	1.0E+11	7.4E+9
Xe-133m	0	0	5.9E+6	3.6E+7	1.4E+8	4.4E+8	7.4E+8	1.4E+9	2.1E+9	1.6E+6
Xe-135	0	0	1.7E+7	1.0E+8	3.5E+8	8.1E+8	1.1E+9	1.0E+9	1.6E+7	1.0E–13
Xe-135m	0	0	1.9E+6	8.5E+5	1.7E+4	1.4E+0	6.3E–5	0	0	0
Xe-137	0	0	2.3E+3	2.7E–1	0	0	0	0	0	0
Xe-138	0	0	6.3E+6	2.1E+6	2.5E+4	6.3E–1	0	0	0	0
Total	0	0.0E+0	2.5E+8	1.4E+9	4.9E+9	1.3E+10	2.2E+10	4.4E+10	1.1E+11	9.6E+9
D.1 Control Room Inventory (megabecquerel)										
Kr-83m	7.7E+1	6.4E+3	1.3E+4	1.1E+4	1.0E+4	5.4E+3	1.1E+3	2.2E+1	0	0
Kr-85	7.5E+0	6.6E+2	1.9E+3	2.2E+3	4.5E+3	1.0E+4	9.6E+3	1.6E+4	1.7E+4	6.4E+3
Kr-85m	1.7E+2	1.4E+4	3.6E+4	3.6E+4	5.3E+4	6.5E+4	3.2E+4	8.1E+3	9.8E-2	0
Kr-87	3.2E+2	2.6E+4	4.7E+4	3.2E+4	2.2E+4	5.6E+3	5.8E+2	1.4E+0	0	0
Kr-88	4.5E+2	3.8E+4	8.9E+4	8.2E+4	1.0E+5	8.6E+4	3.0E+4	2.6E+3	4.7E-5	0
Kr-89	4.5E+2	5.6E+3	2.9E-1	7.0E-7	0	0	0	0	0	0
Xe-131m	3.9E+0	3.4E+2	9.9E+2	1.2E+3	2.3E+3	5.3E+3	4.9E+3	8.0E+3	7.1E+3	5.9E+2
Xe-133	1.4E+3	1.2E+5	3.4E+5	4.0E+5	8.0E+5	1.8E+6	1.6E+6	2.6E+6	1.8E+6	2.3E+4
Xe-133m	5.7E+1	5.0E+3	1.4E+4	1.7E+4	3.3E+4	7.1E+4	6.3E+4	9.1E+4	3.8E+4	4.9E+0
Xe-135	1.8E+2	1.5E+4	4.1E+4	4.5E+4	7.8E+4	1.3E+5	9.2E+4	6.2E+4	2.8E+2	0
Xe-135m	2.5E+2	1.5E+4	4.6E+3	3.8E+2	3.9E+0	2.2E-4	5.1E-9	0	0	0
Xe-137	1.0E+3	1.7E+4	5.6E+0	1.2E-4	0	0	0	0	0	0
Xe-138	1.1E+3	6.2E+4	1.5E+4	9.7E+2	5.6E+0	1.0E-4	7.8E-10	0	0	0
Total	5.4E+3	3.2E+5	6.1E+5	6.3E+5	1.1E+6	2.2E+6	1.9E+6	2.8E+6	1.9E+6	3.0E+4

Table 15.6-11 Noble Gas Activities (Continued)

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
D.2 Control Room Integrated Inventory (megabecquerel-seconds)										
Kr-83m	1.6E+3	1.3E+6	4.8E+7	4.2E+7	7.8E+7	1.2E+8	3.7E+7	1.3E+7	0	0
Kr-85	1.5E+2	1.4E+5	5.7E+6	7.1E+6	2.3E+7	1.1E+8	1.3E+8	5.6E+8	3.9E+9	1.6E+10
Kr-85m	3.3E+3	3.0E+6	1.2E+8	1.3E+8	3.2E+8	9.0E+8	6.3E+8	7.8E+8	1.5E+8	0
Kr-87	6.4E+3	5.4E+6	1.8E+8	1.4E+8	1.9E+8	1.8E+8	3.0E+7	4.2E+6	0	0
Kr-88	9.1E+3	8.0E+6	3.0E+8	3.0E+8	6.6E+8	1.4E+9	7.1E+8	4.9E+8	3.0E+7	0
Kr-89	9.6E+3	2.2E+6	2.9E+6	8.0E+1	0	0	0	0	0	0
Xe-131m	7.9E+1	7.1E+4	3.0E+6	3.7E+6	1.2E+7	5.5E+7	6.8E+7	2.8E+8	1.8E+9	3.6E+9
Xe-133	2.7E+4	2.5E+7	1.0E+9	1.3E+9	4.2E+9	1.9E+10	2.3E+10	9.3E+10	5.2E+11	5.2E+11
Xe-133m	1.1E+3	1.0E+6	4.3E+7	5.3E+7	1.7E+8	7.5E+8	9.0E+8	3.4E+9	1.4E+10	4.9E+9
Xe-135	3.5E+3	3.2E+6	1.3E+8	1.5E+8	4.4E+8	1.6E+9	1.5E+9	3.4E+9	2.4E+9	0
Xe-135m	5.0E+3	3.4E+6	4.8E+7	5.9E+6	5.9E+5	5.9E+3	2.9E-1	0	0	0
Xe-137	2.1E+4	6.0E+6	1.2E+7	1.8E+3	0	0	0	0	0	0
Xe-138	2.2E+4	1.4E+7	1.9E+8	1.8E+7	1.3E+6	7.6E+3	1.2E-1	0	0	0
Total	1.1E+5	7.3E+7	2.1E+9	2.1E+9	6.1E+9	2.4E+10	2.7E+10	1.0E+11	5.4E+11	5.5E+11

Table 15.6-12 Noble Gas Activity Release to Environment

Isotope	1 min	10 min	1 h	2 h	4 h	8 h	12 h	1 day	4 days	30 days
A. Reactor Building Release to Environment (megabecquerel)										
Kr-83m	2.7E+4	2.3E+6	9.3E+6	1.2E+7	1.9E+7	2.8E+7	3.2E+7	3.3E+7	3.3E+7	3.3E+7
Kr-85	2.6E+3	2.3E+5	1.0E+6	1.5E+6	3.6E+6	1.2E+7	2.4E+7	8.1E+7	6.7E+8	5.6E+9
Kr-85m	5.6E+4	5.2E+6	2.1E+7	3.1E+7	5.9E+7	1.3E+8	1.9E+8	2.7E+8	2.9E+8	2.9E+8
Kr-87	1.1E+5	9.3E+6	3.6E+7	4.4E+7	6.3E+7	7.8E+7	8.1E+7	8.1E+7	8.1E+7	8.1E+7
Kr-88	1.6E+5	1.4E+7	5.6E+7	7.8E+7	1.4E+8	2.5E+8	3.1E+8	3.6E+8	3.7E+8	3.7E+8
Kr-89	1.7E+5	4.8E+6	6.7E+6	6.7E+6	6.7E+6	6.7E+6	6.7E+6	6.7E+6	6.7E+6	6.7E+6
Xe-131m	1.3E+3	1.2E+5	5.2E+5	7.8E+5	1.9E+6	5.9E+6	1.3E+7	4.1E+7	3.0E+8	1.4E+9
Xe-133	4.8E+5	4.1E+7	1.8E+8	2.8E+8	6.7E+8	2.1E+9	4.4E+9	1.4E+10	8.9E+10	2.5E+11
Xe-133m	2.0E+4	1.8E+6	7.4E+6	1.1E+7	2.7E+7	8.5E+7	1.7E+8	5.2E+8	2.6E+9	4.1E+9
Xe-135	5.9E+4	5.6E+6	2.3E+7	3.4E+7	7.4E+7	1.9E+8	3.3E+8	6.7E+8	1.0E+9	1.0E+9
Xe-135m	8.5E+4	5.9E+6	1.7E+7	1.8E+7	1.8E+7	1.8E+7	1.8E+7	1.8E+7	1.8E+7	1.8E+7
Xe-137	3.7E+5	1.3E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7	1.9E+7
Xe-138	3.7E+5	2.6E+7	7.4E+7	7.4E+7	7.4E+7	7.4E+7	7.4E+7	7.4E+7	7.4E+7	7.4E+7
Total	1.9E+6	1.3E+8	4.3E+8	5.7E+8	1.2E+9	3.0E+9	5.7E+9	1.6E+10	9.4E+10	2.6E+11
B. Condenser Release to Environment (megabecquerel)										
Kr-83m	0	0	5.9E+3	7.8E+4	4.4E+5	1.3E+6	1.7E+6	1.9E+6	1.9E+6	1.9E+6
Kr-85	0	0	7.4E+2	1.3E+4	1.3E+5	9.3E+5	2.6E+6	1.3E+7	2.3E+8	5.9E+9
Kr-85m	0	0	1.5E+4	2.3E+5	1.8E+6	8.5E+6	1.6E+7	3.0E+7	3.6E+7	3.6E+7
Kr-87	0	0	2.0E+4	2.4E+5	1.1E+6	2.4E+6	2.7E+6	2.8E+6	2.8E+6	2.8E+6
Kr-88	0	0	3.7E+4	5.6E+5	3.7E+6	1.4E+7	2.3E+7	3.1E+7	3.2E+7	3.2E+7
Kr-89	0	0	4.1E+0	4.1E+0	4.1E+0	4.1E+0	4.1E+0	4.1E+0	4.1E+0	4.1E+0
Xe-131m	0	0	4.1E+2	6.7E+3	6.7E+4	4.8E+5	1.3E+6	6.7E+6	1.0E+8	1.3E+9
Xe-133	0	0	1.4E+5	2.4E+6	2.3E+7	1.6E+8	4.4E+8	2.2E+9	3.0E+10	1.8E+11
Xe-133m	0	0	5.6E+3	1.0E+5	9.3E+5	6.7E+6	1.8E+7	8.1E+7	8.1E+8	2.0E+9
Xe-135	0	0	1.7E+4	2.7E+5	2.4E+6	1.4E+7	3.3E+7	9.6E+7	2.0E+8	2.0E+8
Xe-135m	0	0	2.9E+3	1.0E+4	1.3E+4	1.3E+4	1.3E+4	1.3E+4	1.3E+4	1.3E+4
Xe-137	0	0	3.4E+1	3.5E+1	3.5E+1	3.5E+1	3.5E+1	3.5E+1	3.5E+1	3.5E+1
Xe-138	0	0	1.0E+4	3.2E+4	3.7E+4	3.7E+4	3.7E+4	3.7E+4	3.7E+4	3.7E+4
Total	0	0	2.5E+5	3.9E+6	3.4E+7	2.1E+8	5.4E+8	2.5E+9	3.2E+10	1.9E+11

Table 15.6-13 Loss of Coolant Accident Meteorology and Offsite Dose Results

Site Boundary Dose Results

	Meteorology* (s/m^3)	Dist (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
	2.18E–03	max	3	6.4E–02
	1.37E–03	Chp 2	1.9E+00	4.1E–02
	1.18E–03	300	1.6E+00	3.5E–02
	2.19E–04	800	3.0E–01	6.5E–02

* "Max" = maximum meteorology to meet 10CFR100 limitation.

Low Population Zone Boundary Dose Results

Time (h)	Meteorology (s/m^3)	Dist (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
8	3.73E–05	4828	7.3E–02	2.5E–03
24	1.21E–05		9.9E–02	3.5E–03
96	4.27E–06		2.0E–01	4.9E–03
720	9.09E–07		3.4E–01	5.9E–03
8	1.56E-04	Chp 2	3.1E–01	1.0E–02
24	9.61E-05		5.1E–01	1.8E–02
96	3.36E-05		1.3E+00	2.9E–02
720	7.42E-06		2.4E+00	3.8E–02

Table 15.6-14 Loss of Coolant Accident Meteorology and Control Room Dose Results

Time (h)	Meteorology* (s/m^3)	Thyroid† (Sv)	Whole Body† (Sv)	Beta† (Sv)
0–8 h	3.10E–03	3.60E-02	3.50E-03	4.20E-02
8–24 h	1.83E–03	7.20E-02	9.00E-03	1.33E-01
1–4 days	1.16E–03	1.66E-01	1.96E-02	3.20E-01
4–30 days	5.12E–04	2.76E-01	2.67E-02	4.47E-01

* See Subsection 15.6.5.5.3.2 for description of meteorology. Values are for dispersion from Reactor Building. Dispersion values for releases from Turbine Building are a factor of six less than Reactor Building dispersion values.

† These values are cumulative from the beginning to the end of period in the first column.

Table 15.6-15 Sequence of Events for Cleanup Line Break Outside Containment

Time (s)	Event
0	Clean up water line break occurs
0+	Check valves on clean up water line to feedwater line isolate Differential pressure instrumentation initiates delay sequence
45	Differential pressure instrumentation actuates isolation valves
75	Isolation valves complete closure and isolation
1-2 hour	Normal reactor shutdown and cooldown procedure

Table 15.6-16 Cleanup Line Break Accident Parameters

I	Data and assumptions used to estimate source terms.	
	A. Power level	4005 MWt
	B. Number of bundles in core	872
	C. Mass of fluid released	2.8×10^7 g
	D. Mass of fluid flashed to steam	9.9×10^6 g
	E. Duration of accident	< 2 h
II	Data and assumptions used to estimate activity released	
	A. Iodine water concentration	15.6.4.5.1.1, case 1
	B. Iodine spiking	Table 15.6-1, IIB.
	C. Iodine plateout fraction	50%
	D. Building release rate	direct to environment
	E. SGTS filter efficiency	none assumed
III	Dispersion and Dose Data	
	A. Meteorology	Table 15.6-18
	B. Boundary distance	Table 15.6-18
	C. Method of dose calculation	Reference 15.6-2
	D. Dose conversion assumptions	Reference 15.6-2, RG 1.109, and ICRP 30
	E. Activity releases	Table 15.6-17
	F. Dose evaluations	Table 15.6-18

Table 15.6-17 Clean Up Water Line Break Isotopic Releases (megabecquerel)

Isotope	Release
I-131	8.1E+4
I-132	1.9E+5
I-133	2.3E+5
I-134	3.2E+5
I-135	2.5E+5

Table 15.6-18 Clean Up Water Line Break Meteorology* and Dose Results

Meteorology (s/m^3)	Distance (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
2.29E-02	max	3.0E–1	2.8E–3
1.37E-03	Chp 2	1.7E–4	1.7E–4
1.18E-03	300	1.5E–4	1.5E–4
2.19E-04	800	2.7E–5	2.7E–5

* Meteorology calculated using Regulatory Guide 1.145 for a ground level 1.0 m/s, F stability. “Max” = maximum meteorology to meet 10% of 10CFR100 limits.

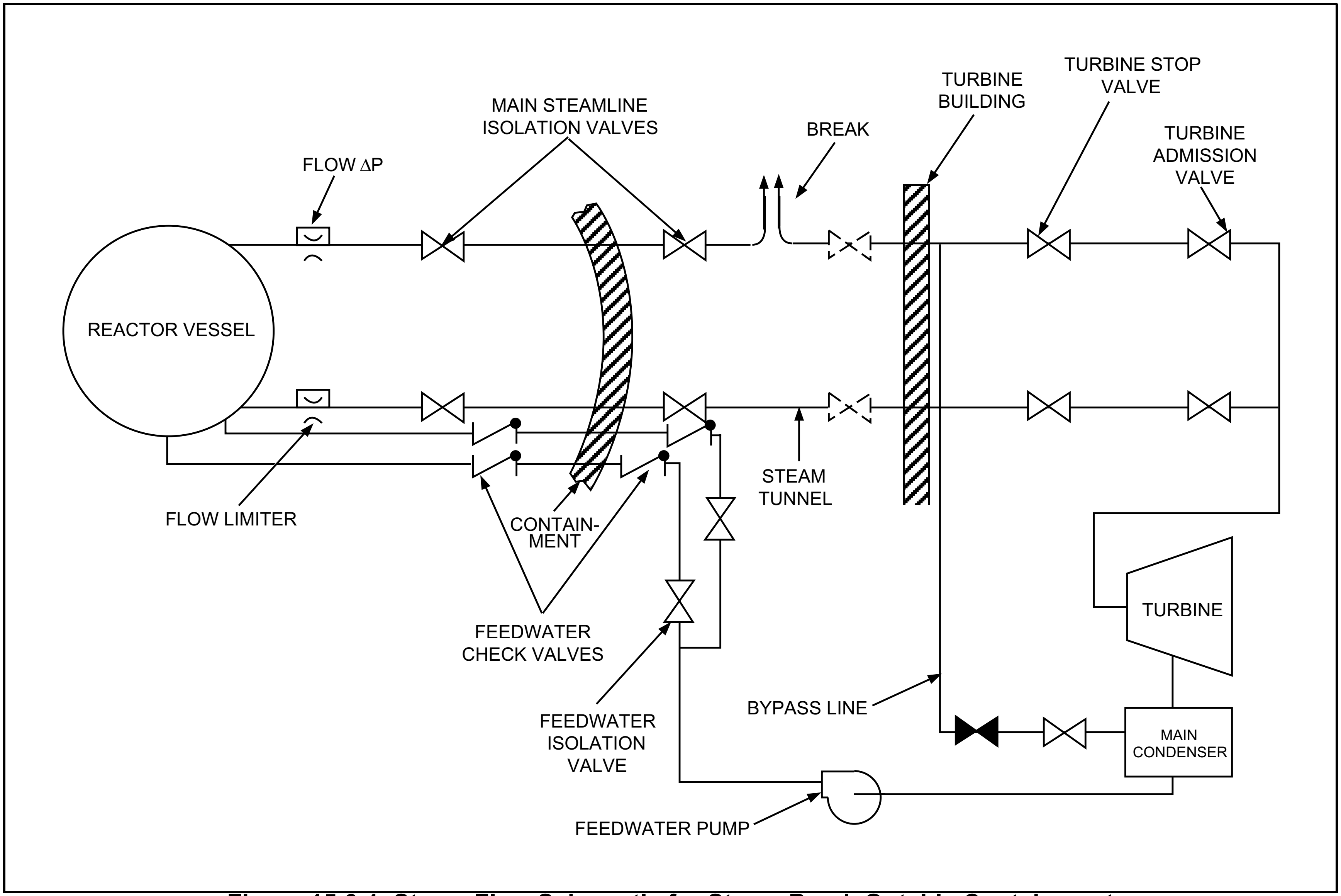


Figure 15.6-1 Steam Flow Schematic for Steam Break Outside Containment

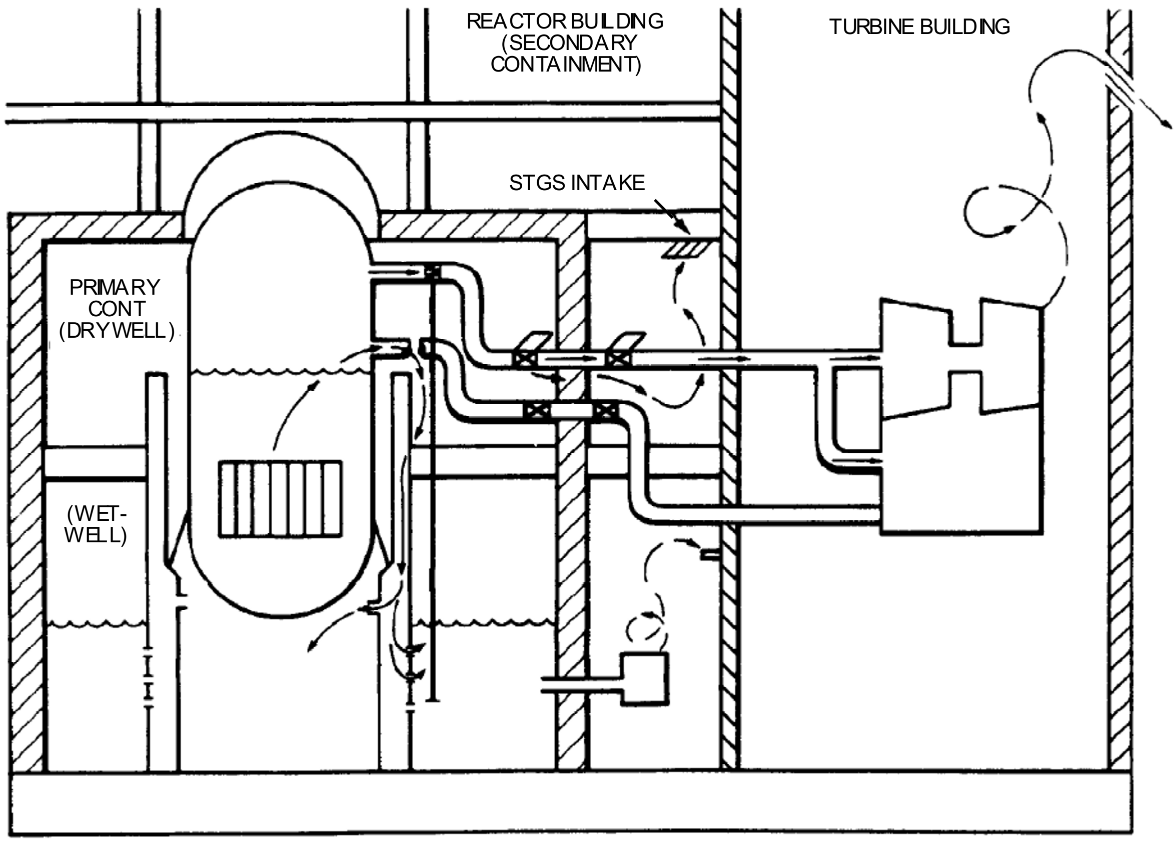


Figure 15.6-2 LOCA Radiological Analysis

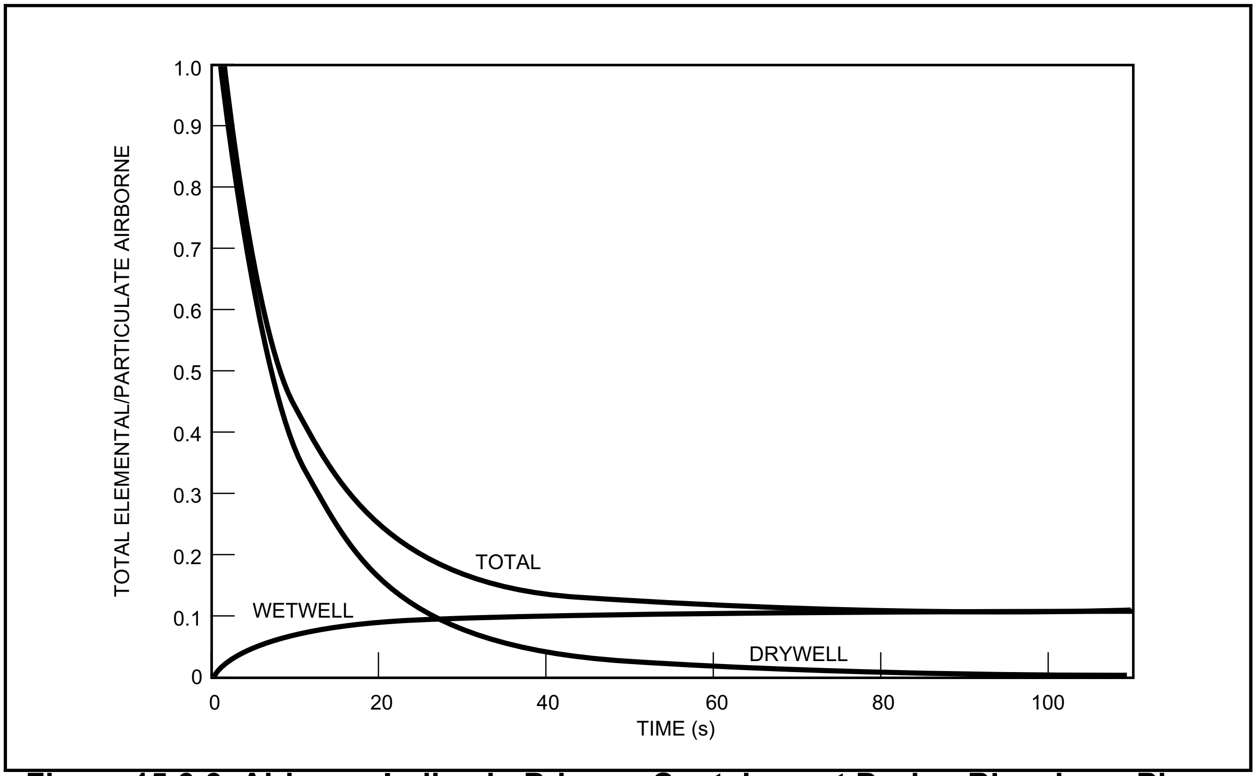


Figure 15.6-3 Airborne Iodine in Primary Containment During Blowdown Phase

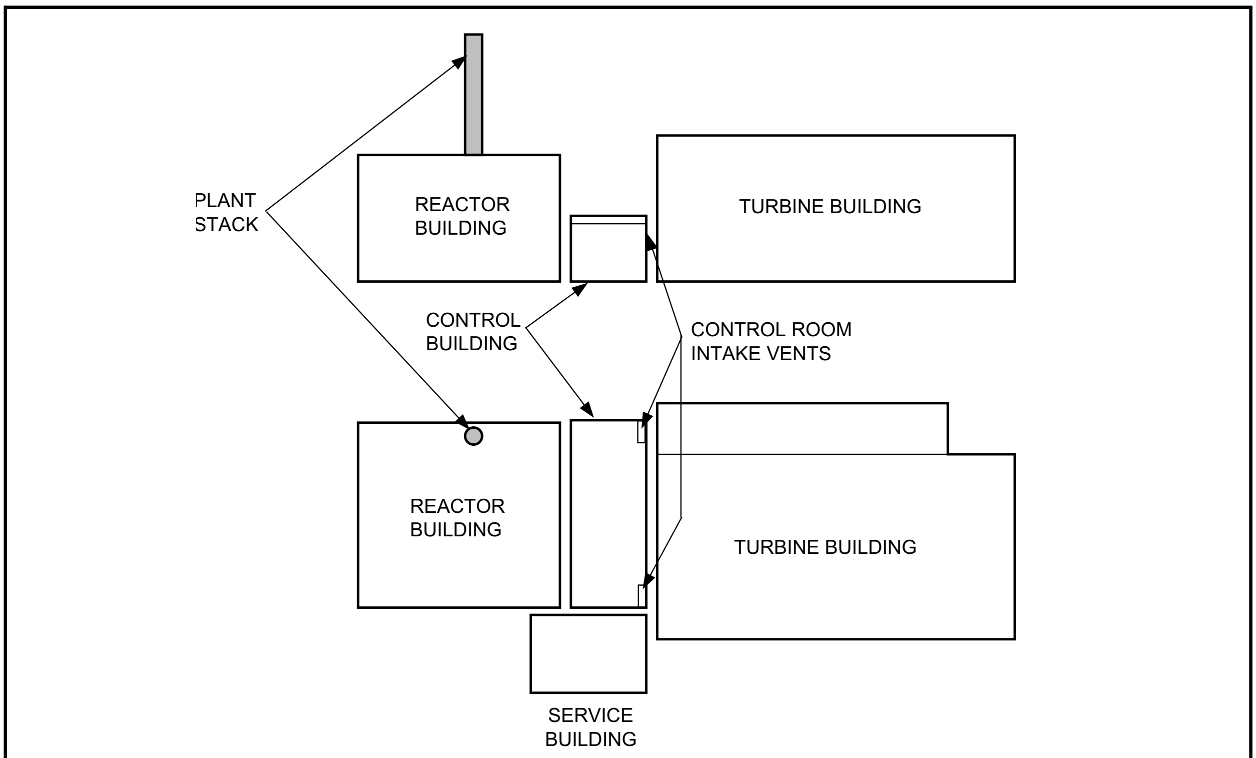


Figure 15.6-4 ABWR Plant Layout

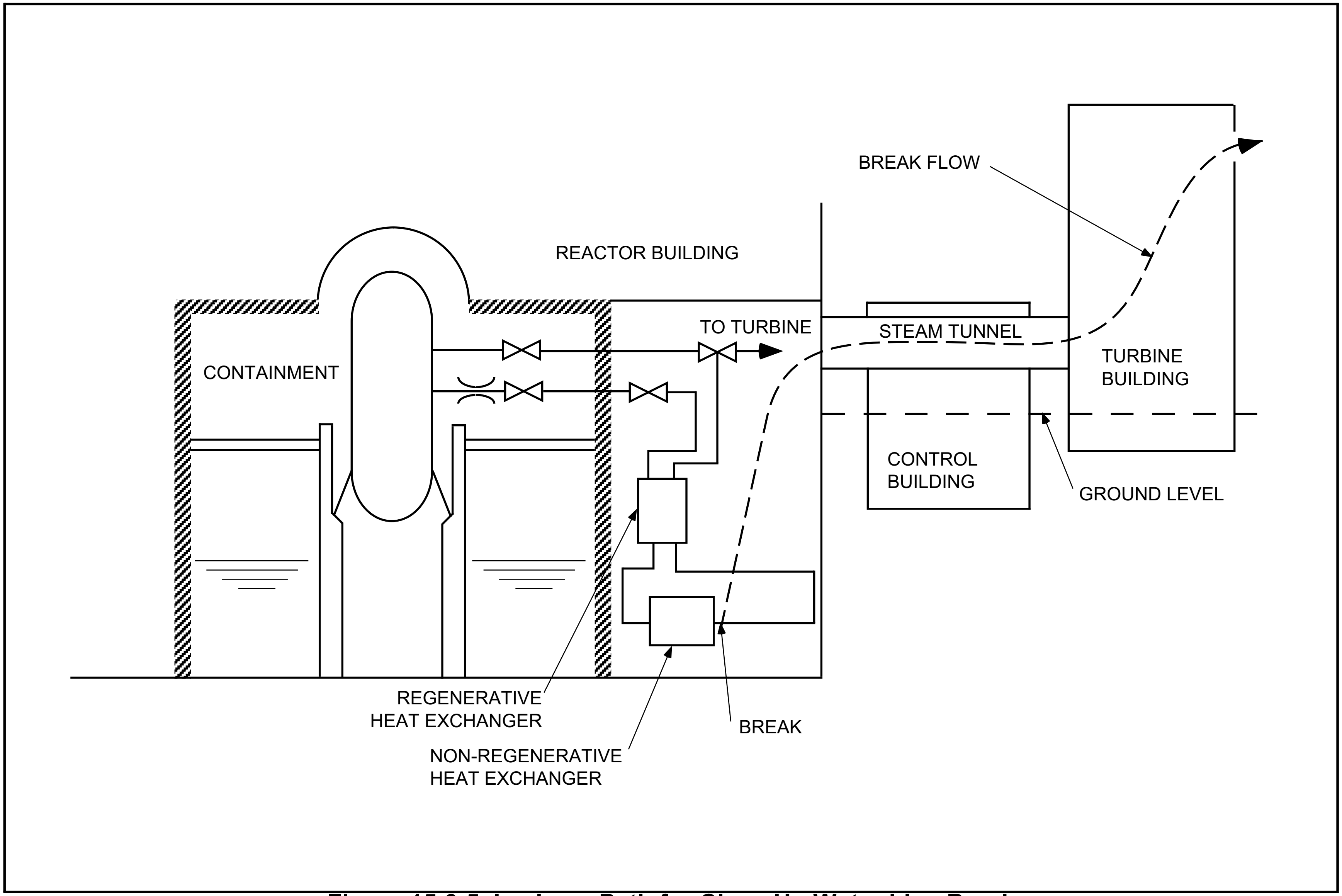


Figure 15.6-5 Leakage Path for Clean Up Water Line Break

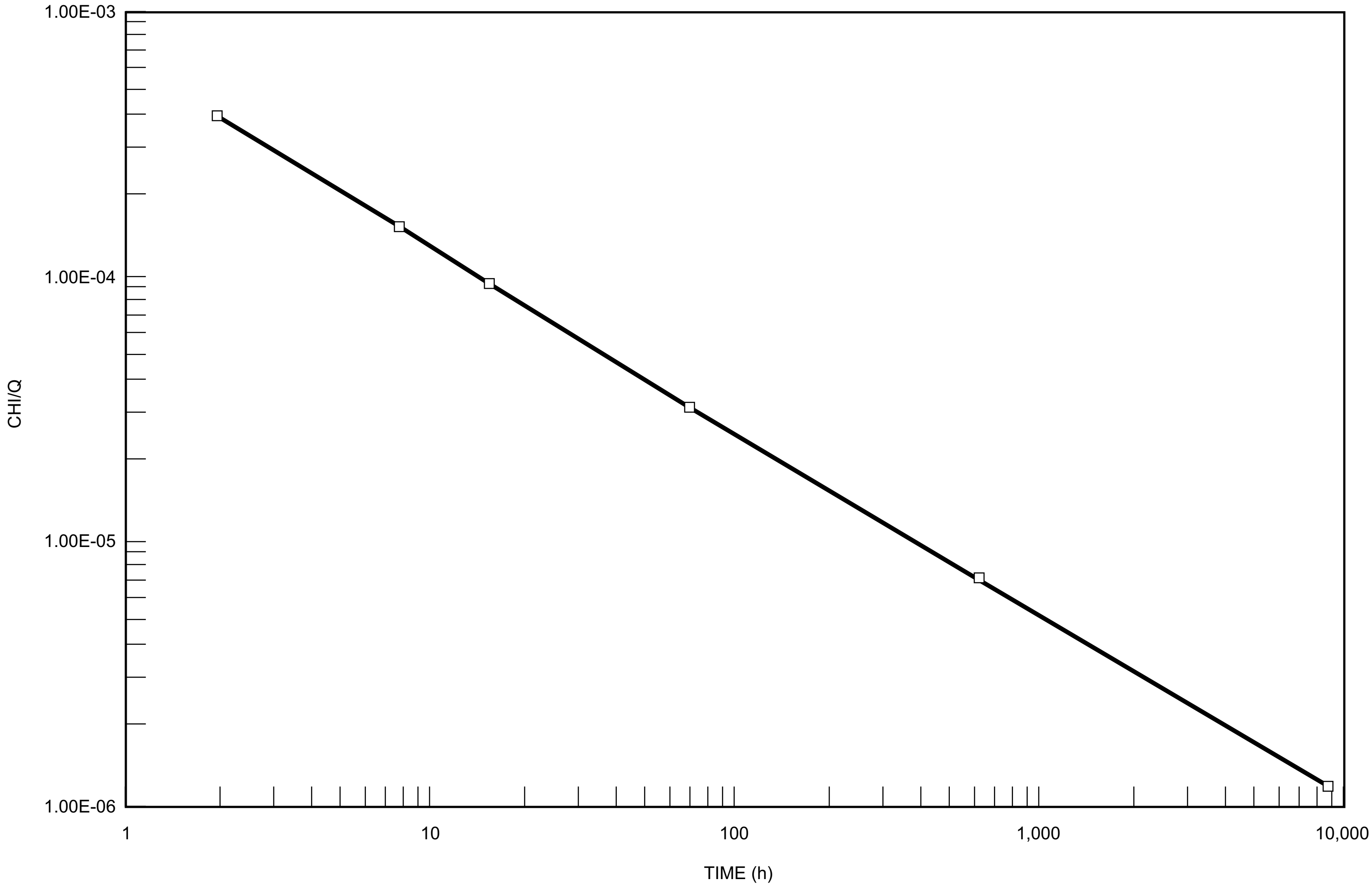


Figure 15.6-6 ABWR Limiting LPZ CHI/Q

15.7 Radioactive Release from Subsystems and Components

15.7.1 Radiological Consequences of a Radioactive Gas Waste System Leak or Failure

15.7.1.1 Basis and Assumptions

The radiological consequences for an offgas system accident as specified in Standard Review Plan 11.3, Branch Technical Position ESTB 11-5 are presented. The branch technical position assumptions were used except as detailed below to evaluate this accident. The results are presented in Tables 15.7-1 through 15.7-3 and show the ABWR design to be compliant with the requirements of the position paper.

The ABWR offgas system is detailed in Subsection 11.3. The system is designed to be both detonation and seismic resistant and meets all criteria of Regulatory Guide 1.143. As such the failure of a single active component leading to a direct release of radioactive gases to the environment is highly unlikely. Therefore, inadvertent operator action with bypass of the delay charcoal beds is analyzed for compliance to ESTB 11-5. A top level diagram of the ABWR offgas system can be found in Figure 11.3-1 (see also Figure 15.7-2) which shows that the ABWR charcoal beds consists of nine charcoal tanks.The first or guard tank contains 4,721 kg of charcoal followed by a flow split into four lines, each line of which leads through 2 massive tanks each containing 12,200 kg of charcoal. Bypass valves exists to direct flow around the (1) guard tank, (2) four series of follow-on tanks or (3) all tanks. To bypass either pathway (1) or (2) above requires the operator to enter a computer command with a required permissive. To bypass all tanks (pathway (3)) requires the operator to key in the command with two separate permissives. Since pathway (3) would require both inadvertent operation upon the operator (keying in the wrong command) plus getting two specific permissives for three incorrect decisions, it is not assumed that pathway (3) is likely to occur. Redundant upon human decision making and downstream of the charcoal beds and the post charcoal bed particle filter shown in Figure 11.3-1 are a series of two redundant radiation monitoring instruments and an air operated isolation valve which will alarm the control room and automatically shut off all flow from the offgas system for radioactivity levels in excess of environmental limits which are defined by 10CFR20 as not greater than $2x10^{-2}$m Sv/h at the site boundary. Therefore, bypass of the charcoal beds during periods with significant radioactive flow through the offgas system will be limited and/or automatically terminated by actuation of the downstream sensors.

To evaluate the potential radiological consequences of an inadvertent bypass of the charcoal beds, it was assumed that operator error or computer error has led to the bypass of the eight follow-on beds in addition to the failure of the automated air operated downstream isolation valve. It is also assumed that during this period, the plant is running at and continues to run at the maximum permissible offgas rate of 14.8 GBq/s (based upon the assumption of 0.0037 GBq/s/MWt as stipulated in Standard Review Plan 11.3) evaluated to a decay time of 30 minutes from the vessel exit nozzle. Even with the failure of the downstream isolation valve, it is not anticipated or assumed that the isolation instrumentation would fail but would instead alarm the control room with a high radiation alarm causing the operator to manually isolate the offgas system (i.e., close suction valves) within 30 minutes of the alarm. Therefore, this analysis differs from the branch technical position on the following points:

(1) Flow is through a single 4,721 kg charcoal tank with an evaluated hold up time given by NUREG-0016, equation 1.5.1.6 using K_d's for Kr and Xe from NUREG-0016.

(2) There is no motive force to remove any significant inventory from the eight follow-on charcoal tanks while in bypass and therefore no activity from these tanks is included in the final release calculations.

(3) With redundant instrumentation, it is expected that operator intervention to either shut off the bypass or isolate the offgas system is predicted to occur within 30 minutes and therefore the total flow from the system is evaluated for 30 minutes and not the 2 hour period stipulated in the branch technical position.

15.7.1.2 Design Basis Accident

The DBA evaluation assumptions are given in Table 15.7-1 with the isotopic flows and releases given in Table 15.7-2, and the meteorology and dose results given in Table 15.7-3.

15.7.1.3 Results

The dose results are given in Table 15.7-3 and are within the limiting 2.5×10^{-2}Sv whole body dose for an infrequent event or the 0.5×10^{-2}Sv frequent event limitation of the Branch Technical Position ESTB-11-5.

15.7.2 Liquid Radioactive System Failure

This section of the Standard Review Plan has been deleted.

15.7.3 Postulated Radioactive Release Due to Liquid Radwaste Tank Failure

15.7.3.1 Identification of Cause and Frequency Classification

An unspecified event causes the complete release of the radioactive inventory in all tanks containing radionuclides in the Liquid Radwaste System. Postulated events that could cause a release of the inventory of a tank are sudden unmonitored cracks in the vessel or operator error. Small cracks and consequent low-level releases are bounded by this analysis and should be contained without any significant release.

The ABWR Radwaste Building is a Seismic Category I structure designed to withstand all credible seismic events. In addition, all compartments containing liquid radwastes are steel-lined up to a height capable of containing the release of all the liquid radwastes into the compartment. Because of these design capabilities, it is considered remote that any major accident involving the release of liquid radwastes into these volumes would result in the release of these liquids to the environment via the liquid pathway. Releases as a result of major cracks would instead result in the release of the liquid radwastes to the compartment and then to the building sump system for containment in other tanks or emergency tanks. A complete description of the Liquid Radwaste System is found in Section 11.2, except for the tank inventories, which are found in Section 12.2.

A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is also provided to prevent inadvertent opening of a drain valve. Should a release of wastes occur, the steel lining would contain the release until the floor drain sump pumps in the building capture and contain such spills.

The probability of a complete tank release is considered low enough to warrant this event as a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

Following a failure, the area radiation alarms would be expected to alarm at one minute with operator intervention following at approximately five minutes after release. However, the rupture of a waste tank would leave little recourse for the operator. Gaseous wastes would be trapped following alarm initiation, since isolation would occur upon alarm initiation; however, no credit for isolation is taken for this aspect and gaseous releases are expected to be purged to the environment.

Liquid release would be contained within the steel liner and would present no immediate threat to the environment leaving the operator sufficient time (on the order of hours) in which to recover systems to pump the release into holding tanks or emergency tanks.

15.7.3.3 Design Basis Accident

Based upon the above discussion, a single pathway is considered for release of fission products to the environment via airborne releases. The liquid pathway is not considered due to the mitigative capabilities of the Radwaste Building.

For the airborne pathway, volatile iodine species in the tanks using the inventories in Section 12.2 are considered. These inventories are based upon the design basis release rates found in Section 11.1. Though isolation is expected within minutes of this occurrence, release of 10% of the iodine inventory is conservatively assumed over a two hour period. Specific values for this analysis are found in Tables 15.7-5 and 15.7-6.

15.7.3.4 Results

No liquid or significant (from airborne species) ground contamination is expected. Airborne doses are given in Table 15.7-7 and are a fraction of 10CFR100 criteria. COL applicants need to update the dose calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.7.6.1 for COL license information.).

15.7.4 Fuel-Handling Accident

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel-handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in dropping a raised fuel assembly onto the reactor core.

15.7.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

Sequence of Events	Elapsed Time (min)
(1) Channeled fuel bundle is being handled by a crane over reactor core. Crane motion changes from horizontal and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	0
(2) Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
(3) Gases pass from the water to the Reactor Building, fuel-handling area.	0
(4) The Reactor Building ventilation system high radiation alarm alerts plant personnel.	10
(5) Operator actions begin.	10

15.7.4.2.2 Identification of Operator Actions

The following actions are carried out:

(1) Initiate the evacuation of the Reactor Building fuel-handling area and the locking of the fuel storage building doors.

(2) The fuel-handling foreman gives instructions to go immediately to the radiation protection personnel decontamination area.

(3) The fuel-handling foreman makes the operations shift engineer aware of the accident.

(4) The shift engineer determines if the normal ventilation system has isolated and the SGTS is in operation.

(5) The shift engineer initiates action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the Reactor Building.

(6) The plant superintendent or delegate determines if the SGTS is performing as designed.

(7) The duty shift engineer posts the appropriate radiological control signs at the entrance of the Reactor Building.

(8) Before entry to the refueling area is made, a careful study of conditions, radiation levels, etc., is performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a reasonable, yet conservative assessment of the consequences.

To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

The fuel assembly is expected to impact on the core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1.35 N•m prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 0.339 kN•m before cladding failure (based on 1% uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of greater than 70% fuel by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are:

(1) The fuel assembly is dropped from the maximum height allowed by the fuel-handling equipment, about 13.4m.

(2) The entire amount of potential energy, referenced to the top of core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, fuel grapple or grapple cable breaks.

(3) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

Regulatory Guide 1.25 provides assumptions acceptable to the NRC that may be used in evaluating the radiological consequences of a postulated fuel-handling accident resulting in damage to the fuel cladding and subsequent release of radioactive materials.

The key implementation assumptions used in the analyses are as follows:

(1) Two-hour site boundary meteorology calculated in accordance with USNRC RG 1.3 and 1.145 for ground level release.

(2) SGTS filter efficiency 99% for all iodine forms.

(3) All activity released to the environment is via the SGTS.

(4) 115 fuel rods are calculated to be damaged.

15.7.4.3.3 Results

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods was determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 279.87 kg and the wet weight of the grapple component is 158.76 kg. The drop distance is 13.38m. The total energy to be dissipated by the first impact is:

$$E = (279.87 + 158.76)(13.38) = 57.56 \text{ kN}\cdot\text{m}$$

One half of the energy was considered to be absorbed by the falling assembly and one half by the impacted assemblies.

No energy was considered to be absorbed by the fuel pellets (i.e., the energy was absorbed entirely by the non-fuel components of the assemblies). The energy available for clad deformation was considered to be proportional to the mass ratio:

$$\frac{\text{mass of cladding}}{(\text{mass of assembly} - \text{mass of fuel pellets})}$$

and is equal to a maximum of 0.519 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is

$$(28.78\ \text{kN}\cdot\text{m})\ (0.519) = 14.93\ \text{kN}\cdot\text{m}$$

Each rod that fails is expected to absorb approximately 0.339 kN•m before cladding failure, based on uniform 1% plastic deformation of the cladding.

The number of rods failed in the impacted assemblies is:

$$N_F = \frac{(14.93\ \text{kN}\cdot\text{m})}{(0.339\ \text{kN}\cdot\text{m})} = 44\ \text{rods}$$

The dropped assembly was considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason, it was assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact was $62 + 44 = 106$.

The assembly was assumed to tip over and impact horizontally on the top of the core. The remaining available energy was used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$E = W_G H_G + \int_0^{H_B} \frac{W_B}{H_B} y\ \text{dy}$$

$$= W_G H_G + \frac{1}{2} W_B H_B$$

$$= (158.76\ \text{kg})(4.06m) + \frac{1}{2}(279.87)(4.06m)$$

$$= 11.9(\text{ kN•m})$$

As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies and the fraction available for clad deformation was 0.519. The energy available to deform clad in the impacted assemblies was:

$$E_c = (0.5)(11.9\text{ kN•m})(0.519)$$

$$= 3.09\text{ kN·m}$$

and the number of failures in the impacted assemblies was

$$N_F = \frac{(3.09\text{ kN•m})}{(0.339\text{ kN•m})} = 9\text{ rods}$$

Since the rods in the dropped assembly were considered to have failed in the initial impact, the total failed rods in both impacts is $106 + 9 = 115$.

15.7.4.4 Not Used

15.7.4.5 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. The analysis is referred to as the “Design Basis Analysis”.

The fission product inventory in the fuel rods assumed to be damaged is based on 1000 days of continuous operation at 4005 MWt. A 24 h period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 h following initiation of reactor shutdown. Figure 15.7-1 shows the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

The Design Basis Analysis is based on Regulatory Guide 1.25. The specific models, assumptions and the program used for computer evaluations are described in Reference 15.7-1. Specific values or parameters used in the evaluation are presented in Table 15.7-8.

15.7.4.5.1.1 Fission Product Release from Fuel

Per the conditions in Regulatory Guide 1.25, The following conditions are assumed applicable for this event:

(1) Power Level—4005 MWt for 3 years

(2) Plenum Activity—10% of the radioactivity for iodine and noble gases except Kr-85 and 30% for Kr-85

(3) Fission Product Peaking Factor—1.5 for those rods damaged

(4) Activity Released to Reactor Building—10% of the noble gas activity and 0.1% for the iodine activity

Based on the above conditions, the activity released to the Reactor Building is presented in Table 15.7-9.

15.7.4.5.1.2 Fission Product Transport to the Environment

Also, per the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the Reactor Building (Table 15.7-9) is released to the environment over a 2 hr period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-10.

15.7.4.5.1.3 Results

The calculated exposures for the Design Basis Analysis are presented in Table 15.7-11 and are within the guidelines of 10CFR100. COL applicants need to update these calculations to conform to the as-designed plant and site-specific parameters. See Subsection 15.7.6.1 for COL license information.

15.7.5 Spent Fuel Cask Drop Accident

15.7.5.1 Identification of Cause

Due to the redundant nature of the crane, the cask drop accident is not believed to be a credible accident. However, the accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

15.7.5.2 Radiological Analysis

The largest size BWR fuel cask is conservatively assumed to be dropped approximately 29 m from the refueling floor level to ground level on transport from the decontamination pit out of the reactor building.

It is conservatively assumed that all fuel rods are damaged and the fission gases in the fuel rod gap space are released to the reactor building and then to the environment over a two-hour period. Table 15.7-12 provides the assumptions for this analysis and Table 15.7-13 radiological consequences. As can be seen from Table 15.7-13, the radiological releases are within guidelines. COL applicants need to update these calculations to conform to the as-designed plant and site specific parameters. See Subsection 15.7.6.1 for COL license information.

15.7.6 COL License Information

15.7.6.1 Radiological Consequences of Non-Line Break Accidents

The COL applicant will evaluate the radiological consequences of the failure of the liquid radwaste tank, the fuel handling accident, and the fuel cask drop accident for the final plant design and site parameters (see Subsections 15.7.3.4, 15.7.4.5.1.3, and 15.7.5.2.).

15.7.7 References

15.7-1 H. Careway, V. Nguyen, et al., "Radiological Accident Evaluation—The CONAC03 Code", December 1981 (NEDO-21143-1).

Table 15.7-1 Offgas System Failure Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms		
A.	Power Level	4005MWt
B.	Offgas Release Rate	14.8 GBq/s (referenced to 30 min)
C.	Charcoal Mass	Guard Tank, 4,721 kg
D.	Charcoal Delay[1]	
	Kr	2.07 h
	Xe	36.9 h
E.	Duration of Release	30 min
F.	Design Basis Rate	3.7 GBq/s
G.	Maximum Technical Specification Rate	14.8 GBq/s
II. Dispersion and Dose Rate		
A.	Meteorology	Table 15.7-3
B.	Dose Methodology	Reference 15.7-1
C.	Dose Conversion Assumptions	Reference 15.7-1, RG 1.109
D.	Activity Releases	Table 15.7-2
E.	Dose Evaluations	Table 15.7-3

Note 1:Charcoal Delay calculated based upon charcoal mass using equation 1.5.1.6 of NUREG-0016 and K_d's taken from 1.5.2.19 and 1.5.2.20 of NUREG-0016.

Table 15.7-2 Isotopic Source and Release to the Environment

	Isotope Flow Rates (Design Basis)			Integrated Releases (at Max Tech Spec)		
	T=0	T=2 min	T=30 min	T=30 min	Charcoal Delay	Total
Isotope	(MBq/s)	(MBq/s)	(MBq/s)	(MBq/s)	(MBq/s)	(MBq/s)
Kr-83m	1.27E+2	1.25E+2	1.05E+2	7.57E+5	4.07E+5	1.16E+6
Kr-85m	2.23E+2	2.22E+2	2.06E+2	1.48E+6	1.15E+6	2.63E+6
Kr-85	6.96E–1	6.96E–1	6.96E–1	5.18E+3	5.18E+3	9.99E+3
Kr-87	7.66E+2	7.51E+2	5.81E+2	4.19E+6	1.69E+6	5.88E+6
Kr-88	7.66E+2	7.59E+2	6.77E+2	4.87E+6	3.23E+6	8.10E+6
Kr-89	4.74E+3	3.05E+3	6.55E+0	4.74E+4	0.00	4.74E+4
Kr-90	1.04E+4	7.96E+2	1.75E–13	0.00	0.00	0.00
Kr-91	1.27E+4	8.92E–1	0	0.00	0.00	0.00
Kr-92	1.27E+4	2.96E–16	0	0.00	0.00	0.00
Kr-93	3.34E+3	1.18E–25	0	0.00	0.00	0.00
Kr-94	8.21E+2	0	0	0.00	0.00	0.00
Kr-95	7.66E+1	0	0	0.00	0.00	0.00
Kr-97	5.00E–1	0	0	0.00	0.00	0.00
Total	4.66E+4	5.71E+3	1.58E+3	1.14E+7	6.48E+6	1.78E+7
Xe-131m	5.44E–1	5.44E–1	5.44E–1	4.07E+3	3.70E+3	7.40E+3
Xe-133m	1.04E+1	1.04E+1	1.04E+1	7.47E+4	4.59E+4	1.21E+5
Xe-133	2.92E+2	2.92E+2	2.92E+2	2.10E+6	1.80E+6	3.90E+6
Xe-135m	9.73E+2	8.92E+2	2.50E+2`	1.80E+6	0.00	1.80E+6
Xe-135	8.36E+2	8.33E+2	8.03E+2	5.79E+6	3.39E+5	6.13E+6
Xe-137	5.44E+3	3.77E+3	2.42E+1	1.74E+5	0.00	1.74E+5
Xe-138	3.20E+3	2.90E+3	7.40E+2	5.33E+6	0.00	5.33E+6
Xe-139	1.04E+4	1.33E+3	4.03E–10	0.00E+0	0.00	0.00
Xe-140	1.11E+4	2.46E+1	0	0.00E+0	0.00	0.00
Xe-141	9.07E+3	9.03E–18	0	0.00E+0	0.00	0.00
Xe-132	2.65E+3	6.51E–27	0	0.00E+0	0.00	0.00
Xe-143	4.44E+2	0	0	0.00E+0	0.00	0.00
Xe-144	2.09E+1	0	0	0.00E+0	0.00	0.00
Total	4.45E+4	1.01E+4	2.12E+3	1.53E+7	2.19E+6	1.75E+7
Kr+XE	9.11E+4	1.58E+4	3.70E+3	2.66E+7	8.66E+6	3.53E+7

Table 15.7-3 Offgas System Failure Meteorology and Dose Results

Meteorology	Whole Body Dose
1.37E-3 s/m3	2.75 mSv

Table 15.7-4 Not Used

Table 15.7-5 Radwaste System Failure Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
	A. Power level	4005 MWt
	B. Source inventory	Table 12.2-13
	C. Fraction of iodine released	10%
	D. Duration of accident	2 h
II	**Dispersion and Dose Data**	
	A. Meteorology	Table15.7-7
	B. Dose commitment distance	800m
	C. Method of dose calculation	Reference 15.7-1
	D. Dose conversion assumptions	Reference 15.7-1, RG 1.109, ICRP 30.
	E. Activity released	Table 15.7-6
	F. Dose evaluations	Table 15.7-7

Table 15.7-6 Isotopic Release to Environment (megabecquerel)

Isotope	1-min	10-min	1-hour	2-hour
I-131	4.8E+4	4.4E+5	1.6E+6	1.9E+6
I-132	4.1E+3	3.5E+4	1.1E+5	1.3E+5
I-133	2.8E+4	2.6E+5	8.9E+5	1.1E+6
I-134	2.6E+3	2.2E+4	6.3E+4	7.0E+4
I-135	1.2E+4	1.1E+5	3.7E+5	4.4E+5
Total I	9.5E+4	8.7E+5	3.0E+6	3.6E+6

Table 15.7-7 Radwaste System Failure Accident Meteorology* and Dose Results

Meteorology (s/m^3)	Distance (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
1.42E–03	Max	3.0E–1	2.4E–4
1.37E-03	Chp 2	2.9E–1	2.3E–4
1.18E–03	300	2.5E–1	2.0E–4
2.19E–04	800	4.6E–2	3.8E–5

* Meteorology calculated using Regulatory Guide 1.145 for ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 10% of 10CFR100.

Table 15.7-8 Fuel-Handling Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
	A. Power level	4005 MWt
	B. Radial peaking factor	1.5
	C. Duration of accident	2 h
	D. No. rods damaged	115 rods
	E. Minimum time to accident	24 h
	F. Peak linear power density	44 kW/m
	G. Average burnup	32,000 MW•d/t
	H. Maximum fuel centerline temperature	1824°C
	I. Fraction of activity released	10% of all isotopes except 30% Kr-85
II	**Data and Assumptions Used to Estimate Activity Released**	
	A. Species fraction	
	(1) Organic iodine	0.25%
	(2) Inorganic iodine	99.75%
	(3) Noble gas	100%
	B. Pool Retention decontamination factor	
	(1) Organic iodine	1
	(2) Inorganic iodine	133
	(3) Noble gas	1
	C. SGTS filtration efficiency*	
	(1) Organic iodine	99%
	(2) Inorganic iodine	99%
	(3) Noble gas	0%
	D. Reactor Building Release Rate	300%/2 h
III	**Dispersion and Dose Data**	
	A. Meteorology	Table 15.7-11
	B. Boundary and LPZ distances	Table 15.7-11
	C. Method of dose calculation	Reference 15.7-1
	D. Dose conversion assumptions	Reference 15.7-1 RG 1.109, ICRP30
	E. Activity inventory/releases	Table 15.7-9, Table 15.7-10
	F. Dose evaluations	Table 15.7-11

* No SGTS filtration for first 20 minutes of accident.

Table 15.7-9 Fuel-Handling Accident Reactor Building Inventory (megabecquerel)

Isotope	1 minute	10minute	1 hour	2 hours
I-131	1.13E+07	8.99E+06	6.88E+06	6.73E+06
I-132	1.45E+07	1.11E+07	6.59E+06	4.77E+06
I-133	1.17E+07	9.25E+06	6.92E+06	6.55E+06
I-134	6.29E–01	4.44E–01	1.77E–01	7.88E–02
I-135	1.91E+06	1.50E+06	1.06E+06	9.32E+05
Total	3.93E+07	3.08E+07	2.14E+07	1.90E+07
Kr-83m	5.99E+05	4.51E+05	2.53E+05	1.69E+05
Kr-85m	7.66E+06	5.96E+06	4.03E+06	3.38E+06
Kr-85	4.18E+07	3.33E+07	2.56E+07	2.50E+07
Kr-87	1.18E+03	8.70E+02	4.22E+02	2.41E+02
Kr-88	2.21E+06	1.70E+06	1.07E+06	8.18E+05
Kr-89	2.32E–05	2.59E–06	3.55E–11	3.70E–16
Xe-131m	7.29E+06	5.81E+06	4.44E+06	4.37E+06
Xe-133m	9.66E+07	7.70E+07	5.85E+07	5.62E+07
Xe-133	2.46E+09	1.96E+09	1.50E+09	1.46E+09
Xe-135m	2.80E+07	1.51E+07	1.26E+06	8.66E+04
Xe-135	5.62E+08	4.44E+08	3.20E+08	2.90E+08
Xe-137	5.22E–05	8.14E–06	7.36E–10	1.04E–14
Xe-138	5.62E–05	2.90E–05	1.93E–06	1.00E–07
Total	3.21E+09	2.54E+09	1.91E+09	1.84E+09

Table 15.7-10 Fuel-Handling Accident Isotopic Release to Environment (megabecquerel)

Isotope	1 minute	10 minute	1 hour	2 hours
I-131	2.85E+05	2.56E+06	4.55E+06	4.55E+06
I-132	3.67E+05	3.22E+06	5.62E+06	5.62E+06
I-133	2.95E+05	2.64E+06	4.70E+06	4.70E+06
I-134	1.60E–02	1.36E–01	2.28E–01	2.28E–01
I-135	4.85E+04	4.29E+05	7.62E+05	7.62E+05
TOTAL	9.96E+05	8.85E+06	1.56E+07	1.56E+07
Kr-83m	1.52E+04	1.32E+05	2.33E+05	2.38E+05
Kr-85m	1.94E+05	1.72E+06	3.08E+06	3.16E+06
Kr-85	1.05E+06	9.47E+06	1.72E+07	1.77E+07
Kr-87	3.00E+01	2.59E+02	4.51E+02	4.55E+02
Kr-88	5.62E+04	4.92E+05	8.81E+05	8.99E+05
Kr-89	6.55E–07	2.77E–06	3.01E–06	3.01E–06
Xe-131m	1.84E+05	1.65E+06	3.00E+06	3.09E+06
Xe-133m	2.44E+06	2.18E+07	3.96E+07	4.07E+07
Xe-133	6.22E+07	5.59E+08	1.01E+09	1.04E+09
Xe-135m	7.25E+05	5.44E+06	8.18E+06	8.18E+06
Xe-135	1.42E+07	1.27E+08	2.29E+08	2.36E+08
Xe-137	1.45E–06	6.77E–06	7.66E–06	7.66E–06
Xe-138	1.46E–06	1.07E–05	1.59E–05	1.59E–05
TOTAL	8.11E+07	7.26E+08	1.32E+09	1.35E+09

Table 15.7-11 Fuel-Handling Accident Meteorologial* Parameters And Radiological Effects

Meteorology (s/m^3)	**Distance (m)**	**Thyroid Dose (Sv)**	**Whole Body Dose (Sv)**
1.37E–03	max/Chp 2	7.5E–01	1.2E–02
1.18E–03	300	6.4E–01	1.1E–02
2.19E–04	800	1.2E–01	2.0E–03

* Meteorology calculated using Regulatory Guide 1.145 for a ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 25% of 10CFR100 limitation.

Table 15.7-12 Fuel Cask Drop Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
A.	Power level of reactor while fuel was in core	4005 MWt
B.	Radial peaking factor while fuel was in core	1.5
C.	Fuel bundles in cask	18
D.	Fuel damaged	1116 rods
E.	Minimum time of fuel in storage prior to accident	120 days
F.	Peak linear power density	44 kW/m
G.	Average burnup of fuel	32,000 MW•d/t
H.	Maximum fuel centerline temperature	1824°C
I.	Fraction of activity released	10% of all isotopes except 30% Kr-85
J.	Time period for Reactor Building release	2 h
K.	Iodine filter efficiency	None
II	**Dispersion and Dose Data**	
A.	Meteorology	Table 15.7-13
B.	Boundary and LPZ distances	Table 15.7-13
C.	Method of dose calculation	Reference 15.7-1
D.	Dose conversion assumptions	Reference 15.7-1 and RG 1.109
E.	Activity inventory/releases	Table 15.7-13
F.	Dose evaluations	Table 15.7-13

Table 15.7-13 Cask Drop Accident Radiological Results Fission Product Releases (megabecquerel)

Isotope	Release to Reactor Building	Release to Environment
I-131	4.0E+5	3.8E+5
Kr-85	4.1E+8	3.7E+8
Xe-131m	2.0E+5	1.9E+5
Xe-133	4.8E+3	4.4E+3

Meteorology* and Dose Results

Meteorology (s/m^3)	Distance (m)	Thyroid (Sv)	Whole Body (Sv)
1.84E–02	max	7.4E–1	1.3E–3
1.37E-02	Chp 2	5.6E–2	1.0E–4
1.18E–03	300	4.8E–2	8.5E–5
2.19E–04	800	8.9E–3	1.6E–5

* Meteorology calculated using Regulatory Guide 1.145 for ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 25% of 10CFR100 limitation.

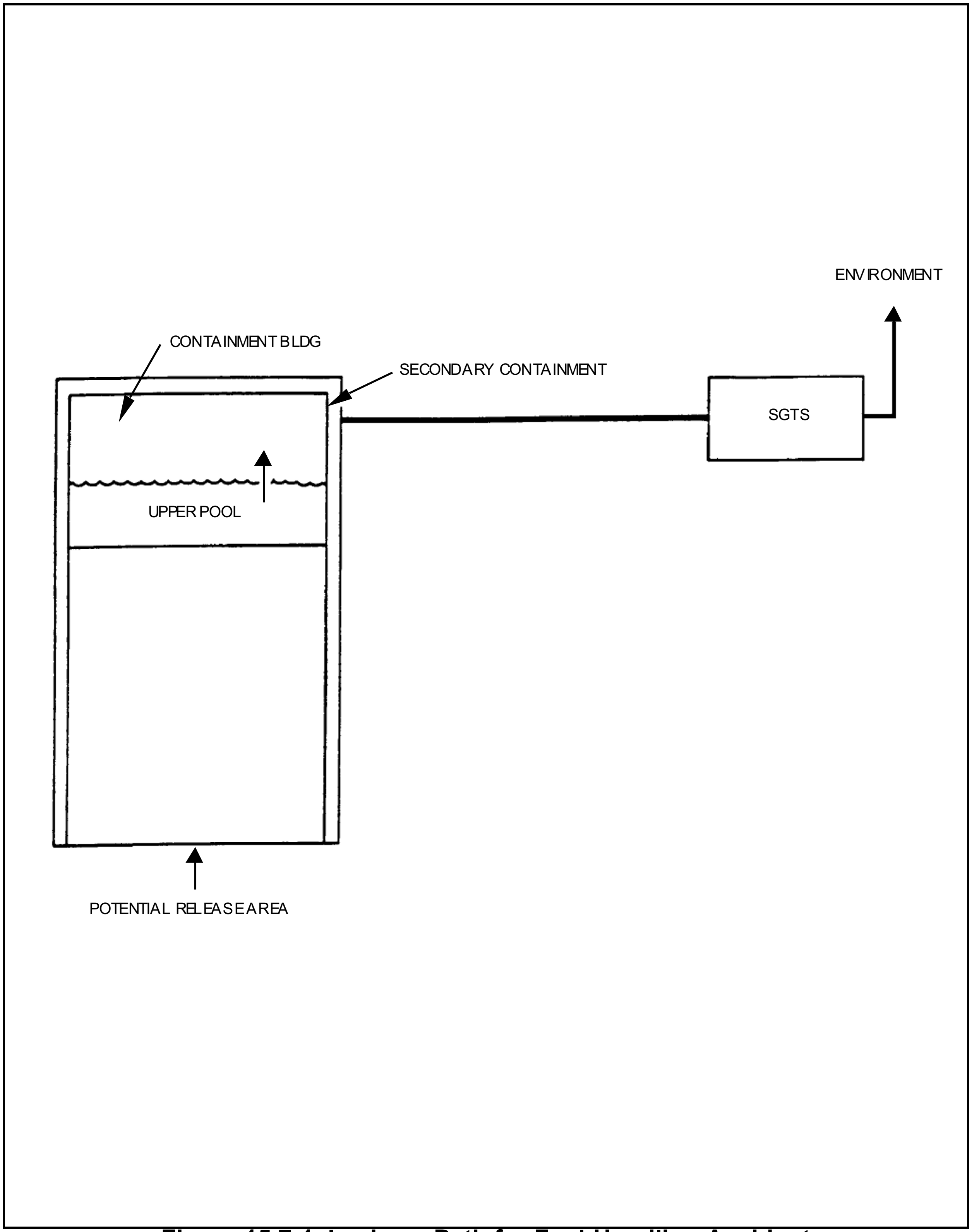


Figure 15.7-1 Leakage Path for Fuel-Handling Accident

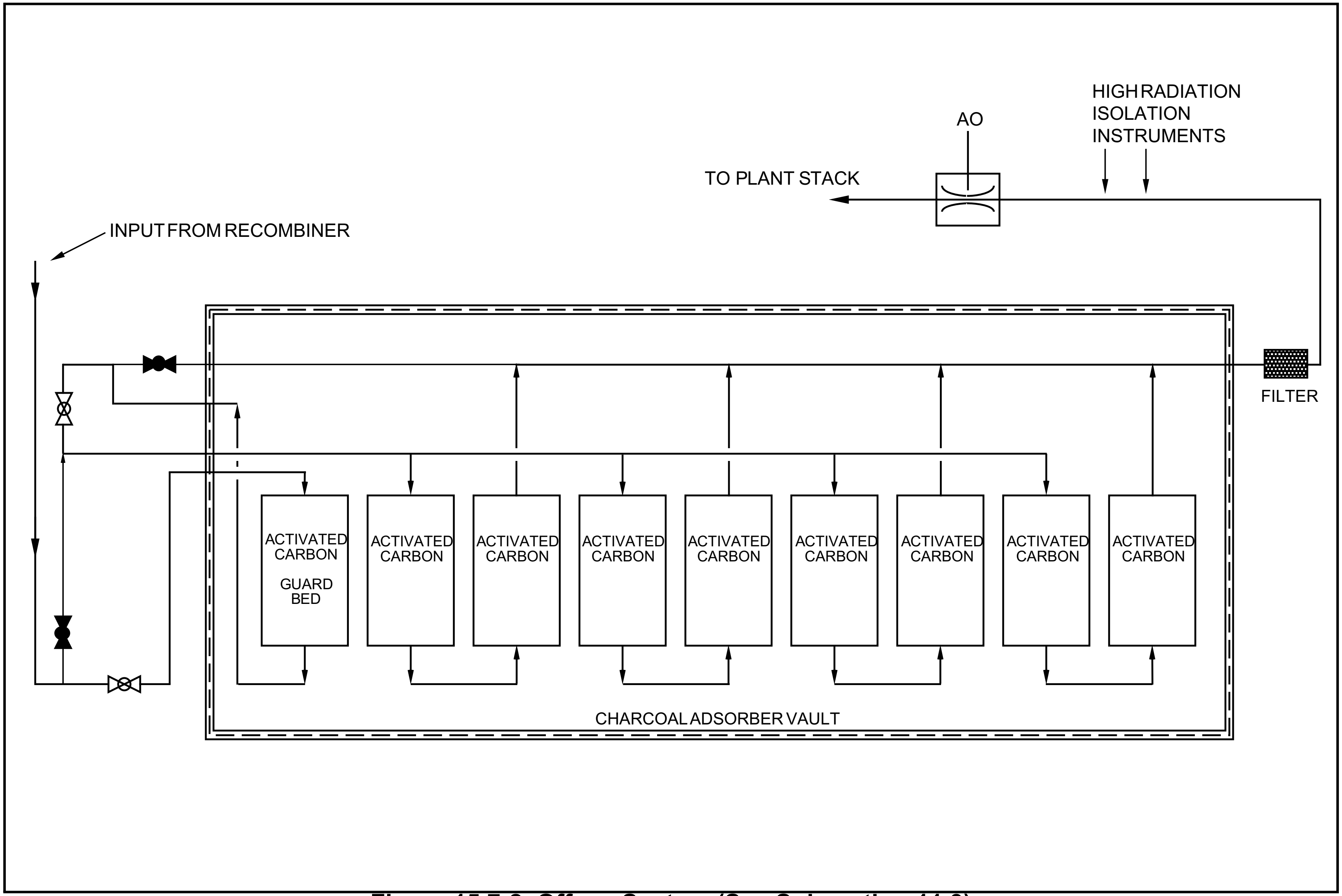


Figure 15.7-2 Offgas System (See Subsection 11.3)

15.8 Anticipated Transients Without Scram

15.8.1 Requirements

SRP 15.8 requires an automatic recirculation pump trip (RPT) and emergency procedures for ATWS. This SRP has been somewhat superseded by the issuance of 10CFR50.62, which requires the BWR to have automatic RPT, an Alternate Rod Insertion (ARI) System and an automatic Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent to19.5 m^3/h of 13 wt% sodium pentaborate solution.

15.8.2 Plant Capabilities

For ATWS prevention/mitigation for ABWR, the following are provided:

(1) An ARI System that utilizes sensors and logic which are diverse and independent of the reactor protection system

(2) Electrical insertion of FMCRDs that also utilize sensors and logic which are diverse and independent of the reactor protection system

(3) Automatic recirculation pump trip under conditions indicative of an ATWS

(4) Automatic initiation of the SLCS with 378 L/min of 13.4wt.% under conditions indicative of an ATWS

The ABWR has the ATWS-RPT feature which prevents reactor vessel overpressure and possible short-term fuel damage for the most limiting postulated ATWS events. The design details of this system are given in Section 7.7. Emergency procedures for ATWS are described in Chapter 18. Thus, SRP 15.8 is satisfied.

The ATWS rule of 10CFR50.62 was written as hardware-specific, rather than functionally, because it clearly reflected the BWR use of locking-piston control rod drives. The ABWR, however, uses a FMCRD design with both hydraulic and electric means to achieve shutdown. This drive design is described in detail in Section 4.6. The use of this design eliminates the common mode failure potentials of the existing locking-piston CRD by eliminating the scram discharge volume (mechanical common mode potential failure) and by having an electric motor run-in diverse from the hydraulic scram feature.

This latter feature allows rod run-in if scram air header pressure is not exhausted because of a postulated common mode electrical failure and simultaneous failure of the ARI system, and therefore satisfies the intent required by 10CFR50.62. Thus, the design does not need an SLCS to respond to an ATWS threatening event.

The SLCS is required by 10CFR50 Appendix A criteria and is described in Section 9. Because the new drive design eliminates the previous common-mode failure potential and because of the very low probability of simultaneous common-mode failure of a large number of drives, a failure to achieve shutdown is deemed incredible. However, automatic initiation of SLCS under conditions indicative of an ATWS is also incorporated in order to meet the rule specified in 10CFR50.62.

Supporting analysis is documented in Appendix 15E.

15A Plant Nuclear Safety Operational Analysis (NSOA)

15A.1 Objectives

The objectives of the Nuclear Safety Operational Analysis (NSOA) are cited below.

15A.1.1 Essential Protective Sequences

Identify and demonstrate that essential protection sequences needed to accommodate the plant normal operations, moderate frequency incidents (anticipated operational transients), infrequent incidents (abnormal operational transients), and limiting faults (design basis accidents) are available and adequate. In addition, each event considered in the plant safety analysis (Chapter 15) is further examined and analyzed. Specific essential protective sequences are identified. The appropriate sequence is discussed for all BWR operating modes.

15A.1.2 Design Basis Adequacy

Identify and demonstrate that the safety design basis of the various structures, systems or components needed to satisfy the plant essential protection sequences are appropriate, available and adequate. Each protective sequence identifies the specific structures, systems or components performing safety or power generation functions. The interrelationships between primary and secondary (or auxiliary equipment) systems in providing these functions are shown. The individual design bases (identified throughout Tier 2 for each structure, system, or component) are brought together by the analysis in this section. In addition to the individual equipment design basis analysis, the plant-wide design bases are examined and presented here.

15A.1.3 System-Level/Qualitative—Type FMEA

Identify a system-level/qualitative-type Failure Modes and Effects Analysis (FMEA) of essential protective sequences to show compliance with the Single Active Component Failure (SACF)or Single Operator Error (SOE) criteria. Each protective sequence entry is evaluated relative to SACF or SOE criteria. Safety classification aspects and interrelationships between systems are also considered.

15A.1.4 NSOA Criteria Relative to Plant Safety Analysis

Identify the systems, equipment, or components' operational conditions and requirements essential to satisfy the nuclear safety operational criteria utilized in the Chapter 15 plant events.

15A.1.5 Technical Specification Operational Basis

Will establish limiting operating conditions, testing and surveillance bases relative to plant Technical Specifications.

15A.2 Approach to Operational Nuclear Safety

15A.2.1 General Philosophy

The specified measures of safety used in this analysis are referred to as “unacceptable consequences.” They are analytically determinable limits on the consequences of different classifications of plant events. The NSOA is thus an “event-consequence” oriented evaluation. Refer to Figure 15A-1 for a description of the systematic process by which these unacceptable results are converted into safety requirements.

15A.2.2 Specific Philosophy

The following guidelines are utilized to develop the NSOA.

(1) Scope and Classification of Plant Events

(a) Normal (Planned)Operations

Normal Operations are planned conditions in the absence of significant abnormalities. Operations subsequent to an incident (transient, accident or special event) are not considered planned operations until the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations. Specific events are presented in Table 15A-8.

(b) Moderate Frequency Incidents (Anticipated (Expected) Operational Transients)

Moderate Frequency Incidents are deviations from normal conditions which are expected to occur at a moderate frequency, and, as such, the design includes the capability to withstand the conditions without operational impairment. Included are incidents that result from a single operator error, control malfunction and others as presented in Table 15A-9.

(c) Infrequent Incidents (Abnormal (Unexpected) Operational Transients)

Infrequent Incidents are infrequent deviations from normal conditions. The design includes a capability to withstand these conditions without operational impairment. Table 15A-10 presents the events included within this classification.

(d) Limiting Faults (Design Basis (Postulated) Accidents)

Limiting Faults are hypothesized accidents the characteristics and consequences of which are utilized in the design of those systems and components pertinent to the preservation of radioactive material barriers and the restriction of radioactive material release from the barriers. The potential radiation exposures resulting from Limiting Faults may be greater than for any similar accident postulated from the same general accident assumptions. Specific events are presented in Table 15A-11.

(e) Special (Hypothetical) Events

Special Events are postulated to demonstrate some special capability of the plant in accordance with NRC requirements. For analyzed events within this classification, see Table 15A-12.

(2) Safety and Power Generation Aspects

Matters identified with “safety” classification are governed by regulatory requirements. Safety functions include:

(a) The accommodation of moderate frequency incidents, infrequent incidents and limiting faults

(b) The maintenance of containment integrity

(c) The assurance of ECCS

(d) The continuance of reactor coolant pressure boundary (RCPB) integrity

Safety classified aspects are related to 10CFR100 dose limits, infrequent and low probability occurrences, SACF criteria, worst-case operating conditions and initials assumptions, automatic (30-min.) corrective actions, significant unacceptable dose and environmental effects, and the involvement of other coincident (mechanistic or non-mechanistic) plant and environmental situations.

Power generation classified considerations are related to continued plant power generation operation, equipment operational matters, component availability aspects and to long-term offsite public effects.

Some matters identified with “power generation” classification are also covered by regulatory guidelines. Power generation functions include:

(a) Accommodation of planned operations and moderate frequency incidents

(b) Minimization of radiological releases to appropriate levels

(c) Assurance of safe and orderly reactor shutdown, and/or return to power generation operations

(d) Continuance of plant equipment design conditions to ensure long-term reliable operation

Power generation is related to 10CFR20 and 10CFR50 Appendix I dose limits.

(3) Frequency of Events

Consideration of the frequency of the initial (or initiating) event is straightforward. Added considerations (e.g., further failures or operator errors) certainly influence the classification grouping. The events in this appendix are initially grouped per initiating frequency occurrence. The imposition of further failures necessitates further classification to a lower frequency category.

The introduction of SACF or single operator error (SOE) into the examination of planned operation, moderate frequency incidents or infrequent incidents evaluations has not been previously considered a design basis or evaluation prerequisite. It is provided and included here to demonstrate the plant's capability to accommodate the requirement.

(4) Conservative Analysis—Margins

The unacceptable consequences established in this appendix relative to the public health and safety are, in themselves, in strict and conservative conformance to regulatory requirements.

(5) Safety Function Definition

First, the essential protective sequences shown for an event in this appendix list the minimum structures and systems required to be available to satisfy the SACF or SOE evaluation aspects of the event. Other protective “success paths” exist in some cases than are shown with the event.

Second, not all the events involve the same natural, environmental or plant conditional assumptions. For example, loss-of-coolant accident (LOCA) and safe shutdown earthquake (SSE) mechanical loads are associated with Event 32. In Event 29, the control rod drop accident (CRDA) is not assumed to be associated with any SSE or operating basis earthquake (OBE) occurrence. Therefore, seismic safety function requirements are not considered for Event 29. Some of the safety function equipment associated with the Event 29 protective sequence are also capable of handling more limiting events, such as Event 32.

Third, containment may have a safety function for some event when uncontained radiological release would be unacceptable, but for other events it may not be applicable (e.g., during refueling). The requirement to maintain the containment in post-accident recovery is only needed to limit doses to less than 10CRF100. After radiological sources are depleted with time, further containment is unnecessary. Thus, the “time domain” and “need for” aspects of a function are taken into account and considered when evaluating the events in this appendix.

Fourth, the operation of engineered safety features (ESF) equipment, for normal operational events, should not be misunderstood to mean that ESF equipment requirements apply to this event category.

Likewise, the interpretation of the use of ESF-SACF capable systems for moderate frequency incidents protective sequences should not imply that these equipment requirements (seismic, redundancy, diversity, testable, IEEE, etc.) are required for moderate frequency incidents.

(6) Envelope and Actual Event Analysis

The event analysis presented in Chapter 15 does not include event frequency considerations, but does present an “envelope analysis” evaluation based on expected situations. Studies of the actual plant occurrences, their frequency and their actual impact are reflected in their categorization in this appendix. This places the plant safety evaluations and their impact into a better perspective by focusing attention on the “envelope analysis” with more appropriate understanding.

15A.2.2.1 Consistency of the Analysis

Figure 15A-2 illustrates three inconsistencies. Panel A shows the possible inconsistency resulting from operational requirements being placed on separate levels of protection for one event. If the second and sixth levels of protection are important enough to warrant operational requirements, then so are the third, fourth and fifth levels. Panel B shows the possible inconsistency resulting from operational requirements being arbitrarily placed on some action thought to be important to safety. In the case shown, scram represents different protection levels for two similar events in one category: if the fourth level of protection for Event B is important enough to warrant an operational requirement, then so is the fourth level for Event A. Thus, to simply place operational requirements on all equipment needed for some action (scram, isolation, etc.) could be inconsistent and unreasonable if different protection levels are represented. Panel C shows the possible inconsistency resulting from operational requirements being placed on some arbitrary level of protection for any and all postulated events. Here the inconsistency is not recognizing and accounting for different event categories based on cause or expected frequency of occurrence.

Inconsistencies of the types illustrated in Figure 15A-2 are avoided in the NSOA by directing the analysis to “event consequences” oriented aspects. Analytical inconsistencies are avoided by (1) treating all the events of a category under the same set of functional rules, (2) applying another set of functional rules to another category, and (3) having a consistent set of rules between categories. Thus, it is valid to compare the results of the analyses of the events in any one category and invalid to compare events of different categories, and thus different rules, to the other category. An example of this is the different rules (limits, assumptions, etc.) of accidents compared to the applicable anticipated transients.

15A.2.3 Comprehensiveness of the Analysis

The analysis must be sufficiently comprehensive in method that (1) all plant hardware is considered and (2) the full range of plant operating conditions is considered. The tendency to be preoccupied with “worst cases” (those that appear to give the most severe consequences) is recognized; however, the protection sequences essential to lesser cases may be different (more or less restrictive) from the worst-case sequence. To assure that operational and design basis requirements are defined and appropriate for all equipment essential to attaining acceptable consequences, all essential protection sequences must be identified for each of the plant safety events examinations. Only in this way is a comprehensive level of safety attained. Thus, the NSOA is also “protection sequence” oriented to achieve comprehensiveness.

15A.2.4 Systematic Approach of the Analysis

In summary, the systematic method utilized in this analysis contributes to both the consistency and comprehensiveness of the analysis. The desired characteristics representative of a systematic approach to selecting BWR operational requirements are as follows:

(1) Specify measures of safety-unacceptable consequences

(2) Consider all normal operations

(3) Systematic event selection

(4) Common treatment analysis of all events of any one type

(5) Systematic identification of plant actions and systems essential to avoiding unacceptable consequences

(6) Emergency operational requirements and limits from system analysis

Figure 15A-1 illustrates the systematic process by which the operational and design basis nuclear safety requirements and technical specifications are derived. The process involves the evaluation of carefully selected plant events relative to the unacceptable consequences (specified measures of safety). Those limits, actions, systems and components found to be essential to achieving acceptable consequences are the subjects of operational requirements.

15A.2.5 Relationship of Nuclear Safety Operational Analysis to Safety Analyses of Chapter 15

One of the main objectives of the operational analysis is to identify all essential protection sequences and to establish the detailed equipment conditions essential to satisfying the nuclear safety operational criteria. The spectrum of events examined in Chapter 15 represents a complete set of plant safety considerations. The main objective of the earlier analyses of Chapter 15, is, of course, to provide detailed worst-case (limiting or envelope) analysis of the plant events. The worst cases are correspondingly analyzed and treated likewise in this appendix, but in light of frequency of occurrence, unacceptable consequences and assumption categories.

The detailed discussion relative to each of the events covered in Chapter 15 is not repeated in this appendix. Refer to the specific section in Chapter 15 as cross- correlated in Tables 15A-8 through 15A-12. These tables provide cross-correlation between the NSOA event, its protection sequence diagram and its safety evaluation in Chapter 15.

15A.2.6 Relationship Between NSOA and Operational Requirements, Technical Specifications, Design Basis, and SACF Aspects

By definition, “an operational requirement” is a requirement or restriction (limit) on either the value of a plant variable or the operability condition associated with a plant system. Such requirements must be observed during all modes of plant operation (not just at full power) to assure that the plant is operated safely (to avoid the unacceptable results). There are two kinds of operational requirements for plant hardware:

(1) Limiting Condition for Operation: the required condition for a system while the reactor is operating in a specified state

(2) Surveillance Requirements: the nature and frequency of tests required to assure that the system is capable of performing its essential functions

Operational requirements are systematically selected for one of two basic reasons:

(1) To assure that unacceptable consequences are mitigated following specified plant events by examining and challenging the system design

(2) To assure the consequences of a transient or accident is acceptable with the existence of a SACF or SOE criteria

The individual structures and systems which perform a safety function are required to do so under design basis conditions, including environmental consideration and under single active component failure assumptions. The NSOA confirms the previous examination of the individual equipment (see the “Evaluations” subsection) requirement conformance analyses.

15A.2.7 Unacceptable Consequences Criteria

Tables 15A-1 through 15A-5 identify the unacceptable consequences and capability considerations associated with different event categories. To prevent or mitigate them, they are recognized as the major bases for identifying system operational requirements as well as the bases for all other safety analyses versus criteria throughout Tier 2.

15A.2.8 General Nuclear Safety Operational Criteria

The nuclear safety operational criteria used to select operational requirements are described in Table 15A-6.

The unacceptable consequences associated with the different categories of plant operations and events are dictated by:

(1) Probability of occurrence

(2) Allowable limits (per the probability) related to radiological, structural, environmental, etc., aspects

(3) Coincidence of other related or unrelated disturbances

(4) Time domain of event and consequences consideration

15A.3 Method of Analysis

15A.3.1 General Approach

The NSOA is performed on the plant as designed. The end products of the analysis are the nuclear safety operational requirements and the restrictions on plant hardware and its operation that must be observed (1) to satisfy the nuclear safety operational criteria and (2) to show compliance of the plant safety and power generation systems with plant wide requirements. Figure 15A-2 shows the process used in the analysis. The following inputs are required for the analysis of specific plant events:

(1) Unacceptable Consequences Criteria (Subsection 15A.2.7)

(2) General Nuclear Safety Operational Criteria (Subsection 15A.2.8)

(3) BWR Operating States (Subsection 15A.3.2)

(4) Selection of Events for Analysis (Subsection 15A.3.3)

(5) Guidelines for Event Analysis (Subsection 15A.3.5)

With this information, each selected event can be evaluated to systematically determine the actions, systems and limits essential to avoiding the defined unacceptable consequences. The essential plant components and limits so identified are then considered to be in agreement with and subject to nuclear operational, design basis requirements and technical specification restrictions.

15A.3.2 BWR Operating States

Four BWR operating states in which the reactor can exist are defined in Subsection 15A.6.2.4 and summarized in Table 15A-7. The main objective in selecting operating states is to divide the BWR operating spectrum into sets of initial conditions to facilitate consideration of various events in each state.

Each operating state includes a wide spectrum of values for important plant parameters. Within each state, these parameters are considered over their entire range to determine the limits on their values necessary to satisfy the nuclear safety operational criteria. The plant parameters to be considered in this manner include the following:

(1) Reactor coolant temperature

(2) Reactor vessel water level

(3) Reactor vessel pressure

(4) Reactor vessel water quality

(5) Reactor coolant forced circulation flow rate

(6) Reactor power level (thermal and neutron flux)

(7) Core neutron flux distribution

(8) Feedwater temperature

(9) Containment temperature and pressure

(10) Suppression pool water temperature and level

(11) Spent fuel pool water temperature and level

15A.3.3 Selection of Events for Analysis

15A.3.3.1 Normal Operation

Operations subsequent to an incident (transient, accident or additional plant capability event) are not considered planned operations until the actions taken or equipment used in the plant are identical to those that would be used had the incident not occurred. As defined, the planned operations can be considered as a chronological sequence: refueling outage --> achieving criticality --> heatup --> power operation --> achieving shutdown --> cooldown --> refueling outage.

The normal operations are defined below.

(1) Refueling Outage: Includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is taken critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

(a) Planned, physical movement of core components (fuel, control rods, etc.)

(b) Refueling test operations (except criticality and shutdown margin tests)

(c) Planned maintenance

(d) Required inspection

(2) Achieving Criticality: Includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

(3) Heatup: Begins when achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine-generator.

(4) Power Operation: Begins when heatup ends and includes continued plant operation at power levels in excess of heatup power.

(5) Achieving Shutdown: Begins when the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) following power operation.

(6) Cooldown: Begins when achieving nuclear shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of RPV temperature and pressure.

The exact point at which some of the planned operations end and others begin cannot be precisely determined. It will be shown later that such precision is not required, for the protection requirements are adequately defined in passing from one state to the next. Dependence of several planned operations on the one rod subcritical condition provides an exact point on either side of which protection (especially scram) requirements differ. Thus, where a precise boundary between planned operations is needed, the definitions provide the needed precision.

Together, the BWR operating states and the planned operations define the full spectrum of conditions from which transients, accidents and special events are initiated. The BWR operating states define only the physical condition (pressure, temperature, etc.) of the reactor; the planned operations define what the plant is doing. The separation of physical conditions from the operation being performed is deliberate and facilitates careful consideration of all possible initial conditions from which incidents may occur.

15A.3.3.2 Moderate Frequency Incidents (Anticipated Operational Transients)

To select moderate frequency incidents (anticipated operational transients), eight nuclear system parameter variations are considered as potential initiating causes of threats to the fuel and the reactor coolant pressure boundary. The parameter variations are as follows:

(1) Reactor pressure vessel pressure increase

(2) Reactor pressure vessel water (moderator) temperature decrease

(3) Control rod withdrawal

(4) Reactor pressure vessel coolant inventory decrease

(5) Rector core coolant flow decrease

(6) Reactor core coolant flow increase

(7) Core coolant temperature increase

(8) Excess of coolant inventory

These parameter variations, if uncontrolled, could result in damage to the reactor fuel or reactor coolant pressure boundary, or both. A nuclear system pressure increase threatens to rupture the reactor coolant pressure boundary from internal pressure. A pressure increase also collapses voids in the moderator, causing an insertion of positive reactivity that threatens fuel damage as a result of overheating. A reactor vessel water (moderator) temperature decrease results in an insertion of positive reactivity as density increases. This could lead to fuel overheating. Positive reactivity insertions are possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten fuel damage caused by overheating. Both a reactor vessel coolant inventory decrease and a reduction in coolant flow through the core threatens the integrity of the fuel as the coolant becomes unable to adequately remove the heat generated in the core. An increase in coolant flow through the core reduces the void content of the moderator and results in an insertion of positive reactivity. Core coolant temperature increase threatens the integrity of the fuel; such a variation could be the result of a heat exchanger malfunction during operation in the shutdown cooling mode. An excess of coolant inventory could be the result of malfunctioning water level control equipment; such a malfunction can result in a turbine trip, which causes an expected increase in nuclear system pressure and power.

Moderate frequency incidents (anticipated operational transients) are defined as transients resulting from a single active component failure (SACF) or single operator error (SOE) that can be reasonably expected (moderate probability of occurrence once per year to once in 20 years) during any mode of plant operation. Examples of single operation failures or operator errors in this range of probability are:

(1) Opening or closing any single valve (a check valve is not assumed to close against normal flow)

(2) Starting or stopping any single component

(3) Malfunction or maloperation of any single control device

(4) Any single electrical failure

(5) Any single operator error

An operator error is defined as an active deviation from nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single reasonably expected erroneous decision. The set of actions is limited as follows:

(1) Those actions that could be performed by only one person

(2) Those actions that would have constituted a correct procedure had the initial decision been correct

(3) Those actions that are subsequent to the initial operator error and that affect the designed operation of the plant, but are not necessarily directly related to the operator error

The various types of a single operator error or a single active component failure are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the radioactive material barriers.

15A.3.3.3 Infrequent Incidents (Abnormal Operational Transients)

To select infrequent incidents, eight nuclear system parameter variations are considered as potential initiating causes of gross core-wide fuel failures and threats of the reactor coolant pressure boundary. The parameter variations are as follows:

(1) Reactor pressure vessel pressure increase

(2) Reactor pressure vessel water (moderator) temperature decrease

(3) Control rod withdrawal

(4) Reactor vessel coolant inventory decrease

(5) Reactor core coolant flow decrease

(6) Reactor core coolant flow increase

(7) Core coolant temperature increase

(8) Excess of coolant inventory

The eight parameter variations listed above include all effects within the nuclear system caused by abnormal operational transients that threaten gross core-wide reactor fuel integrity or seriously affect reactor coolant pressure boundary. Variation of any one parameter may cause a change in another listed parameter; however, for analysis purposes, threats to barrier integrity are evaluated by groups according to the parameter variation originating the threat.

Infrequent incidents (abnormal operational transient) are defined as incidents resulting from single or multiple equipment failure and/or single or multiple operator errors that are not reasonably expected (less that one event in 20 years to one in 100 years) during any mode of plant operation. Examples of single or multiple operational failure and/or single or multiple operator errors are:

(1) Failure of major power generation equipment components

(2) Multiple electrical failures

(3) Multiple operator errors

(4) Combinations of equipment failure and an operator error

Operator error is defined as an active deviation from nuclear plant standard operating practices. A multiple operator error is the set of actions that is a direct consequence of several unexpected erroneous decisions.

The various types of a single errors and/or single malfunctions are applied to various plant systems with a consideration for a variety of plant conditions to discover events directly resulting in an undesired parameter variation. Once discovered, each event is evaluated for the threat it poses to the integrity of the various radioactive material barriers.

15A.3.3.4 Limiting Faults (Design Basis Accidents)

Limiting faults (accidents) are defined as hypothesized events that affect the radioactive material barriers and are not expected during plant operations. These are plant events, equipment failures, combinations of initial conditions which are of extremely low probability (once in 100 years or longer). The postulated accident types considered are as follows:

(1) Mechanical failure of a single component leading to the release of radioactive materials from one or more barriers. The components referred to here are not those

that act as radioactive material barriers. Examples of mechanical failure are breakage of the coupling between a control rod drive and the control rod.

(2) Arbitrary rupture of any single pipe up to and including complete severance of the largest pipe in the reactor coolant pressure boundary. This kind of accident is considered only under conditions in which the nuclear system is pressurized.

For purposes of analysis, accidents are categorized as those events that result in releasing radioactive material:

(1) From the fuel with the reactor coolant pressure boundary, Reactor Building initially intact

(2) Directly to the containment

(3) Directly to the Reactor or Turbine Buildings with the containment initially intact

(4) Directly to the Reactor Building with the containment not intact

(5) Directly to the spent fuel containing facilities within the Reactor Building

(6) Directly to the Turbine Building

(7) Directly to the environs

The effects of various accident types are investigated, with consideration for the full spectrum of plant conditions, to examine events that result in the release of radioactive material.

15A.3.3.5 Special Events

A number of additional events are evaluated to demonstrate plant capabilities relative to special arbitrary nuclear safety criteria. These special events involve extremely low probability occurrence situations. As an example, the adequacy to the redundant reactivity control system is demonstrated by evaluating the special event: “reactor shutdown without control rods.” A similar example, the capability to perform a safe shutdown from outside the main control room, is demonstrated by evaluating the special event: “reactor shutdown from outside the main control room.”

15A.3.4 Applicability of Events to Operating States

The first step in performing an operational analysis for a given “incident” (transient, accident or special event) is to determine in which operating states the incident can occur. An incident is considered applicable within an operating state if the incident can be initiated from the physical conditions that characterize the operating state. Applicability of the “normal operations” to the operating states follows from the definitions of planned operations. A planned operation is considered applicable within an operating state if the planned operation can be conducted when the reactor exists under the physical conditions defining the operating state.

15A.3.5 Guidelines for Event Analysis

The following functional guidelines are followed in performing SACF, operational design basis analyses for the various plant events:

(1) An action, system, or limit shall be considered essential only if it is essential to avoiding an unacceptable result or satisfying the nuclear safety operational criteria.

(2) The full range of initial conditions (as defined in Subsection 15A.3.5(3)) shall be considered for each event analyzed so that all essential protection sequences are identified. Consideration is not limited to worst cases because lesser cases sometimes may require more restrictive actions or systems different from the worst cases.

(3) The initial conditions for transients, accidents and additional plant capability events shall be limited to conditions that would exist during planned operations in the applicable operating state.

(4) For normal operations, consideration shall be made only for actions, limits, and systems essential to avoiding the unacceptable consequences during operation in that state (as opposed to transients, accidents and additional plant capability events, which are followed through to completion). Normal operations are treated differently from other events because the transfer from one state to another during planned operations is deliberate. For events other than normal operations, the transfer from one state to another may be unavoidable.

(5) Limits shall be derived only for those essential parameters that are continuously monitored by the operator. Parameter limits associated with the required performance of an essential system are considered to be included in the requirement for the operability of the system. Limits on frequently monitored process parameters are called “envelope limits,” and limits on parameters associated with the operability of a safety system are called “operability limits.” Systems associated with the control of the envelope parameters are considered nonessential if it is possible to place the plant in a safe condition without using the system in question.

(6) For transients, accidents and special events, consideration shall be made for the entire duration of the event and aftermath until some planned operation is resumed.

Normal operation is considered resumed when the procedures being followed or equipment being used are identical to those used during any one of the defined planned operations. Where “Extended Core Cooling” is an immediate integral part of the event, it will be included in the protection sequence. Where it may be an eventual part of the event, it will not be directly added but, of course, can be implied to be available.

(7) Credit for operator action shall be taken on a case-by-case basis, depending on the conditions that would exist at the time operator action would be required. Because transients, accidents and special events are considered through the entire duration of the event until normal operation is resumed, manual operation of certain systems is sometimes required following the more rapid or automatic portions of the event. Credit for operator action is taken only when the operator can reasonably be expected to accomplish the required action under the existing conditions.

(8) For transients, accidents and special events, only those actions, limits and systems shall be considered essential for which there arises a unique requirement as a result of the event. For instance, if a system that was operating prior to the event (during planned operation) is to be employed in the same manner following the event, and if the event did not affect the operation of the system, then the system would not appear on the protection sequence diagram.

(9) The operational analyses shall identify all the support of auxiliary systems essential to the functioning of the frontline safety systems. Safety system auxiliaries whose failure results in safe failure of the frontline safety systems shall be considered nonessential.

(10) A system or action that plays a unique role in the response to a transient, accident or special event shall be considered essential unless the effects of the system or action are not included in the detailed analysis of the event.

15A.3.6 Steps in an Operational Analysis

All information needed to perform an operational analysis for each plant event has been presented (Figure 15A-1). The procedure for performing an operational analysis for a given event (selected according to the event selection criteria) is as follows:

(1) Determine the BWR operating states in which the event is applicable.

(2) Identify all the essential protection sequences (safety actions and frontline safety systems) for the event in each applicable operating state.

(3) Identify all the safety system auxiliaries essential to the functioning of the frontline safety systems.

These three steps are performed in Section 15A.6.

To derive the operational requirements and technical specifications for the individual components of a system included in any essential protection sequence, the following steps are taken:

(1) Identify all the essential actions within the system (intrasystem actions) necessary for the system to function to the degree necessary to avoid the unacceptable consequences.

(2) Identify the minimum hardware conditions necessary for the system to accomplish the minimum intrasystem actions.

(3) If the single-failure criterion applies, identify the additional hardware conditions necessary to achieve the plant safety actions (e.g., scram, pressure relief, isolation, cooling) in spite of single failures. This step gives the nuclear safety operational requirements for the plant components so identified.

(4) Identify surveillance requirements and allowable repair times for the essential plant hardware (Subsection 15A.5.2).

(5) Simplify the operational requirements determined in steps (3) and (4) so that a technical specification may be obtained that encompass the true operational requirements and are easily used by plant operations and management personnel.

15A.4 Display of Operational Analysis Results

15A.4.1 General

To fully identify and establish the requirements, restrictions and limitations that must be observed during plant operation, plant systems and components must be related to the needs for their actions in satisfying the nuclear safety operational criteria. This section displays these relationships in a series of block diagrams.

Tables 15A-7 and 15A-8 through 15A-12 indicate which operating states each event is applicable. For each event, a block diagram is presented showing the conditions and systems required to achieve each essential safety action. The block diagrams show only those systems necessary to provide the safety actions such that the nuclear safety operational and design basis criteria are satisfied. The total plant capability to provide a safety action is generally not shown, only the minimum capability essential to satisfying the operational criteria. It is very important to understand that only enough protective equipment is cited in the diagram to provide the necessary action. Many events can utilize many more paths to success than are shown. These operational analyses involve the minimum equipment needed to prevent or avert an unacceptable consequence. Thus, the diagrams depict all essential protection sequences for each event with the least amount of protective equipment needed. Once all of these protection sequences are identified in block diagram form, system requirements are derived by considering all events in which the particular system is employed. The analysis considers the following conceptual aspects:

(1) The BWR operating state

(2) Types of operations or events that are possible within the operating state

(3) Relationships of certain safety actions to the unacceptable consequences and to specific types of operations and events

(4) Relationships of certain systems to safety actions and to specific types of operations and events

(5) Supporting or auxiliary systems essential to the operation of the frontline safety systems

(6) Functional redundancy (the single-failure criterion applied at the safety action level; this is, in effect, a qualitative, system-level, FMEA-type analysis)

Each block in the sequence diagrams represents a finding of essentiality for the safety action, system or limit under consideration. Essentiality in this context means that the safety action, system or limit is needed to satisfy the nuclear safety operational criteria. Essentiality is determined through an analysis in which the safety action, system or limit being considered is completely disregarded in the analyses of the applicable operations or events. If the nuclear safety operational criteria are satisfied without the safety action, system or limit, then the safety action, system or limit is not essential, and no operational nuclear safety requirement would be indicated. When disregarding a safety action, system or limit results in violating one or more nuclear safety operational criteria, the safety action, system or limit is considered essential, and the resulting operational nuclear safety requirements can be related to specific criteria and unacceptable consequences.

15A.4.2 Protection Sequence and Safety System Auxiliary Diagrams

Block diagrams illustrate essential protection sequences for each event requiring unique safety actions. These protection sequence diagrams show only the required frontline safety systems. The format and conventions used for these diagrams are shown in Figure 15A-3.

The auxiliary systems essential to the correct functioning of frontline safety systems are shown on safety system auxiliary diagrams. The format used for these diagrams is shown in Figure 15A-4. The diagram indicates that auxiliary systems A, B, and C are required for proper operations of frontline safety system X.

Total plant requirements for an auxiliary system or the relationships of a particular auxiliary system to all other safety systems (frontline and auxiliary) within an operating state are shown on the commonality of auxiliary diagrams. The format used for these diagrams is shown in

Figure 15A-5. The convention employed in Figure 15A-5 indicates that auxiliary system A is required:

(1) To be single-failure proof relative to system γ in State A-events X, Y; State B-events X, Y; State C-events X, Y, Z; State D-events X, Y, Z.

(2) To be single-failure proof relative to the parallel combination of systems α and β in State A-events U, V, W; State B-events V, W; State C-events U, V, W, X; State D-events U, V, W, X.

(3) To be single-failure proof relative to the parallel combination of system ¼ and ε in series with the parallel combination of systems Epsilon and Chi in State C-events Y, W; State D-events Y, W, Z. As noted, system ε is part of the combination but does not require auxiliary system A for its proper operation.

(4) For system δ in State B-events Q, R; State D-events Q, R, S.

With these three types of diagrams, it is possible to determine for each system the detailed functional requirements and conditions to be observed regarding system hardware in each operating state. The detailed conditions to be observed regarding system hardware include such nuclear safety operational requirements as test frequencies and the number of components that must be operable.

15A.5 Bases for Selecting Surveillance Test Frequencies and Allowable Outage Times

15A.5.1 Normal Surveillance Test Frequencies

After the essential nuclear safety systems and engineered safeguards have been identified by applying the nuclear safety operational criteria, surveillance requirements are selected for these systems. In this selection process, the various systems are considered in terms of relative availability, test capability, plant conditions necessary for testing and engineering experience with the system type. Surveillance test frequencies are determined using models developed in the Probabilistic Risk Assessment (PRA).

15A.5.2 Allowable Outage Times

Allowable outage times are selected by computation using models developed in the PRA. The resulting maximum average allowable outage times assure that a system's long-term availability, including allowance for repair and test, is not reduced below a specified availability.

15A.5.3 Outage Time Rule

A safety system can be repaired or tested while the reactor is in operation if the repair and test time is equal to or less than the maximum allowable average outage time. If repair or test is not complete when the allowable outage time expires, the plant must be placed in its safest mode (with respect to the protection lost) in accordance with the Technical Specifications.

To maintain the validity of the assumptions used to establish the previously noted repair rule, the following restrictions must be observed:

(1) The allowable outage time is only used as needed to restore failed equipment to operation or to perform required surveillance tests, not for routine maintenance. Routine maintenance should be scheduled when the equipment is not needed.

(2) At the conclusion of the repair, the repaired component must be retested and placed in service.

(3) Once the need for repair of a failed component is discovered, repairs should proceed as quickly as possible consistent with good craftsmanship.

15A.6 Operational Analyses

Results of the operational analyses are discussed in the following paragraphs and displayed on Figures 15A-6 through 15A-70 and in Tables 15A-8 through 15A-12.

15A.6.1 Safety System Auxiliaries

Figures 15A-6 and 15A-7 show the safety system auxiliaries essential to the functioning of each frontline safety system. Commonality of auxiliary diagrams are shown in Figures 15A-65 through 15A-70.

15A.6.2 Normal Operations

15A.6.2.1 General

Requirements for the normal or planned operations normally involve limits (L) on certain key process variables and restrictions (R) on certain plant equipment. The control block diagrams for each operating state (Figures 15A-8 through 15A-11) show only those controls necessary to avoid unacceptable safety consequences (1-1 through 1-4 of Table 15A-1). Table 15A-8 summarizes additional information for Normal Operation.

Following is a description of the planned operations (Events 1 through 6) as they pertain to each of the four operating states. The description of each operating state contains a definition of that state, a list of the planned operations that apply to that state and a list of the safety actions that are required to avoid the unacceptable safety consequences.

15A.6.2.2 Event Definitions

Event 1—Refueling Outage

Refueling outage includes all the planned operations associated with a normal refueling outage except those tests in which the reactor is made critical and returned to the shutdown condition. The following planned operations are included in refueling outage:

(1) Planned, physical movement of core components (e.g., fuel, control rods, etc.)

(2) Refueling test operations (except for the criticality and the shutdown margin tests)

(3) Planned maintenance

(4) Required inspections

Event 2—Achieving Criticality

Achieving criticality includes all the plant actions normally accomplished in bringing the plant from a condition in which all control rods are fully inserted to a condition in which nuclear criticality is achieved and maintained.

Event 3—Reactor Heatup

Heatup begins where achieving criticality ends and includes all plant actions normally accomplished in approaching nuclear system rated temperature and pressure by using nuclear power (reactor critical). Heatup extends through warmup and synchronization of the main turbine generator.

Event 4—Power Operation—Electric Generation

Power operation begins where heatup ends and continued plant operation at power levels in excess of heatup power or steady-state operation. It also includes plant maneuvers such as:

(1) Daily electrical load reduction and recoveries

(2) Electrical grid frequency control adjustment

(3) Control rod movements

(4) Power generation surveillance testing involving:

  (a) Turbine stop valve closing

  (b) Turbine control valve adjustments

  (c) Main Steam Isolating Valve (MSIV) exercising

Event 5—Achieving Reactor Shutdown

Achieving shutdown begins where the main generator is unloaded and includes all plant actions normally accomplished in achieving nuclear shutdown (more than one rod subcritical) after power operation.

Event 6—Reactor Cooldown

Cooldown begins where achieving shutdown ends and includes all plant actions normal to the continued removal of decay heat and the reduction of nuclear system temperature and pressure.

15A.6.2.3 Required Safety Actions/Related Unacceptable Consequences

The following paragraphs describe the safety actions for planned operations. Each description includes a selection of the operating states that apply to the safety action, the plant system affected by limits or restrictions and the unacceptable consequence that is avoided. The four operating states are defined in Table 15A-7. The unacceptable consequences criteria are tabulated in Table 15A-1.

15A.6.2.3.1 Radioactive Material Release Control

Radioactive materials may be released to the environs in any operating state; therefore, radioactive material release control is required in all operating states. Because of the significance of preventing excessive release of radioactive materials to the environs, this is the only safety action for which monitoring systems are explicitly shown. The offgas vent radiation monitoring system provides indication for gaseous release through the main vent. Gaseous releases through other vents are monitored by the ventilation monitoring system. The process liquid radiation monitors are not required because all liquid wastes are monitored by batch sampling before a controlled release. Limits are expressed on the offgas vent system, liquid radwaste system and solid radwaste system so that the planned release of radioactive materials comply with the limits given in 10CFR20, 10CFR50, and 10CFR71 (related unacceptable safety result 1-1 Table 15A-1).

15A.6.2.3.2 Core Coolant Flow Rate Control

In State D, when above approximately 10% Nuclear Boiler (NB) rated power, the core coolant flow rate must be maintained above certain minimums (i.e., limited) to maintain the integrity of the fuel cladding (1-2) and assure the validity of the plant safety analysis (1-4).

15A.6.2.3.3 Core Power Level Control

The plant safety analyses of accidental positive reactivity additions have assumed as an initial condition that the neutron source level is above a specified minimum. Because a significant positive reactivity addition can only occur when the reactor is less than one rod subcritical, the assumed minimum source level need be observed only in States B and D. The minimum source level assumed in the analyses has been related to the counts/s readings on the startup range neutron monitors (SRNM); thus, this minimum power level limit on the fuel is expressed as a required SRNM count level. Observing the limit assures validity of the plant safety analysis (1-4). Maximum core power limits are also expressed for operating States B and D to maintain fuel integrity (1-2) and remain below the maximum power levels assumed in the plant safety analysis (1-4).

15A.6.2.3.4 Core Neutron Flux Distribution Control

Core neutron flux distribution must be limited in State D; otherwise, core power peaking could result in fuel failure (1-2). Thermal limits are applied in this state, because the core neutron flux distribution must be maintained within the envelope of conditions considered by plant safety analysis (1-4).

15A.6.2.3.5 Reactor Vessel Water Level Control

In any operating state, the reactor vessel water level could, unless controlled, drop to a level that will not provide adequate core cooling; therefore, reactor vessel water level control applies to all operating states. Observation of the reactor vessel water level limits protects against fuel failure (1-2) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.6 Reactor Vessel Pressure Control

Reactor vessel pressure control is not needed in states A and B because vessel pressure cannot be increased above atmospheric pressure. In State C, a limit is expressed on the reactor vessel to assure that it is not hydrostatically tested until the temperature is above the NDT temperature plus 33.3°C; this prevents excessive stress (1-3). Also, in States C and D a limit is expressed on the Residual Heat Removal (RHR) System to assure that it is not operated in the shutdown cooling mode when the reactor vessel pressure is greater than approximately 0.689 MPaG (0.932 MPaG limit); this prevents excessive stress (1-3). In States C and D, a limit on the reactor vessel pressure is necessitated by the plant safety analysis (1-4).

15A.6.2.3.7 Nuclear System Temperature Control

In operating States C and D, a limit is expressed on the reactor vessel to prevent the reactor vessel head bolting studs from being in tension when the temperature is less than 21°C to avoid excessive stress (1-3) on the reactor vessel flange. This limit does not apply in States A and B because the head will not be bolted in place during criticality tests or during refueling. In all operating states, a limit is expressed on the reactor vessel to prevent an excessive rate of change of the reactor vessel temperature to avoid excessive stress (1-3). In States C and D, where it is planned operation to use the Feedwater System, a limit is placed on the reactor fuel so that the feedwater temperature is maintained within the envelope of conditions considered by the plant safety analysis (1-4). For State D, a limit is observed on the temperature difference between the bottom head drain and the reactor vessel saturation to prevent the starting of the reactor internal pumps. This operating restriction and limit prevents excessive stress in the reactor vessel (1-3).

15A.6.2.3.8 Nuclear System Water Quality Control

In all operating states, water of improper chemical quality could produce excessive stress as a result of chemical corrosion (1-3). Therefore, a limit is placed on reactor coolant chemical quality in all operating states. For all operating states where the nuclear system can be pressurized (States C and D), an additional limit on reactor coolant activity assures the validity of the analysis of the main steamline break accident.

15A.6.2.3.9 Nuclear System Leakage Control

Because excessive nuclear system leakage could occur only while the reactor vessel is pressurized, limits are applied only to the reactor vessel in States C and D. Observing these limits prevents vessel damage due to excessive stress (1-3) and assures the validity of the plant safety analysis (1-4).

15A.6.2.3.10 Core Reactivity Control

In State A during refueling outage, a limit on core loading (fuel) to assure that core reactivity is maintained within the envelope of conditions considered by the plant safety analysis (1-4). In all states, limits are imposed on the Control Rod Drive (CRD) System to assure adequate control of core reactivity so that core reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4).

15A.6.2.3.11 Control Rod Worth Control

Any time the reactor is not shut down and is generating less than 20% power (State D), a limit is imposed on the control rod pattern to assure that control rod worth is maintained within the envelope of conditions considered by the analysis of the control rod drop accident (1-4).

15A.6.2.3.12 Refueling Restriction

By definition, planned operation event 1 (refueling outage) applies only to State A. Observing the restrictions on the reactor fuel and on the operation of the CRD System within the specified limit maintains plant conditions within the envelope considered by the plant safety analysis (1-4).

15A.6.2.3.13 Containment and Reactor Building Pressure and Temperature Control

In States C and D, limits are imposed on the suppression pool temperature to maintain containment pressure within the envelope considered by plant safety analysis (1-4). These limits assure an environment in which instruments and equipment can operate correctly within the containment. Limits on the pressure suppression pool apply to the water temperature and water level to assure that it has the capability of absorbing the energy discharged during a safety/relief valve blowdown.

15A.6.2.3.14 Stored Fuel Shielding, Cooling and Reactivity Control

Because both new and spent fuel will be stored during all operating states, stored fuel shielding, cooling and reactivity control apply to all operating states. Limits are imposed on the spent fuel pool storage positions, water level, fuel-handling procedures and water temperature. Observing the limits on fuel storage positions assures that spent fuel reactivity remains within the envelope of conditions considered by the plant safety analysis (1-4). Observing the limits on water level assures shielding in order to maintain conditions within the envelope of conditions considered by the plant safety analysis
(1-4) and provides the fuel cooling necessary to avoid fuel damage (1-2). Observing the limit on water temperature avoids excessive fuel pool stress (1-3).

15A.6.2.4 Operational Safety Evaluations

State A

In State A, the reactor is in a shutdown condition, the vessel head is off and the vessel is at atmospheric pressure. The applicable events for planned operations are refueling outage, achieving criticality, and cooldown (Events 1, 2, and 6, respectively).

Figure 15A-8 shows the necessary safety actions for planned operations, the corresponding plant systems and the event for which these actions are necessary. As indicated in the diagram, the required safety actions are as follows:

- Safety Action
  - Radioactive material release control
  - Reactor vessel water level control
  - Nuclear system temperature control
  - Nuclear system water quality control
  - Core reactivity control
  - Refueling restrictions
  - Stored fuel shielding, cooling and reactivity control

State B

In State B, the reactor vessel head is off, the reactor is not shutdown and the vessel is at atmospheric pressure. Applicable planned operations are achieving criticality and achieving shutdown (Events 2 and 5, respectively).

Figure 15A-9 presents the necessary safety actions for planned operations, the plant systems and the event for which the safety actions are necessary. The required safety actions for planned operations in State B are as follows:

- Safety Actions
  - Radioactive material release control
  - Core power level control
  - Reactor vessel water level control
  - Nuclear system temperature control
  - Nuclear system water quality control
  - Core reactivity control
  - Rod worth control
  - Stored fuel shielding, cooling and reactivity control

State C

In State C, the reactor vessel head is on and the reactor is shutdown. Applicable planned operations are achieving criticality and cooldown (Events 2 and 6, respectively).

Sequence diagrams relating safety actions for planned operations, plant systems and applicable events are shown in Figure 15A-10. The required safety actions for planned operation in State C are as follows:

- Safety Actions
  - Radioactive material release control
  - Reactor vessel pressure control
  - Reactor vessel water level control
  - Nuclear system temperature control
  - Nuclear system water quality control
  - Nuclear system leakage control
  - Core reactivity control
  - Containment building pressure and temperature control

– Spent fuel shielding, cooling and reactivity control

State D

In State D, the reactor vessel head is on, and the reactor is not shutdown. Applicable planned operations are achieving criticality, heatup, power operation and achieving shutdown (Events 2, 3, 4, and 5, respectively).

Figure 15A-11 presents the necessary safety actions for planned operations, corresponding plant systems and events for which the safety actions are necessary. The required safety actions for planned operations in State D are as follows:

- Safety Actions
  - Radioactive material release control
  - Core cooling flow rate control
  - Core power level control
  - Core neutron flux distribution control
  - Reactor vessel water level control
  - Reactor vessel pressure control
  - Nuclear system temperature control
  - Nuclear system water quality control
  - Nuclear system leakage control
  - Core reactivity control
  - Rod worth control
  - Containment and reactor building pressure and temperature control
  - Stored fuel shielding, cooling and reactivity control

15A.6.3 Moderate Frequency Incidents (Anticipated Operational Transients)

15A.6.3.1 General

The safety requirements and protection sequences for moderate frequency incidents (anticipated operational transients) are described in the following subsections for Events 7 through 22. The protection sequence block diagrams show the sequence of frontline safety systems (Figures 15A-12 through 15A-27). The auxiliaries for the frontline safety systems are presented in the auxiliary diagrams (Figures 15A-6 and 15A-7) and the commonality of auxiliary diagrams (Figures 15A-65 through 15A-70).

15A.6.3.2 Required Safety Actions/Related Unacceptable Consequences

The following list presents the safety actions for anticipated operational transients to mitigate or prevent the unacceptable safety consequences. Refer to Table 15A-2 for the unacceptable consequences criteria.

Safety Action	Related Unacceptable Consequences Criteria	Reason Action Required
Scram and/or trip of 4 RIPs	2-2, 2-3	To prevent fuel damage and to limit RPV system pressure rise.
Pressure relief	2-3	To prevent excessive RPV pressure rise.
Core and containment cooling	2-1, 2-2, 2-4	To prevent fuel and containment damage in the event that normal cooling is interrupted.
Reactor vessel isolation	2-2	To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.
Restore AC power	2-2	To prevent fuel damage by restoring AC power to systems essential to other safety actions.
Prohibit rod motion	2-2	To prevent exceeding fuel limits during transients.
Containment Isolation	2-1, 2-4	To minimize radiological effects.

15A.6.3.3 Event Definitions and Operational Safety Evaluations

Event 7—Manual and Inadvertent SCRAM

The deliberate manual or inadvertent automatic SCRAM due to single operator error is an event which can occur under any operating conditions. Although assumed to occur here for examination purposes, multi-operator error or action is necessary to initiate such an event.

While all the safety criteria apply, no unique safety actions are required to control the planned-operation-like event after effects of the subject initiation actions. In all operating states, the safety criteria are therefore met through the basic design of the plant systems. Figure 15A-12 presents the protection sequences for this event.

Event 8—Loss-of-Plant Instrument or Service System Air

Loss of all plant instrument or service air system causes reactor shutdown and the closure of air-operated isolation valves. Although these actions occur, they are not a requirement to prevent unacceptable consequence in themselves. Multi-equipment failures would be necessary to cause the deterioration of the subject system to the point that the components supplied with instrument or service air cease to operate “normally” and/or “fail-safe.”

Figure 15A-13 shows how scram is accomplished by loss of air to scram solenoid valves of the Reactor Protection System and the CRD System. The nuclear system pressure relief system provides pressure relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall. Either high-pressure core cooling system supplies water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.

Adequate reserve service air supplies are maintained exclusively for the continual operation of the Automatic Depressurization Subsystem (ADS) safety/relief valves until reactor shutdown is accomplished.

Event 9—Recirculation Flow Control Failure (Increasing Flow)—One RIP Runout

A recirculation flow control failure causing one RIP to runout is applicable in States C and D. The resulting increase in core flow is detected by the RFCS, which reduces the flow through the remaining RIPs, as shown in Figure 15A-14.

Event 10—Recirculation Flow Control Failure (Decreasing Flow)—One RIP Runback

This flow control malfunction causes a decrease in core coolant flow. This event is not applicable to States A and B because the reactor vessel head is off and the reactor internal pumps normally would not be in use. Figure 15A-15 shows that no protection sequence is needed for this event.

Event 11—Trip of Three Reactor Internal Pumps (RIPs)

The trip of three reactor internal pump produces a mild transient of flow and power reduction followed by a select control rod run-in action by the RFCS on detection of this trip. This event is not applicable in States A and B because the reactor vessel head is off and the recirculation pumps normally would not be in use. The trip could occur in States C and D. Figure 15A-16 presents the protection sequence for this event.

Event 12,13—Isolation of One or All Main Steamlines

Isolation of the main steamlines can result in a transient for which some degree of protection is required only in operating States C and D. In operating States A and B, the main steamlines are continuously isolated.

Isolation of all main steamlines is most severe and rapid in operating State D during power operation.

Figure 15A-17 shows how scram is accomplished by main steamline isolation through the actions of the Reactor Protection and CRD Systems. The nuclear system pressure relief system provides pressure relief. Pressure relief, combined with loss of feedwater flow, causes reactor vessel water level to fall and the RCIC System supplies water to maintain water level and to protect the core until normal steam flow (or other planned operation) is established.

Isolation of one main steamline causes a significant transient only in State D during high power operation. Scram, if it occurs, is the only unique action required to avoid fuel damage and nuclear system overpressure. Because the feedwater system and main condenser remain in operation following the event, no unique requirement arises for core cooling.

As shown in Figure 15A-18, the scram safety action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection and CRD Systems.

Event 14—Loss of All Feedwater Flow

A loss of feedwater flow results in a net decrease in the coolant inventory available for core cooling. A loss of feedwater flow can occur in States C and D. Appropriate responses to this transient include a reactor scram on low water level and restoration of reactor water level by the RCIC System.

As shown in Figure 15A-19, the Reactor Protection and CRD Systems effect a scram on low water level. The RCIC System maintains adequate water level for initial core cooling and to restore and maintain water level. For long-term shutdown and extended core coolings, containment/suppression pool cooling systems are manually or automatically initiated.

Event 15—Loss of a Feedwater Heater

Loss of a feedwater heater must be considered with regard to the nuclear safety operational criteria only in operating State D because significant feedwater heating does not occur in any other operating stage.

A loss of more the 16.7°C of feedwater heating causes an alarm to be initiated by the Feedwater Control System (FWCS). Therefore, the most severe case is a loss of 16.7°C of feedwater heating, just below alarm initiation. This 16.7°C loss in feedwater heating results in a minimal 4% power increase and no scram is expected. The operator can control this minimal increase in power. The protection sequence for this event is shown on Figure 15A-20.

Event 16—Feedwater Controller Failure—Runout of One Feedwater Pump

A feedwater controller failure, causing runout of one feedpump, is possible in all operating states. In operating States A and B, no safety actions are required, because the vessel head is removed and the moderator temperature is low. In operating State D, the FWCS reduces flow from the other feedpump to maintain constant feed flow. Steady-state operation may continue, as no scram or turbine trip is expected as shown on Figure 15A-21.

Event 17—Pressure Regulator Failure—One Bypass Valve Failed Open

A pressure regulator failure in the open direction, causing the opening of one turbine control or bypass valve, applies only in operating States C and D, since in other states the pressure regulator is not in operation. An opening of a bypass valve is more severe than opening of a control valve. In either case, the pressure regulator slightly closes the remaining control valves to maintain set pressure. Steady-state operation may continue as shown in Figure 15A-22.

Event 18—Pressure Regulator Failure—One Control Valve Failed

A pressure regulator failure in the closed direction (or downscale), causing the closing of a turbine control valve, applies only in operating States C and D because in other states the pressure regulator is not in operation.

The pressure regulator slightly opens the remaining control valves or bypass valves to maintain set pressure. This action may not be fast enough to mitigate the event. A high neutron flux scram due to the increasing pressure is expected for initial rated power operation. The protection sequence is shown in Figure 15A-23.

Event 19—Main Turbine Trips (With Bypass System Operation)

A main turbine trip can occur only in operating State D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power because the initial power level is less than 40%, thus minimizing the effects of the transient and enabling return to planned operations via the bypass system operation. For a turbine trip above 40% power, a scram occurs via turbine stop valve closure, as will a trip of four RIPs. Subsequent relief valve actuation occurs. Figure 15A-24 presents the protection sequences required for main turbine trips. Main turbine trip and load rejection events are similar anticipated operational transients having the same required safety actions.

Event 20—Loss of Main Condenser Vacuum

A loss of vacuum in the main turbine condenser can occur any time steam pressure is available and the condenser is in use; it is applicable to operating States C and D. However, scram protection in State C is not needed, because the reactor is not coupled to the turbine system.

For State D above 40% power, loss of condenser vacuum initiates a turbine trip with its attendant stop valve closures (which leads to SCRAM) and a trip of four RIPs and also initiates isolation, pressure relief valve and RCIC actuation. Below 40% power (State D) scram is initiated by a high neutron flux or high vessel pressure signal. Figure 15A-25 shows the protection sequences. Decay heat necessitates extended core and suppression pool cooling. When the RPV depressurizes sufficiently, the operation of RHR System shutdown cooling is achieved.

Event 21—Generator Load Rejection, Bypass On

A main generator load rejection with bypass system operation can occur only in operating State D (during heatup or power operation). Fast closure of the main turbine control valves is initiated whenever an electrical grid disturbance occurs, which results in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the main turbine-generator rotor. Closure of the turbine control valves causes a sudden reduction in steam flow, which results in an increase in system pressure. Above 40% power, scram occurs as a result of fast control valve closure, as will a trip of four RIPs. A generator load rejection during heatup (<40%) is not severe because the turbine bypass system can accommodate the decoupling of the reactor and the turbine-generator unit, thus minimizing the effects of the transient and enabling return to planned operations.
Figure 15A-26 presents the protection sequences required for a generator load rejection. Main generator load rejection event and main turbine trip are similar events having the same required safety actions.

Event 22—Loss of Unit Auxiliary Transformer

The loss of the unit auxiliary transformer causes a generator trip, a scram, a trip of four RIPs, a loss of feedwater flow and a loss of condenser vacuum.

Figure 15A-27 shows the protection sequence for this event, including a scram, a trip of four RIPs, a vessel isolation, pressure relief, and core and containment cooling. This event is applicable only in States C and D, because normal AC power in States A and B is supplied from the grid.

Event 23—Inadvertent HPCF Pump Start (Coolant/Moderator Temperature Decrease)

An inadvertent pump start (temperature decrease) is defined as an unintentional start of any nuclear system pump that adds sufficient cold water to the reactor coolant inventory to cause a measurable decrease in moderator temperature. This event is considered in all operating states because it can potentially occur under any operating condition. Since the HPCF pump operates over nearly the entire range of the operating states and delivers the greatest amount of cold water to the vessel, the following analysis will describe its inadvertent operation rather than other NSSS pumps (e.g., RCIC, RHR).

While all the safety criteria apply, no unique safety actions are required to control the effects of such a pump start. In operating States A and C, the safety criteria are met through the basic design of the plant systems, and no safety action is specified. In States B and D, where the reactor is not shutdown, the pressure and temperature will decrease. The operator or the plant normal control system can control any power changes in the normal manner of power control.

Figure 15A-28 illustrates the protection sequence for the subject event.

15A.6.3.4 Other Event Definitions and Operational Safety Evaluations

The following events should be classified as either infrequent or limiting faults. However, criteria for moderate frequency incidents are conservatively applied.

Event 26—Main Turbine Trips with Failure of One Bypass Valve

A main turbine trip can occur only in operating State D (during heatup or power operation). A turbine trip during heatup is not as severe as a trip at full power because the initial power level is less than 40%, thus minimizing the effects of the transient and enabling return to planned operations via the bypass system operation. For a turbine trip above 40% power, a scram occurs via turbine stop valve closure, as will a trip of four RIPs. Subsequent relief valve actuation occurs. Figure15A-31 presents the protection sequences required for main turbine trip with a failure of one bypass valve. The response of the plant to a turbine trip or a generator load rejection with a failure of one bypass valve is similar to that with a full bypass operation; protection sequences for these cases are the same.

Event 27—Generator Load Rejection with Failure of One Bypass Valve

A main generator load rejection with failure of one bypass valve can occur only in operating State D (during heatup or power operation). Fast closure of the main turbine control valves is initiated whenever an electrical grid disturbance occurs, which results in significant loss of electrical load on the generator. The turbine control valves are required to close as rapidly as possible to prevent excessive overspeed of the main turbine-generator rotor. Closure of the turbine control valves causes a sudden reduction in steam flow, which results in an increase in system pressure. Above 40% power, scram occurs as a result of fast control valve closure, as will a trip of four RIPs.

Prolonged shutdown of the turbine-generator unit necessitates extended core and containment cooling. Figure 15A-32 presents the protection sequences required for a main generator load rejection. Main generator load rejection with a failure of one bypass valve is similar to a load rejection with a full bypass operation. Therefore, the required safety actions for both are the same sequence.

Event 38—Abnormal Startup of One Reactor Internal Pump (RIP)

The abnormal startup of a reactor internal pump (RIP) can occur in any state and is most severe and rapid for those operating states in which the reactor may be critical (States B and D).

Occurrence of this event is prevented by a Recirculation Flow Control System (RFCS) interlock that prevents a pump start unless all remaining pumps are at their minimum speeds. For this case of multiple failures and operator errors, the large flow reversal and associated starting pump inverter overcurrent activates a protective logic that trips the two or three RIPs on the bus. In that case, the event is covered by Event 11. Figure 15A-45 shows the protective sequence for this event.

Event 39—Recirculation Flow Control Failure (Increasing Flow)—Runout of All RIPs

A recirculation flow control failure, causing runout of all RIPs, is applicable in States C and D. In State D, the resulting increase in power level is limited by a reactor scram. As shown in Figure 15A-46, the scram safety action is accomplished through the combined actions of the Neutron Monitoring, Reactor Protection and FMCRD Systems.

Event 40—Recirculation Flow Control Failure (Decreasing Flow)—Runback of All RIPs

This recirculation flow control malfunction causes a decrease in core coolant flow. This event is not applicable to States A and B because the reactor vessel head is off and the reactor internal pumps normally would not be in use. Figure 15A-47 shows that no protection sequences are required for this event.

Event 44—Feedwater Controller Failure—Runout of Two Feedwater Pumps

A feedwater controller failure, causing an excess of coolant inventory in the reactor vessel, is possible in all operating states. Feedwater controller failures considered are those that would give failures of automatic flow control, manual flow control, or feedwater bypass valve control. In operating States A and B, no safety actions are required, since the vessel head is removed and the moderator temperature is low. In operating State D, any positive reactivity effects of the reactor caused by cooling of the moderator can be mitigated by a scram. As shown in Figure 15A-51, the accomplishment of the scram safety action is satisfied through the combined actions of the Neutron Monitoring, Reactor Protection and FMCRD Systems. Due to the increasing water level and the resulting L-8 turbine trip, pressure relief is required in States C and D and is achieved through the operation of the RPV pressure relief system. Initial restoration of the core water level is by the RCIC or HPCF Systems.

Event 45—Pressure Regulator Failure—Opening of All Turbine Control and Bypass Valves

A pressure regulator failure in the open direction, causing the opening of all turbine control and bypass valves, applies only in operating States C and D because in other states the pressure regulator is not in operation. A pressure regulator failure is most severe and rapid in operating State D at low power.

The various protection sequences giving the safety actions are shown in Figure 15A-52. Depending on plant conditions existing prior to the event, scram is initiated either on main steamline isolation, main turbine trip or reactor vessel low water level. The sequence resulting in reactor vessel isolation also depends on initial conditions. With the mode switch in RUN, isolation is initiated when main steamline pressure decreases to 5.2 MPaG. After isolation is completed, decay heat causes reactor vessel pressure to increase until limited by the operation of the relief valves. Core cooling following isolation is provided by the RCIC or HPCF Systems. Shortly after reactor vessel isolation, normal core cooling is re-established via the main condenser and feedwater systems or, if prolonged isolation is necessary, extended core and containment cooling will be manually actuated.

Event 48—Main Turbine Trip (Without Bypass System Operation)
A main turbine trip without bypass can occur only in operating State D (during heatup of power operation). Figure 15A-55 presents the protection sequences required for main turbine trips. Plant operation with bypass system operation above or below 40% power, due to bypass system failure, results in the same transient effects: a scram, a trip of four RIPs, and subsequent relief valve actuation. After initial shutdown, extended core and containment cooling is required as noted previously in Event 19.

Turbine trips without bypass system operation results in more severe thermohydraulic impacts on the reactor core than with bypass system operation. The allowable limit or acceptable calculational techniques for this event is less restrictive, because the event is of lower probability of occurrence than the turbine trip with a bypass operation event.

Event 49—Generator Load Rejection with Failure of All Bypass Valves
A main generator trip without bypass system operation can occur only in operating State D (during heatup or power operation). A generator trip during heatup without bypass operation results in the same situation as the power operation case. Figure 15A-56 presents the protection sequences required for a generator load rejection with failure of all bypass valves. The event is basically the same as described in Event 21 at power levels above 40%. A scram, trip of four RIPs, and relief valve operation immediately results in prolonged shutdown, which follows the same pattern as Event 21.

The thermohydraulic and thermodynamic effects on the core, of course, are more severe than with the bypass operating. Because the event is of lower probability than Event 21, the unacceptable consequences are less limiting.

15A.6.4 Infrequent Incidents (Abnormal Operational Transients)

15A.6.4.1 General

The safety requirements and protection sequences for infrequent incidents (abnormal operational transients) are described in the following paragraphs for Events 23 through 27. The protection sequence block diagrams show the sequence of frontline safety systems (Figures 15A-28 through 15A-32). The auxiliaries for the frontline safety systems are indicated in the auxiliary diagrams (Figures 15A-6 and 15A-7) and the commonality of auxiliary diagrams (Figures 15A-65 through 15A-70).

15A.6.4.2 Required Safety Actions/Related Unacceptable Consequences

Table 15A-13 relates the safety actions for infrequent incidents to mitigate or prevent the unacceptable safety consequences cited in Table 15A-3.

15A.6.4.3 Event Definition and Operational Safety Evaluation

Event 24—Inadvertent Opening of a Safety/Relief Valve

The inadvertent opening of a safety/relief valve is possible in any operating state. The protection sequences are shown in Figure 15A-29. In States A and B, the water level cannot be lowered far enough to threaten fuel damage; hence, no safety actions are required.

In States C and D, there is a slight decrease in reactor pressure following the event. The pressure regulator closes the main turbine control valves enough to stabilize pressure at a level slightly below the initial value. There are no unique safety system requirements for this event.

If the event occurs when the Feedwater System is not active, a scram is initiated by a low water level signal and core cooling is accomplished by the RCIC System, which are automatically initiated by the Nuclear Boiler Instrumentation System (NBIS). The Automatic Depressurization System (ADS) or the Manual Relief Valve System remain as the backup depressurization system, if needed. After the vessel has depressurized, long-term core cooling is accomplished by the RHR System. Containment and suppression pool cooling are automatically or manually initiated.

Event 25—Control Rod Withdrawal Error During Refueling and Startup Operations

Because a control rod withdrawal error resulting in an increase of positive reactivity can occur under any operating condition, it must be considered in all operating states.

- Refueling

No unique safety action is required in operating State A for the withdrawal of one control rod because the core is more than one control rod subcritical. Withdrawal of more than one control rod is precluded by the protection sequence shown in Figure 15A-30. During core alterations, the mode switch is normally in the REFUEL position, which allows the refueling equipment to be positioned over the core and also inhibits more than one control rod withdrawal.

Moreover, mechanical design of the control rod assembly prevents physical removal of the control rod blade from the top without removing the adjacent fuel assemblies.

- Startup

During startup, while pulling control rods in States C, the reactor is subcritical by more than one rod. Therefore, no protection sequence is needed for this condition.

During low power operation (States B and D), the RPS initiates SCRAM on short period or high neutron flux in addition to a short period rod block as shown on Figure 15A-30.

15A.6.5 Limiting Faults (Design Basis Accidents)

15A.6.5.1 General

The safety requirements and protection sequences for limiting faults (accidents) are described in the following paragraphs for Events 28 through 52. The protection sequence block diagrams show the safety actions and the sequence of frontline safety systems used for the accidents (Figures 15A-33 through15A-59). The auxiliaries for the frontline safety systems are presented in the auxiliary diagrams (Figures 15A-6 and 15A-7) and the commonality of auxiliary diagrams (Figures 15A-45 through
15A-70).

15A.6.5.2 Required Safety Actions/Unacceptable Consequences

Table 15A-14 presents the safety actions for design basis accident to mitigate or prevent the unacceptable consequences cited in Table 15A-4.

15A.6.5.3 Event Definition and Operational Safety Evaluations

Event 28—Control Rod Ejection Accident
A control rod ejection accident for the fine motion control rod drive design is not a credible event. Therefore, no protection sequence is required.

Event 29—Control Rod Drop Accident (CRDA)
A control rod drop accident for the fine motion control rod drive design is not a credible event. Therefore, no protection sequence is required.

Event 30—Control Rod Withdrawal Error (During Power Operation)
During power operation in State D, the Automated Rod Block Monitoring System (ARBM) of the Rod Control and Information System prevents control rod withdrawals that would result in thermal limit violations. Therefore, this event is not a credible event and no protection sequence is required as shown in Figure 15A-35.

Event 31—Fuel-Handling Accident
Because a fuel-handling accident can potentially occur any time when fuel assemblies are being manipulated, either over the reactor core or in a spent fuel pool, this accident is considered in all operating states. Considerations include mechanical fuel damage caused by drop impact and a subsequent release of fission products. The protection sequences pertinent to this accident are shown in Figure 15A-36. Containment and/or Reactor Building isolation and standby gas treatment operation are automatically initiated by the respective building, pool and/or ventilation radiation monitoring systems.

Event 32—Loss-of-Coolant Accidents (LOCA) Resulting from Postulated Piping Breaks Within RPCB Inside Primary Containment
Pipe breaks inside the primary containment are considered only when the nuclear system is significantly pressurized (States C and D). The result is a release of steam and water into the

containment. Consistent with NSOA criteria, the protection requirements consider all size line breaks, including liquid pipe breaks down to small steam instrument line breaks. The most severe cases are the circumferential break of the high pressure core flooder (liquid) system injection line and the circumferential break of the largest (steam) main steamline.

As shown in Figures 15A-37 and 15A-38, in operating State C (reactor shut down, but pressurized), a pipe break accident up to the largest pipe break can be accommodated within the nuclear safety operational criteria through the various operations of the MSIVs, Emergency Core Cooling Systems (HPCF, ADS, RHR-LPFL, RCIC), Leak Detection and Isolation System, Standby Gas Treatment System, main control room heating, cooling and ventilation system, plant protection system (RHR heat exchangers) and the Nuclear Boiler Instrumentation System. For small pipe breaks inside the containment, pressure relief is effected by the nuclear system pressure relief system, which transfers decay heat to the suppression pool. For large breaks, depressurization takes place through the break itself. In State D (reactor not shut down, but pressurized), the same equipment is required as in State C but, in addition, the Reactor Protection System and the FMCRD System must operate to scram the reactor. The limiting items, on which the operation of the above equipment is based, are the allowable fuel cladding temperature and the containment pressure capability. The FMCRD housing supports are considered necessary whenever the system is pressurized to prevent excessive control rod movement through the bottom of the reactor pressure vessel following the postulated rupture of one FMCRD housing (a lesser case of the design basis LOCA and a related preventive of a postulated rod ejection accident).

After completion of the automatic action of the above equipment, manual operation of the RHR (suppression pool, drywell and wetwell cooling modes) and ADS or relief valves operation (controlled depressurization) may be required to maintain containment pressure and fuel cladding temperature within limits during extended core cooling.

Event 33—Loss-of-Coolant Accidents (LOCA) Resulting from Postulated Pipe Breaks—Outside Primary Containment

Pipe break accidents outside the primary containment are assumed to occur any time the nuclear system is pressurized (States C and D). This accident is most severe during operation at high power (State D). In State C, this accident becomes a subset of the State D sequence.

The protection sequences for the various possible pipe breaks outside the containment are shown in Figures 15A-39 and 15A-40. The sequences also show that for small breaks (breaks not requiring immediate action), the reactor operator can use a large number of process indications to identify the break and isolate it.

In Operating State D (reactor not shut down, but pressurized), scram is accomplished through operation of the Reactor Protection System and the FMCRD System. Reactor vessel isolation is accomplished through operation of the main steamline isolation valves and the Leak Detection and Isolation System.

For a main steamline break, initial core cooling is accomplished by either the HPCF or RCIC, or the Automatic Depressurization System (ADS) or manual relief valve operation in conjunction with RHR-LPFL. These systems provide parallel paths to effect initial core cooling, thereby satisfying the single-failure criterion. Extended core cooling is accomplished by the single-failure proof, parallel combination of HPCF and RHR LPFL Systems. The ADS or relief valve system operation and the RHR suppression pool cooling, wetwell and drywell spray modes are required to maintain containment temperature, pressure, and fuel cladding temperature within limits during extended core cooling. Subsequent to isolation of the break and depressurization of the vessel, RHR shutdown cooling mode may be operated for long term decay heat removal from the core.

Event 34—Gaseous Radwaste System Leak or Failure

It is assumed that the line leading to the steam jet air ejector fails near the main condenser. This results in activity normally processed by the Offgas Treatment System being discharged directly to the Turbine Building and subsequently through the ventilation system to the environment. This failure results in a loss-of-flow signal to the Offgas System. This event is applicable only in States A, B, C and D, and is shown in Figure 15A-41.

The reactor operator initiates a normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result (timing dependent on leak rate) in a main turbine trip, a vessel isolation that terminates the steam and activity outflow from the reactor, and ultimately a reactor shutdown. Refer to Event 20 for reactor protection sequence (Figure 15A-25).

Event 35—Augmented Offgas Treatment System Failure

An evaluation of those events which could cause a gross failure in the Offgas System has resulted in the identification of a postulated seismic event, more severe than the one for which the system is designed, as the only conceivable event which could cause significant damage.

The detected gross failure of this system will result in manual isolation of this system from the main condenser. The isolation results in high main condenser pressure and ultimately a main turbine trip and associated reactor scram and vessel isolation (that terminates the steam and activity discharge from the vessel). Protective sequences for the event are shown in Figure 15A-42. The loss of vacuum in the main condenser transient has been analyzed in Event 20 (Figure 15A-25).

Event 36—Liquid Radwaste Leak or Failure

Releases which could occur inside and outside of the containment, not covered by Events 28, 29, 30, 33, 35 and 36, include small spills and equipment leaks of radioactive materials inside structures housing the subject process equipment. Conservative values for leakage have been assumed and evaluated in the plant under routine releases. The offsite dose that results from any small spill which could occur outside containment is negligible in comparison to the dose resulting from the accountable (expected) plant leakages. The protective sequences for this event are presented in Figure 15A-43.

Event 37—Liquid Radwaste System—Storage Tank Failure

An unspecified event causes the complete release of the average radioactivity inventory in the storage tank containing the largest quantities of significant radionuclides from the Liquid Radwaste System. This is assumed to be one of the concentrator waste tanks in the Radwaste Building. The airborne radioactivity released during the accident passes directly to the environment via the Radwaste Building vent.

The postulated events that could cause release of the radioactive inventory of the concentrator waste tank include cracks in the vessels and operator error. The possibility of small cracks and consequent low-level release rates receives primary consideration in system and component design. The concentrator waste tank is designed to operate at atmospheric pressure and 93.3^{o}C maximum temperature so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is provided to prevent inadvertent opening of a drain valve. Should a release of liquid radioactive wastes occur, floor drain sump pumps in the floor of the Radwaste Building will receive a high water level alarm, activate automatically and remove the spilled liquid to a contained storage tank. The protective sequences for this event are presented in Figure 15A-44.

Event 41—Trip of All Reactor Internal Pumps (RIPs)

This event is not applicable in States A and B because the reactor vessel head is off and the RIPs normally would not be in use. The trip could occur in States C and D. A trip of all RIPs results in a scram and may cause a high water level trip of the main turbine and the feedpump turbines. Figure 15A-48 provides the protection sequence for this event. A simultaneous trip of all RIPs may cause some fuel cladding heatup due to momentary transition boiling. The cladding heatup is insignificant, its temperature is below 1204°C, the fuel enthalpy is lower than 1.17 kJ/g and event consequences are acceptable.

Event 42—Loss of Shutdown Cooling

Loss of shutdown cooling is applicable in States A, B, C and D, during normal shutdown and cooldown. Because each of the three RHR loops may be lined up independently in the shutdown cooling mode, a simultaneous loss of all three loops is not a credible event and therefore no protection sequence is required as shown in Figure 15A-49.

Event 43—RHR Shutdown Cooling—Increased Cooling

An RHR shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered in States C and D if RPV system pressure is too high to permit operation of the shutdown cooling (RHRS) (Figure 15A-50). No unique safety actions are required to avoid the unacceptable safety consequences for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers.

In States B and D, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

Event 46—Pressure Regulator Failure—Closure of All Turbine Control and Bypass Valves

A pressure regulator failure in the close direction (or downscale), causing the closing of all turbine control and bypass valves, applies only in operating States C and D because in other states the pressure regulator is not in operation. The protection sequence shown on Figure 15A-53 includes a high neutron flux scram by the Neutron Monitoring, Reactor Protection and FMCRD Systems, a high pressure trip of four RIPs, pressure relief and core and containment cooling.

Event 50—Misplaced Fuel Bundle Accident

Operation with a fuel assembly in the improper position is shown in Figure 15A-57 and can occur in all operating states. No protection sequences are necessary relative to this event. Calculated results of worst fuel-handling loading error does not cause fuel cladding integrity damage. It requires three independent equipment/operator errors to allow this situation to develop.

Event 51—Reactor Internal Pump (RIP) Seizure

An RIP seizure event considers the instantaneous stoppage of the pump motor shaft of one RIP. The case involves operation at design power in State D. Because a seizure of one out of ten RIPs produces a flow disturbance of less than 10%, consequences of a RIP seizure are mild and no scram occurs. Therefore, normal operation may continue and no protection sequence is required as shown in Figure 15A-58.

Event 52—Reactor Internal Pump (RIP) Shaft Break

An RIP shaft break event considers the degraded, delayed stoppage of the pump motor shaft of one RIP. The case involves operation at design power in State D. The consequences of this event are bounded by Event 51—RIP Seizure. Normal operation may continue and no protection sequence is required as shown in Figure 15A-59.

15A.6.6 Special Events

15A.6.6.1 General

Additional special events are postulated to demonstrate that the plant is capable of accommodating off-design occurrences (Events 53 through 56). As such, these events are beyond the safety requirements of the other event categories. The safety actions shown on the sequence diagrams (Figures 15A-60 through 15A-63) for the additional special events follow directly from the requirements cited in the demonstration of the plant capability.

Auxiliary system support analyses are shown in Figures 15A-6 and 15A-7, and 15A-65 through 15A-70.

15A.6.6.2 Required Safety Action/Unacceptable Consequences

Table 15A-15 relates the safety actions for special events to demonstrate the capabilities cited in Table 15A-5.

15A.6.6.3 Event Definitions and Operational Safety Evaluation

Event 53—Shipping Cask Drop and Spent Fuel Cask Drop Accident

Due to the redundant nature of the crane, the cask drop accident is not a credible accident. However, the accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

The largest size of BWR fuel cask is conservatively assumed to be dropped approximately 29 m from the refueling floor level to ground level on transport from the decontamination pit out of the Reactor Building. Some of the coolant in the outer cask structure may leak from the cask.

The reactor operator ascertains the degree of cask damage and, if possible, makes the necessary repairs and refill the cask coolant to its normal level if coolant has been lost.

It is assumed that if the coolant is lost from the external cask shield, the operator establishes forced cooling of the cask by introducing water into the outer structure annulus or by spraying water on the cask exterior surface. Maintaining the cask in a cool condition therefore ensures no fuel damage as a result of a temperature increase due to decay heat.

Because the cask is still within the Reactor Building volume, any activity postulated to be released can be accommodated by the secondary containment and Standby Gas Treatment System. The protective sequences for this event are provided in Figure 15A-60.

Event 54—Reactor Shutdown—ATWS

This event is applicable in States B, C and D. Figure 15A-61 shows the protection sequence for this extremely improbable and demanding event in each operating state.

State D is the most limiting case. Upon initiation of the plant transient situation (MSIV closure), a scram is initiated. The scram using hydraulic force is assumed to fail. However, the control rods can still be moved by the electric motors. This FMCRD insertion is sufficient to shut down a reactor. The reactor internal pumps are tripped; a trip of 4 RIPs on high pressure or turbine trip signals (or at Level 3) and a trip of 6 RIPs at Level 2. These trips cause a power decrease if the vessel becomes isolated from the main condenser, reactor power can be transferred from the reactor to the suppression pool via the relief valves. The Nuclear Boiler Instrumentation System initiates operation of the RCIC and HPCF Systems on low water level, which maintains reactor vessel water level. The Standby Liquid Control System is manually initiated and the transition from low reactor power to decay heat occurs. The RHR suppression pool cooling, and drywell and wetwell spray modes are used to remove the reactor power and decay heat from the suppression pool and primary containment as required. When RPV pressure falls to 0.689 to 1.38 MPaG level, the RHR shutdown cooling mode is started and continued until reaching cold shutdown.

Event 55—Reactor Shutdown from Outside Main Control Room

Reactor shutdown from outside the main control room is an event investigated to evaluate the capability of the plant to be safely shutdown and cooled to the cold shutdown state from outside the main control room. The event is applicable in any operating States A, B, C and D.

Figure 15A-62 shows the protection sequences for this event in each operating state. In State A, no sequence is shown because the reactor is already in the condition finally required for the event. In State C, only cooldown is required, since the reactor is already shutdown.

A scram from outside the main control room can be achieved by opening the AC supply breakers for the Reactor Protection System. If the nuclear system becomes isolated from the main condenser, decay heat is transferred from the reactor to the suppression pool via the relief valves. The Nuclear Boiler Instrumentation System initiates the operation of the RCIC and HPCF Systems on low water level which maintains reactor vessel water level, and the RHR suppression pool cooling mode is used to remove the decay heat from the suppression pool if required. When reactor pressure falls below the shutdown cooling interlock pressure, the RHR shutdown cooling mode is started.

Event 56—Reactor Shutdown Without Control Rods

Reactor shutdown without control rods is an event requiring an alternate method of reactivity control—the Standby Liquid Control System (SLCS). By definition, this event can occur only when the reactor is not already shutdown. Therefore, this event is considered only in operating States B and D.

The SLCS must operate to avoid unacceptable consequence criteria 5-3. The design bases for the SLCS result from these operating criteria when applied under the most severe conditions (State D at rated power). As indicated in Figure 15A-63, the SLCS is manually initiated and controlled in States B and D.

15A.7 Remainder of NSOA

With the information presented in the protection sequence block diagrams, the auxiliary diagrams and the commonality of auxiliary diagrams, it is possible to determine the exact functional and hardware requirements of each system. This is done by considering each event in which the system is employed and deriving a limiting set of operational requirements. This limiting set of operational requirements establishes the lowest acceptable level of performance for a system or component, or the minimum number of components or portions of a system that must be operable in order that plant operation may continue.

The operational requirements derived using this process may be complicated functions of operating states, parameter ranges, and hardware conditions. The final step is to simplify these complex requirements into technical specifications that encompass the operational requirements that can be used by plant operations and management personnel.

15A.8 Conclusions

It is concluded that the nuclear safety operational and plant design basis criteria are satisfied when the plant is operated in accordance with the nuclear safety operational requirements determined by the method presented in this appendix.

Table 15A-1 Unacceptable Consequences Criteria Plant Event Category: Normal Operation

Unacceptable Consequences
1-1 Release of radioactive material to the environs that exceed the limits of either 10CFR20 or 10CFR50.
1-2 Fuel failure to such an extent that were the freed fission products released to the environs via the normal discharge paths for radioactive material, the limits of 10CFR20 would be exceeded.
1-3 Nuclear system stress in excess of that allowed for planned operation by applicable industry codes.
1-4 Existence of a plant condition not considered by plant safety analyses.

Table 15A-2 Unacceptable Consequences Criteria Plant Event Category: Moderate Frequency Incidents (Anticipated Operational Transients)

Unacceptable Consequences
2-1 Release of radioactive material to the environs that exceed the limits of 10CFR20.
2-2 Reactor operation induced fuel cladding failure.
2-3 Nuclear system stress exceeding that allowed for transients by applicable industry codes.
2-4 Containment stresses exceeding that allowed for transients by applicable industry codes.

Table 15A-3 Unacceptable Consequences Criteria Plant Event Category: Infrequent Incidents (Abnormal Operational Transients)

Unacceptable Consequences
3-1 Radioactive material release exceeding of a small fraction of 10CFR100.
3-2 Fuel damage that would preclude resumption of normal operation after a normal restart.
3-3 Generation of a condition that results in consequential loss of function of the reactor coolant system.
3-4 Generation of a condition that results in a consequential loss of function of a necessary containment barrier.

Table 15A-4 Unacceptable Consequences Criteria Plant Event Category: Limiting Faults (Design Basis Accidents)

Unacceptable Consequences
4-1 Radioactive material release exceeding the guideline values of 10CFR100.
4-2 Failure of the fuel barrier which would cause changes in core geometry such that core cooling would be inhibited.
4-3 Nuclear system stresses exceeding that allowed for transients by applicable industry codes.
4-4 Containment stresses exceeding that allowed for transients by applicable industry codes when containment is required.
4-5 Overexposure to radiation of plant main control room personnel (in excess of 0.05 Sv whole body, 0.3 Sv inhalation and 0.75 Sv skin).

Table 15A-5 Capability Consequences Plant Event Category: Special Events

Special Events Considered
A. Reactor shutdown from outside control room.
B. Reactor shutdown without control rods.
C. Reactor shutdown with anticipated transient without scram (ATWS).
D. Shipping Cask Drop.
Capability Demonstration
5-1 Ability to shut down reactor by manipulating controls and equipment outside the main control room
5-2 Ability to bring the reactor to the cold shutdown condition from outside the main control room.
5-3 Ability to shut down the reactor independent of control rods.
5-4 Ability to contain radiological contamination.
5-5 Ability to limit radiological exposure

Table 15A-6 General Nuclear Safety Operational Criteria

Applicability	Nuclear Safety Operaton Criteria
Planned operation moderate frequency and infrequent incidents limiting faults and additional special plant capability events.	The plant shall be operated so as to avoid unacceptable consequences.
Moderate frequency and infrequent incidents and design basis accidents.	The plant shall be designed and operated in such a manner that no single active component failure can prevent (1) safety-related core activity control, (2) safety-related core and containment heat removal, (3) reactor coolant pressure boundary integrity, (4) safety-related containment isolation and (5) safety-related containment atmosphere control and cleanup.

Table 15A-7 BWR Operating States*

	States			
Conditions	**A**	**B**	**C**	**D**
Reactor vessel head off	X	X		
Reactor vessel head on			X	X
Shutdown	X		X	
Not Shutdown		X		X
Definition				
Shutdown: K_{eff} sufficiently less than 1.0 such that the full withdrawal of one control rod pair (with the same HCU) or one control rod of maximum worth could not produce criticality under the most restrictive conditions of temperature, pressure, core age and fission product concentrations.				

* Further discussion is provided in Subsection 15A.6.2.4.

Table 15A-8 Normal Operation

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
1	Refueling	15A-8	—	X			
2	Achieving Criticality	15A-8, 15A-9 15A-10, 15A-11	—	X	X	X	X
3	Heatup	15A-11	—				X
4	Power Operation—Electric Generation - Steady State - Daily Load Reduction and Recover—Grid Frequency Control Responses—Control Rod Sequence Exchanges - Power Generation Surveillance Testing • Turbine Control Valve Surveillance Tests • Turbine Stop Valve Surveillance Tests • MSIV Surveillance Tests	15A-11	—				X
5	Achieving Shutdown	15A-9, 15A-11			X		X
6	Cooldown	15A-8 15A-10		X		X	

Table 15A-9 Moderate Frequency Accidents (Anticipated Operational Transients)

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
7	Manual or Inadvertent SCRAM	15A-12	7.2	X	X	X	X
8	Loss of Plant Instrument Service Air Systems	15A-13	9.3.1	X	X	X	X
9	Recirculation Flow Control Failure—One RIP Runout	15A-14	15.4.5			X	X
10	Recirculation Flow Control Failure—One RIP Runback	15A-15	15.3.2			X	X
11	Three RIPs Trip	15A-16	15.3.1			X	X
12	All MSIV Closure	15A-17	15.2.4			X	X
13	One MSIV Closure	15A-18	15.2.4			X	X
14	Loss of All Feedwater Flow	15A-19	15.2.7			X	X
15	Loss of a Feedwater Heating	15A-20	15.1.1				X
16	Feedwater Controller Failure—Runout of One Feedwater Pump	15A-21	15.1.2	X	X	X	X
17	Pressure Regulator Failure—Opening of One Bypass Valve	15A-22	15.1.3			X	X
18	Pressure Regulator Failure—Opening of One Control Valve	15A-23	15.2.1			X	X
19	Main Turbine Trip with Bypass System Operational	15A-24	15.2.3				X
20	Loss of Main Condenser Vacuum	15A-25	15.2.5			X	X
21	Generator Load Rejection with Bypass System Operational	15A-26	15.2.2				X
22	Loss of Unit Auxiliary Transformer	15A-27	15.2.6			X	X
23	Inadvertent Startup of HPCF Pump	15A-28	15.5.1	X	X	X	X
26*	Main Turbine Trip with One Bypass Valve Failure	15A-31	15.2.3				X
27*	Generator Load Rejection with One Bypass Valve Failure	15A-32	15.2.2				X
38*	Abnormal Startup System Reactor Internal Pump	15A-45	15.4.4	X	X	X	X

Table 15A-9 Moderate Frequency Accidents (Anticipated Operational Transients) (Continued)

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
39*	Recirculation Flow Control Failure—All RIPs Runout	15A-46	15.4.5			X	X
40*	Recirculation Flow Control Failure—All RIPs Runback	15A-47	15.3.2			X	X
44*	Feedwater Controller Failure Runout of Two Feedwater Pumps	15A-51	15.1.2	X	X	X	X
45*	Pressure Regulator Failure—Opening of all Bypass and Control Valves	15A-52	15.1.3			X	X
48*	Main Turbine Trip with Bypass Failure	15A-55	15.2.3				X
49*	Generator Load Rejection with Bypass Failure	15A-56	15.2.2				X

* This event should be classified as an infrequent event or a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

Table 15A-10 Infrequent Accidents (Abnormal Operational Transients)

NSOA Event No.	Event Description	NSOA Event Figure No.	Safety Analysis Section No.	BWR Operating State A	B	C	D
24	Inadvertent Opening of a Safety/Relief Valve	15A-29	15.1.4			X	X
25	Control Rod Withdrawal Error—Startup and Refueling Operations	15A-30	15.4.1	X	X	X	X

Table 15A-11 Limiting Faults (Design Basis Accidents)

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
28	Control Rod Ejection Accident	15A-33	15.4.8	X	X	X	X
29	Control Rod Drop Accident	15A-34	15.4.9	X	X	X	X
30	Control Rod Withdrawal Error—Power Operation	15A-35	15.4.2				X
31	Fuel-Handling Accident	15A-36	15.7.4	X	X	X	X
32	Loss-of-Coolant Accident Resulting from Spectrum of Postulated Piping Breaks Within the RCPB Inside Containment	15A-37 and 15A-38	15.6.5			X	X
33	Small, Large, Steam and Liquid Piping Breaks Outside Containment	15A-39 and 15A-40	15.6.4			X	X
34	Gaseous Radwaste System Leak or Failure	15A-41	15.7.1	X	X	X	X
35	Augmented Offgas Treatment System Failure	15A-42	15.7.1	X	X	X	X
36	Liquid Radwaste System Leak or Failure	15A-43	15.7.2	X	X	X	X
37	Liquid Radwaste System Storage Tank Failure	15A-44	15.7.3	X	X	X	X
41	Trip of All RIPs	15A-48	15.3.1			X	X
42	Loss of RHR Shutdown Cooling	15A-49	15.2.9	X	X	X	X
43	RHR Shutdown Cooling Increased Cooling	15A-50	15.1.6	X	X	X	X
46	Pressure Regulator Failure—Closure of all Bypass and Control Valves	15A-53	15.2.1			X	X
50	Misplaced Fuel Bundle Accident	15A-57	15.4.7	X	X	X	X
51	Reactor Internal Pump Seizure	15A-58	15.3.3				X
52	Reactor Internal Pump Shaft Break	15A-59	15.3.4				X

Table 15A-12 Special Events

NSOA Event No.	Event Description	NSOA Event Figure No	Safety Analysis Section No.	BWR Operating State A	B	C	D
53	Shipping Cask Drop Spent Radwaste Spent Fuel New Fuel	15A-60	15.7.5	X	X	X	X
54	Reactor Shutdown From Anticipated Transient Without SCRAM (ATWS)	15A-61	15.8	X	X	X	X
55	Reactor Shutdown From Outside Control Room	15A-62	7.5	X	X	X	X
56	Reactor Shutdown Without Control Rods	15A-63	9.3.5	X	X	X	X

Table 15A-13 Safety Actions for Infrequent Incidents

Safety Action	Related Unacceptable Consequences	Reason Action Required
Scram and/or trip of four RIPs	3-2 3-3	To limit gross core-wide fuel damage and to limit nuclear system pressure rise.
Pressure relief	3-3	To prevent excessive nuclear system pressure rise.
Core, suppression pool and containment cooling	3-2 3-4	To limit further fuel and containment damage in the event that normal cooling is interrupted.
Reactor vessel isolation	3-2	To limit further fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level.
Restore AC power	3-2	To limit initial fuel damage by restoring AC power to systems essential to other safety actions.
Containment isolation	3-1	To limit radiological effects.

Table 15A-14 Safety Actions for Design Basis Accidents

Safety Action	Related Unacceptable Consequences	Reason Action Required
Scram	4-2 4-3	To prevent fuel cladding failure* and excessive nuclear system pressures.
Pressure relief	4-3	To prevent excessive nuclear system pressure.
Core cooling	4-2	To prevent fuel cladding failure.
Reactor vessel isolation	4-1	To limit radiological effects to not exceed the guideline values of 10CFR100.
Containment isolation	4-1	To limit radiological effects to not exceed the guideline values of 10CFR100.
Containment cooling	4-4	To prevent excessive pressure in the containment when containment is required.
Stop rod ejection	4-2	To prevent fuel cladding failure.
Restrict loss of reactor coolant (passive)	4-2	To prevent fuel cladding failure.
Main Control Room environmental control	4-5	To prevent overexposure to radiation of plant personnel in the control room.
Limit reactivity insertion rate	4-2 4-3	To prevent fuel cladding failure and to prevent excessive nuclear system pressure.

* Failure of the fuel barrier includes fuel cladding fragmentation (LOCA) and excessive fuel enthalpy (CRDA).

Table 15A-15 Safety Actions for Special Events

Safety Action	Related Unacceptable Consequences	Reason Action Required
A. Main Control Room Considerations		
Manually initiate all shutdown controls from local panels	5-1 5-2	Local panel control has been provided and is available outside main control room.
Manually initiate SLCS	5-3	Standby Liquid Control System to control reactivity to assure cold shutdown is available.
B. Shipping Cask Considerations See Subsection 9.1.4		

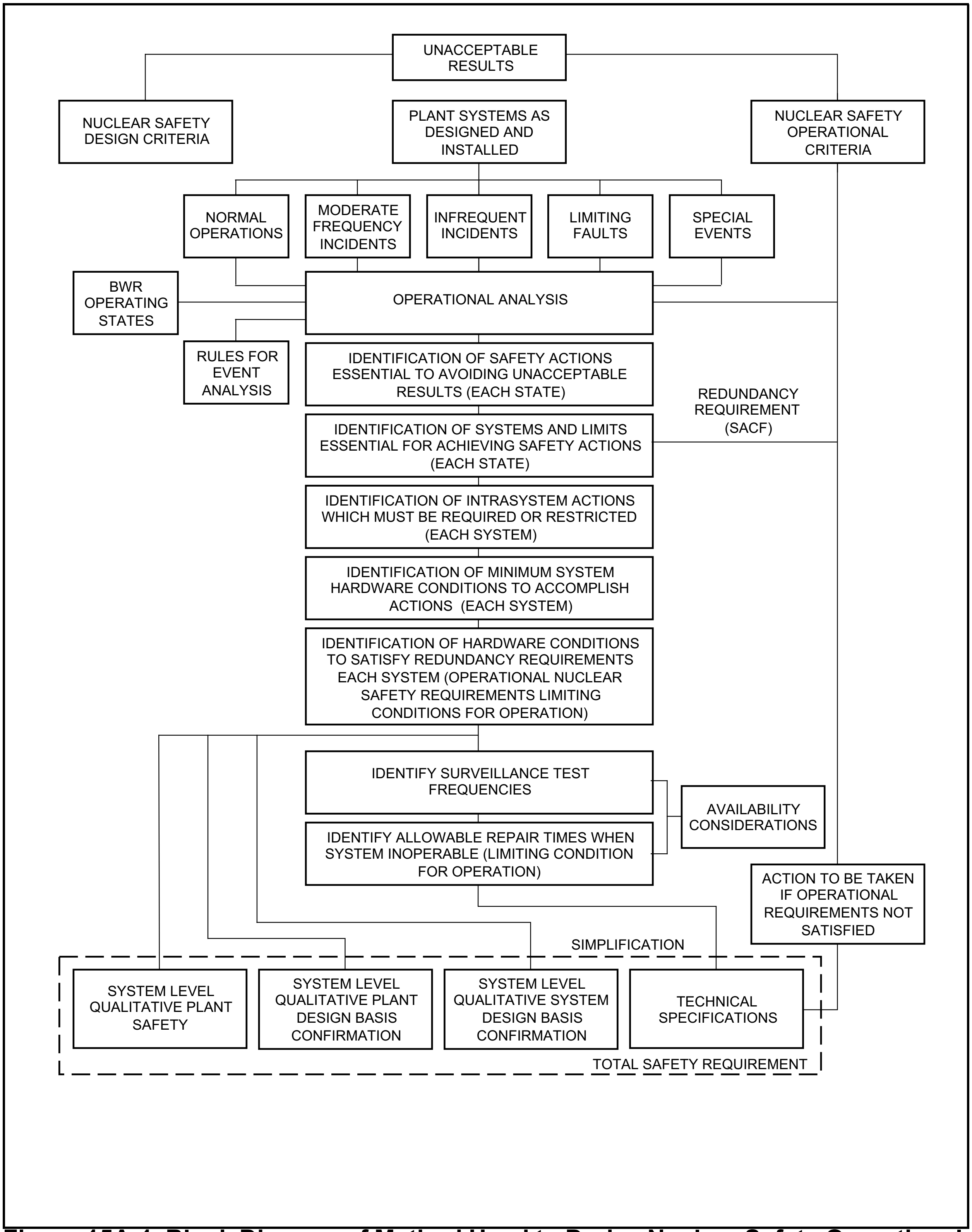


Figure 15A-1 Block Diagram of Method Used to Derive Nuclear Safety Operational Requirements System-Level Qualitative Design Basis Confirmation Audits and Technical Specifications

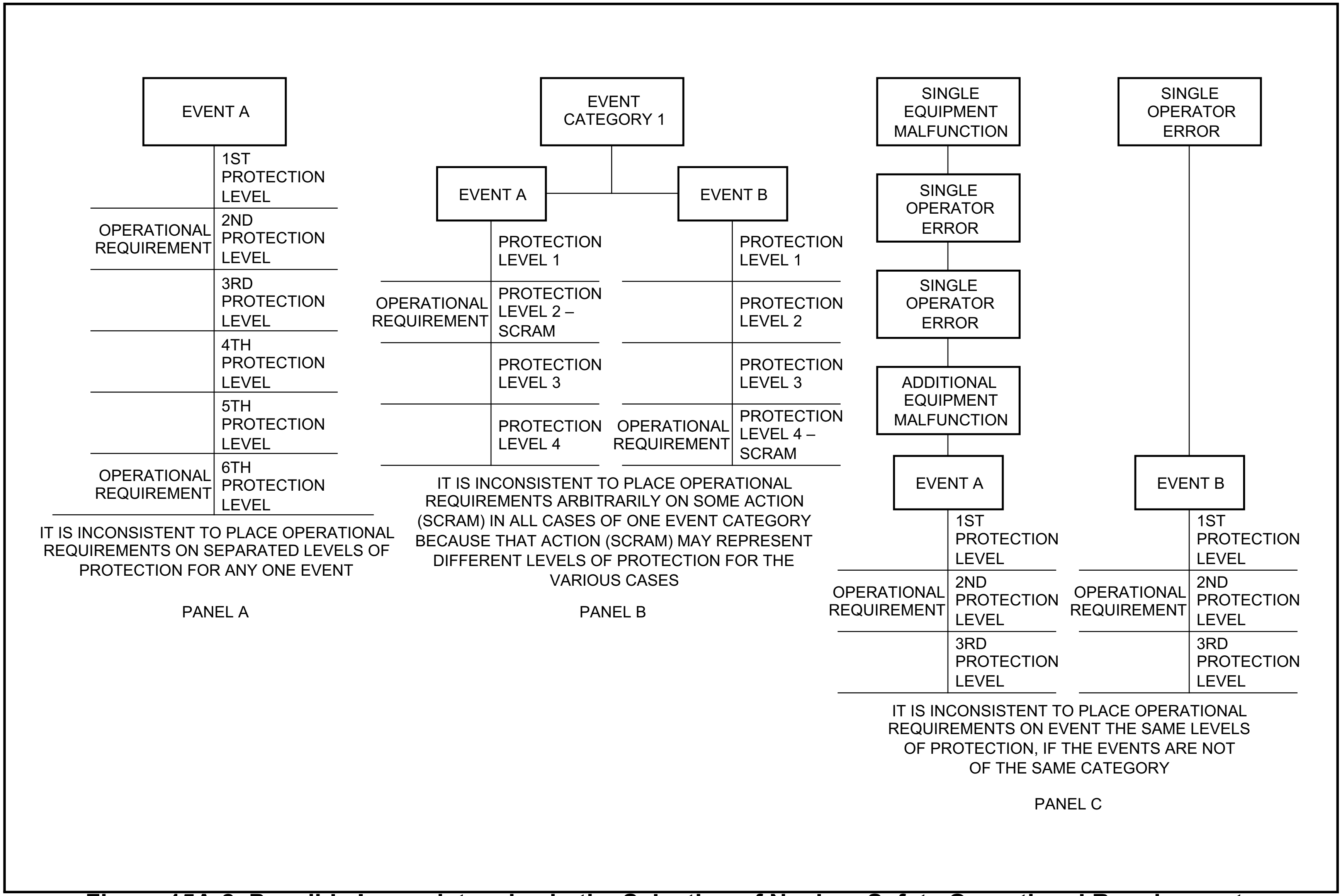


Figure 15A-2 Possible Inconsistencies in the Selection of Nuclear Safety Operational Requirements

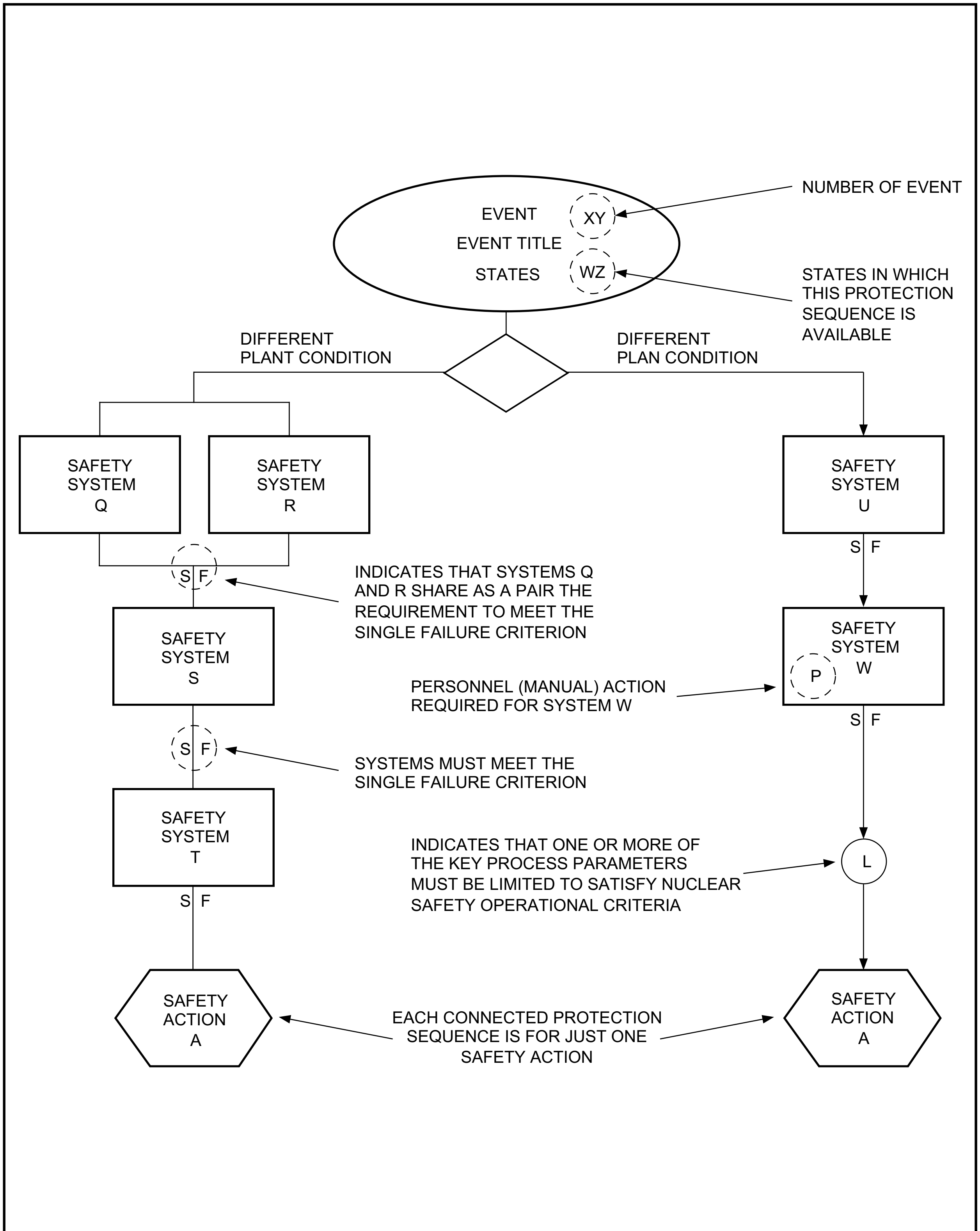


Figure 15A-3 Format for Protection Sequence Diagrams

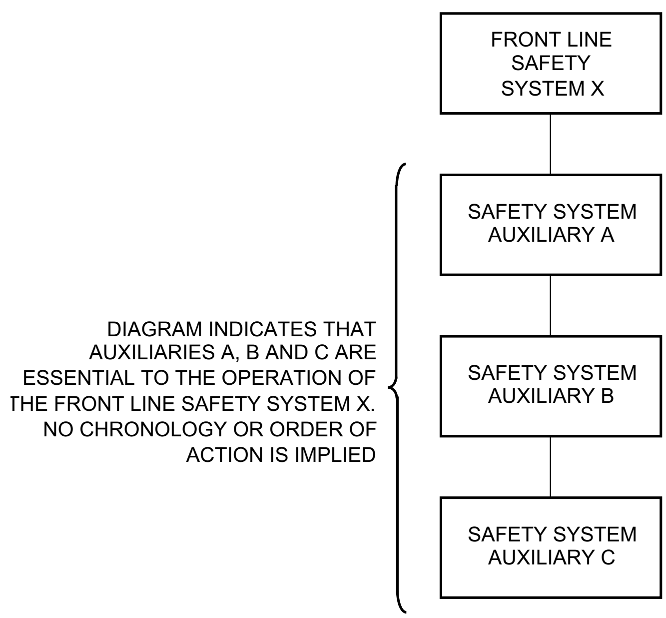


Figure 15A-4 Format for Safety System Auxiliary Diagrams

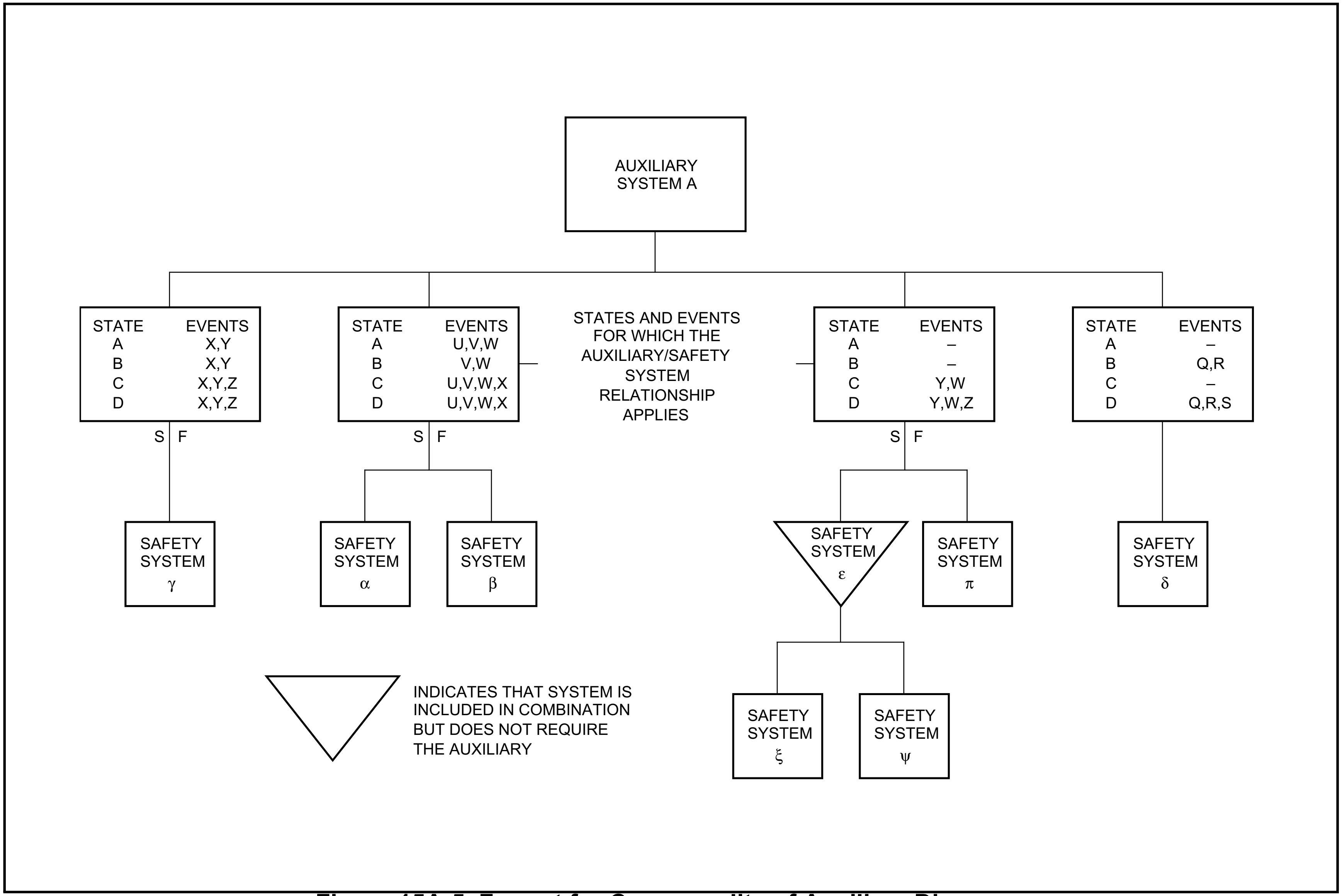


Figure 15A-5 Format for Commonality of Auxiliary Diagrams

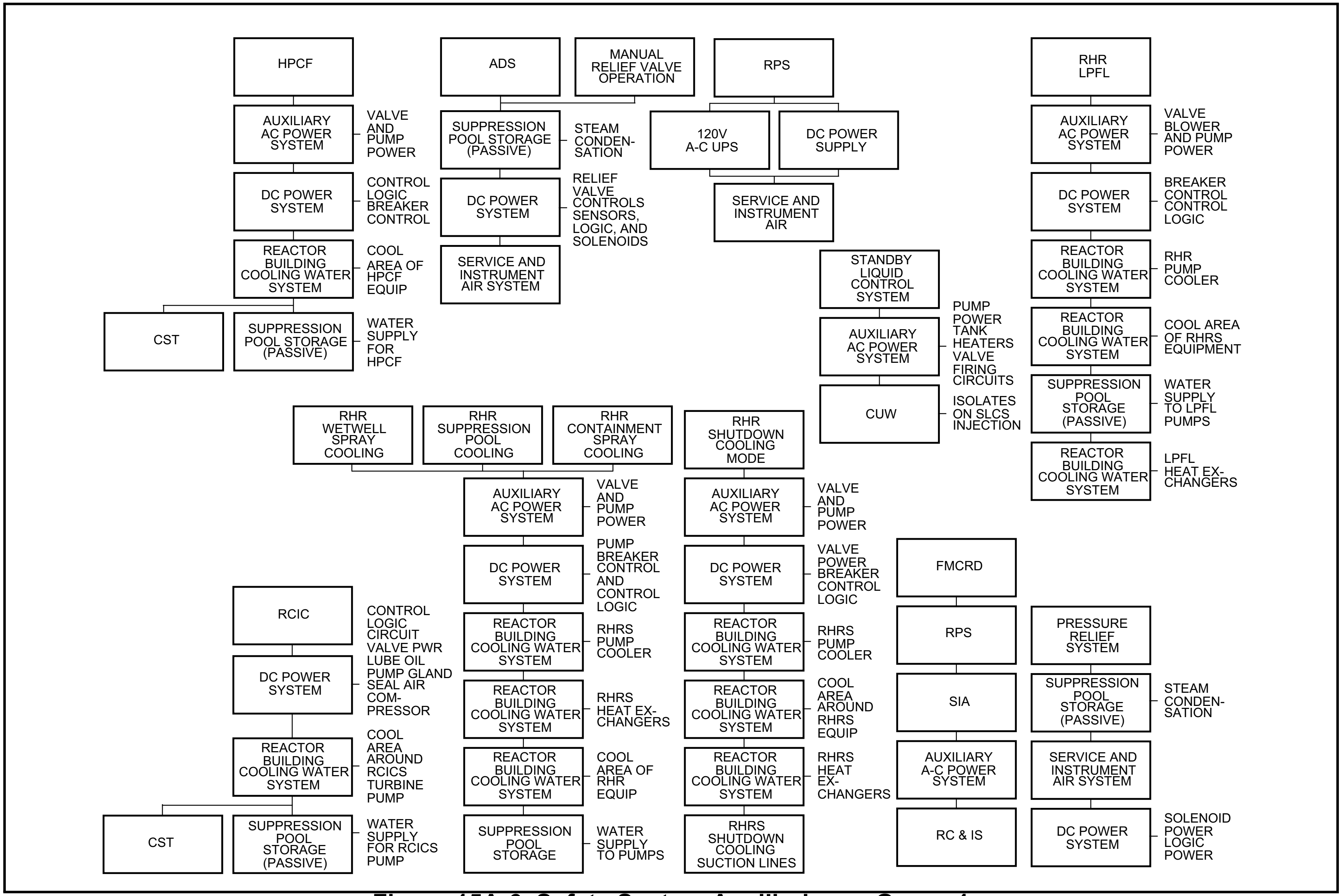


Figure 15A-6 Safety System Auxiliaries — Group 1

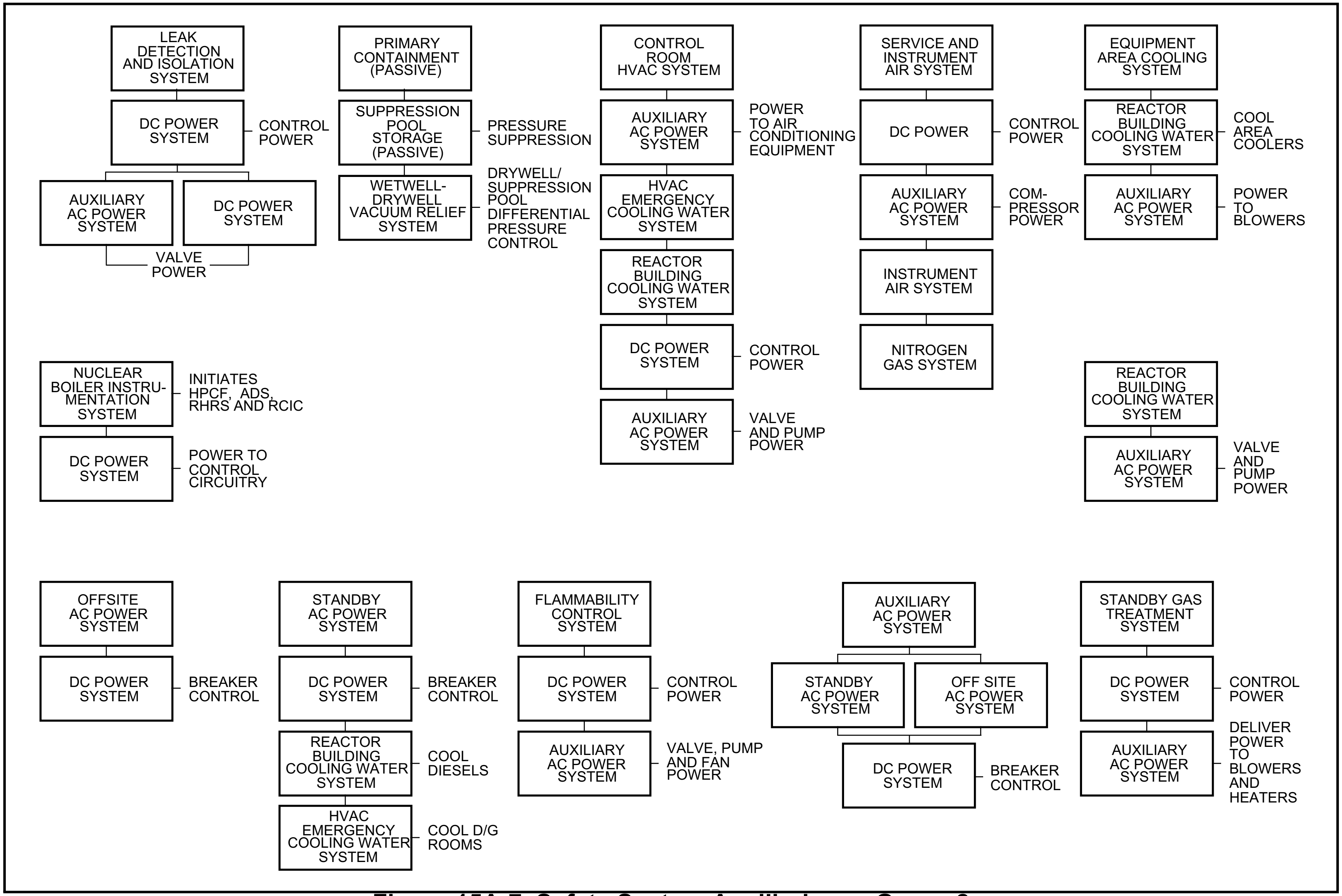


Figure 15A-7 Safety System Auxiliaries — Group 2

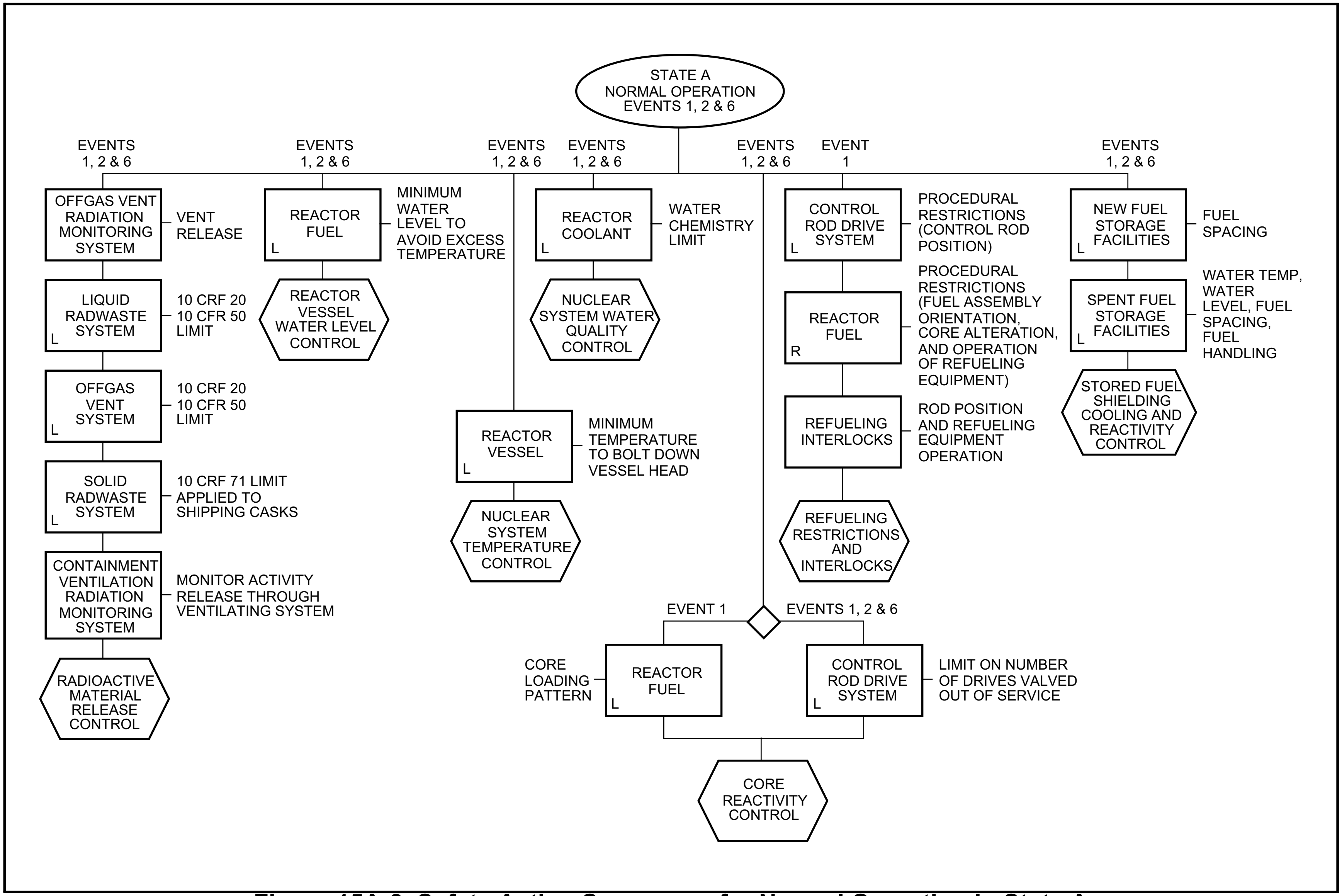


Figure 15A-8 Safety Action Sequences for Normal Operation in State A

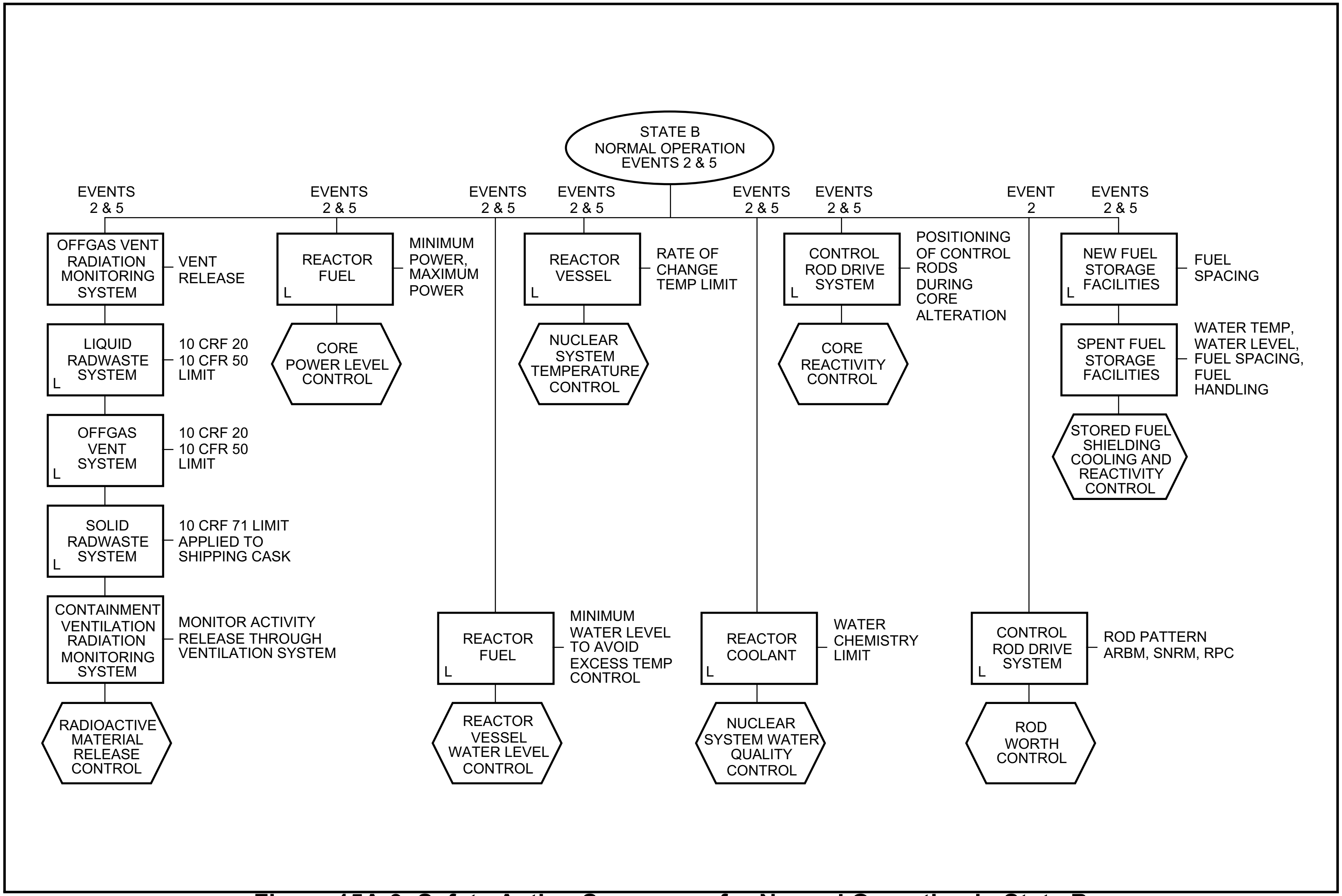


Figure 15A-9 Safety Action Sequences for Normal Operation in State B

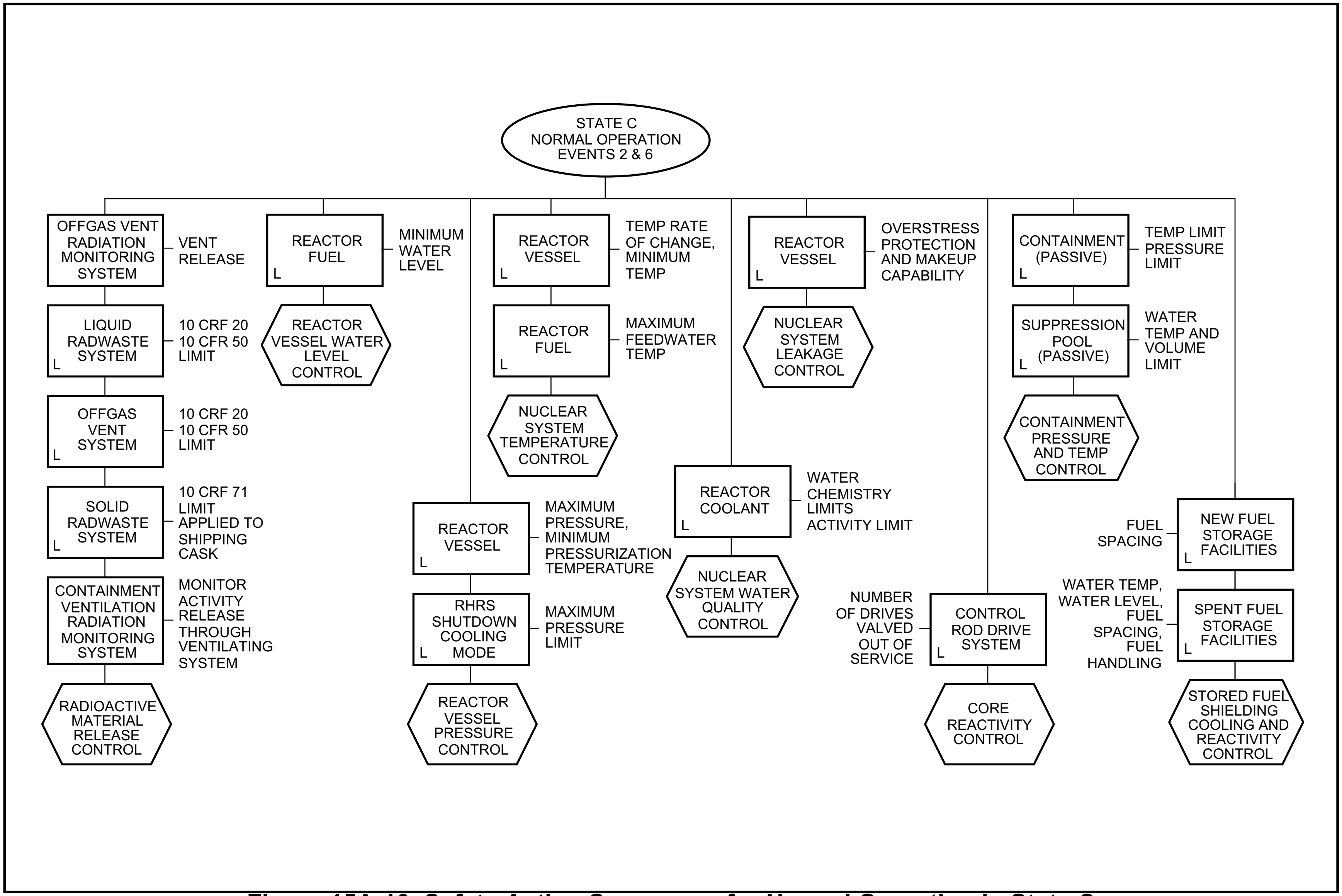


Figure 15A-10 Safety Action Sequences for Normal Operation in State C

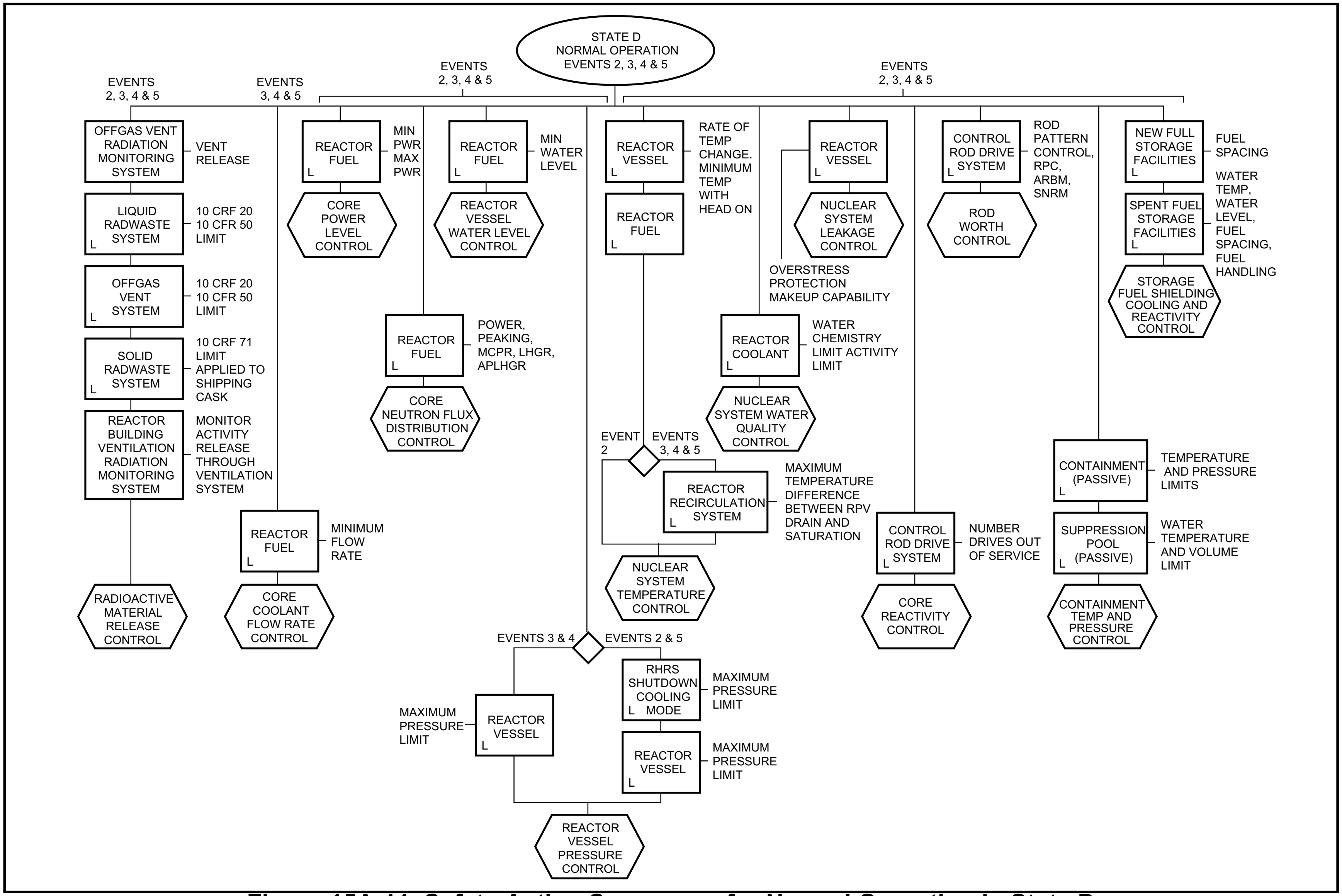


Figure 15A-11 Safety Action Sequences for Normal Operation in State D

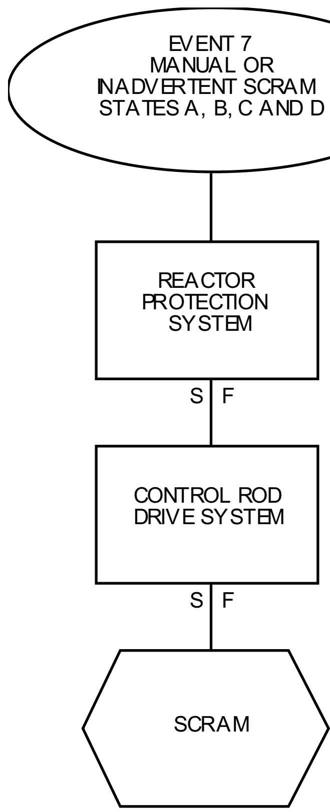


Figure 15A-12 Protection Sequence for Manual or Inadvertent Scram

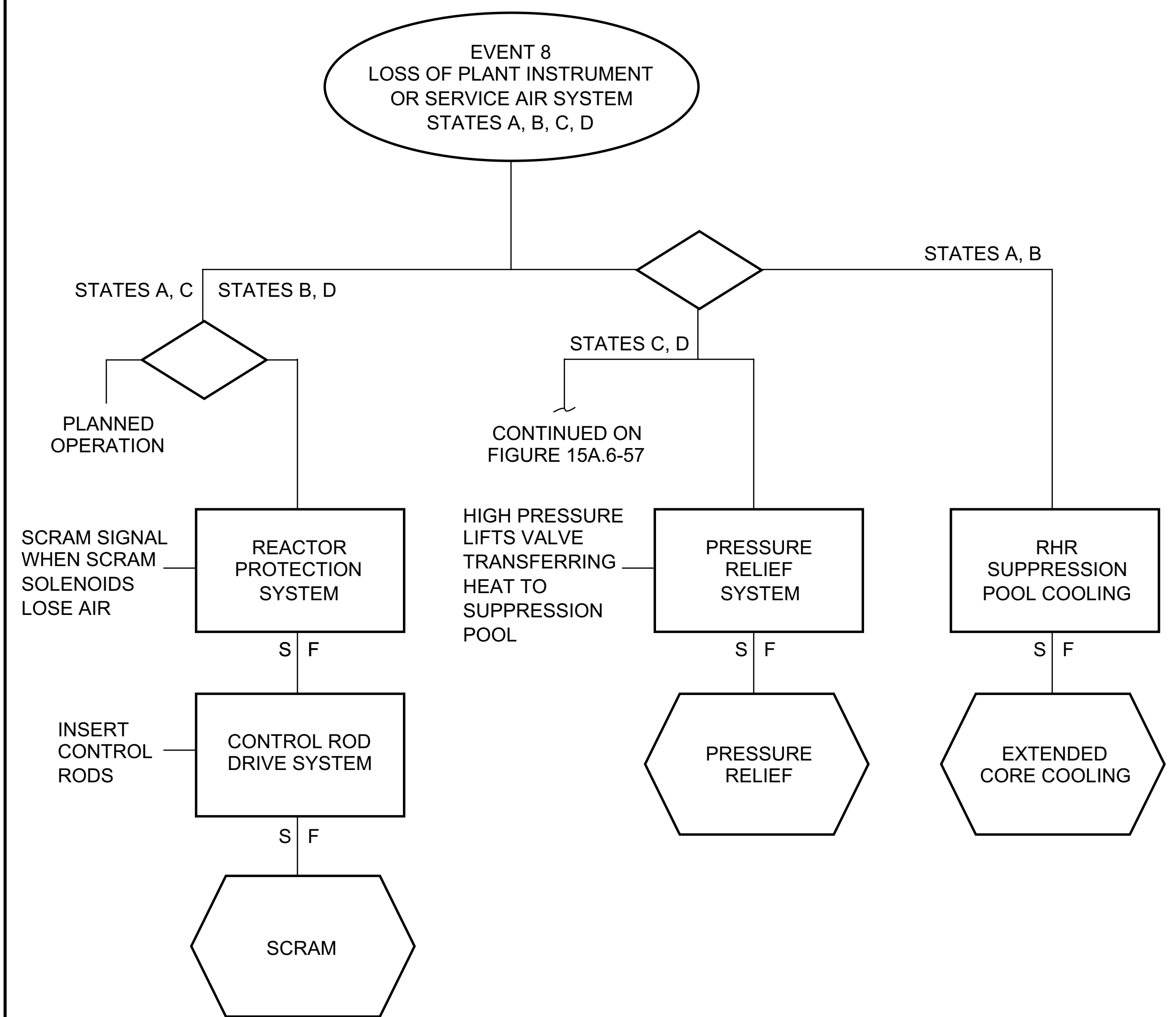


Figure 15A-13 Protection Sequence for Loss of Plant Instrument or Service Air System

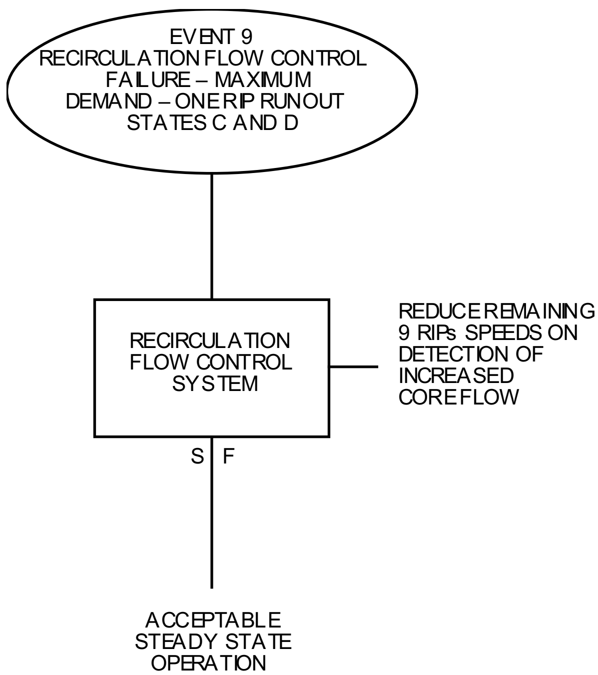


Figure 15A-14 Protection Sequence for Recirculation Flow Control Failure—Maximum Demand—One Reactor Internal Pump (RIP) Runout

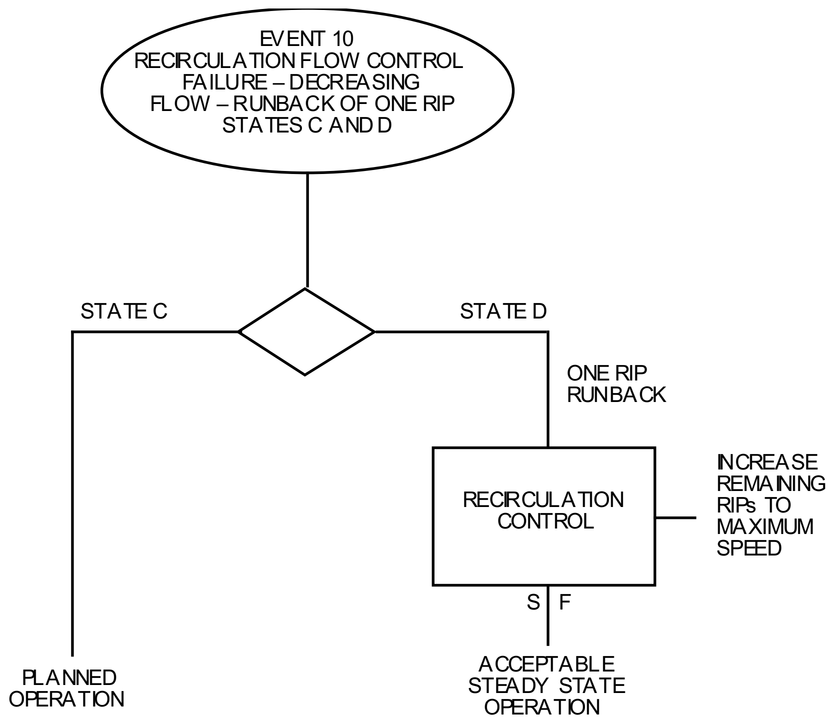


Figure 15A-15 Protection Sequence for Recirculation Flow Control Failure—Decreasing Flow Runback of One Reactor Internal Pump (RIP)

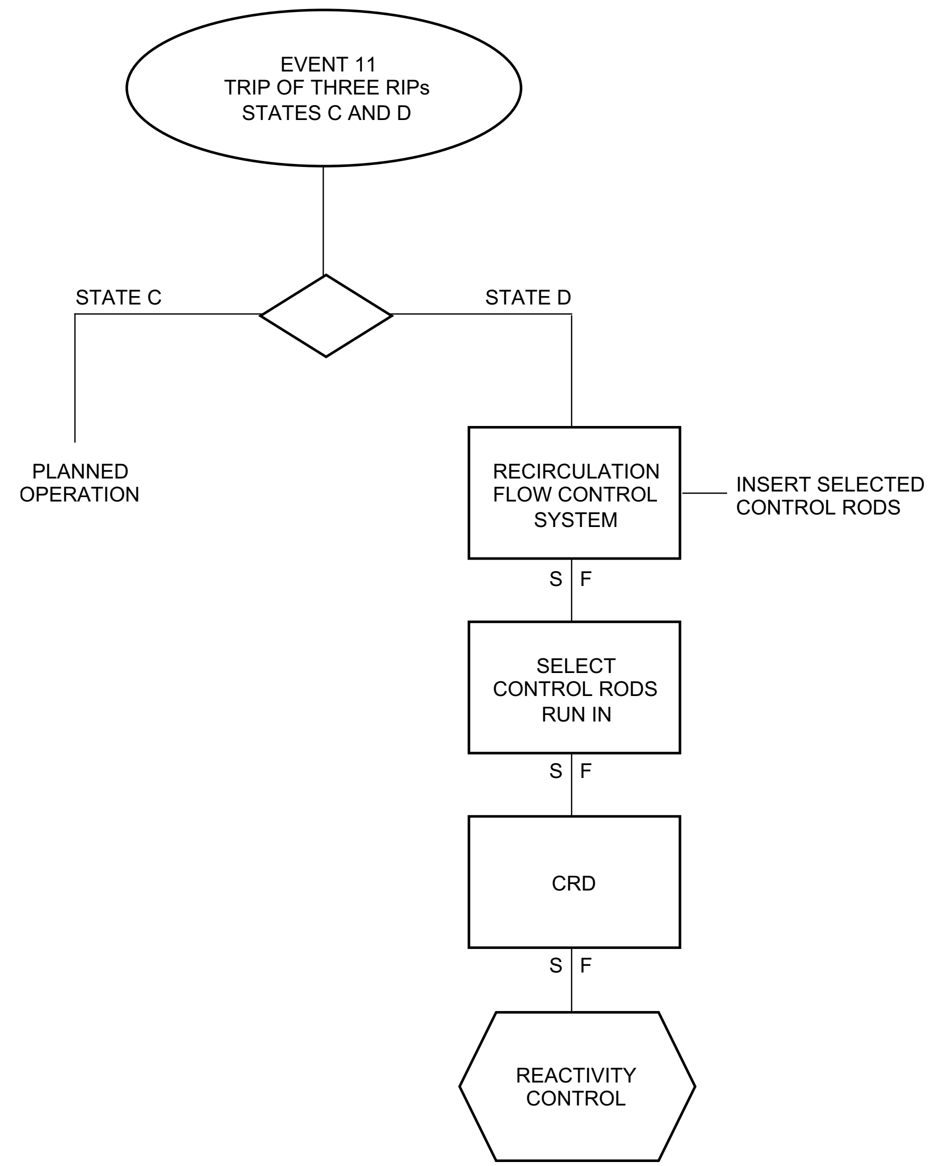


Figure 15A-16 Protection Sequence for Trip of Three Reactor Internal Pumps (RIPs)

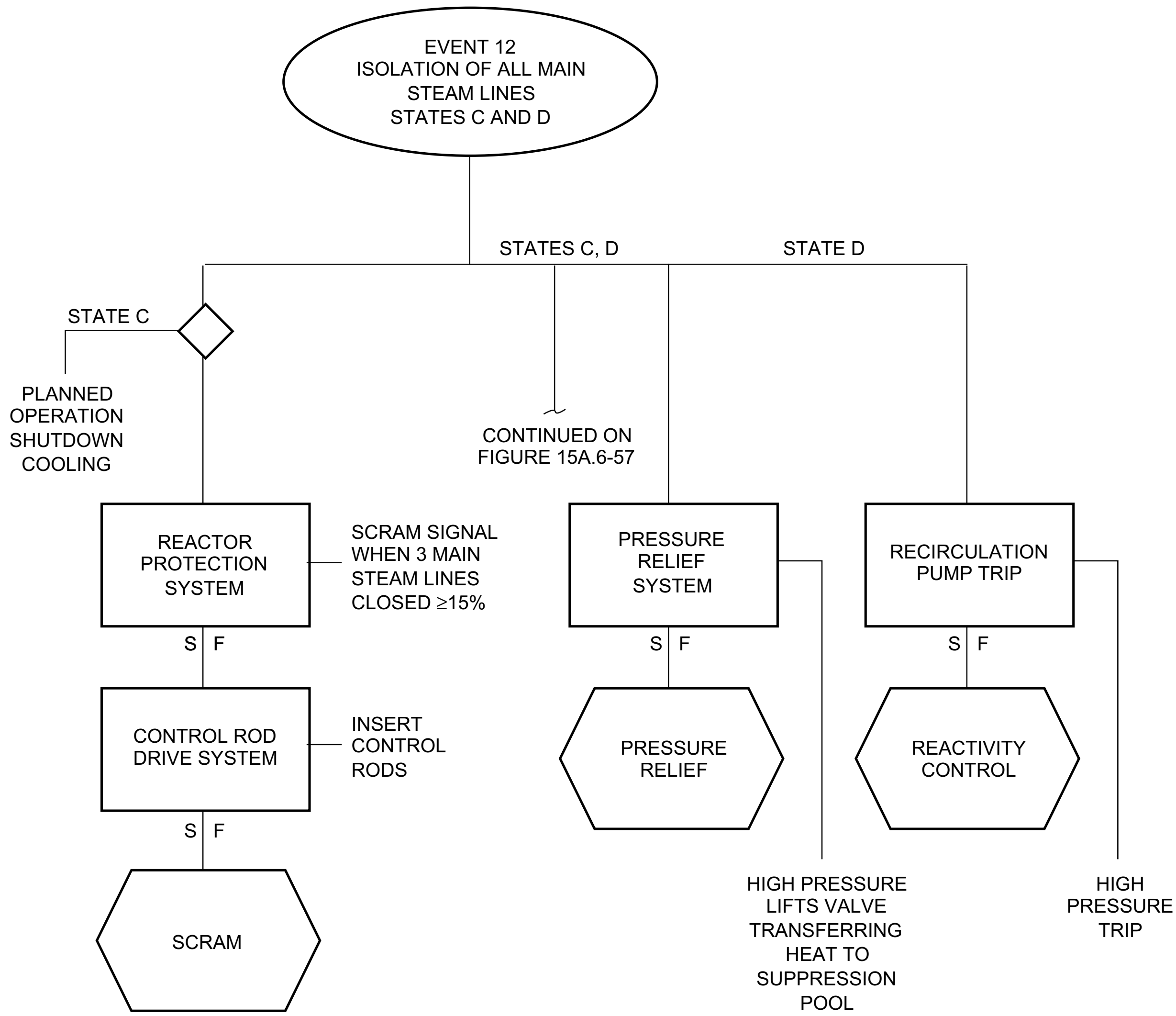


Figure 15A-17 Protection Sequences for Isolation of All Main Steamlines

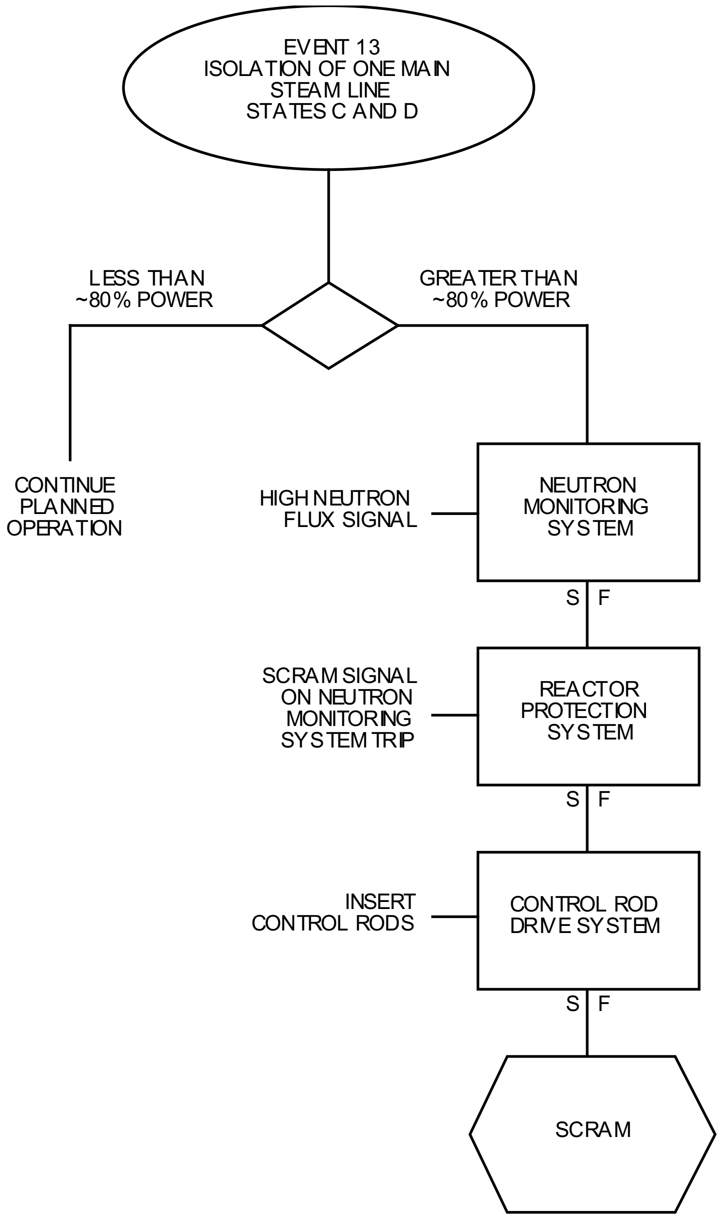


Figure 15A-18 Protection Sequences for Isolation of One Main Steamline

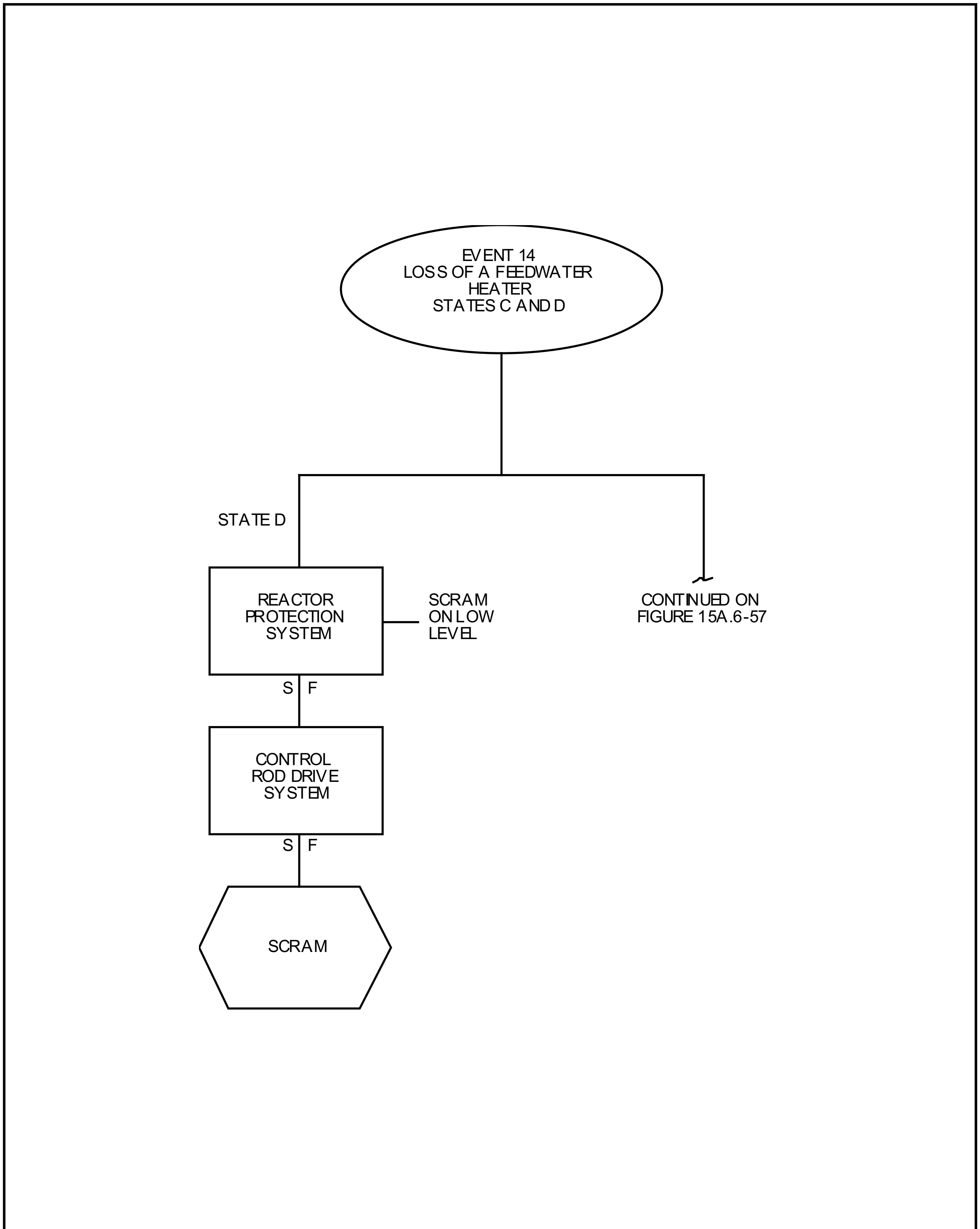


Figure 15A-19 Protection Sequence for Loss of All Feedwater Flow

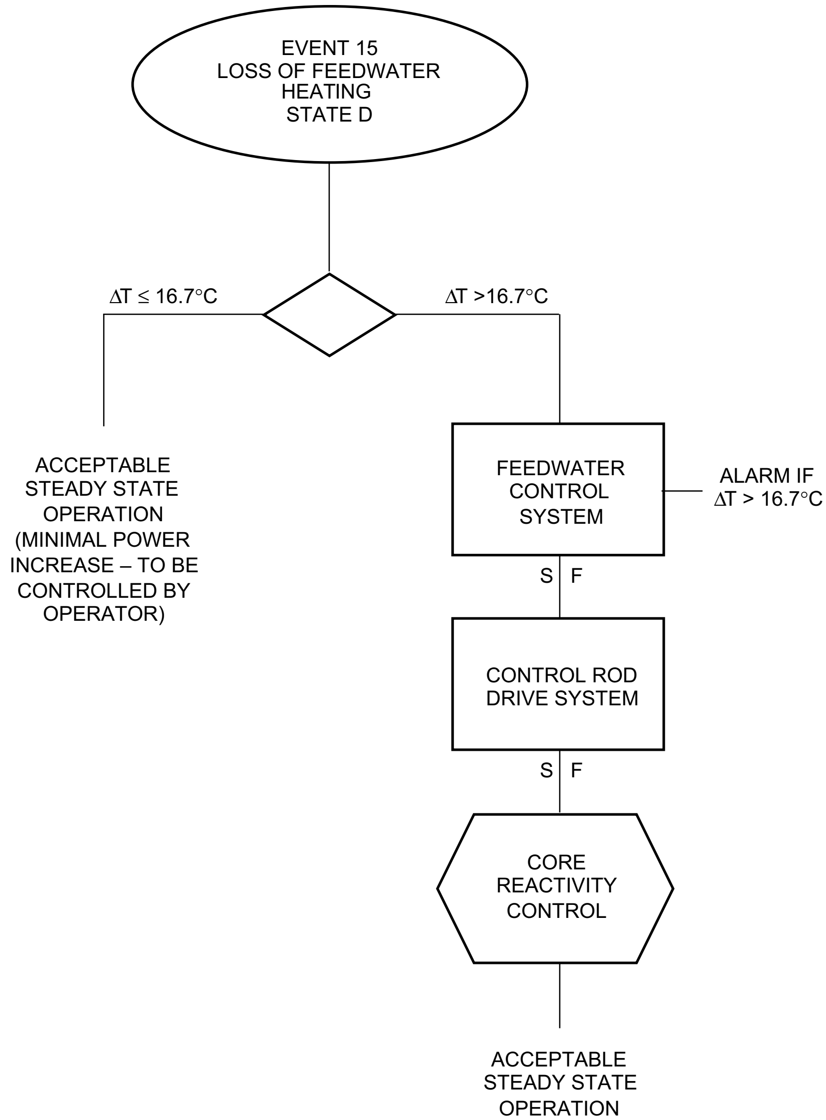


Figure 15A-20 Protection Sequence for a Loss of Feedwater Heating

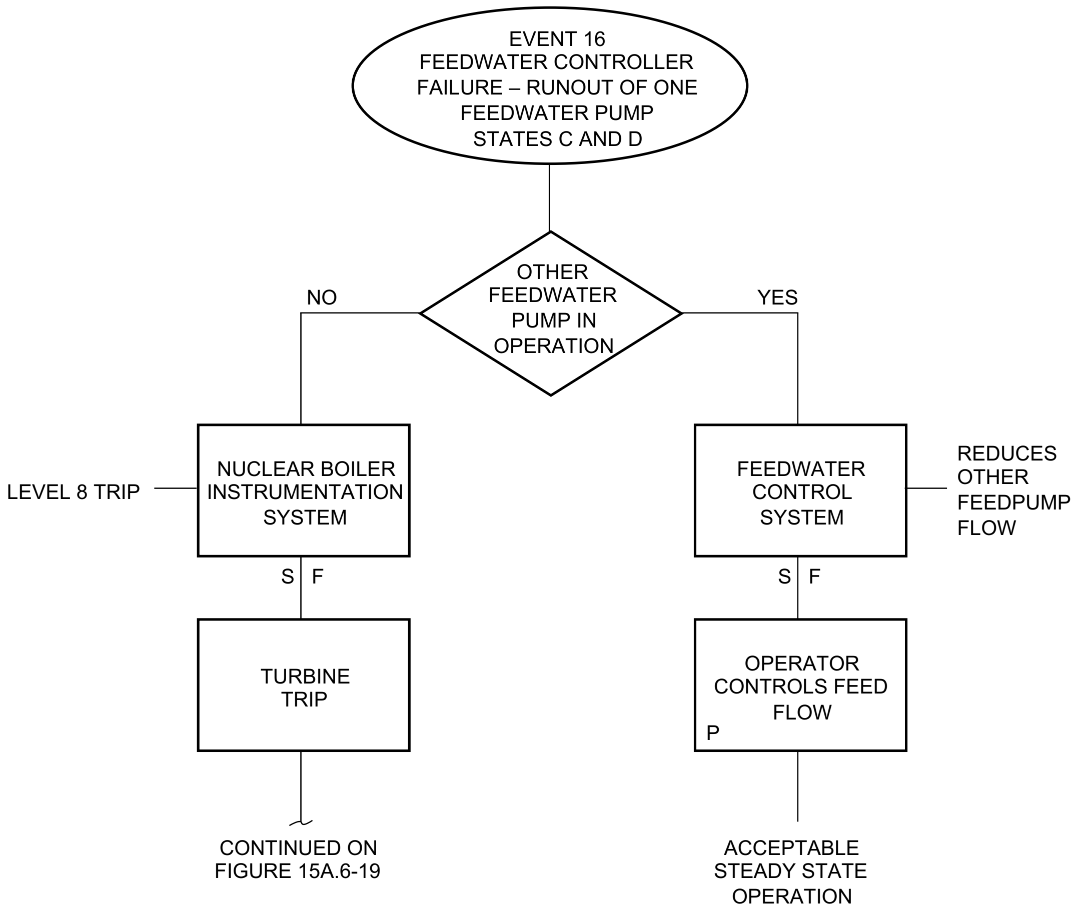


Figure 15A-21 Protection Sequence for Feedwater Controller Failure—Runout of One Feedwater Pump

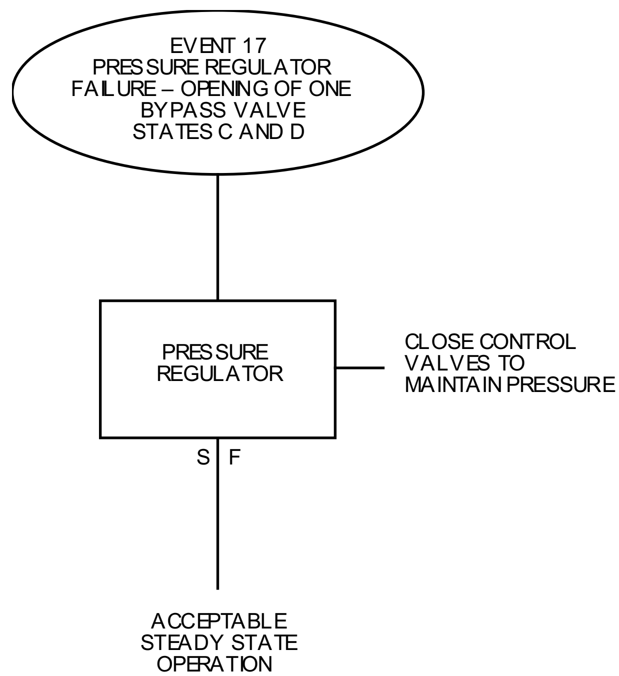


Figure 15A-22 Pressure Regulator Failure—Opening of One Bypass Valve

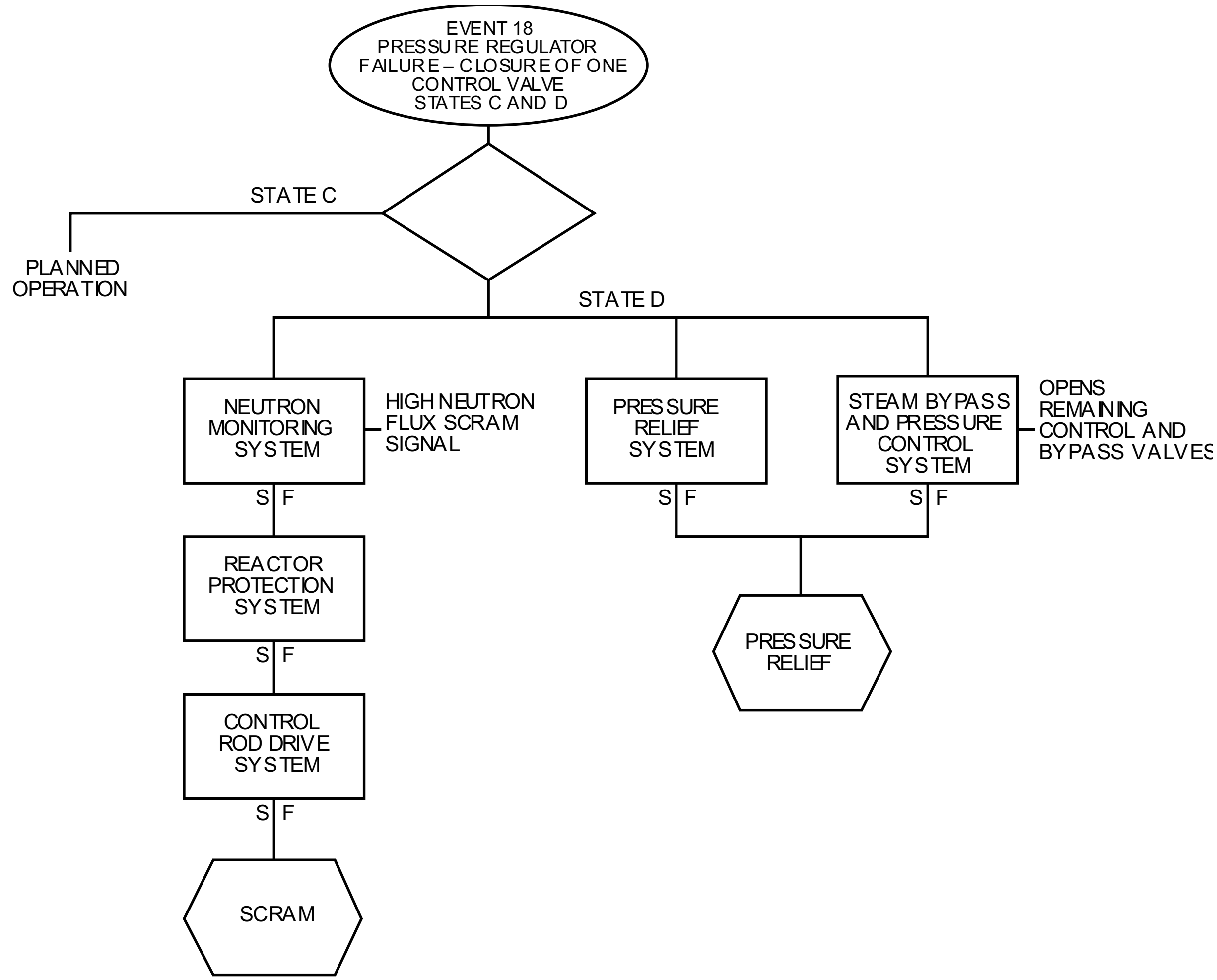


Figure 15A-23 Pressure Regulator Failure—Closure of One Control Valve

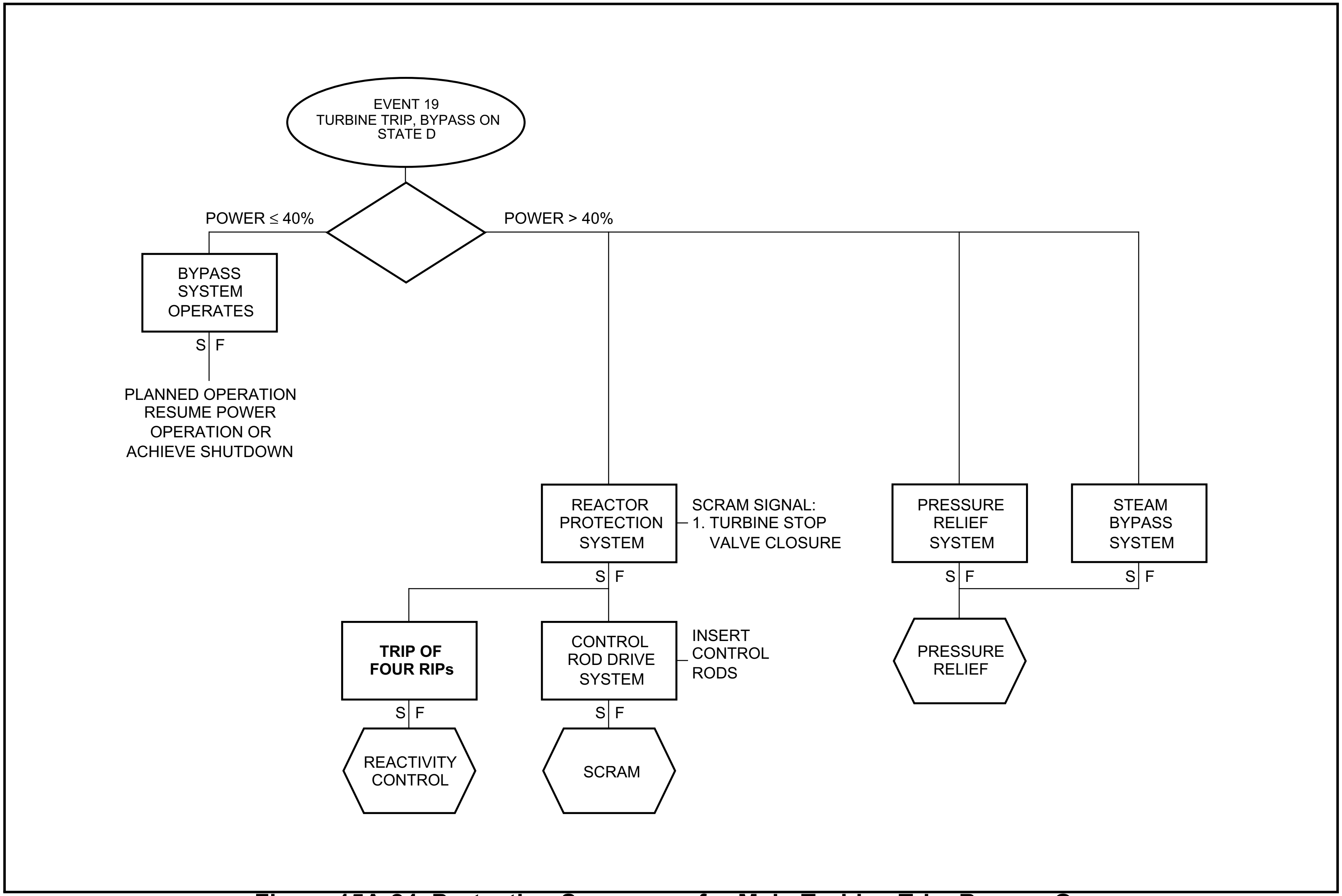


Figure 15A-24 Protection Sequences for Main Turbine Trip, Bypass On

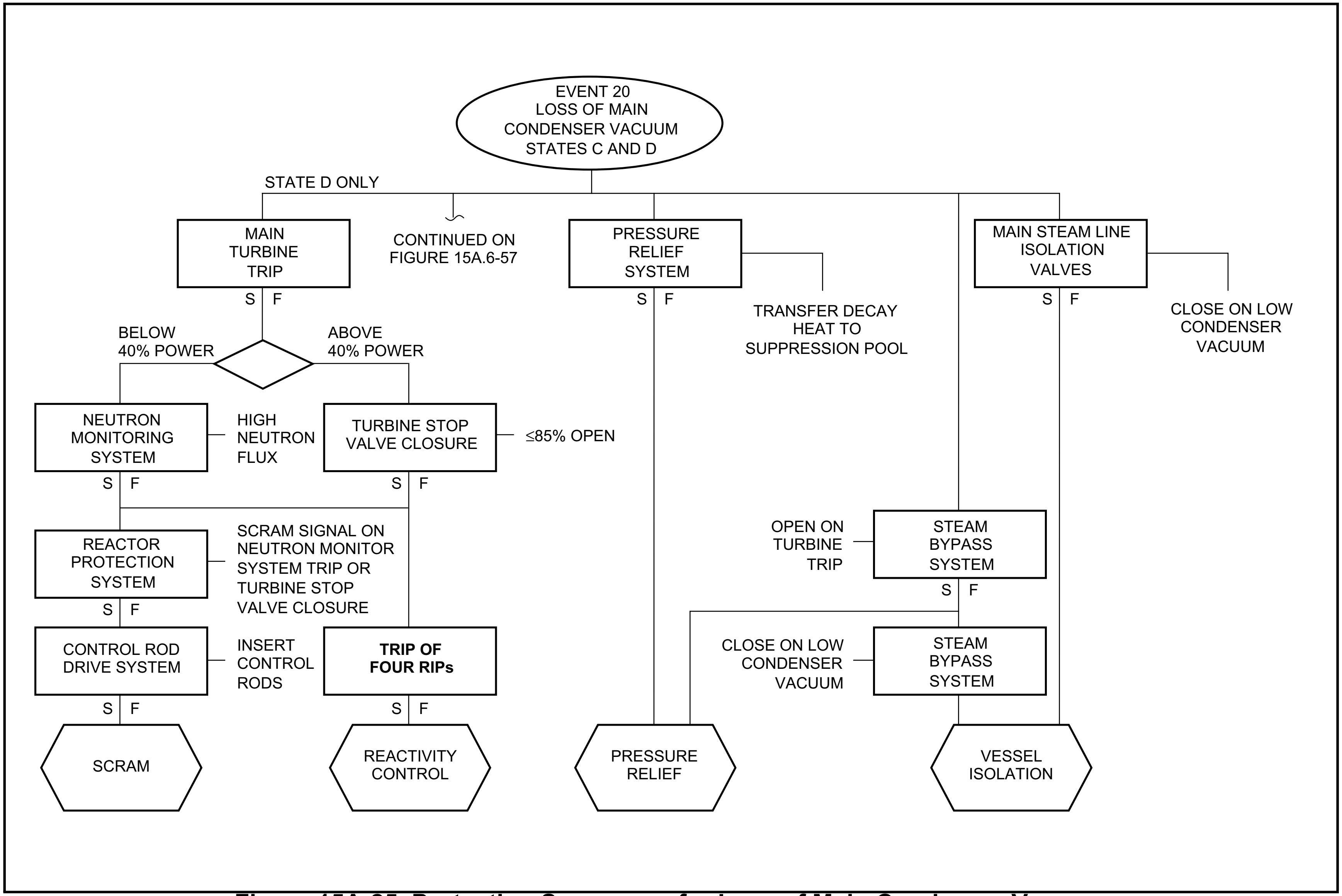


Figure 15A-25 Protection Sequences for Loss of Main Condenser Vacuum

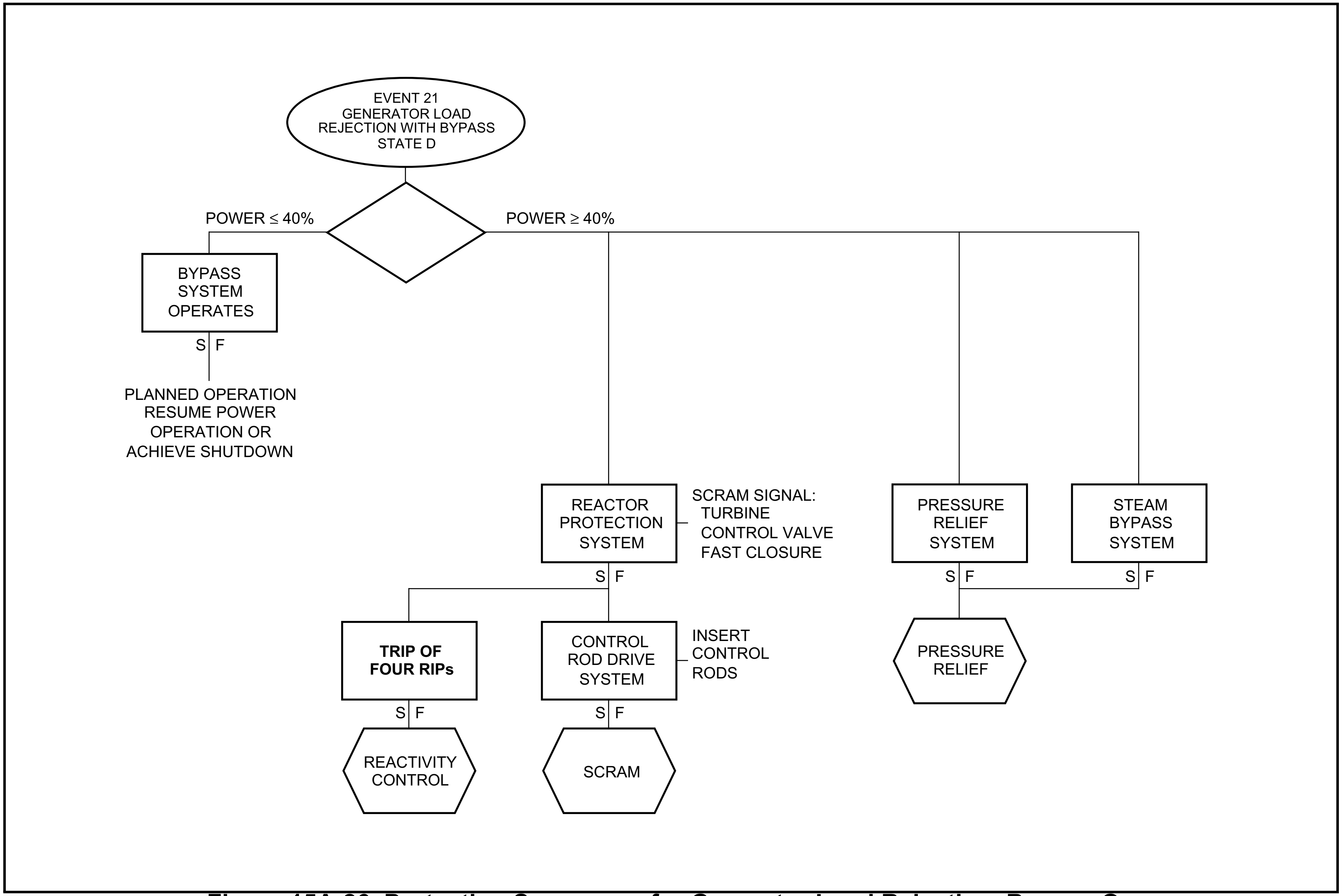


Figure 15A-26 Protection Sequences for Generator Load Rejection, Bypass On

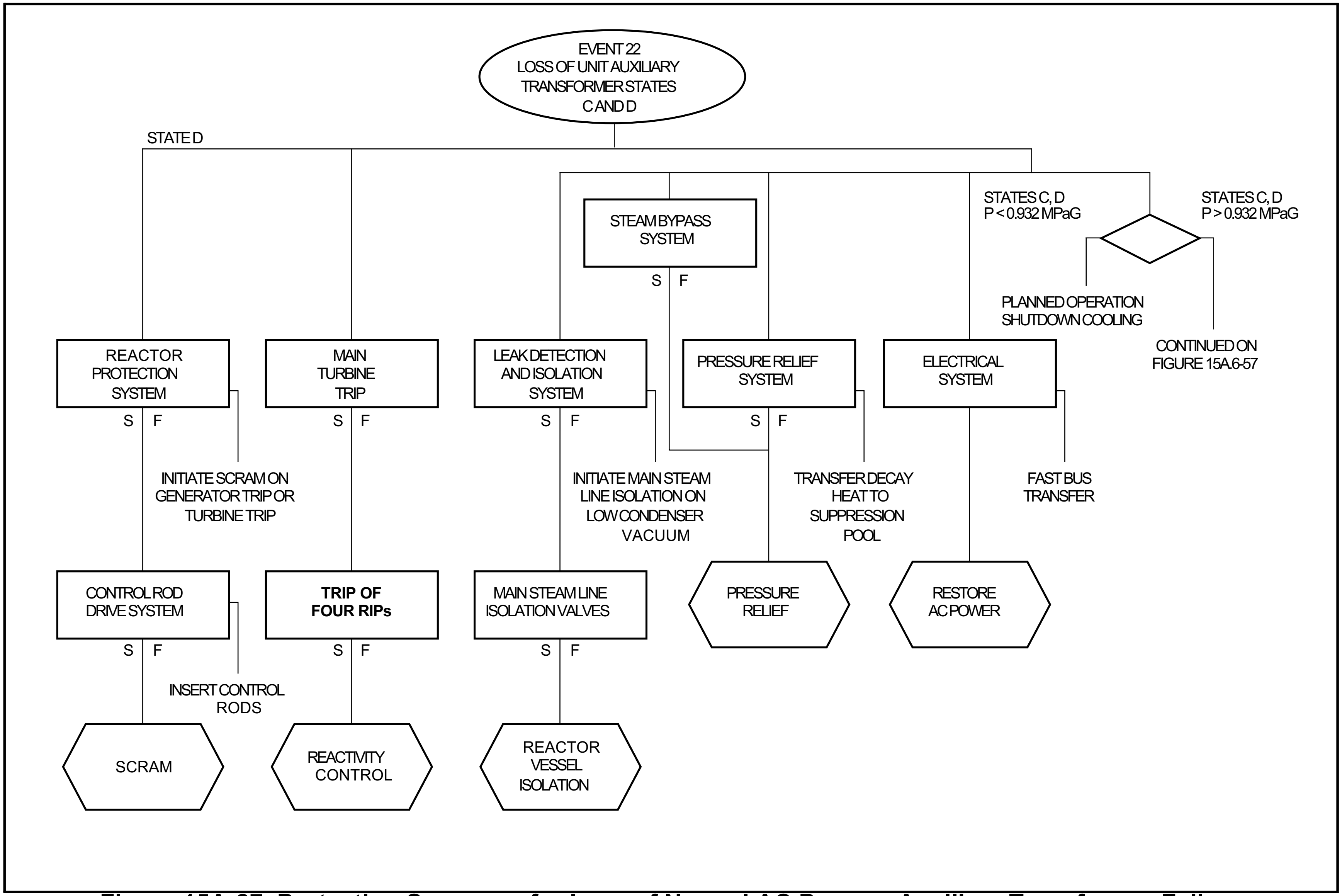


Figure 15A-27 Protection Sequence for Loss of Normal AC Power—Auxiliary Transformer Failure

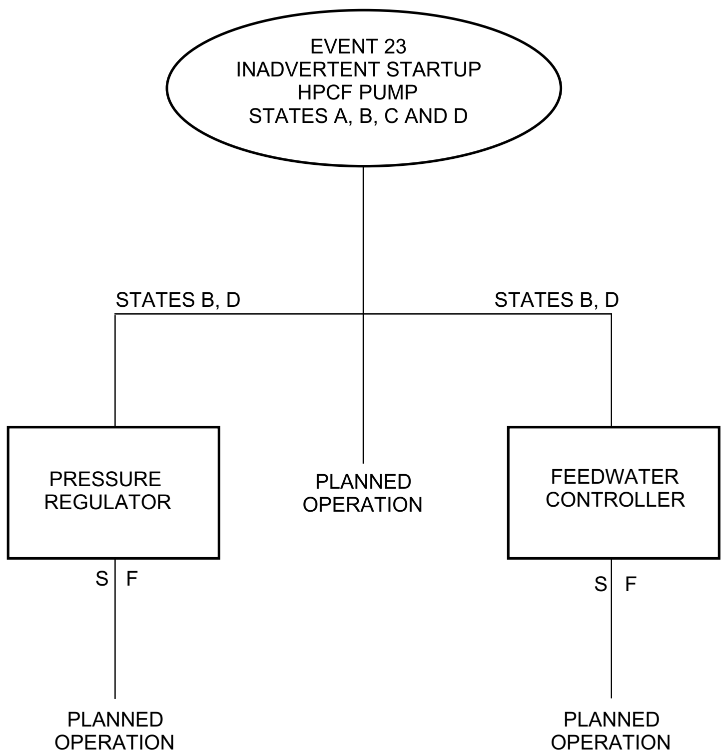


Figure 15A-28 Protection Sequence for Inadvertent Startup of HPCF Pumps

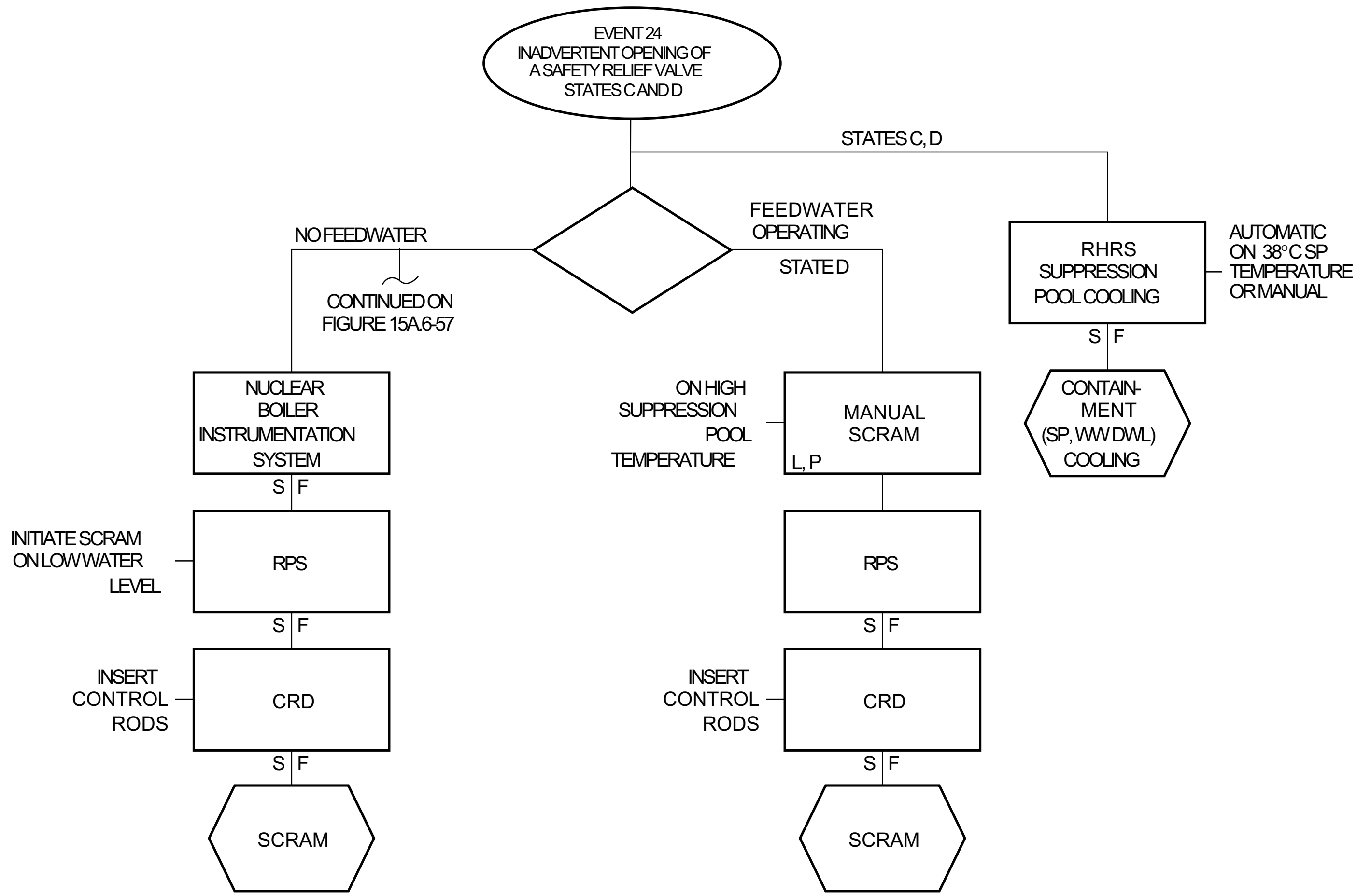


Figure 15A-29. Protection Sequences for Inadvertent Opening of a Safety Relief Valve

Figure 15A-29 Protection Sequences for Inadvertent Opening of a Safety Relief Valve

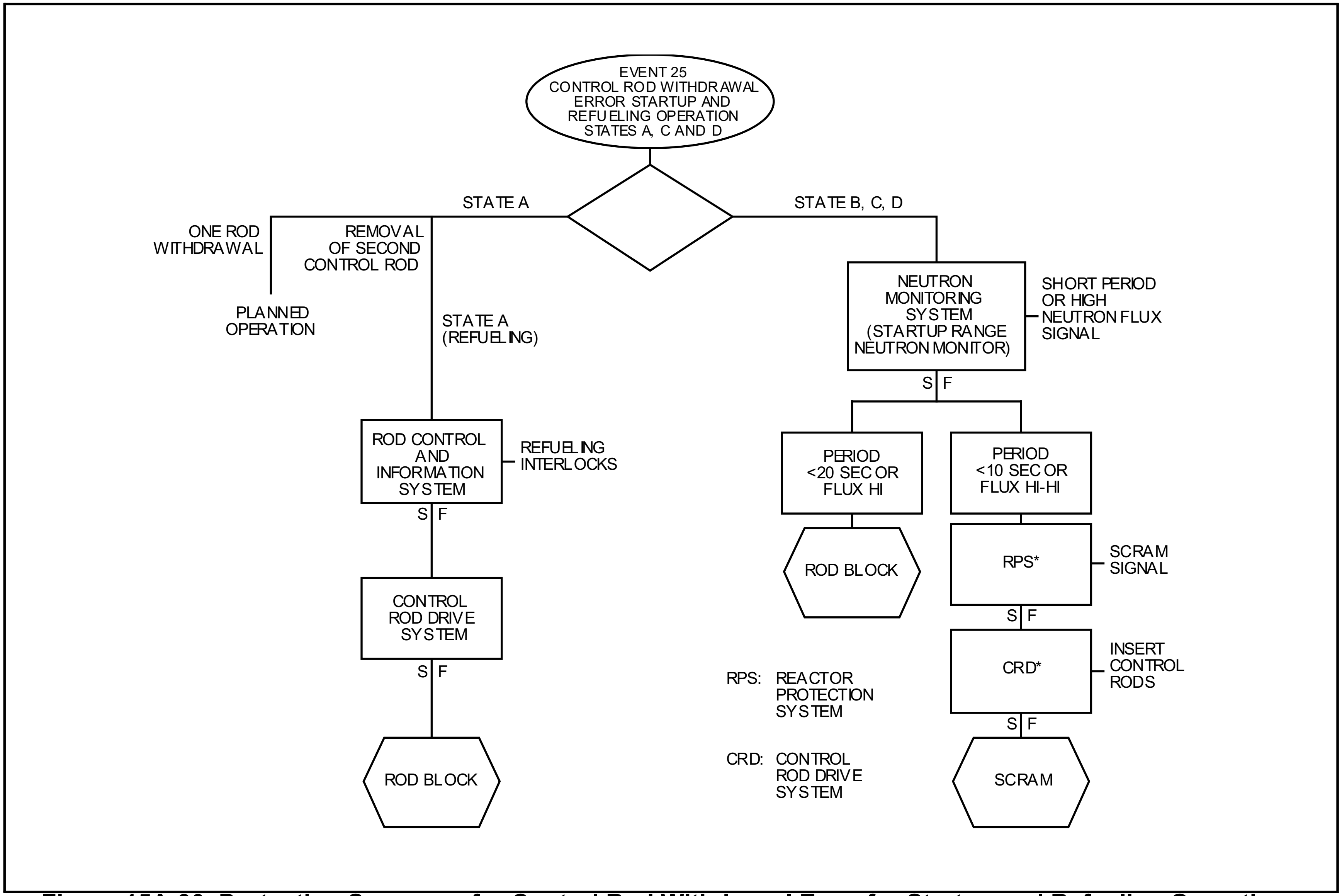


Figure 15A-30 Protection Sequence for Control Rod Withdrawal Error for Startup and Refueling Operations

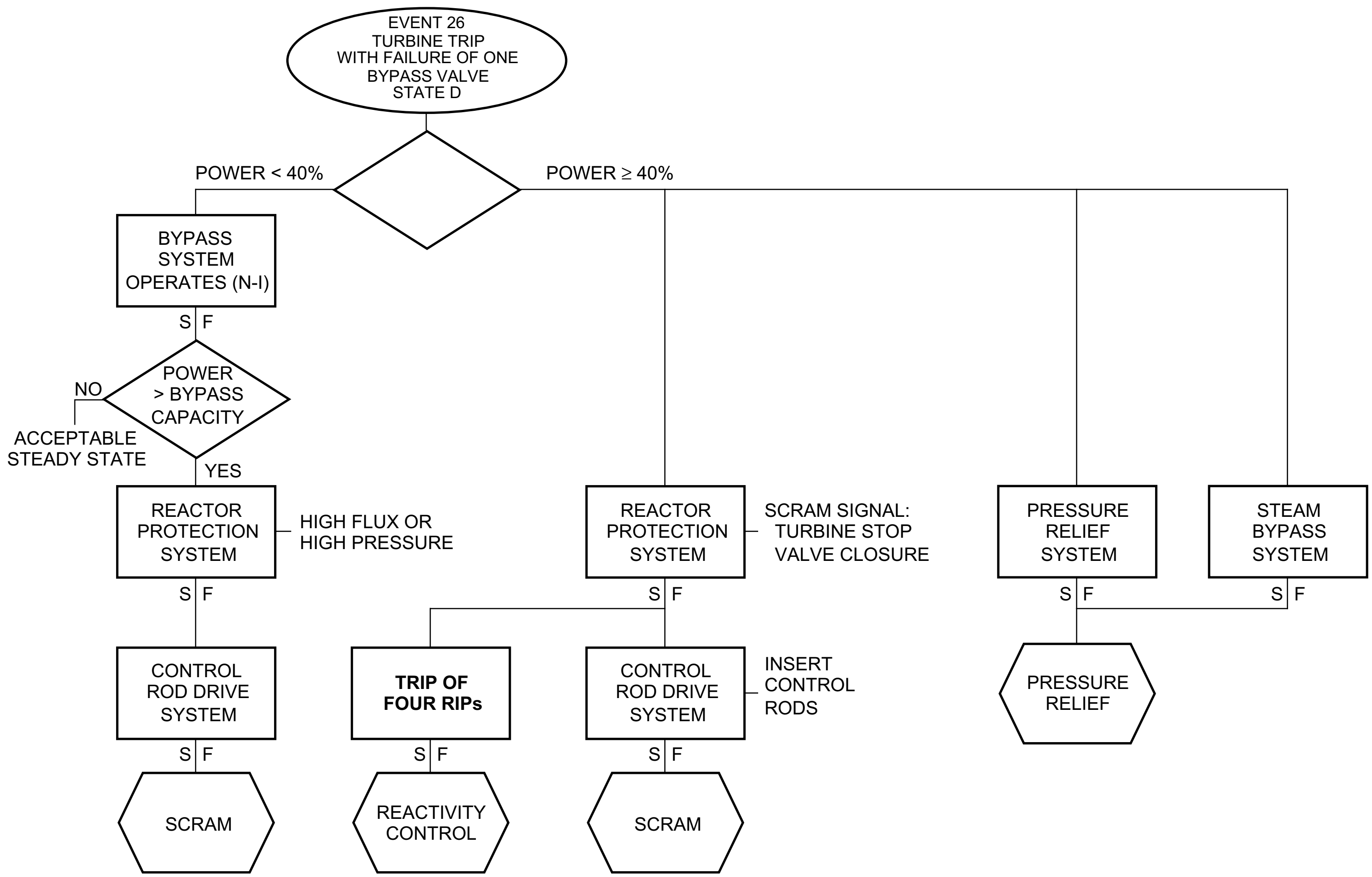


Figure 15A-31 Protection Sequences for Main Turbine Trip with Failure of One Bypass Valve

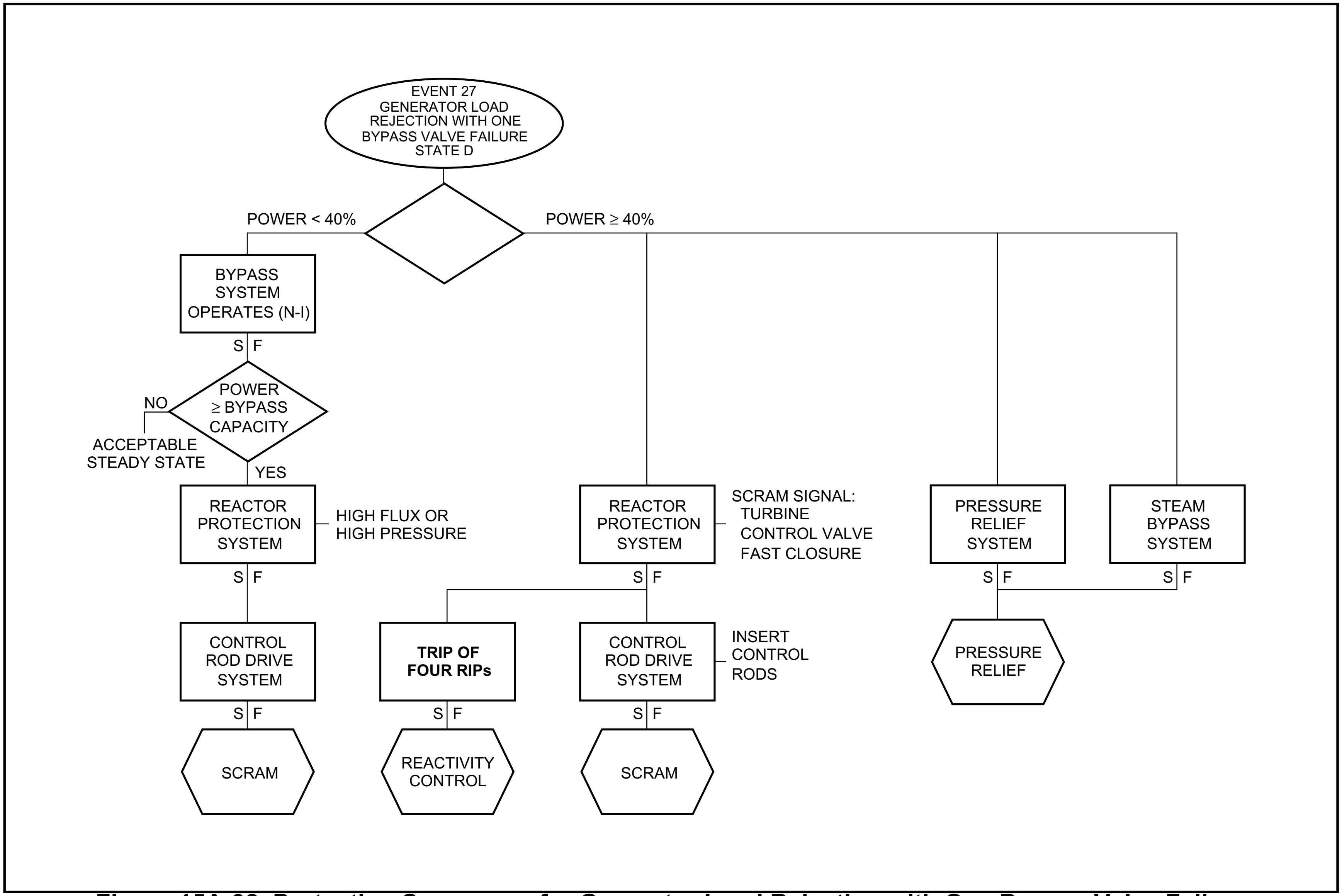


Figure 15A-32 Protection Sequences for Generator Load Rejection with One Bypass Valve Failure

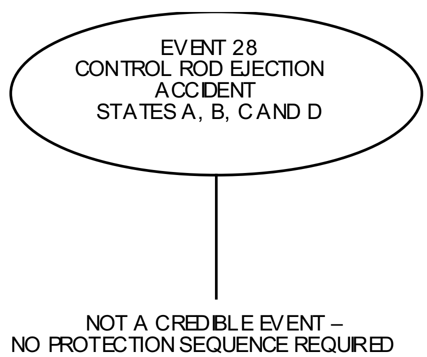


Figure 15A-33 Protection Sequence for Control Rod Ejection Accident

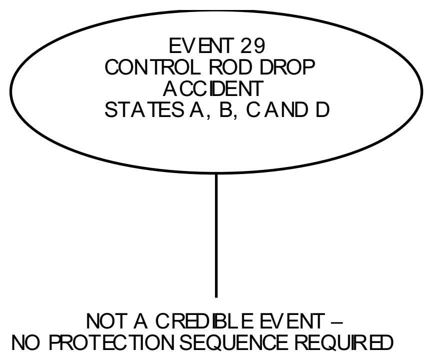


Figure 15A-34 Protection Sequence for Control Rod Drop Accident

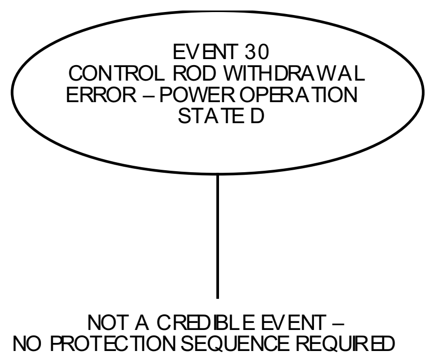


Figure 15A-35 Protection Sequence for a Control Rod Withdrawal Error During Power Operation

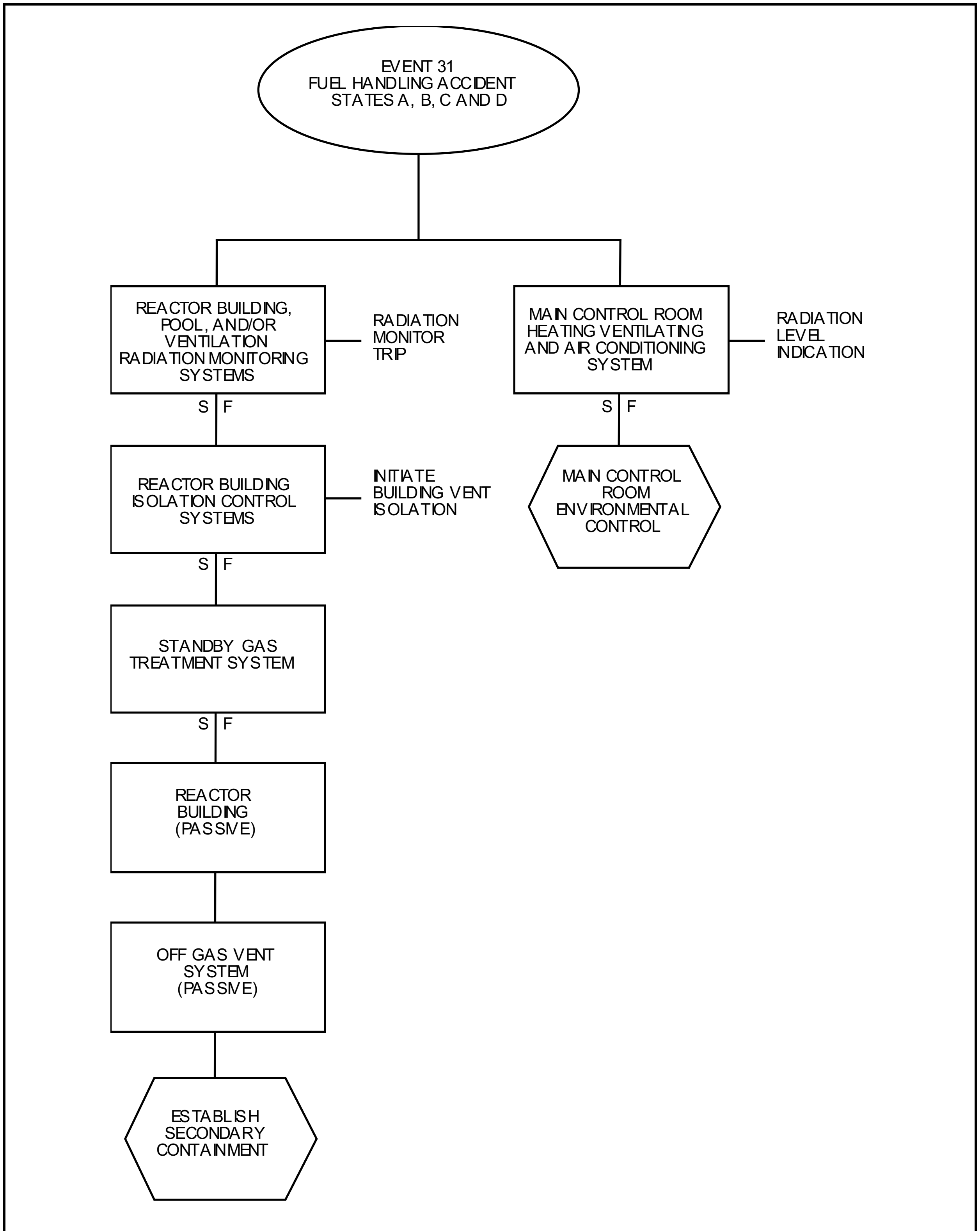


Figure 15A-36 Protection Sequences for Fuel-Handling Accident

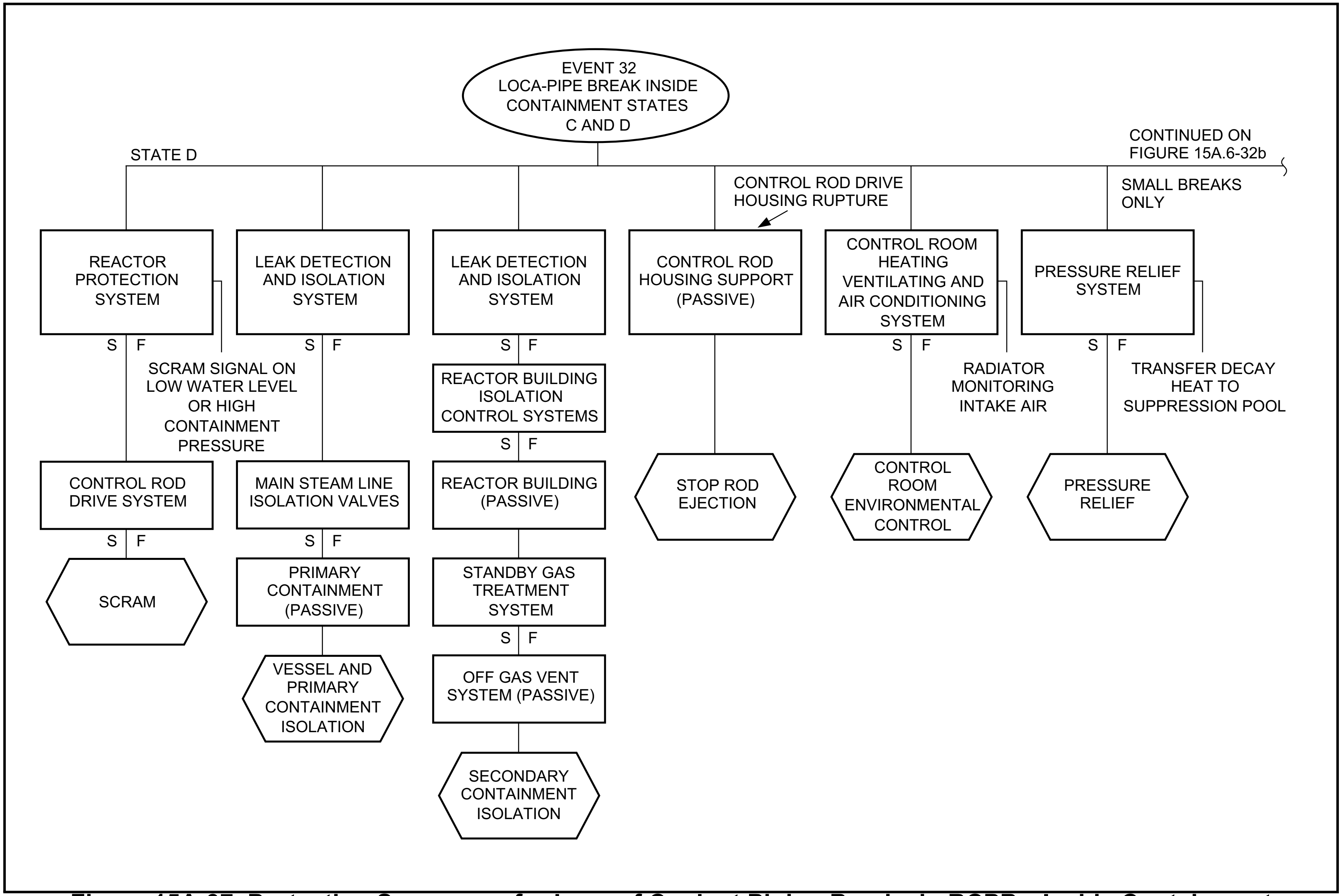


Figure 15A-37 Protection Sequences for Loss of Coolant Piping Breaks in RCPB—Inside Containment

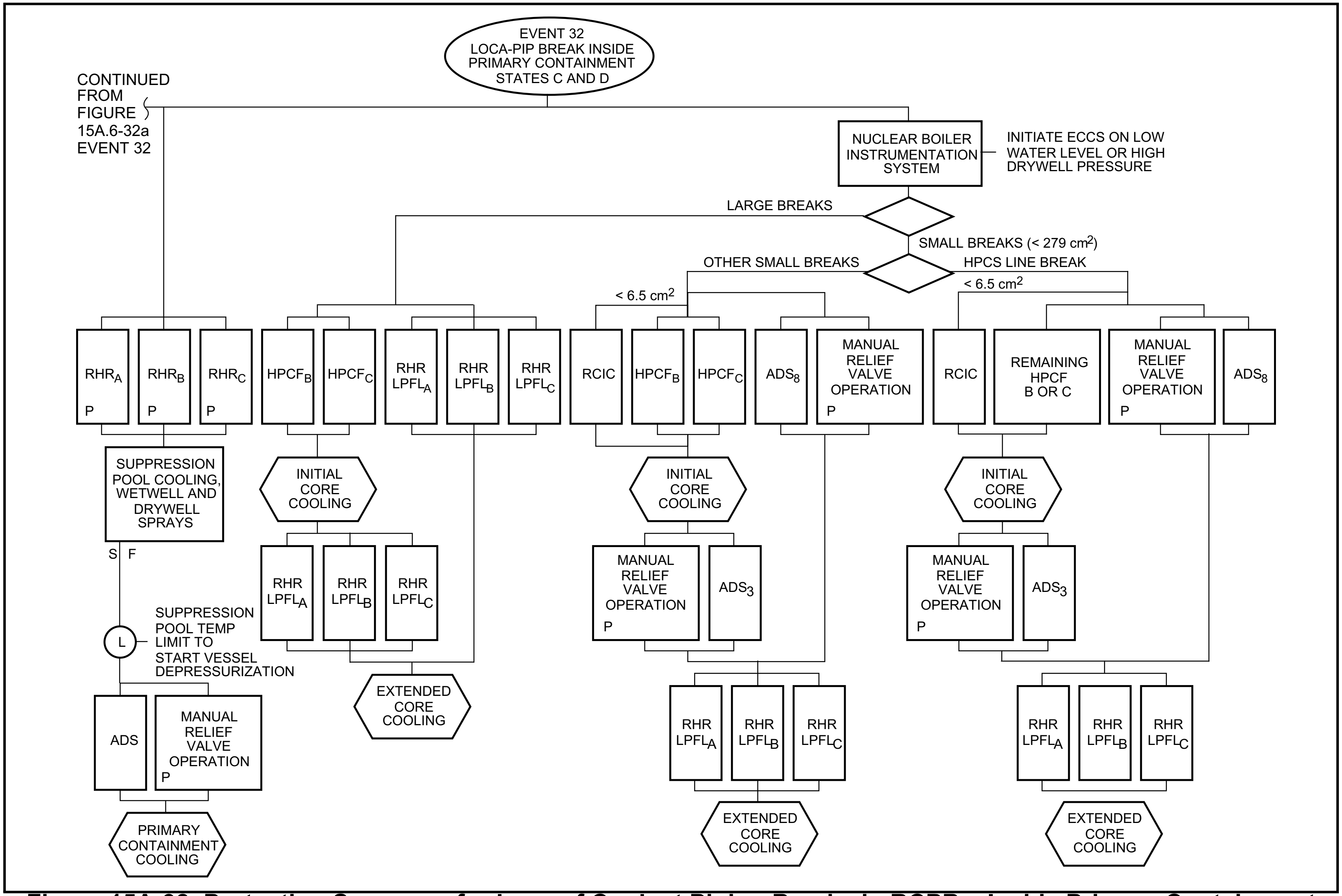
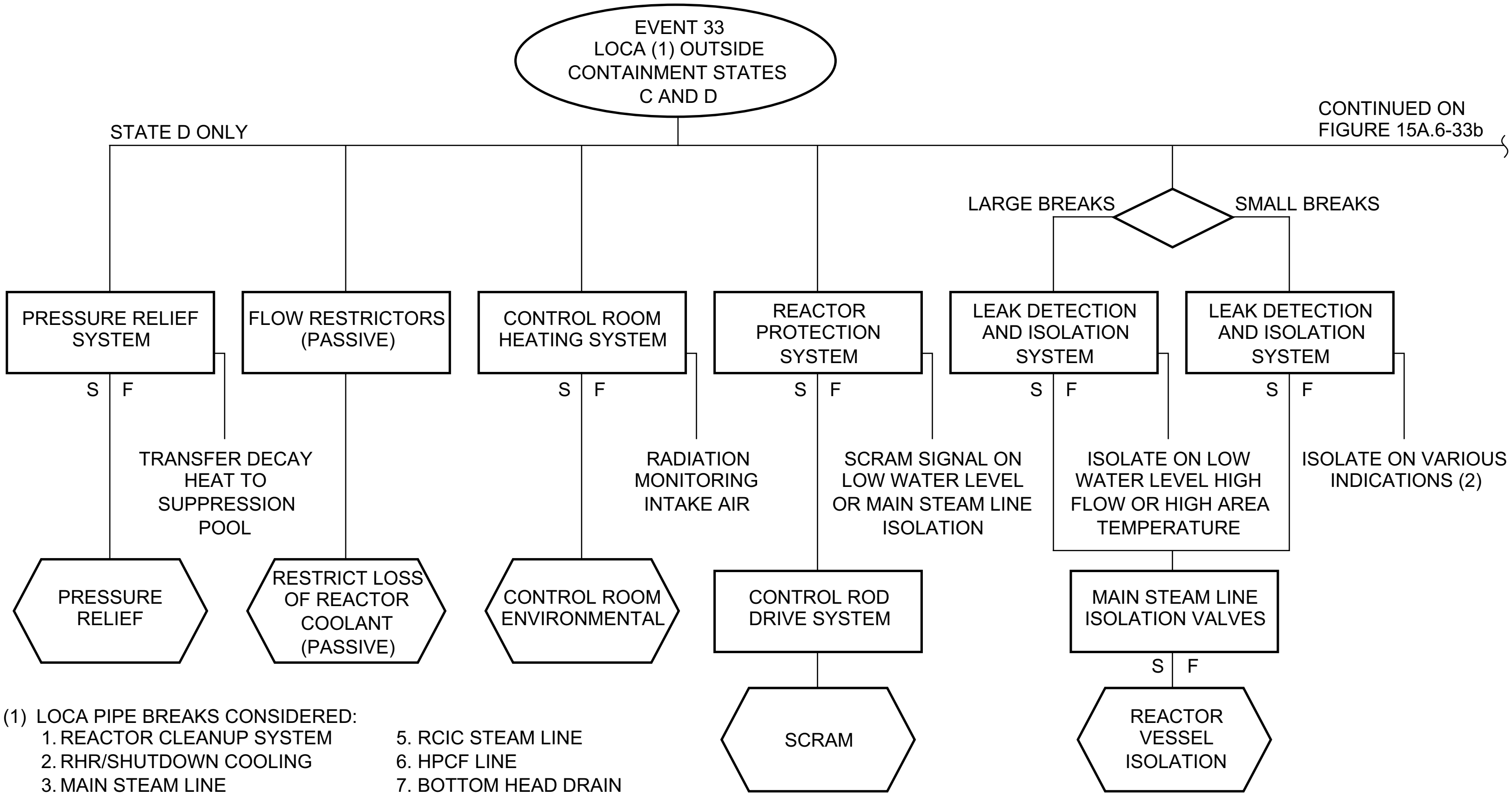


Figure 15A-38 Protection Sequence for Loss of Coolant Piping Breaks in RCPB – Inside Primary Containment



(1) LOCA PIPE BREAKS CONSIDERED:
1. REACTOR CLEANUP SYSTEM
2. RHR/SHUTDOWN COOLING
3. MAIN STEAM LINE
4. FEEDWATER LINE
5. RCIC STEAM LINE
6. HPCF LINE
7. BOTTOM HEAD DRAIN

(2) VARIOUS INDICATIONS:
1. FEED SIGNALS TO PUMPS
2. FEED TEMPERATURE
3. SPACE TEMPERATURE
4. FLOW INDICATIONS
5. REACTOR VESSEL WATER FLOW
6. FEEDWATER FLOW – STEAM LINE
7. HOT WELL LEVEL
8. VISUAL INSPECTION
9. LEAKAGE INDICATIONS

Figure 15A-39 Protection Sequences for Liquid and Steam, Large and Small Piping Breaks Outside Containment

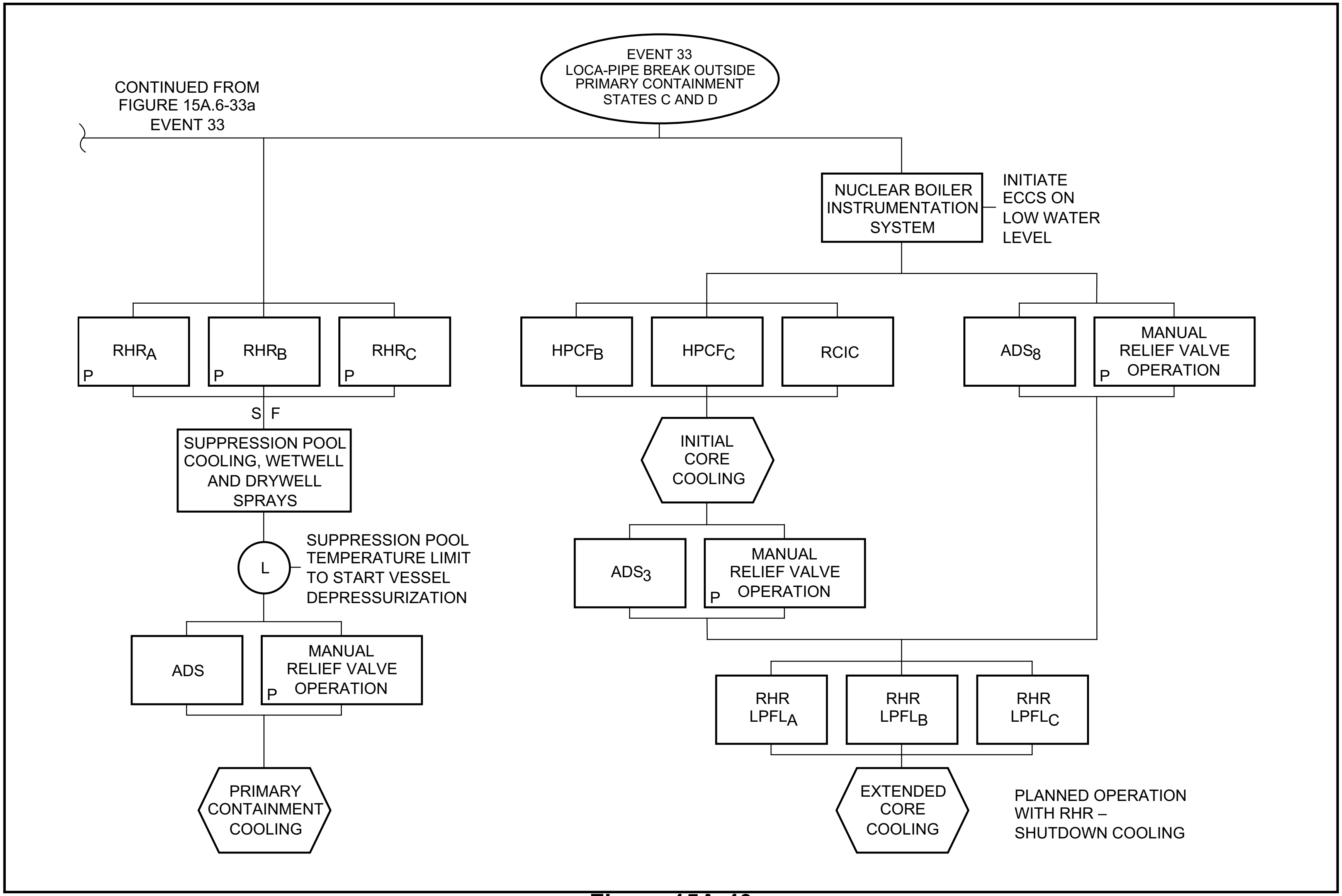


Figure 15A-40

Protection Sequence for Liquid and Steam, Large and Small Piping Breaks Outside Primary Containment

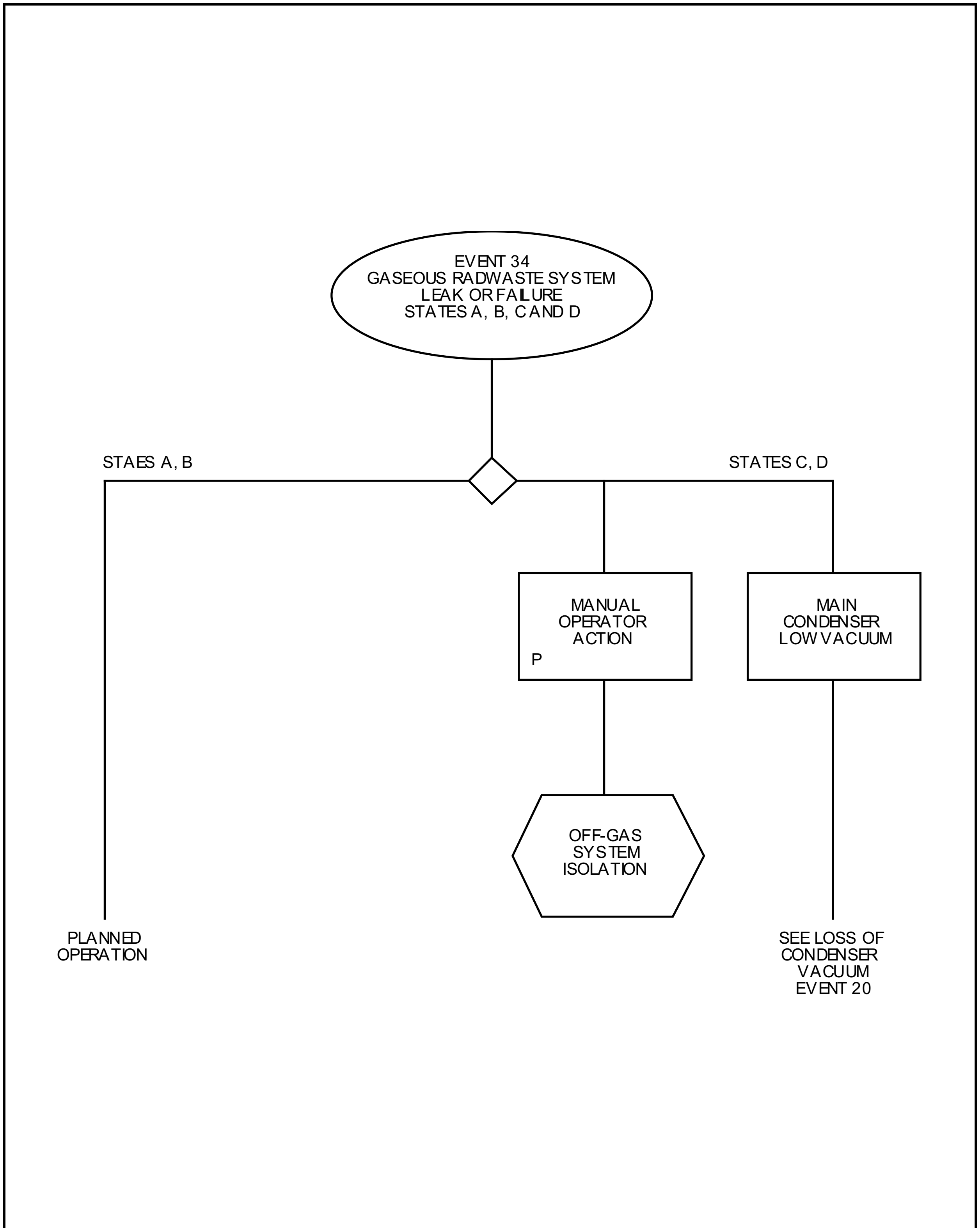


Figure 15A-41 Protection Sequence for Gaseous Radwaste System Leak or Failure

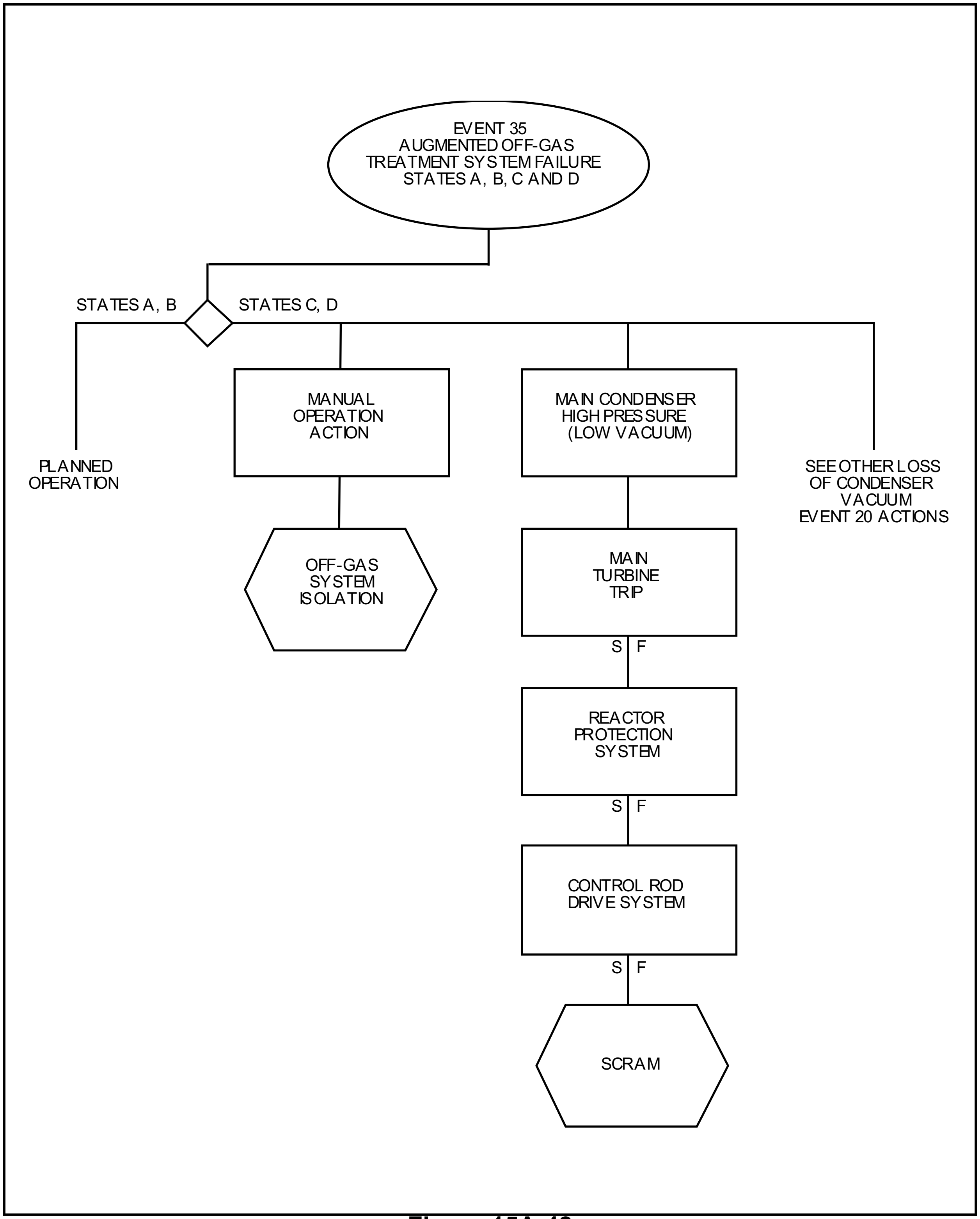


Figure 15A-42
Protection Sequence for Augmented Offgas Treatment System Failure

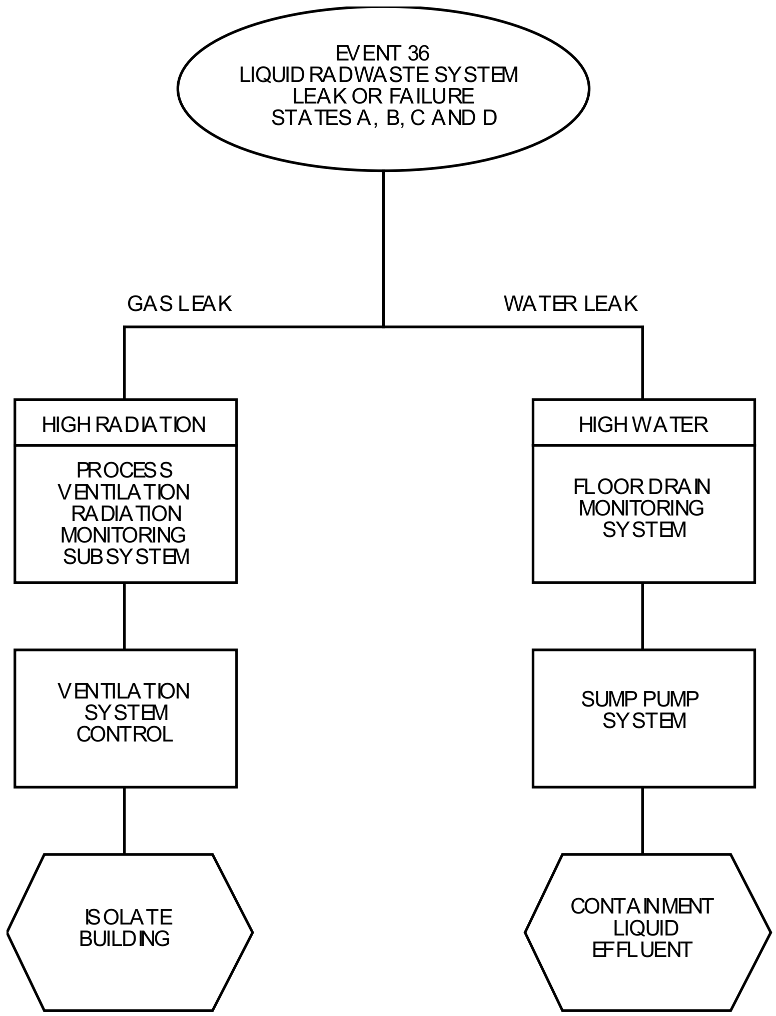


Figure 15A-43 Protection Sequence for Liquid Radwaste System Leak or Failure

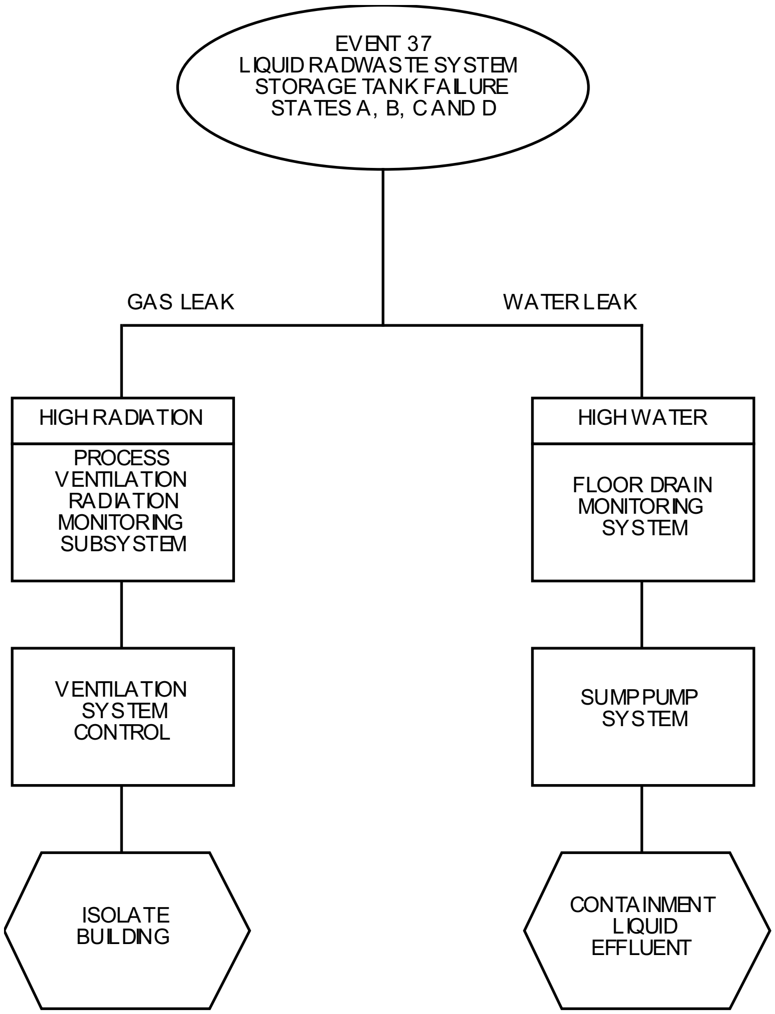


Figure 15A-44
Protection Sequence for Liquid Radwaste System Storage Tank Failure

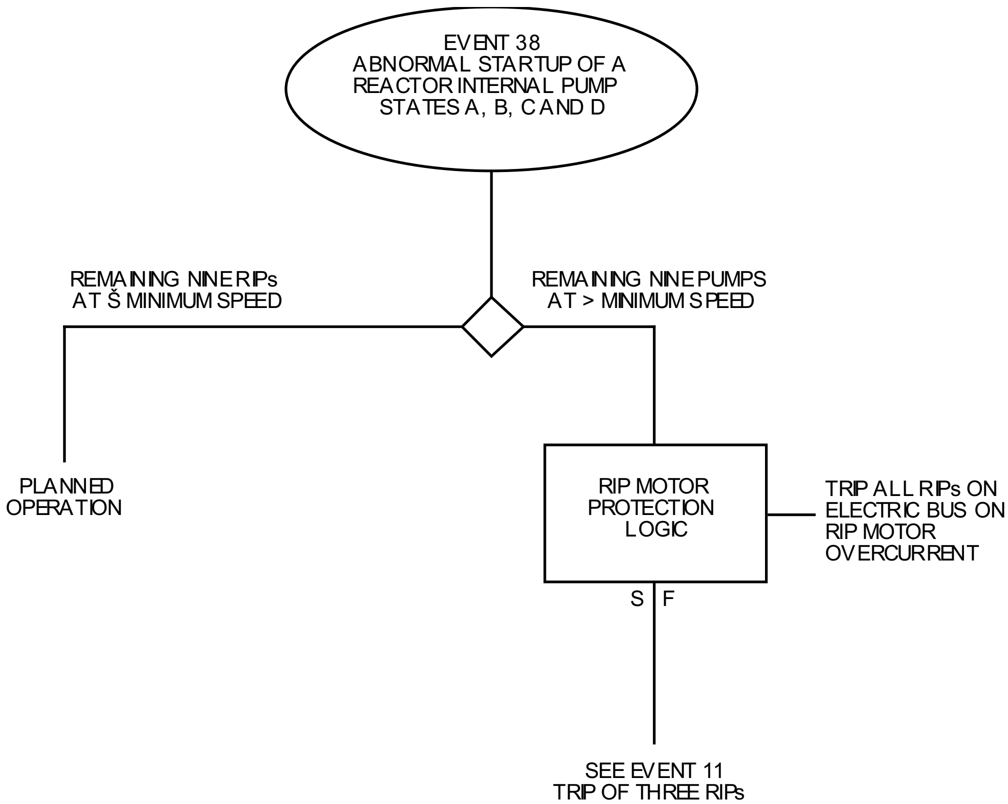


Figure 15A-45
Protection Sequence for Abnormal Startup of a Reactor Internal Pump

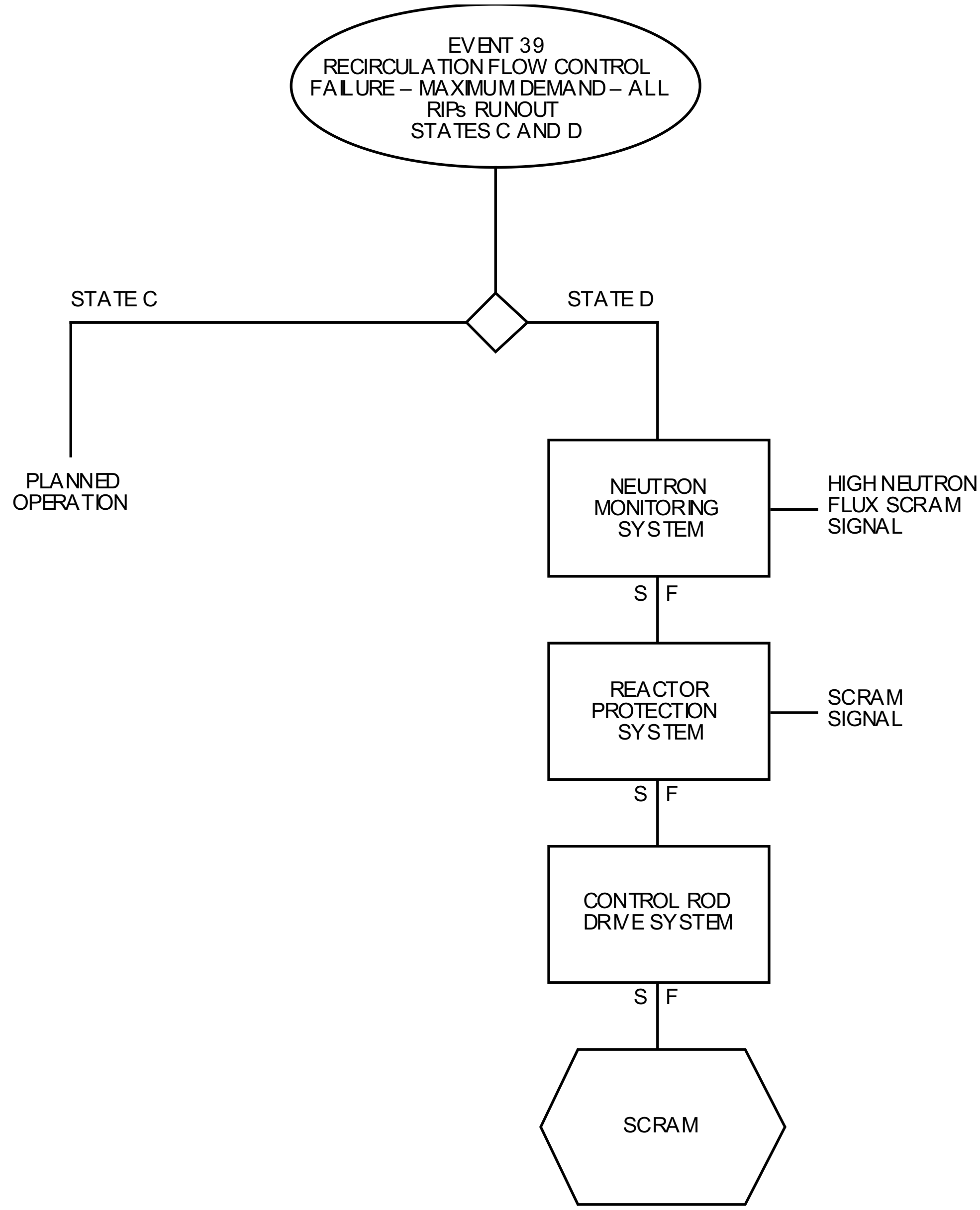


Figure 15A-46 Protection Sequence for Recirculation Flow Control Failure—Maximum Demand—All Reactor Internal Pumps (RIPs) Runout

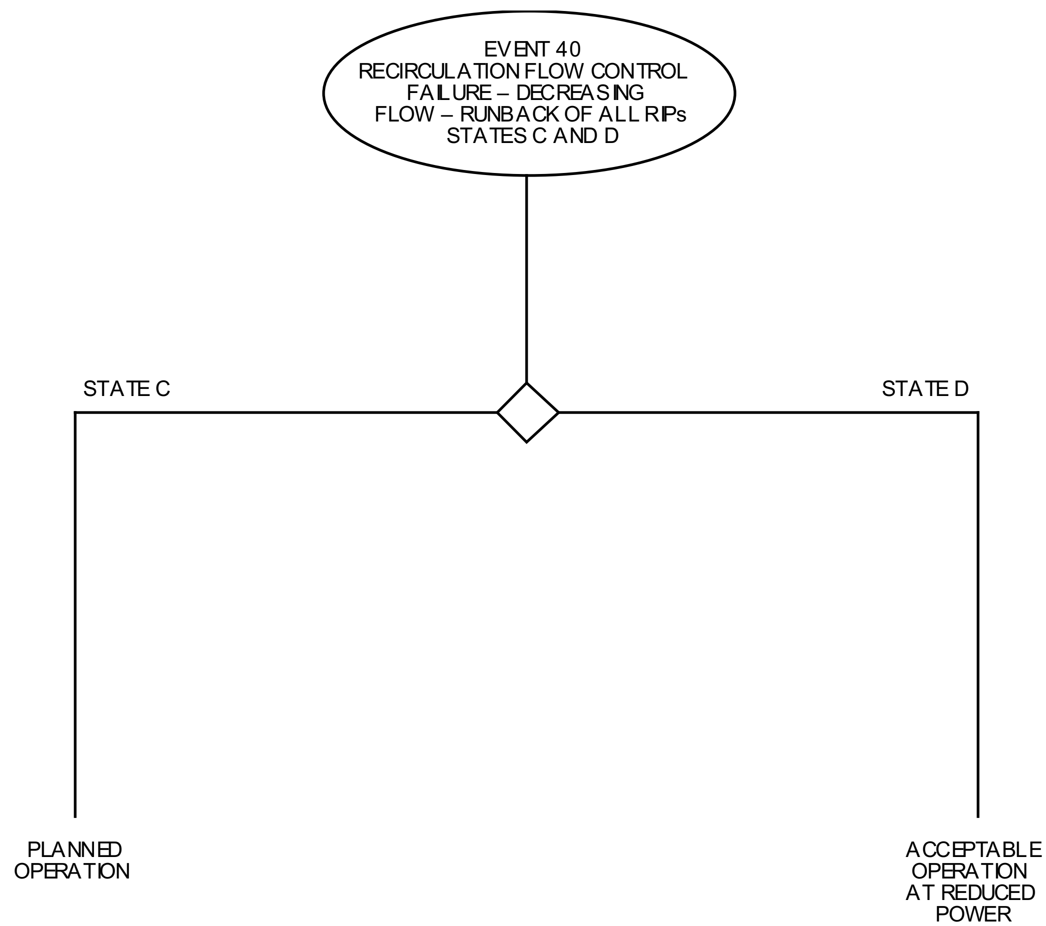


Figure 15A-47 Protection Sequence for Recirculation Flow Control Failure—Decreasing Flow—Runback of All Reactor Internal Pumps (RIPs)

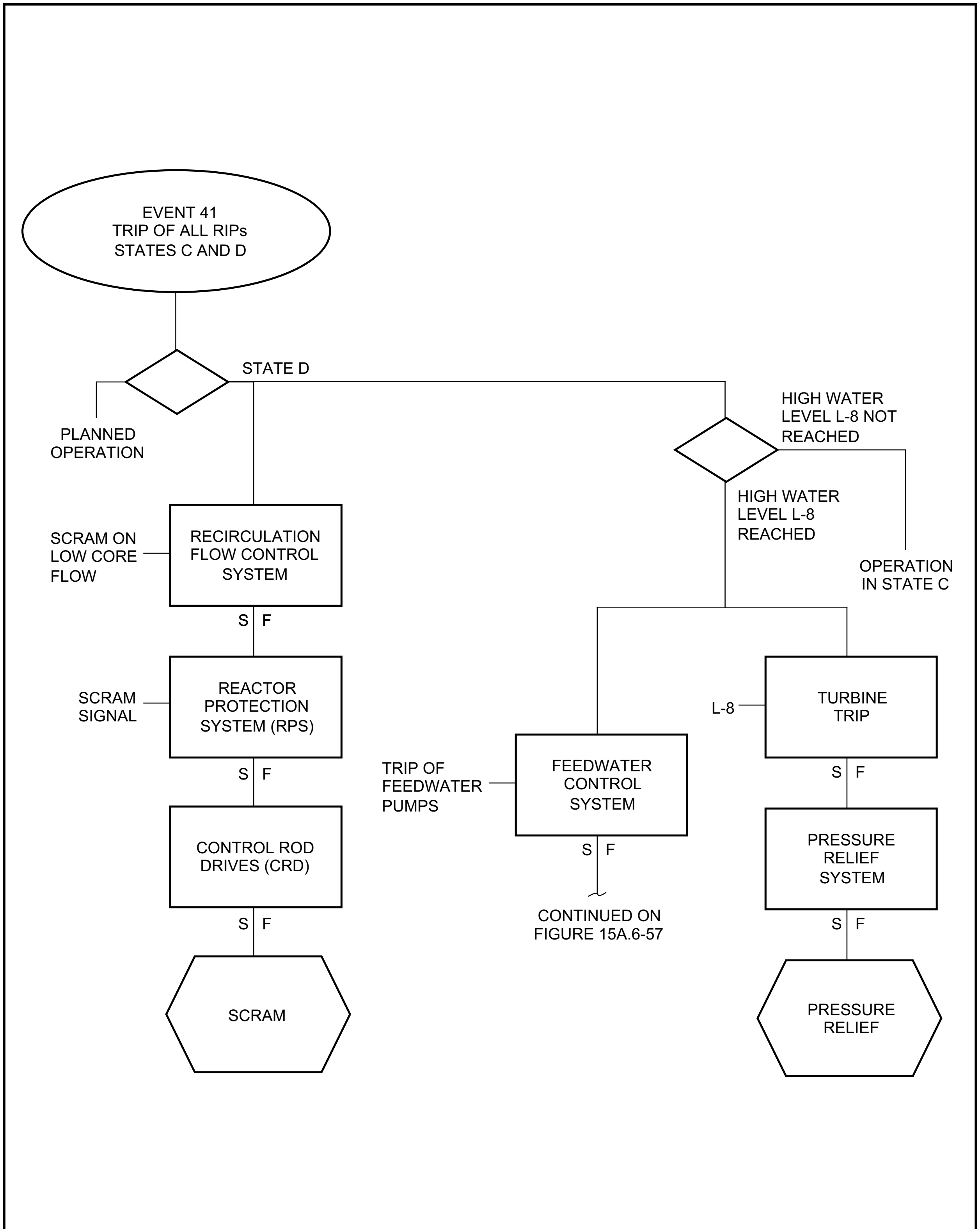


Figure 15A-48 Protection Sequence for Trip of All Reactor Internal Pumps (RIPs)

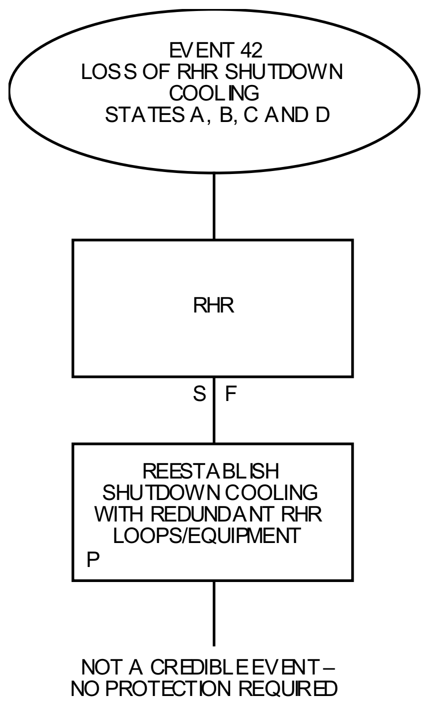


Figure 15A-49 Protection Sequence for RHR—Loss of Shutdown Cooling

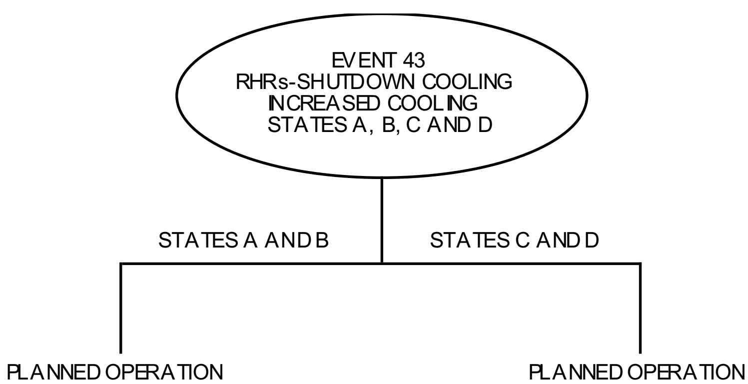


Figure 15A-50 RHR—Shutdown Cooling Failure—Increased Cooling

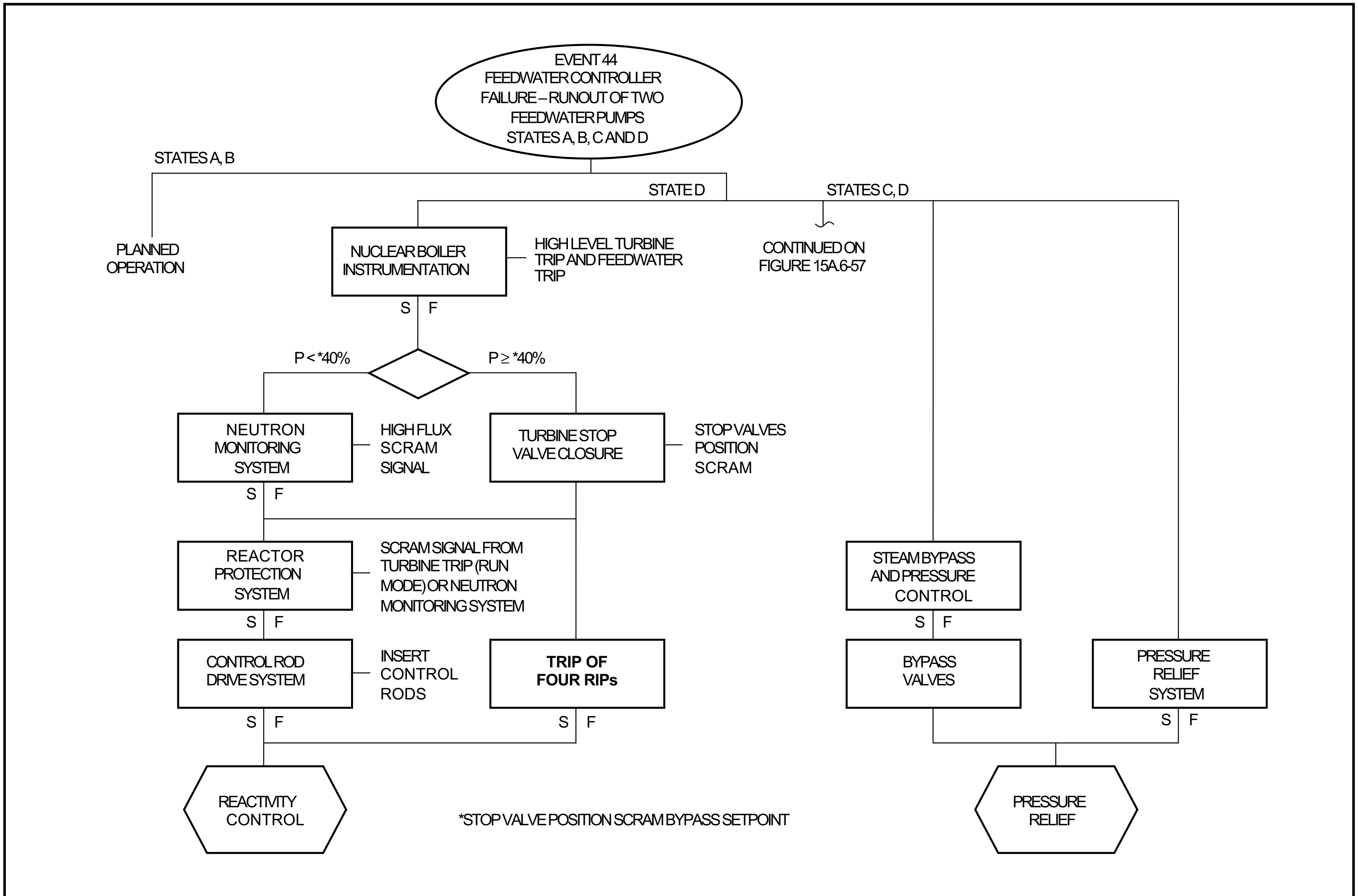


Figure 15A-51 Protection Sequences for Feedwater Controller Failure—Runout of Two Feedwater Pumps

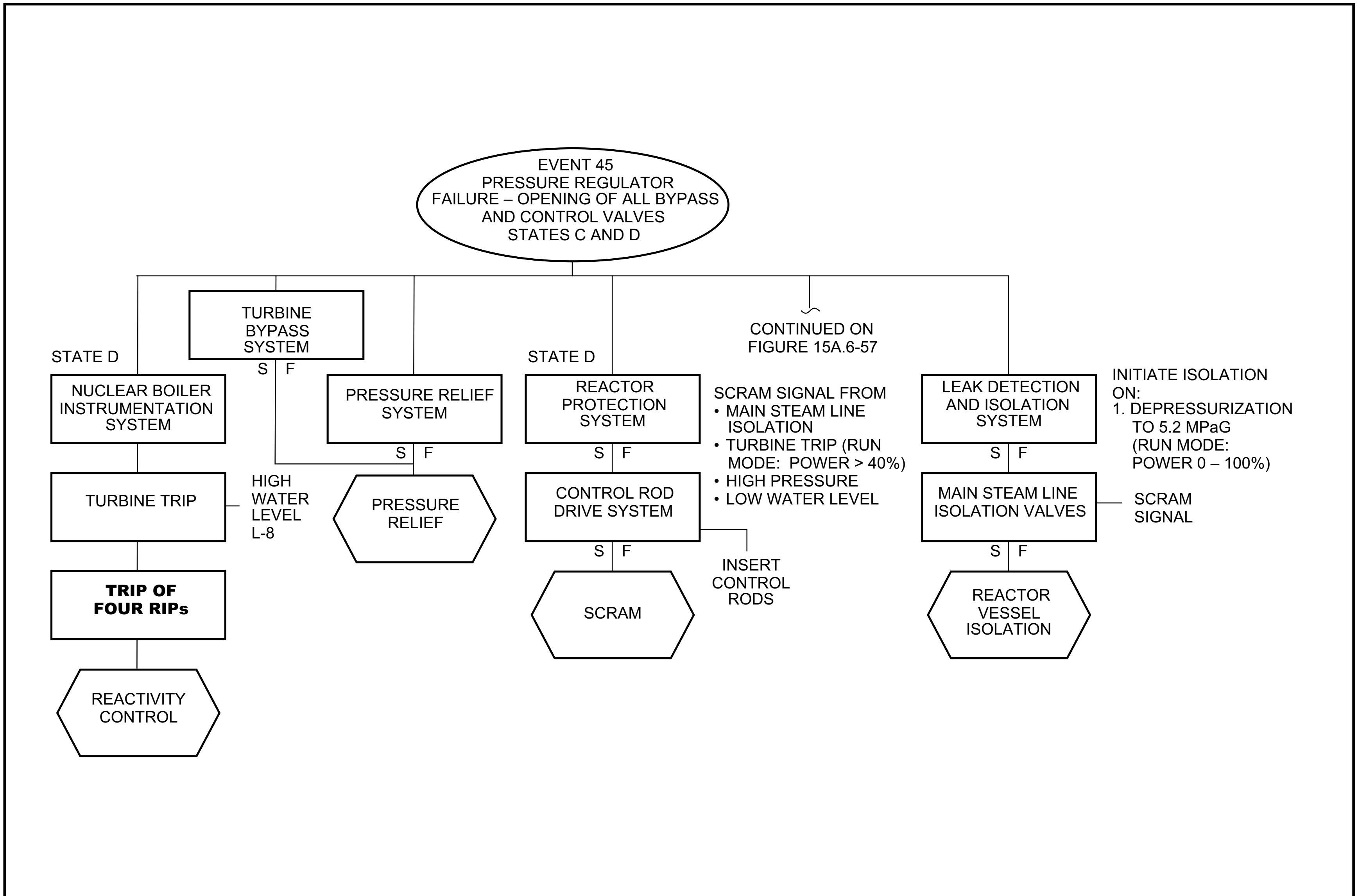


Figure 15A-52 Protection Sequences for Pressure Regulator Failure—Opening of All Bypass and Control Valves

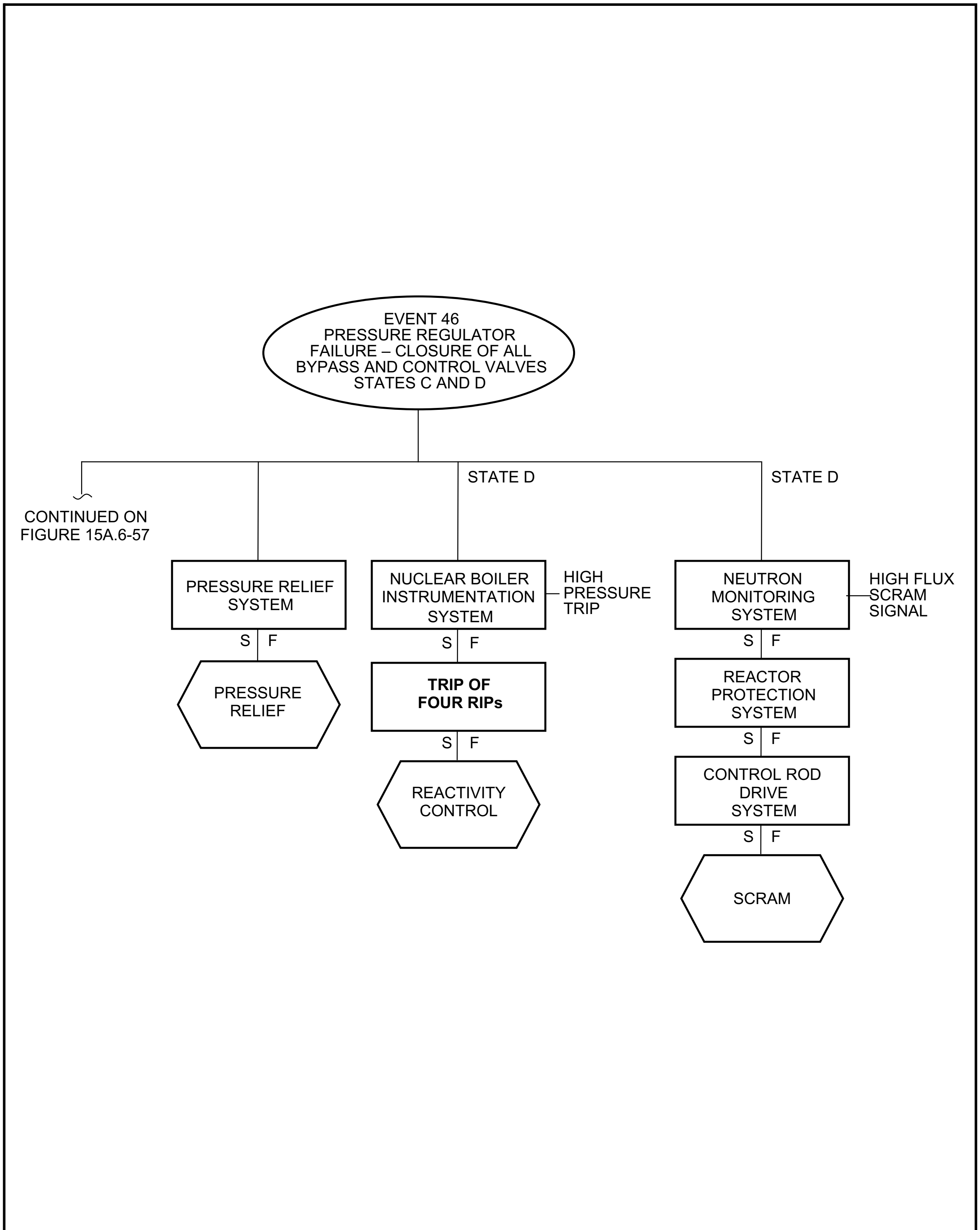


Figure 15A-53 Pressure Regulator Failure—Closure of All Bypass Valves and Control Valves

Figure 15A-54 Not Used

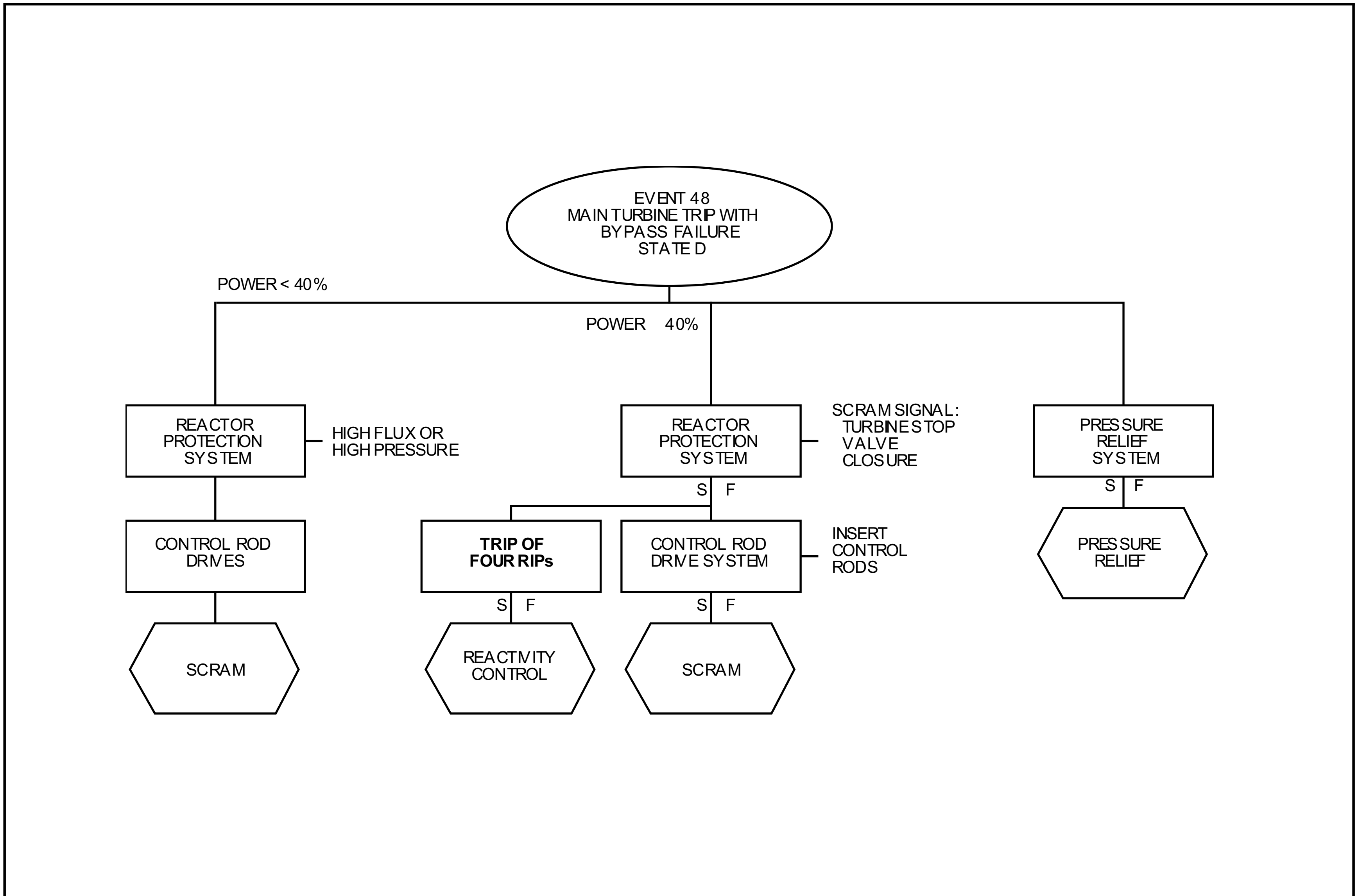


Figure 15A-55 Protection Sequences Main Turbine Trip—with Bypass Failure

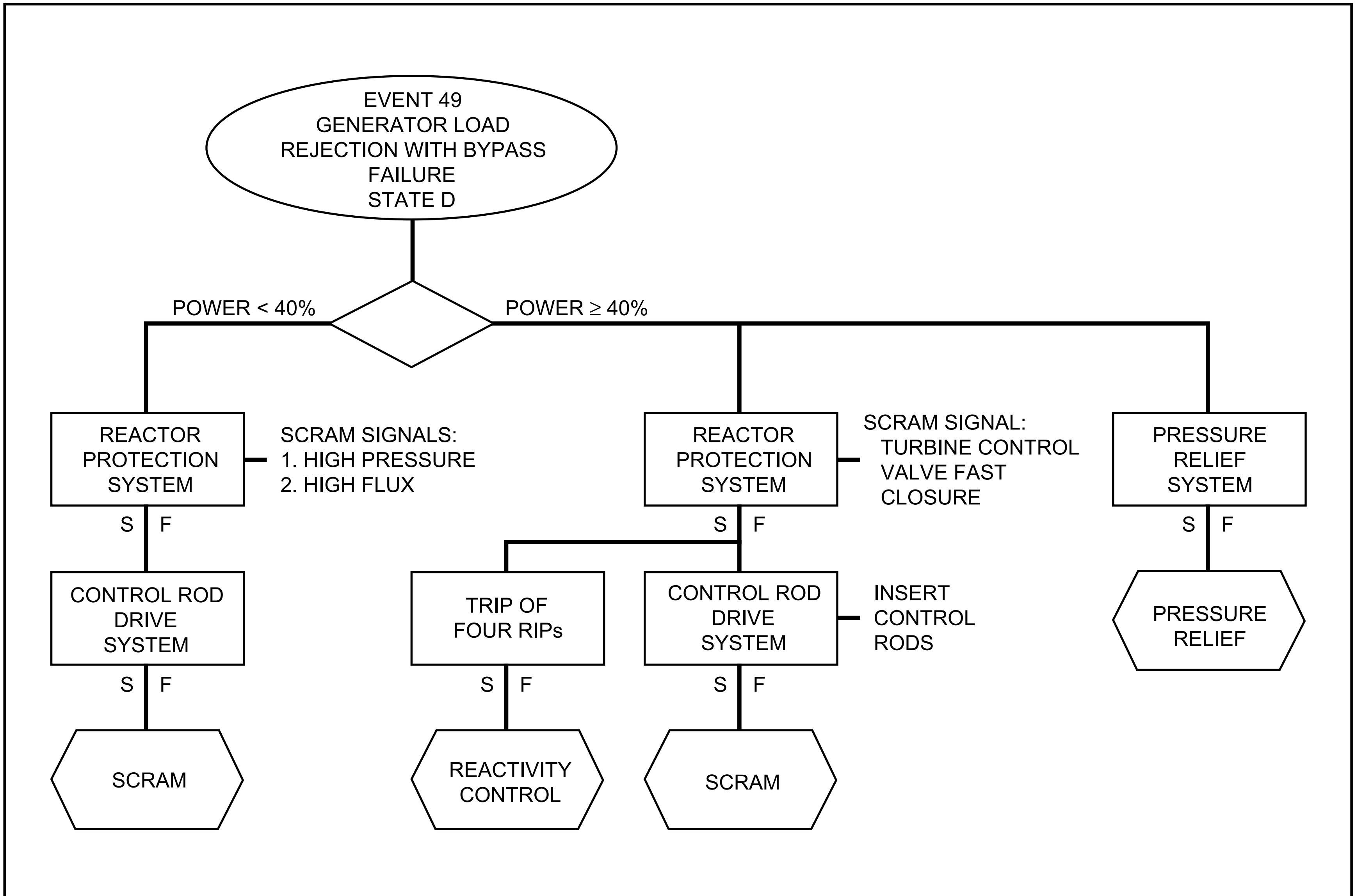


Figure 15A-56 Protection Sequences Main Generator Load Rejection—with Bypass Failure

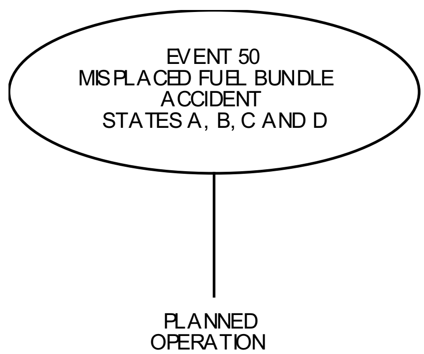


Figure 15A-57 Protection Sequence for Misplaced Fuel Bundle Accident

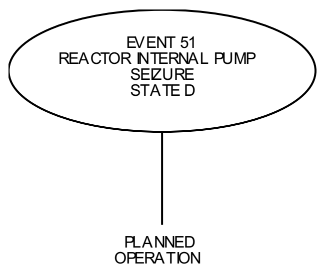


Figure 15A-58 Protection Sequence for Reactor Internal Pump Seizure

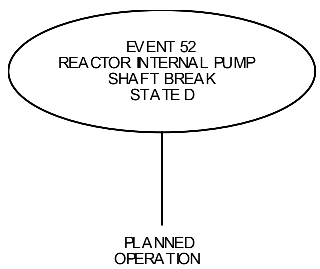


Figure 15A-59 Protection Sequence for RIP Shaft Break

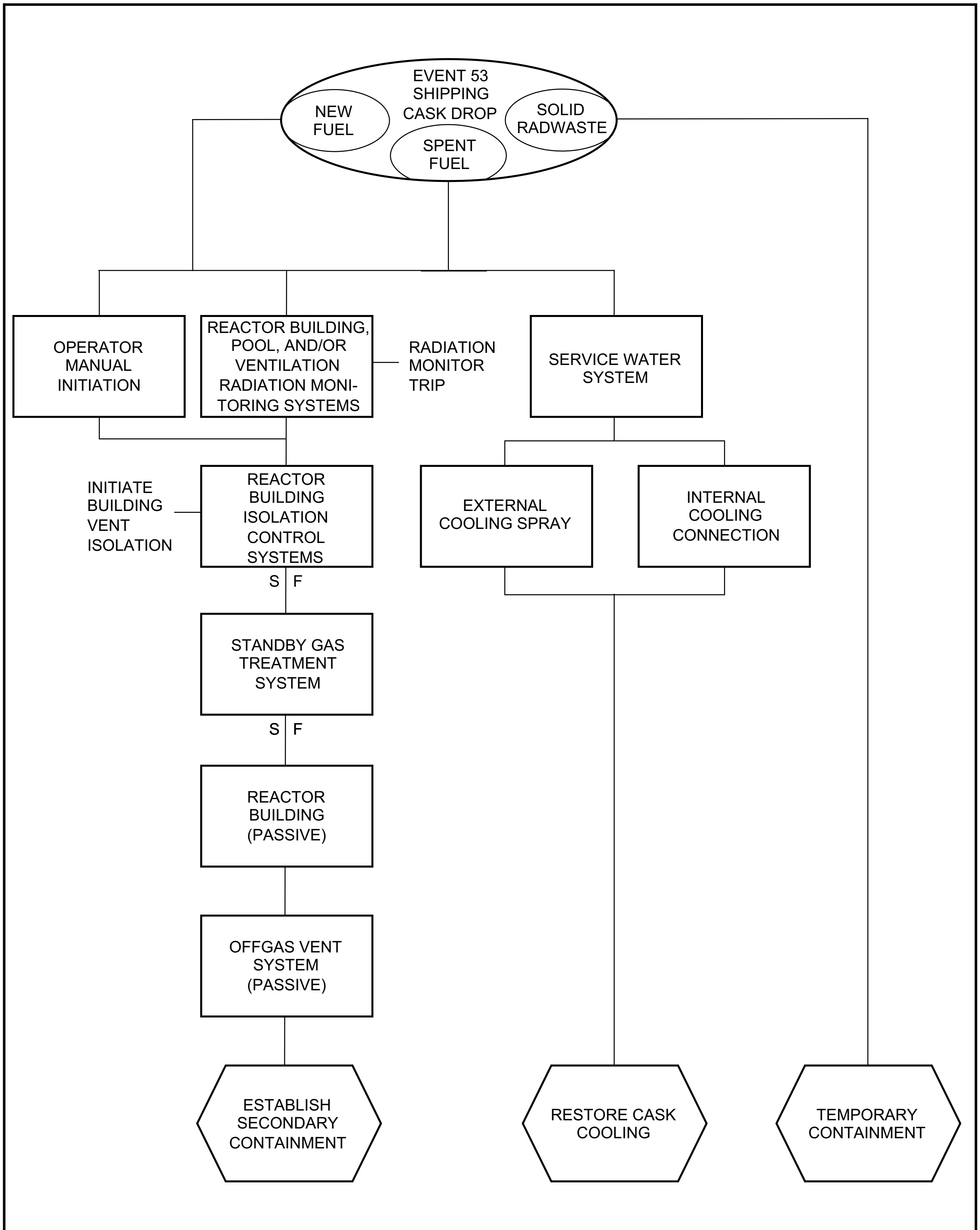


Figure 15A-60 Protection Sequence for Shipping Cask Drop

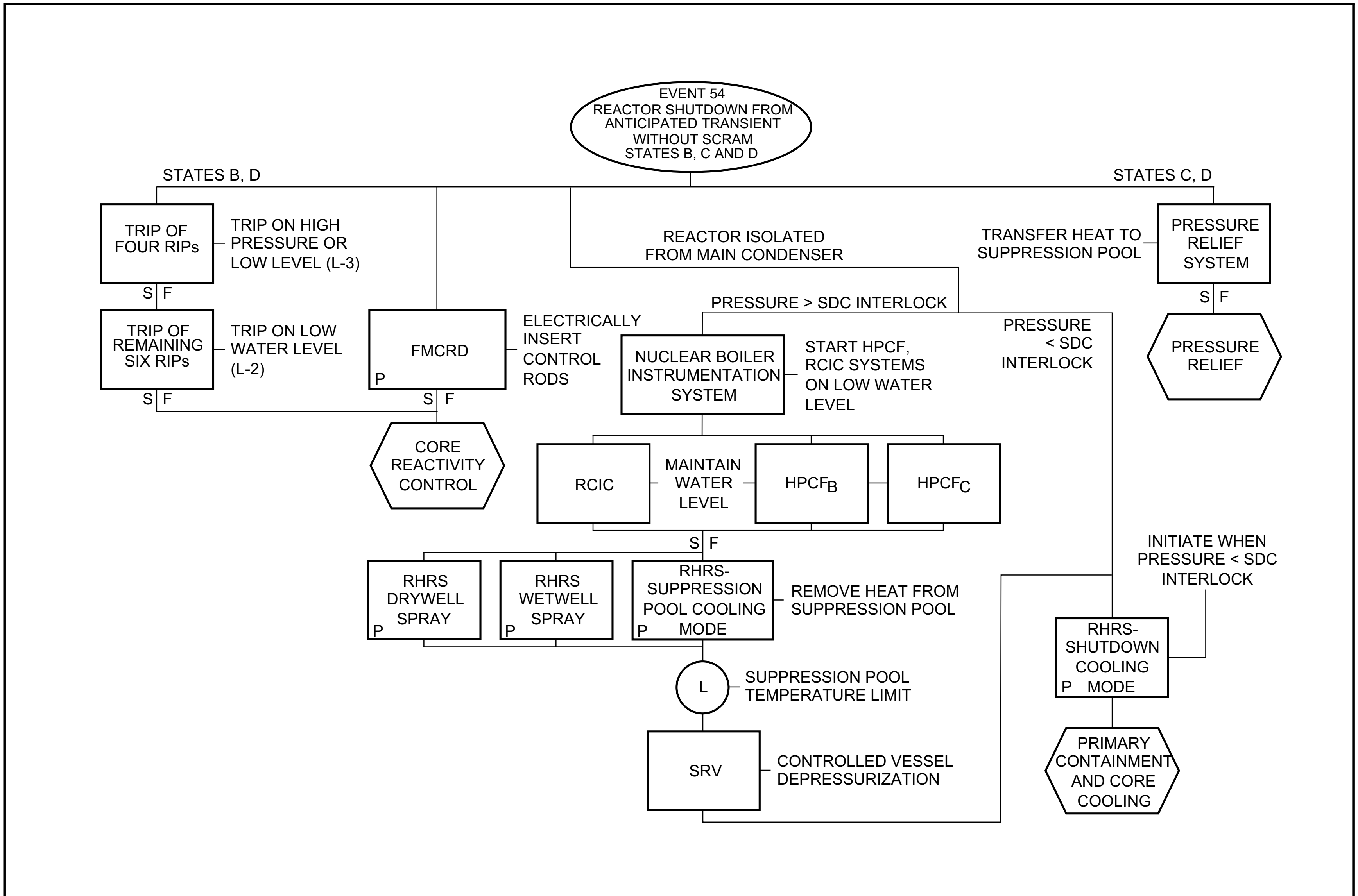


Figure 15A-61 Protection Sequence for Reactor Shutdown—from Anticipated Transient Without Scram

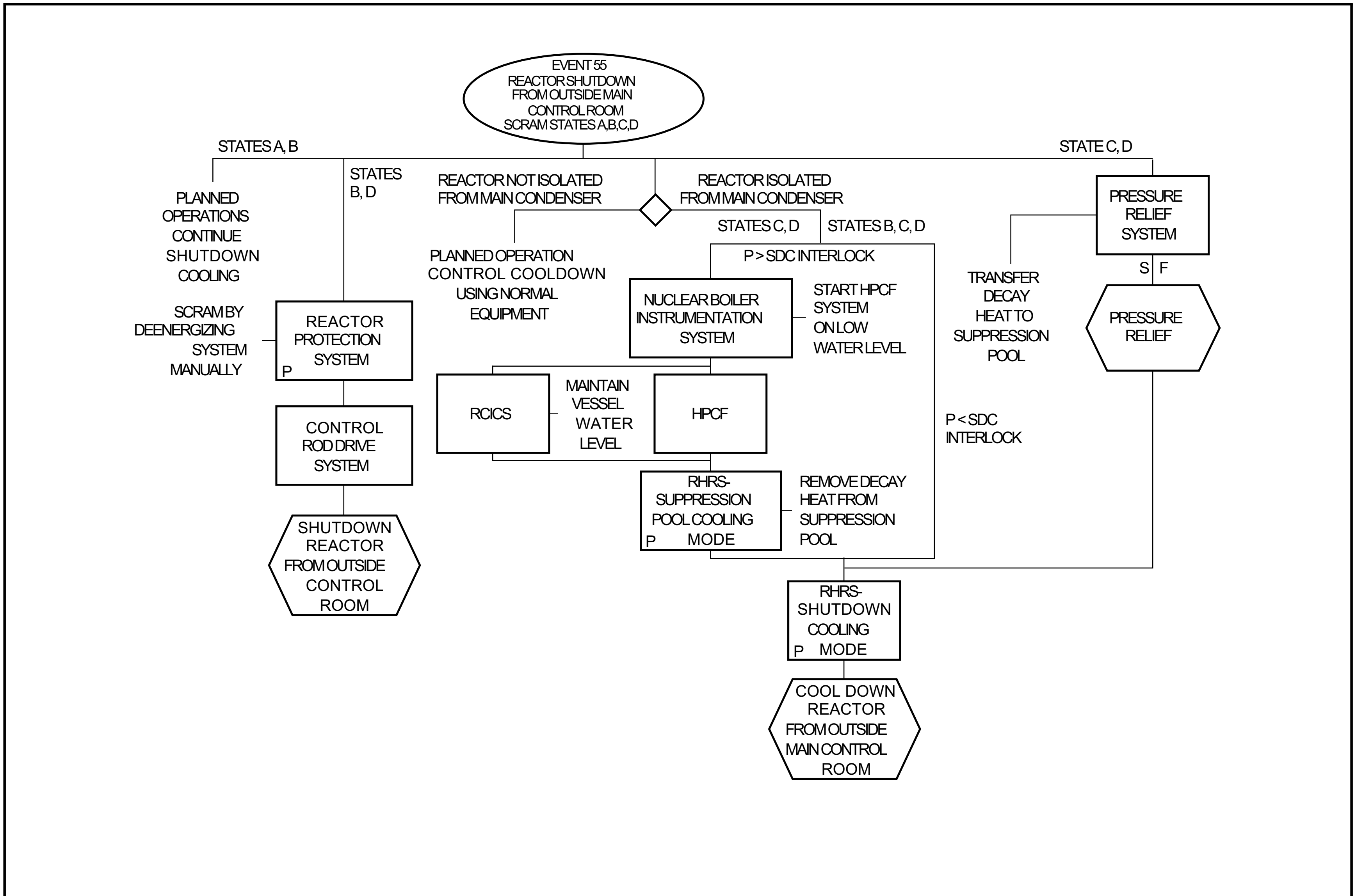


Figure 15A-62 Protection Sequence for Reactor Shutdown—from Outside Main Control Room

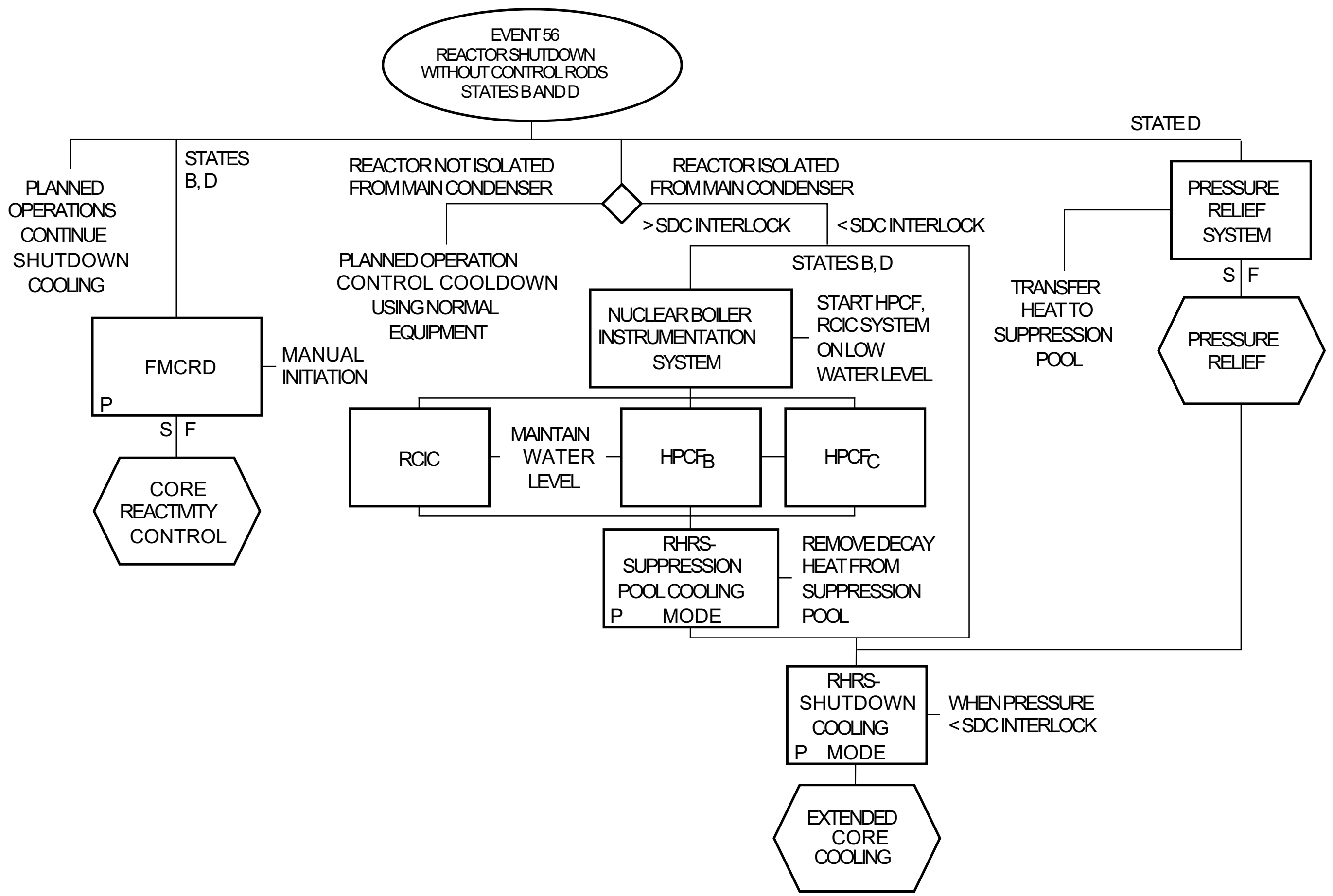


Figure 15A-63 Protection Sequence for Reactor Shutdown—Without Control Rods

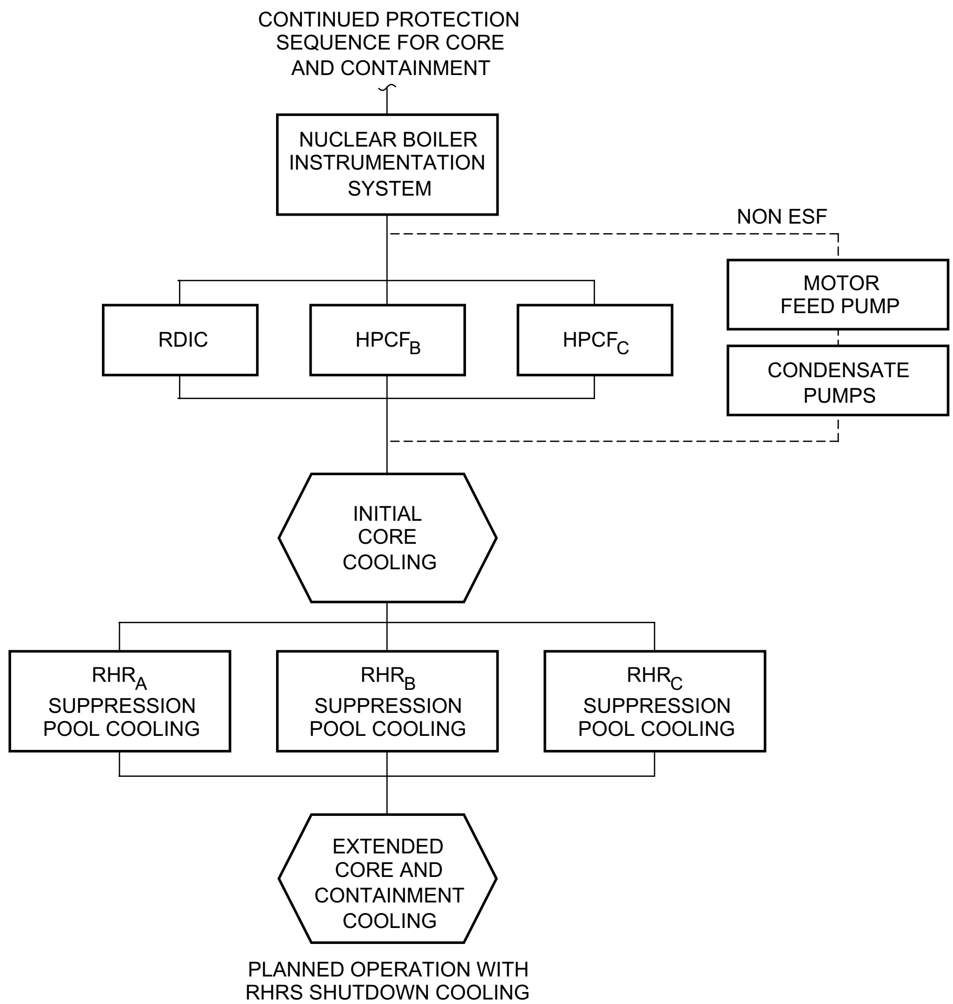
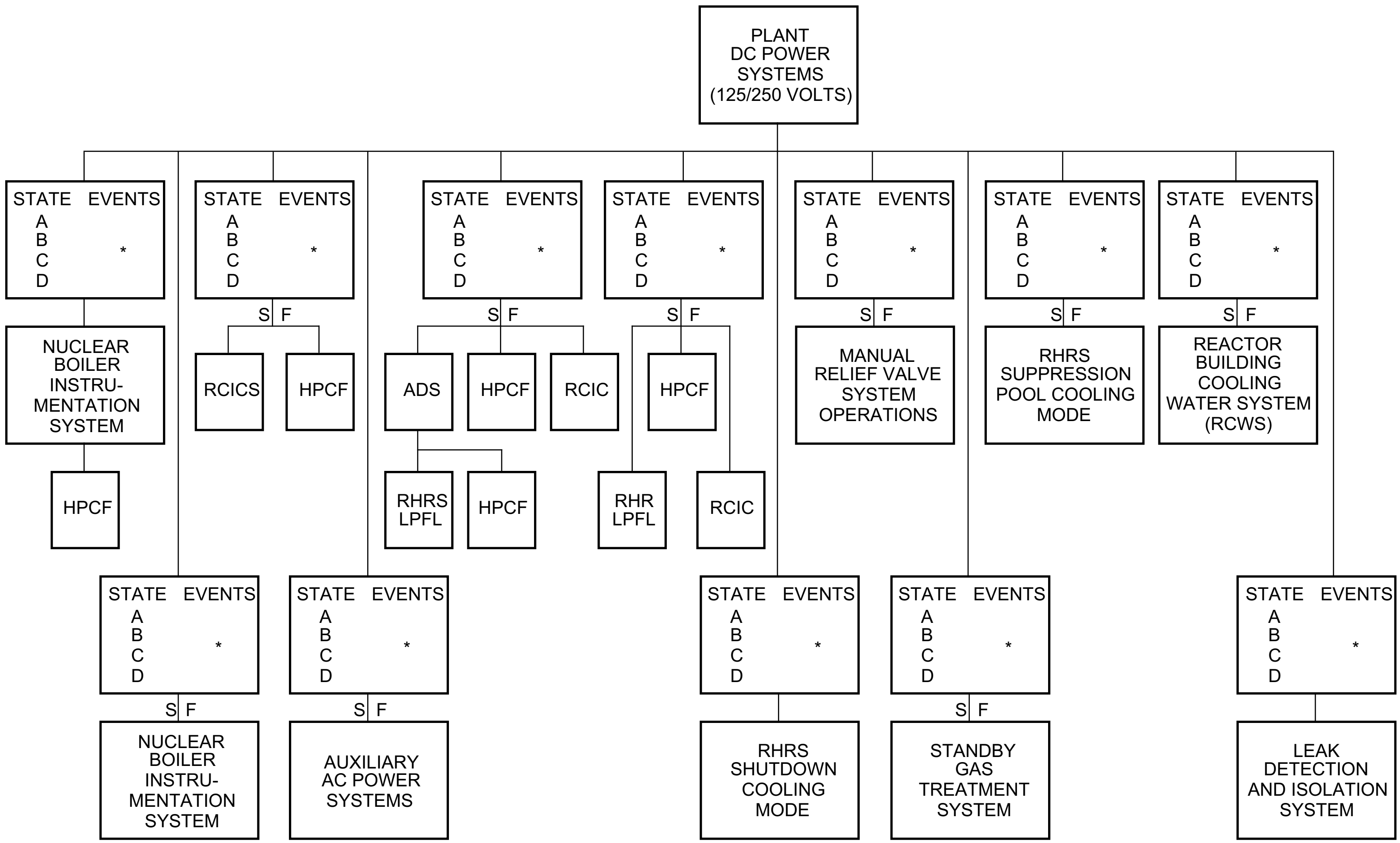


Figure 15A-64 Protection Sequence for Core and Containment Cooling for Loss of Feedwater and Vessel Isolations



* APPLICABLE EVENTS (TABLES 15A.6-2 THROUGH 15A.6-5)

Figure 15A-65 Commonality of Auxiliary Systems—DC Power Systems (125/250 Volts)

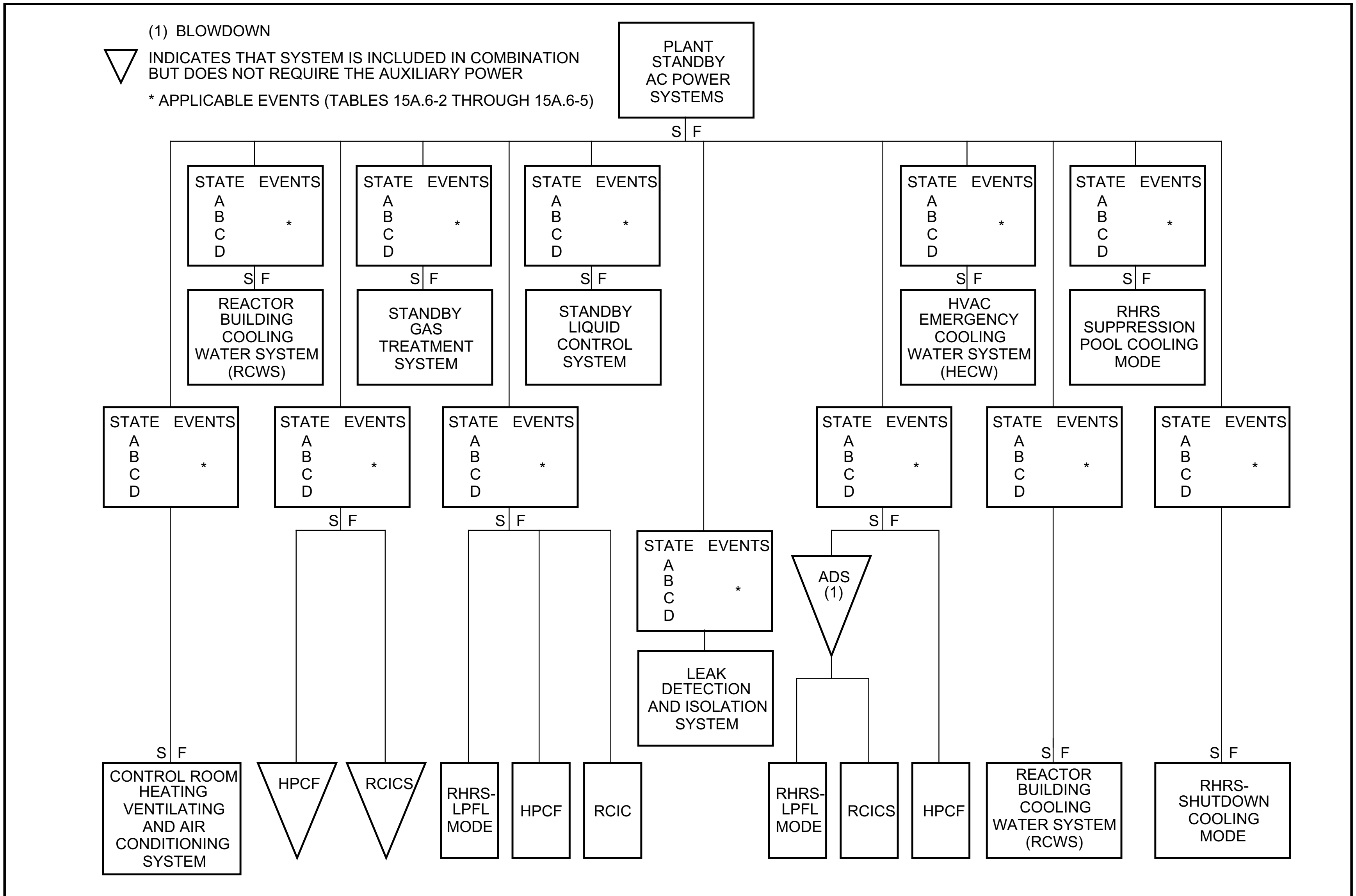
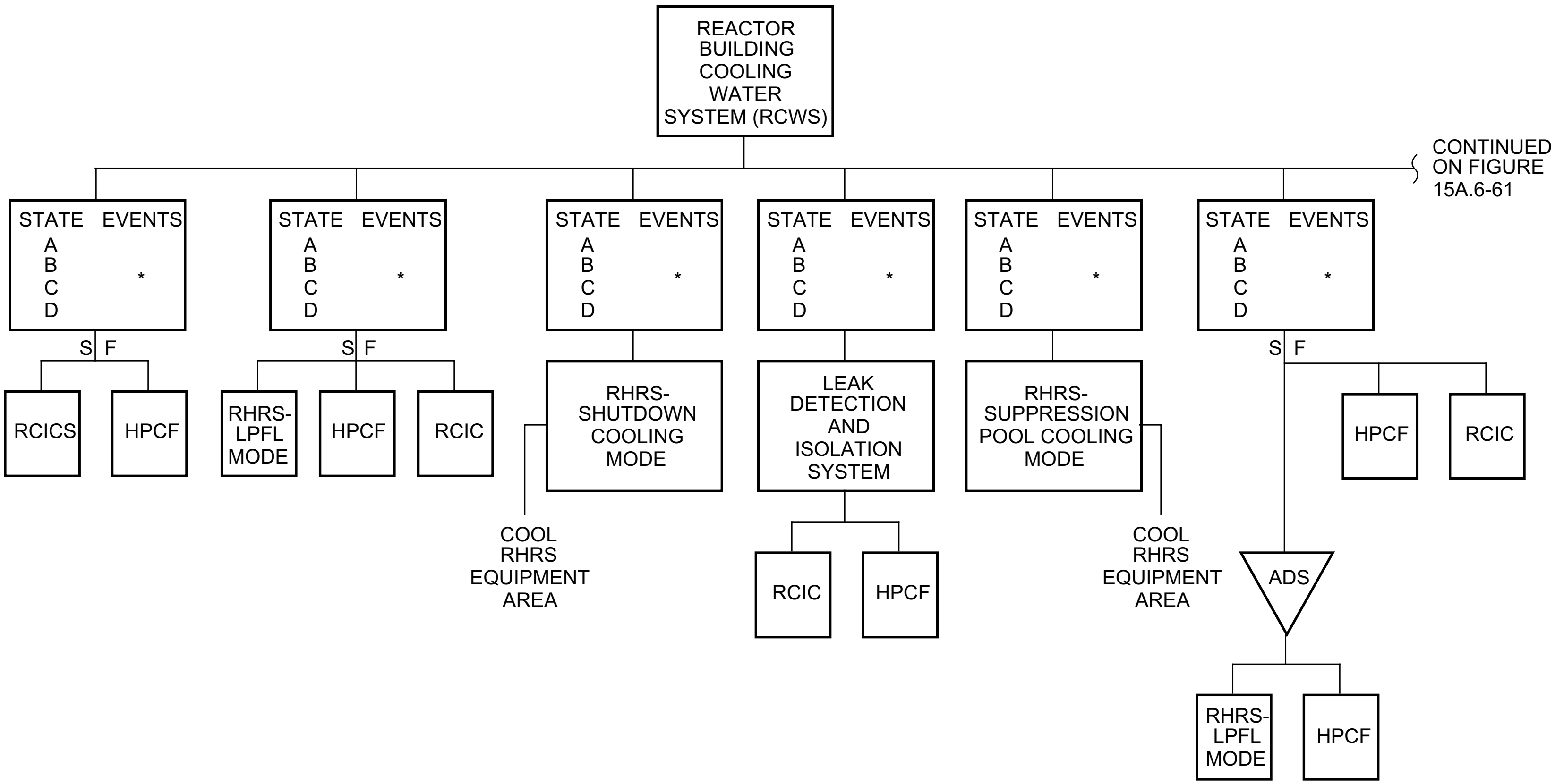


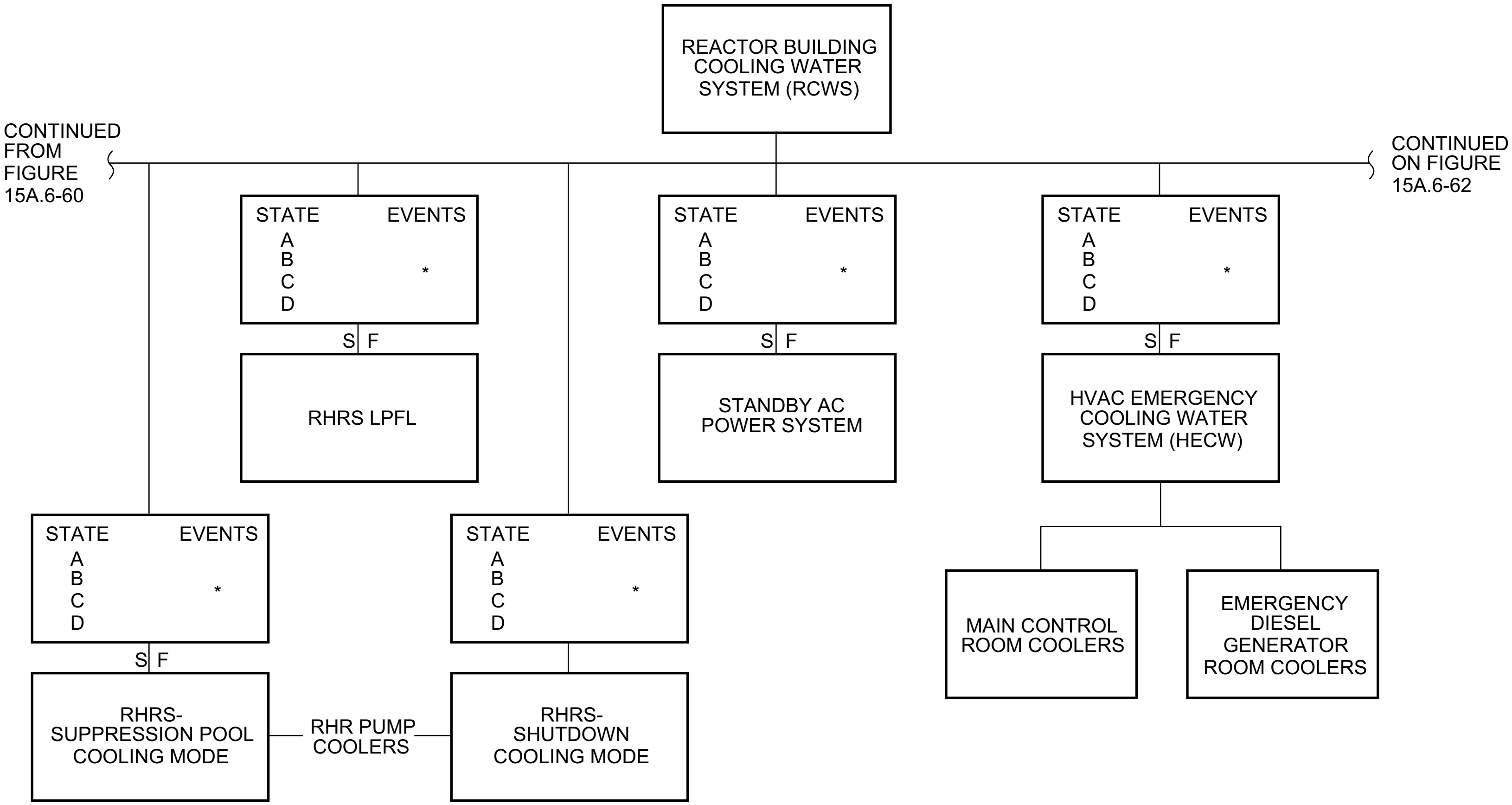
Figure 15A-66 Commonality of Standby AC Power Systems (120/480/6900 Volts)



* APPLICABLE EVENTS (TABLES 15A.6-2 THROUGH 15A.6-5)

NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 54, 55 AND 56

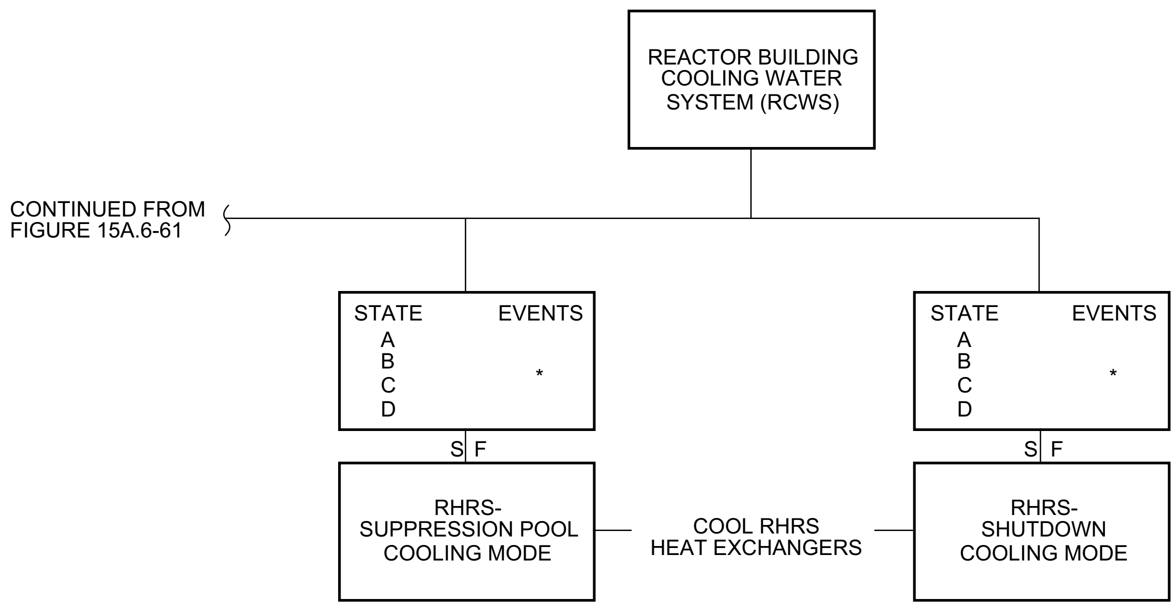
Figure 15A-67 Commonality of Auxiliary Systems—Reactor Building Cooling Water System (RCWS)



* APPLICABLE EVENTS (TABLES 15A.6-2 THROUGH 15A.6-5)

NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 54, 55 AND 56

Figure 15A-68 Commonality of Auxiliary Systems—Reactor Building Cooling Water System (RCWS) (Continued)



* APPLICABLE EVENTS (TABLES 15A.6-2 THROUGH 15A.6-5)

NOTE: SF REQUIREMENT NOT APPLICABLE IN EVENTS 54, 55 AND 56

Figure 15A-69 Commonality of Auxiliary Systems—Reactor Building Cooling Water System (RCWS) (Continued)

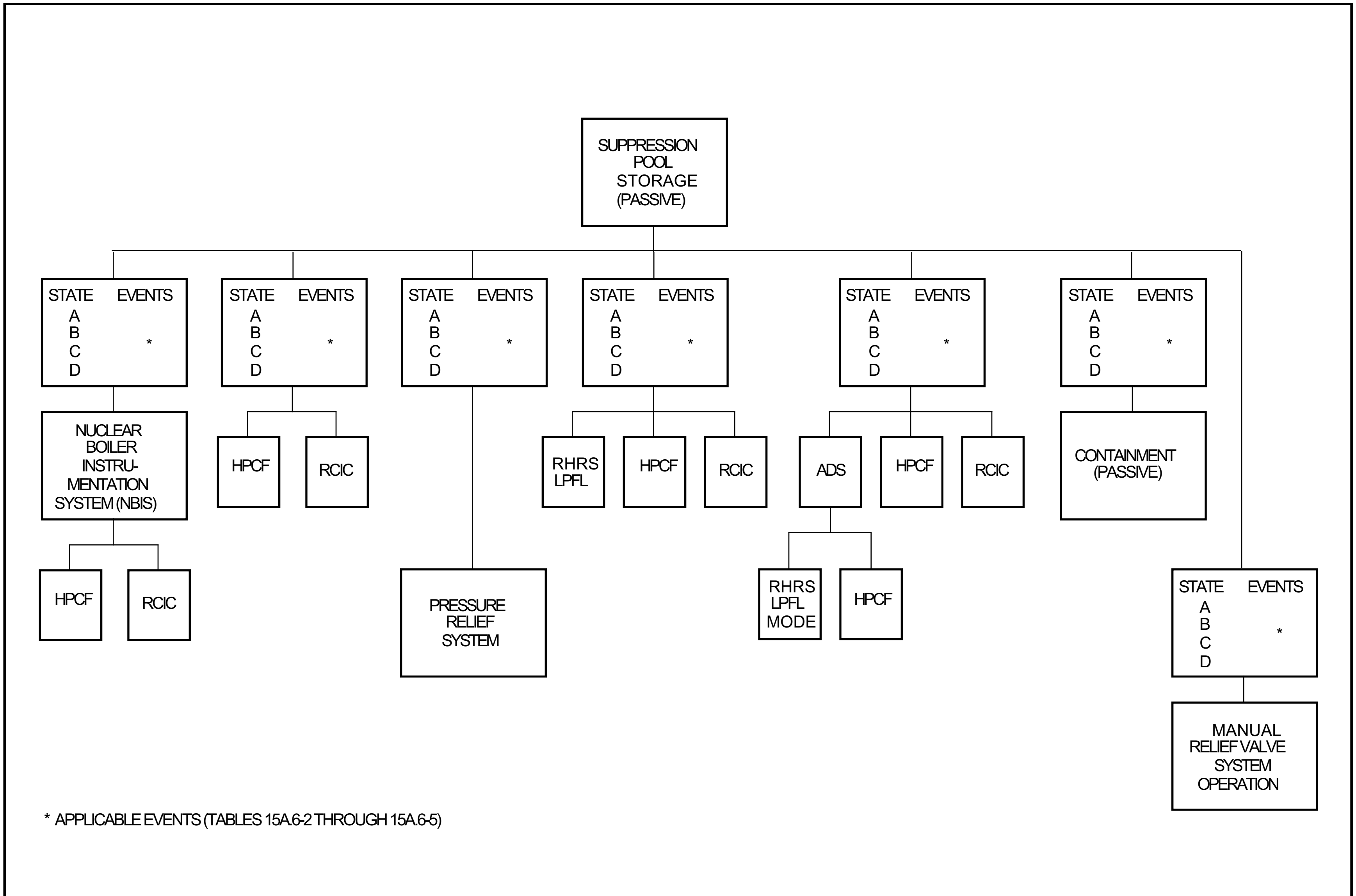


Figure 15A-70 Commonality of Auxiliary Systems—Suppression Pool Storage

15B Failure Modes and Effects Analysis (FMEA)

15B.1 Introduction

This appendix provides failure modes and effects analyses (FMEAs) for two ABWR systems and one major component which represent a significant change from past BWR designs. Specifically, FMEAs are provided for the following:

(1) Control Rod Drive System (with emphasis on the fine motion control rod drive)

(2) Essential Multiplexing system

(3) Reactor internal pump

Regulatory Guide 1.70 requires FMEAs to be performed on selected subsystems of Chapters 6, 7 and 9. GEH considers that the plant nuclear safety operational analysis (NSOA) of Appendix 15A and the probabilistic evaluations of Appendix 19D adequately address single failures for those systems and components which are similar to past BWR designs and resources are best directed to conducting and reporting FMEAs for new systems and components noted above.

15B.2 Control Rod Drive System

15B.2.1 Introduction

The Control Rod Drive (CRD) System is comprised of the fine motion control rod drives (FMCRD), the hydraulic control units (HCUs), and the control rod drive (CRD) pumps. This analysis is focused on the FMCRD because the HCU and CRD pump equipment do not include substantial departure from the earlier BWR designs. Extensive FMEAs and reliability analyses have been performed on the earlier designs and many reactor years experience have accumulated. The key elements of the HCUs are included in the discussion for completeness.

The interfaces of the CRD System are identified and the potential impact of those interfaces is part of this analysis.

15B.2.2 Conclusion

The finding of this analysis is that there are no single failures which can prevent the CRD System from performing its safety functions. The FMEA is presented in Tables 15B-1 and 15B-2.

15B.2.3 Description

A simplified CRD System process flow diagram is shown in Figure 15B-1. CRD System water is taken from the condensate, feedwater and condensate air extraction system, or Condensate Storage Tank (CST) through a suction filter by a centrifugal pump and discharged through a drive water filter to the HCUs. (During shutdown the CST is the primary source.) Each of these components is independently redundant and only one of each is in operation at any one time. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the CST. The pumped water is directed to the HCU to provide hydraulic scram and to furnish purging to the drive. This system also provides purge water for the reactor internal pumps and the reactor water cleanup pumps.

The HCUs are all supplied by the same operating CRD pump, but the HCUs are divided into four banks, A & D on one side of the reactor and B & C on the other side of the reactor. Each HCU serves two FMCRDs. The HCU P&ID is shown in Figure 15B-2. The purge water enters the HCU through valve 104, passes through a filter, a restricting orifice, and a check valve to the scram line. The flow passes into the FMCRD at a pressure slightly higher than vessel pressure and up through the drive to the vessel. This flow provides cooling for the drive and serves to prevent debris from entering the drive from the vessel. The charging water enters the HCU through valve 113, passes through a check valve, fills an accumulator against nitrogen pressure and is stopped from entering the FMCRD by an air-operated scram valve, 126. The accumulator capacity is adequate to scram two FMCRDs.

The scram valve is held closed by instrument air. The scram valve is controlled by a double solenoid pilot valve, 139. The solenoids are normally energized and both must be de-energized to scram the drive. The pilot valve is shown in the de-energized state. When energized, the pilot valve exhaust port is closed and the instrument air is applied to the scram valve diaphragm, holding the scram valve closed. De-energization of the pilot valve shuts off the instrument air and opens the scram valve diaphragm to exhaust, allowing the valve to open and apply accumulator pressurized water to a pair of FMCRDs. Scram is effected when the pressurized water is applied to the hollow piston of the FMCRDs. Another set of valves, the Air Header Dump Valves, also dump the air pressure during normal scrams. Under ATWS conditions, the instrument air header pressure can also be discharged by the Alternate Rod Insertion (ARI) valves.

The FMCRDs have three safety functions and one normal operating function. The safety functions are:

(1) Scram

(2) Rod Drop Prevention

(3) Rod Ejection Prevention

The normal operating function is the positioning of the control rod in response to the Rod Control and Information System (RCIS). The FMCRD also feeds back rod status and position information to the RCIS for performance monitoring by the RCIS.

The FMCRD assembly drawing is shown in Figure 15.B-3. There are two major parts to the FMCRD: (1) the hydraulic scram actuation system and (2) the electric motor drive, which inserts or withdraws the control rod in response to the RCIS signals. The electric motor drive also fully inserts the rod as a backup to the hydraulic scram. During normal operation, the insertion and withdrawal of the FMCRD is under the direction of the RCIS. The FMCRD stepping motor turns a spindle (screw) which causes the vertical motion of a ball-nut. This linear motion is transferred to the control rod via a hollow piston which rests on the ball-nut. Thus, the piston and control rod are raised or lowered depending on the direction of rotation of the FMCRD motor and spindle. One design feature of the FMCRD is the automatic run-in of the ball-nut by the electric motor drive following the hydraulic scram. This use of the electric motor provides a backup to the hydraulic accumulator scram.

On loss of electric power to both scram pilot valve solenoids, the associated HCU applies insert forces to its respective drives using the precharged accumulator water contained within the HCU. Water enters the FMCRD through the scram port; the pressure differential between the hollow piston and the reactor vessel drives the piston upward. The water displaced from the drive is discharged into the reactor vessel through a labyrinth seal in the throttling sleeve at the buffer. During a scram, the hollow piston separates from the ball-nut as the control rod is driven into the core. Spring-loaded latch fingers in the hollow piston expand and engage notches in the guide tube. The fingers support the hollow piston and the control blade until the ball-nut can be driven up to support the hollow piston and release the latch finger.

A provision is made for integral, internal blow-out support to prevent the FMCRD ejection if failure of the FMCRD housing occurs at any of various locations. The drive motor brake and a ball check valve at the flange where the accumulator piping meets the FMCRD both provide protection against rod ejection. The valve prevents control rod ejection in case of a failure in the scram piping. If a scram line failure were to occur, a large pressure differential across the hollow piston could result in the ejection of the control rod. The ball check valve would be seated by the reverse flow through the scram port and ejection would be prevented. The FMCRD electromechanical brake is keyed to the motor shaft. The brake is normally engaged by spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RCIS. The brake prevents a high pressure differential across the hollow piston from causing the reverse rotation of the lead screw and “run-out” of the control rod.

Interfaces

Required inputs:

(1) Water from the condensate, feedwater and condensate air extraction system and from the CST

(2) Instrument air

(3) Signals from RPS channels A & B

(4) Electrical power to the FMCRD motors and brakes

Outputs:

(1) Purge flow water into the vessel

(2) Rod position signal from the synchro

(3) Rod position indication signal from reed switches

(4) Rod separation signal from reed switches

(5) Scram full insert signal

The only substantive problem which has occurred in any of the interfaces in history has been the disabling of scram solenoid valves by contaminated instrument air. The contaminates caused the deterioration of the valve seats and prevented the valves from opening. This problem was corrected by the incorporation of Viton-A seat material which is impervious to the contaminates. Viton-A has been specified for the ABWR solenoid valve seats.

15B.2.4 FMCRD Failure Modes Evaluation

The following evaluation and discussion of failure modes which threaten the ability of the FMCRD to perform its safety functions is presented as extensive expansion on the FMEA and system description above.

15B.2.4.1 Evaluation of Failures Relating to Scram

There are no known single failures/malfunctions that result in a loss of scram function for more than one pair of ganged control rod drives. High scram reliability is a result of a number of features of the CRD System. For example:

(1) Each accumulator provides sufficient stored energy to scram two CRDs at any reactor pressure.

(2) Each pair of drive mechanisms has its own scram valve and a dual solenoid scram pilot valve; therefore, only a single scram valve needs to open for scram to be initiated. Both pilot valve solenoids must be de-energized to initiate a scram.

(3) The Reactor Protection System (RPS) and the HCUs are designed so that the scram signal and mode of operation override all others.

(4) The FMCRD hollow piston and guide tube are designed so they will not restrain or prevent control rod insertion during scram.

(5) The electric motor drive insertion of each control rod is initiated simultaneously with the initiation of hydraulic fast scram. This provides a diverse means to assure control rod insertion.

Failures in the pressure boundary of an individual FMCRD or scram insert line can, at most, result in loss of scram capability only for the two drives sharing the associated ganged accumulator. The plant is capable of achieving cold shutdown under this failure condition. Additionally, the HCUs located in each quadrant will be physically separated into two groups. One group consists of the A-sequence HCUs (HCUs connected to the A-sequence rods only) and the other group consists of the B-sequence HCUs (HCUs connected to the B-sequence rods only). With this separation arrangement, the potential for the failure of two HCUs (one failing as a consequence of the other failing first) resulting in the failure of two facial adjacent rods within the core is avoided. This assures the capability to achieve hot shutdown with two HCUs failed (one HCU failed plus an adjacent HCU failed due to consequential effects).

Failures in individual HCUs which lead to low charging pressure on the nitrogen side are alarmed if pressure in the HCU drops below a predetermined setpoint. In this case, only the two drives grouped to the affected HCU are potentially incapable of scramming when required. As described above, the failure of two drives connected with one HCU to scram does not prevent the plant from achieving cold shutdown. However, a loss of charging water header pressure, resulting from a failure of the header piping or a CRD pump, affects the charging capability of all HCUs. Instrumentation is provided on the charging water header to monitor line pressure. In the event of loss of charging pressure, this instrumentation sends signals to the RPS which, in turn, generates a scram initiating signal.

The low pressure scram setpoint is set high enough to assure adequate charge pressure is available in the individual HCUs to complete the scram, but low enough to minimize unwanted scrams from normal pressure fluctuations in the line.

15B.2.4.2 Evaluation of Failure Relating to Rod Drop

The failure paths resulting in a rod drop accident (RDA) are shown in Figure 15B-4. The combination of multiple failures of protective features to reach a control rod drop condition by any failure path is considered to be so low in probability that RDA can be categorized as an incredible event for the FMCRD design. Some of these protective features are described as follows:

(1) Two redundant and separate Class 1E switches are provided to detect the separation of the hollow piston from the ball-nut. This means two sets of reed switches physically separated from one another with their cabling run through separate conduits. The separation switch is classified Class 1E, since its function detects a detached control rod and causes a rod block, thereby preventing a rod drop accident.

The principle of operation of the control rod separation mechanism is illustrated in Figure 15B-5. During normal operation, the weight of the control rod and hollow piston resting on the ball-nut causes the spindle assembly to compress a spring on which the lower half of the splined coupling between the drive shaft and spindle assembly rests (the lower half of the splined coupling is also known as the “weighing table"). When the hollow piston separates from the ball-nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and spindle assembly upward. This action causes a magnet in the weighing table to operate the Class 1E reed switches located in a probe outside the lower housing.

(2) Two redundant, spring-loaded latches on the hollow piston open to engage in windows in the guide tube within the FMCRD to catch the hollow piston if separation from the ball-nut were to occur. These latches open to support the hollow piston (and control rod) following scram until the ball-nut is run up to provide the normal support for the hollow piston (and control rod).

(3) A bayonet coupling between the control rod and FMCRD is provided. The coupling spud at the top end of the hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a
45 degree rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that must be manually unlocked by specific procedures before the components can be separated.

Coupling integrity is verified by pull test of the control rod upon initial coupling at refueling and by an "overtravel" test in which the ball-nut is driven down beyond the "full out" position into overtravel. After the weighing spring has raised the spindle to the limit of its travel, further rotation of the spindle in the withdraw direction will drive the ball-nut down away from the piston (assuming the coupling is engaged). Piston movement, if any, can then be detected by a reed switch at the overtravel position.

The control rod can only be uncoupled from the FMCRD by relative rotation, which is not possible during operation. The control rod cannot rotate, since it is always constrained between four fuel assemblies and the hollow piston has rollers which operate in a track within the FMCRD. Only structural failure would permit or result in control rod-to-FMCRD uncoupling.

(4) An automatic rod block is provided in the RCIS. Each channel of the RCIS monitors one of the Class 1E separation switches. If control rod separation is indicated by either switch, the associated RCIS channel will initiate a rod withdrawal block. Both channels of the RCIS would have to fail for a rod withdrawal operation to continue under these conditions. Additionally, a Class 1E indication and alarm is provided in the control room to alert the operator of a separation.

Because of the features described above, it is evident from Figure 15B-4 that multiple component/structural failures would have to occur before an RDA is possible. The most severe scenario, with respect to uncontrolled insertion of reactivity, is the case where the blade becomes separated from the hollow piston and sticks in the core as the hollow piston is withdrawn. If the blade subsequently unsticks, the rate of drop could exceed acceptable reactivity insertion rates. However, to reach this point requires several failures: (1) an undetected miscoupling during assembly or a structural failure of the coupling, (2) a sticking of the blade, and (3) a double failure of the separation switches or a double failure of the automatic rod block logic and failure of the operator to acknowledge the separation alarm. For the case where the blade remains coupled to the hollow piston and they stick as an assembly, the subsequent drop velocity is below the maximum allowable reactivity insertion rate. This scenario also requires multiple failures: (1) a sticking of the blade, (2) a double failure of the separation switches or a double failure of the automatic rod block logic and failure of the operator to acknowledge the alarm, and (3) a double failure of the latches on the hollow piston.

The number of failures associated with each event described above is considered to be so numerous as to result in a probability of occurrence low enough for rod drop to be categorized as an incredible event for the ABWR design.

15B.2.4.3 Evaluation of Failures Relating to Rod Ejection

15B.2.4.3.1 Drive Housing Failures

The bottom head of the reactor vessel has a penetration for each CRD location. A drive housing is raised into position inside each penetration and fastened by welding to a stub tube (Figure 15B-6). The drive is raised into the drive housing and bolted to a flange at the bottom of the housing.

In an unlikely event that a failure occurs of (a) the drive housing below the vessel/stub tube attachment weld, or (b) the weld itself but not the housing, ejection of the CRD and attached control rod is prevented by the integral internal blowout support. The postulated failure locations are identified by points A and B schematically in Figure 15B-6. With failure assumed at point A or B, the mechanical load plus the pressure load acting on the drive and housing would tend to eject the drive. The details of this support, which replaces the support structure of beams, hanger rods, grids, and support bars below the vessel used in previous product lines, are described in the following paragraphs.

The internal blowout support consists of the bayonet type support internal to the housing (Figure 15B-6). The internal blowout support catches the ejecting outer tube if failure (a) defined above occurs. This tube (which is welded at its lower end to the drive middle flange), is attached as shown in Figure 15B-6 at the top to the support, which is bayonet locked to the control rod guide tube base. The guide tube base, being supported by the housing extension, prevents downward movement of the outer tube and the drive. The internal blowout support catches the cap of the ejecting housing if failure (b) defined above occurs, and becomes a part of support chain consisting of the guide tube base, the guide tube and core plate, as shown in Figure 15B-7.

The internal blowout support prevents ejection of a CRD and attached control rod in the unlikely event of a drive housing failure. In both cases, the FMCRD motor brake function

(Section 15B.2.4.3.3) would be unimpaired and the motor spindle would not rotate and allow descent of the rod.

15B.2.4.3.2 Total Failure of All Drive Flange Bolts or Lower Housing

If a failure were to occur in the flange bolts or the spool piece (points C and D on Figure 15B-6), the drive would be prevented from ejecting downward also by the integral internal blowout support. The drive middle flange welded to the outer tube is prevented from ejecting by the internal support (similar to case (a) above). The middle flange retains the drive as described below.

The FMCRD design provides an anti-rotation device which engages when the lower housing (spool piece) is removed for maintenance. This device prevents rotation of the spindle which, in turn, holds the control rod in position when the spool piece is removed. The two components of the anti-rotation device are (1) the upper half of the coupling between the lower housing drive shaft and ball spindle, and (2) the back seat of the middle flange (Figure 15B-6). The coupling of the lower housing drive shaft to the ball spindle is splined to permit removal of the lower housing. The under side of the upper coupling piece has a circumferentially splined surface which engages with a mating surface on the middle flange back seat when the spindle is lowered during spool piece removal. When engaged, spindle rotation is prevented. In addition to preventing rotation, this device also provides sealing of leakage from the drive while the spool piece is removed.

In the unlikely event of the total failure of all the drive flange bolts, the anti-rotation device will engage the middle flange back seat, thus preventing rod ejection. The middle flange welded to the outer tube is supported by the internal support at the top as described in Subsection 15B.2.4.3.1.

15B.2.4.3.3 Rupture of Hydraulic Line to Drive Housing Flange

The FMCRD design provides single-failure-proof protection against the consequences of a scram line break by incorporating two diverse means for prevention of rod ejection. The first is a testable ball check valve located in the FMCRD flange. Under conditions of a scram line break, reverse flow will cause the ball to lift and seal the scram inlet port, thereby preventing rod ejection. The second feature is a testable, electromechanical brake located between the FMCRD motor and the synchromechanism. The electromechanical brake is designed to be a “safe-as-is” component that is normally in an engaged position when de-energized (rod ejection prevented), except when normal motor-driven rod movement is required. The brake is released (disengaged) when the motor is energized. The risk of a rod ejection occurring during rod motion is judged as acceptable due to the low probability of a coincident scram line failure and check valve failure occurring during the time the brake is disengaged.

15B.3 Reactor Internal Pump

15B.3.1 Introduction

Reactor internal pumps (RIPs) were first put in use by the Swedish NSSS supplier ASEA-ATOM in the late 1960s. At the present, six plants with a total of 44 RIPs are in operation. These RIPs have become the reference design for the ABWR RIPs. This FMEA addresses the following major aspects of potential failures:

(1) RIP impeller missiles

(2) RIP seizure

(3) RIP motor housing break, including consideration of shaft ejection

(4) RIP motor housing external loads

(5) Loss of RIP purge flow including purge pipe break

(6) Loss of secondary flow (reactor cooling water — RCW) to RIP heat exchanger

(7) Loss of primary RIP motor cooling including primary cooling water pipe break

(8) RIP loose parts

15B.3.2 Conclusions

The finding of this analysis is that there is no single failure which would impact the safety of the plant.

15B.3.3 Description

15B.3.3.1 Overall

The Reactor Recirculation System (RRS) P&ID is shown in Figure 5.4-4. The RRS is comprised of 10 pumps that collectively provide forced circulation of the reactor coolant through the lower plenum of the reactor and up through the lower grid, the reactor core, steam separators, and back down the downcomer annulus.

In addition to the RIPs, several subsystems are also included as part of the RRS to provide closely related, or closely supporting, functions to the RRS in composite or to the RIPs as individual components. The subsystems and reactor coolant pressure boundary (RCPB) are also shown on Figure 5.4-4. These subsystems are:

(1) Recirculation motor cooling (RMC) subsystem

(2) Recirculation motor purge (RMP) subsystem

(3) Recirculation motor inflatable shaft seal (RMISS) subsystem

The RIP and its auxiliary components have one safety function which is pressure retention (passive).

15B.3.3.2 RIP

The RIP consists of pumping components (impeller and diffuser) which are located inside the RPV and the driving component (motor), which is housed inside a casing. The casing is an extension of the RPV. The pumping unit and the motor have one common shaft. The shaft penetrates the RPV and extends into the motor's hollow rotor. The pump impeller and the motor rotor are assembled by various fasteners.

In order to reduce the bypass leakage of the pump, the piston rings are incorporated in the RIP between the outside of the diffuser and the pump deck. An optional diffuser wear ring may be provided on the diffuser.

15B.3.3.3 Adjustable Speed Drives

The adjustable speed drives (ASD) will be used to supply variable voltage/variable frequency electrical power to the reactor recirculation pumps. The recirculation pumps are single stage, vertical pumps driven by three-phase, four-pole, wet-type, squirrel cage, AC induction motors. Each ASD will supply power to one recirculation pump motor. The ASD receives electrical power from a supply bus at a relatively constant AC voltage and frequency. The ASD converts this constant supply power to a variable frequency/variable voltage output which is supplied to the recirculation pump motor. The output frequency is modulated in response to a demand signal from the system controller in order to vary pump speed.

15B.3.4 RIP Failure Modes Evaluation

The following evaluations and discussions of failure modes which are relevant to the safety of the plant are presented here as summary of detailed analyses.

15B.3.4.1 Missiles Generation

Since the parts of the RIP (impeller) are rotating inside the reactor pressure vessel (RPV), an evaluation has been made to assess the integrity of the RPV should an "impeller missile" occur. Although the rated speed for the RIPs is 157 rad/s, an initial speed of 188.5 rad/s is used for this evaluation. For unidentified reasons, the RIP impeller located approximately 3m below the reactor core bottom is assumed to disintegrate.

The acceptance criterion for a missile striking the RPV cylindrical shell or reactor core shroud is that the kinetic energy (KE) of the missile is less than the critical energy (CE) of the shell and shroud and, therefore, the missile will not degrade the integrity of the core or pressure boundary. The acceptance values are:

(1) RPV shell CE -9.41 MN·m

(2) Core shroud CE -0.24 MN·m

Calculations show that the energy of the impeller missile is:

$$0.09 \text{ MN·m} \tag{15B-1}$$

Comparing the information above, the impeller missile KE is approximately one-half the shroud CE and one-tenth the RPV shell CE.

In conclusion, the integrity of the core and RCPB are maintained in the event of a RIP impeller disintegration.

15B.3.4.2 Pump Seizure

Pump seizure causes rapid reduction of core flow and torsional loads on the RIP casing, RPV RIP nozzle, and RIP motor bottom flange. Several modes of pump seizure have been considered.

The RIP is assumed to be operating at 157 rad/s and for unidentified reasons the following seizures are assumed to occur:

(1) Impeller to diffuser seizure

(2) Rotor winding to stator winding seizure

(3) Thrust bearing seizure

(4) Radial bearing seizure

Any of these seizures will trip off the motor power and transfer the rotating kinetic energy of the impeller and motor rotor shaft into the RPV bottom head RIP nozzle directly or up through the motor housing into the nozzle.

The acceptance criterion for this event is that the torque load resulting from the seizure be less than value specified as the design basis for this event in the reactor vessel loading specification. This value is 42 T-M.

Depending on the location of the seizure in the pump or motor, the impeller-shaft kinetic energy will shear off one set of several bolts and pins in the motor structure. The torque load which shears the bolts and pins is transferred into the bottom flange of the motor housing and up through the housing cylinder into the RPV bottom head RIP nozzle.

In conclusion, any of the calculated torque loads transferred into the RPV RIP nozzles by a RIP or motor seizure are more than a factor of 4 less than the (42 T-M) design torque load specified by the reactor vessel loading specification for this faulted condition. The pump seizure torque will produce stresses in the motor housing and RPV RIP nozzle which are significantly less than Code allowable stresses.

15B.3.4.3 RIP Motor Housing Break

The motor housing and bottom flange are part of the RCPB and therefore are designed not to fail or rupture during normal, upset, emergency, or faulted plant conditions. Regardless of these criteria, and for the purpose of this evaluation, it is assumed that the housing fails creating a temporary small LOCA.

First it is assumed that the RIP impeller and shaft remain intact. The vertical blowout restraint rods prevent the motor and broken housing from being ejected from the RPV and damaging FMCRD piping and other equipment. The restraints are designed to elongate enough to close the 6 mm clearance of the impeller nozzle back seat and stop the discharge of reactor coolant out of the housing break.

Even if the impeller does not back seat, the discharge of reactor coolant will be restricted by the annular flow area between the pump shaft and stretch tube, etc. The ejection of the pump shaft is not credible because the pressure force resulting from a motor housing break pushes the shaft downward, and its upper diameter is larger than the penetration. The motor housing also prevents shaft ejection because, even when the housing has a complete circumferential break, the vertical restraints will not allow it to move away from the penetration.

The acceptance criterion for this event from the viewpoint of nuclear plant safety is that equivalent break size not exceed 20 cm^2, which is the design basis bottom break. The actual flow area is 20 cm^2 around the gap between the upper part of stretch tube and pump shaft. This small LOCA is detected by temperature, pressure, and/or level instrumentation for the RPV, drywell and/or RIP motor cooling circuit.

There are several different seals and sealed penetrations of the RIP motor housing which could be assumed to fail during reactor operation and would result in a very small LOCA. These seals include the RIP motor bottom flange, including the smaller auxiliary cover, motor power terminals, and motor speed detector. The failure of any of these seals would result in hot reactor coolant flowing down through the motor windings and damaging the winding insulation. This motor damage is not a nuclear safety problem.

In conclusion, the RPV RIP nozzle motor housing and associated seals, housing restraint system, and the normal makeup systems and ECCS are adequately designed to mitigate the consequences of a RIP motor housing break or housing seal failures.

15B.3.4.4 RIP Motor Housing External Loads

The motor housing, connected piping, and RIP motor heat exchanger are considered part of the RCPB and are therefore designed in accordance with the same codes and standards as the RPV. The housing is subjected to external loads from cooling water piping reactions or lateral seismic restraints (if they are used) during certain plant design conditions i.e., safe shutdown earthquake.

The RIP to Hx piping is designed with adequate flexibility between the fixed RIP motor heat exchanger and the motor housing to limit the loads and moments applied to the motor housing and consequently into the RPV bottom head to those specified in the reactor vessel loading specification.

Likewise, if lateral motor seismic restraints are incorporated in the design, the loads and moments applied to the motor housing will not exceed the values specified in the reactor vessel loading specification.

With the above criteria, the integrity of the RCPB can be assured under any plant conditions.

15B.3.4.5 Loss of Purge Flow

The RIPs are equipped with a shaft purge system which will provide a very small flow of clean demineralized CRD System water upward along the rotating RIP shaft (inside the stretch tube) into the RPV. The purpose of the purge system is to prevent the migration of radioactive reactor water down into the RIP motor. The purge flow enters the RIP shaft from two locations as shown in Figure 5.4-4.

Purge system piping from the RIP motor housing out to and including an outside containment isolation excess flow check valve is designed to maintain its integrity for all plant conditions, including safe shutdown earthquake. However, for the purposes of this evaluation, the following events are analyzed which result in loss of purge flow:

(1) Break of the purge piping inside or outside the containment

(2) Infrequent shutdown of the CRD pumps, including loss of power accident (LOPA)

(3) Inadvertent closure of valves in the purge supply flow path

Purge line break inside the containment is treated as a very small size LOCA. The event is mitigated by the normal ABWR coolant makeup systems to maintain proper RPV coolant inventory. The acceptance criterion for this event from the viewpoint of nuclear plant safety is that equivalent break size not exceed 20 cm^2, which is the design basis bottom break. The actual flow area of the double purge line break is 6 cm^2. This small LOCA is detected by temperature, pressure, and/or level instrumentation for the RPV, drywell and/or RIP motor cooling circuit. The normal makeup systems are designed to mitigate the consequences of this small LOCA.

Purge flow stoppage by CRD pumps stopping or purge line valve closure may result in damaging of the secondary seal, which would be replaced during the next scheduled maintenance of the RIP(s). The loss of purge flow could result in radioactive contamination of the motor which would be decontaminated during the next scheduled maintenance of the RIP(s). Purge flow stoppage will not result in additional stresses in the RPV nozzle.

In conclusion, the failure of the purge flow to the RIP will be mitigated by the normal makeup or normal maintenance procedures for secondary seal replacement.

15B.3.4.6 RIP Heat Exchanger Secondary Water Flow Loss

The RIPs are designed to operate normally in the following situations which are the acceptance criteria for these events:

(1) **Failure of Secondary Cooling Water—**The RIP motor shall be capable of continued rated power operation for 5 minutes following failure of the RCW. This time period allows corrective action to prevent an all-pump trip.

(2) **Hot Standby Without RCW—**With the RIP stopped, the motor shall withstand hot standby conditions for one hour with the RCW to the RIP motor heat exchanger (RMHx) shut off. This allows adequate time to take corrective action.

The evaluation of the RCW cooling water failure shows the motor water temperature increase will be as follows:

Time (min.)	Temp. (°C)	Status
0	55	RIP at maximum rated power and cooling water is shut off
2	60	Alarm
4	65	RIP auto runback and trip
65	70	Maximum motor cooling outlet temperature

The entire RIP motor housing, RIP motor heat exchanger, and interconnecting piping is designed for minimum 302°C at 8.62 MPa pressure. Therefore, an indefinite loss of RCW to the RIP motor heat exchanger will not affect the integrity of the RCPB.

The operator will receive a low RCW flow alarm and RMHx primary side inlet and outlet water temperature high alarm. If the RCW cannot be restored to the tripped motor, some damage to the winding insulation and/or secondary shaft seal, may occur. These components can be replaced according to normal RIP maintenance procedures.

15B.3.4.7 RIP Primary Cooling Water Loss

The RIP motor housing, RIP motor heat exchanger, and connecting piping are designed in accordance with the same codes and standards as the RPV. This design precludes the rupture of any of the RCPB components during any plant service condition. Regardless of this design criterion and for the purpose of this evaluation, it is assumed that a rupture of the 65A motor cooling water piping occurs or the RIP motor heat exchanger tubes fail.

Rupture of the motor cooling water piping will result in a small LOCA. This discharge of reactor coolant from the pipe break is restricted by the annulus between the pump shaft and the stretch tube. The acceptance criterion for this event from the viewpoint of nuclear plant safety is that equivalent break size not exceed 20 cm^2, which is the design basis bottom break. The actual flow area of the cooling water piping is restricted by the lower part of the stretch tube flow area is 10 cm^2. This small LOCA is detected by temperature, pressure, and/or level instrumentation for the RPV, drywell and/or RIP motor cooling circuit. The normal makeup systems are designed to mitigate the consequences of this small LOCA.

An RIP motor heat exchanger tube break will result in reactor coolant being discharged into the Reactor Cooling Water (RCW) System. This event will be detected by high motor cooling water temperatures, high RCW temperatures, high RCW surge tank level and/or high RCW radioactivity levels. The radioactivity will be contained in the RCW system and not discharged to the environment. As the reactor is being shut down, the discharge of reactor coolant into the RCW can be terminated by closing the primary containment RCW isolation valves after the RIPs have been stopped.

The heat exchanger tube leak rate will be the same as or less than the leak rate for motor cooling the pipe break. This is due to the fact that the leak rate is controlled by the annulus between the shaft and stretch tube.

It is assumed that any cause of RIP motor primary cooling water due to a rupture in the motor coolant circuit will damage the RIP motor winding insulation by the 278°C RPV water entering the motor. The motor can be replaced according to normal RIP maintenance procedures.

In conclusion, the ABWR RIP motor cooling system and normal ABWR coolant makeup systems are designed to detect and mitigate the consequences of a loss of RIP primary cooling water and consequent loss of reactor coolant.

15B.3.4.8 ABWR RIP Loose Part Prevention and Monitoring

The ABWR RIP is an assembly of many parts, some of which are inside the RPV. The parts in a majority of cases are held together by threaded fasteners such as studs, bolts, nuts, and screws. Although these types of fasteners make disassembly possible, they can become loose due to random vibration of the running pump and lead to gross failure of the other parts. Fragments of broken components can be transferred to the reactor internals and fuel. Due to criticality of loose parts, the RIP fasteners are engineered to be positively locked as described below:

(1) A lock sleeve and pin prevent loosening and disassembly of the impeller.

(2) Coupling stud has counter rotation thread to make it self-tightening. A locking mechanism prevents loosening and disassembly of the shaft-impeller-thrust bearing disk subassembly.

(3) The stretch tube, which has the function of securing the diffuser to the RIP nozzle, is tightened with hydraulic tensioning. The preload of the stretch tube is maintained by the stretch tube nut. The stretch tube nut is locked in place by a locking sleeve to the stretch tube.

(4) The optional diffuser wear ring is held in place by a retaining ring which is captured inside a groove in the diffuser.

(5) Piston rings are retained with grooves on the outside diameter of the diffuser.

In addition to positively locking of the most likely sources of loose parts, the ABWR RIP is adequately instrumented to provide early warning to the operator that failures within the RIP may be developing. The RIP is equipped with the following sensors/detectors:

(1) Vibration sensors which can detect effects of loosening, wear, unbalance, and dynamic changes.

(2) Motor cooling temperature sensors which can detect effects of abnormal load on the motor.

(3) Speed sensors which can detect effects of excessive wear, unbalance, and dynamic changes.

(4) Electrical power input (current and voltage) which provides the information about the overall performance of the RIP motor.

(5) Acoustic monitor—A high frequency response accelerometer is attached to the RIP motor casing which will provide signal of impacts and rubs within the motor.

15B.4 Essential Multiplexing System

The FMEA is described by the PRA fault tree analyses in Chapter 19 (see Subsections 19D.6.4.3 and Section 19Q.5) and the analysis of common-cause failure of multiplexer equipment in Appendix 19N. The system configuration fault definitions and provisions for fault tolerance are discussed and analyzed in the PRA.

Table 15B-1 Failure Mode and Effects Analysis for FMCRD

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
1.	CRD housing	Provides CRD pressure boundary and mounting	Rupture, inside or outside vessel	Stress corrosion; weld failure	Forced outage	Possible rod ejection. May have scram failure on affected rod	Rod ejection protection by integral internal blowout support	Drywell leakage; failure to scram
2.	Middle housing	Houses mechanisms of CRD						
2.a	Ball check valve	Prevents rod ejection if scram line breaks	Stick open	Foreign object; misassembly	Insert rod and render inoperative	Loss of rod ejection prevention function	Brake	Surveillance test
2.b	O-ring	Seal joint between middle and lower housing	Leaks	Misassembly; age	Possible outage extension for repair	None	Dual O-rings	Drywell leakage
3.a	Lower housing	House shaft and seal assembly	Rupture	Stress corrosion	Forced outage	Possible rod ejection. May have scram failure on affected rod	Rod ejection protection by engagement of anti-rotation device with backseat of middle flange; brake	Drywell leakage
			Distortion	Residual stress	Possible outage extension for repair	Minor; may have scram failure on affected rod	Not required	Inspection

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
3.b	Flange bolts	Couple housings	Break	Overstress; material failure	Forced outage	Possible rod ejection; failure of affected rod to scram	Rod ejection protection by engagement of anti-rotation device with backseat of middle flange	Drywell leakage
4.	Seal housing	Support and house bearings, seals, shaft	Distortion	Residual stress	Reduced CRD life; outage extension for repair	None	Not required	Inspection
4.a	Lower radial ball bearings	Support drive shaft	Wear, ball or race failure	Misassembly; dirt, material defect	Reduced CRD life; outage extension for repair	None	Dual bearing	High motor current, inspection
4.b	Upper radial ball bearing	Support drive shaft	Wear, ball or race failure	Misassembly; dirt, material defect	Thrust bearing loaded radially; increased friction	None	None	High motor current, inspection
4.c	Thrust bearings	Carry rotating assembly weight	Wear, ball or race failure	Misassembly; dirt, material defect	Radial bearings thrust loaded; increased wear	None	None	High motor current, inspection
4.d	Drive shaft and seal system	Connect motor and spindle; seal reactor pressure	Wear	Dirt; aging	Possible outage extension for drive repair	None	Seal drain; dual seals	Drywell leak rate; inspection

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
4.e	Seal rings	Compress seals	Break	Stress corrosion	Possible outage extension for drive repair	None	Seal drain; dual seals	Drywell leak rate, inspection
4.f	Seal retainer pins	Prevent seal rotation	Break	Misassembly	Possible outage extension for drive repair	None	None	Leakage around drain path, inspection
5.	Drive shaft	Couples motor and spindle	Break	Misassembly; stress corrosion	Insert rod and render inoperative	Possible loss of drive-in capability. Does not affect scram function.	Only one rod affected	Rod position indication
6.	Key R, pins	Couples motor, shaft	Break, shear	Misassembly; faulty part	Insert rod and render inoperative	Possible loss of drive-in capability. Does not affect scram function.	Only one rod affected	Rod position indication
7.	Key B	Couples motor, shaft, spindle	Break, shear	Misassembly; faulty part	Insert rod and render inoperative	Possible loss of drive-in capability. Does not affect scram function.	Only one rod affected	Rod position indication

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
8.	Thrust bearing locknut	Takes vertical load of rod, rotary drive parts	Unscrew	Misassembly	Possible extended outage for drive repair	Possible loss of drive-in capability. Does not affect scram function	Only one rod affected	Inspection, radial bearing wear
9.	Spring washers, withdraw buffer	Absorb impact of full rod withdrawal	Break	Stress corrosion	Possible extended outage for drive repair	None	None	Inspection
10.	Weigh spring	Part of rod separation detection system	Break; loss of separation signal	Stress corrosion; low cycle fatigue	Insert rod and render inoperative	Possible rod drop	Latches on hollow piston	Rod position indication, scram
11.	Spindle adapter	Couples spindle to driving system	Outer keyway jams key. Loss of or false separation signal	Crud, corrosion, galling	Insert rod and render inoperative	Possible rod drop	Rod scram, latches on hollow piston	Rod position indication, scram
12.	Spindle adapter seat	Spindle backseat and lock when mechanism is removed	Splines shear or otherwise damaged	Misassembly	Whole drive must be removed, requiring rod withdrawal; possible extended outage	Loss of rod ejection protection function for total failure of all flange bolts	Low probability of total failure of all flange bolts coincident with backseat spline failure	On drive removal, spindle does not seal; high Rx leak

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
13.	Weigh spring retainer	Restrains the weigh spring	Breaks; possible loss of separation signal	Stress corrosion; misassembly	Not detectable until separation is indicated by position indication anomaly	Slight separation w/o indication; possible rod drop	Latches on hollow piston	Rod position indication error
14.	Ball nut and hollow piston rollers	Support ball nut	Breaks, seizes, increases friction	Impact at scram	Possible rod insertion and switchout	None	Excess motor torque available	High drive motor current
15.	Lead screw (spindle)	Drives ball nut & rod	Distorts; increased friction & wear	Residual stress	Insert rod and switchout drive	None	Excess torque available	High drive motor current
16.	Ball nut	Translate spindle rotation to rod linear motion	Balls jam, friction, wear	Ball failure, crud, foreign object	Insert rod and switchout drive	None	Three ball paths; excess torque available	High drive motor current
16.a	Ball nut return tube	Retain and recirculate balls	Breaks, balls released	Stress corrosion; over–tension at assembly	Rotation interference; insert and switchout drive	None	Redundant return tubes	High motor current
16.b	Ball nut return tube	Retain and recirculate balls	Breaks, balls released	Stress corrosion; overtension at assembly	Interference with weighing system, loss of separation signal	Possible slight separation w/o signal; possible rod drop	Latches on hollow piston	Rod position indication anomaly
17.	Hollow piston	Piston for hydraulic scram	Tube distorts	Residual stress	Increased friction; insert rod and switch out	Possible increase in scram time	Shutdown margin	Slow scram, friction test; rod separated during withdrawal

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
17a	Hollow piston	Piston for hydraulic scram	Binds in labyrinth seal	Trapped crud	Possible forced outage	Possible failure to scram	Shutdown margin	Position indication, scram time; rod separates during withdrawal
18.	Latch fingers	Support hollow piston (and control rod) after scram	Jam due to crud/foreign object	Crud, foreign object	Rod fallback after scram; forced outage	Fail to maintain scram on one rod	Redundant fingers; shutdown margin	No scram indication; rod position
18.a	Latch fingers	Support hollow piston (and control rod) after scram	Break	Overstress; material problem	Rod fallback after scram; forced outage	Fail to maintain scram on one rod	Redundant fingers; shutdown margin	No scram indication; rod position
19.	Latch springs	Position latch fingers to hold rod in scram position	One or more break	Low cycle fatigue; misassembly	Rod fallback after scram; forced outage	Fail to maintain scram on one rod	Triple redundant springs on each latch	No scram indication; rod position
20.	Hollow piston assembly screws	Attach fittings to hollow piston	Loosen, jam against guide tube, slow scram	Vibration misassembly	Possible outage extension for drive repair	None	Shutdown margin	Scram time; drive motor current
21.	Screw, tie bar	Mount for rod position magnet	Break, loss of rod scram indication	Low cycle fatigue; misassembly	Possible outage extension for drive repair	None	Rod position indication	No scram confirmation
22.	Scram buffer springs	Absorb impact of scram stroke	Break, lower spring constant	Low cycle fatigue, stress corrosion	Possible outage extension for drive repair	None	Drive designed for inoperative buffer	Overhaul; inspection

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
22.a	Scram buffer springs	Absorb impact of scram stroke	Jam in the compressed position, reduced buffering	Foreign material	Possible outage extension for drive repair	None	Drive designed for inoperative buffer	Overhaul; inspection
23.	Scram switch	Provide confirming scram completion signal	Fail open, loss of full insertion and scram signal	Bad contacts; broken parts	Possible outage extension for drive repair	None	Redundant switches; drive synchro position indication	Loss of signal
23.a	Scram switch	Provide confirming scram completion signal	Fail closed, continuous full insertion signal	Stuck contacts	Possible outage extension for drive repair	None	Detected on rod withdrawal	Position indication anomaly
24.	Separation switch	Indicates hollow piston/ball nut separation	Fail open, loss of separation signal	Bad contacts broken parts	Rod insertion required; possible outage extension to repair drive	Precursor to rod drop	Redundant switches	Position indication anomaly
24.a	Separation switch	Indicates hollow piston/ball nut separation	Fail closed, false separation signal	Stuck contacts	Rod withdrawal block; rod insertion required; possible outage extension to repair drive	None	Fail safe mode	Position indication anomaly
25.	Upper housing roller	Hollow piston guide	Freeze on shaft	Crud; corrosion	None immediate	None	Other close clearances	Scram time; motor current

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
26.	Spindle roller	Stabilizes spindle rotation	Freeze on shaft; wear	Crud; corrosion	None immediate	None	Redundant on hollow piston	Scram time; motor current
27.	Spindle bushing	Supports spindle roller assembly	Seizes or binds on bolt	Crud, improper heat treatment	Drive replacement required	None	None	Motor current; in extreme, motor stalls
28.	Spindle adapter bolt	Attaches spindle to spindle adapter	Loosens	Backlash in drive train; vibration	Drive replacement required	None	None	Position indication anomaly
29.	Guide tube	Provides cylinder for hollow piston	Distort; higher friction	Residual stress	Reduced drive life	None	Excess motor torque available	Motor current
30.	Guide rail	Align ball nut and hollow piston	Becomes loose	Misassembly; fatigue	Drive replacement may be required	None	None	Unable to withdraw rod
31.	Labyrinth seal	Forms seal between reactor pressure and drive pressure	Distort, friction increase	Residual stress	Possible outage extension for drive repair	Possible increase in scram time	Excess drive scram water pressure and motor torque; shutdown margin	Scram time; friction test
32.	Motor	Drive spindle to set rod position	Stall	Short, open winding; bearing seizure	Insert rod and switch out	Motor-driven insertion function following scram lost on affected drive	Only one rod affected; shutdown margin	Rod position indication

Table 15B-1 Failure Mode and Effects Analysis for FMCRD (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
33.	Synchro	Generate and transmit rod position information	Electrical failure	Short/open	Insert rod and switch out	None	Only one rod affected; shutdown margin	Rod position anomaly; signal lost
34.	Brake	Hold rod drive spindle to prevent rod drift	Lockup	Brake electrical failure; jam	Insert rod and render inoperative	Motor-driven insertion function following scram lost on affected drive	Only one rod affected; shutdown margin	Rod position indication; brake surveillance test
34.a	Brake	Hold rod drive spindle to prevent rod drift	Fail to brake	Wear, wet, mechanism jammed	Insert rod and render inoperative	Possible rod ejection	Ball check valve	Rod position indication, brake surveillance test
35.	Screw (ring flange)	Attaches shaft bearing retainer to shaft housing	Break; seal and bearing shoot-out	Overtorque; material/ manufacturing flaw	Possible outage extension for drive repair	None	None	Inspection

Table 15B-2 Failure Mode and Effects Analysis for HCU Charging Water

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
1.	Charging water header accumulator	Maintain charging water header pressure	Leak, rupture	Material failure, overstress	Loss of charging pressure causes scram	None	Two pumps; low pressure alarm	Charging water header pressure alarm
2.	Suction filter	Filter suction water before CRD pump	Plug	Contamination	Loss of charging pressure requires scram	None	Redundant, independent filters	Differential pressure
3.	CRD pump	Provides purge and charging water for CRDs	Seize, stall	Trash, motor failure	Loss of charging water requires scram	None	Alarmed, redundant, independent pumps	Charging pressure, purge flow
4.	CRD pump discharge filter	Filters charging and purge water	Plug	Contamination	Loss of charging water requires scram	None	Redundant, independent filters	Differential pressure
5.	Flow element	Measure total flow to HCUs	Blocked	Trash	May require plant shutdown for repair	None	Low pressure alarm	Loss of flow measurement
6.	Purge water flow control valve	Control purge water flow	Fail closed	Crud; controls	May require plant shutdown for repair	None	Redundant, independent flow control valves	CRD flow
6.a.	Purge water flow control valve	Control purge water flow	Fail open	Crud; controls	May require plant shutdown for repair	None	Flow restricting orifices in each HCU; redundant, independent flow control valves	CRD flow
7.	Filters, check valves, accumulators within HCU	Various	All	All	Loss of scram or purge water on two drives; rod insert and switch out may be required	None	Shutdown margin	Instrumented and alarmed parameters

Table 15B-2 Failure Mode and Effects Analysis for HCU Charging Water (Continued)

Item	Component Identification	Function	Failure Modes	Causes of Failure Mode	Effect on Availability	Effect on Safety	Compensating Provisions	How Detected
8.	Scram valve	Initiate hydraulic scram	Scram solenoid pilot valve fails closed	Contaminated air, debris accumulation	Rod insertion and switch out	None, only two rods affected on individual HCU	Shutdown margin	Position indication
			Scram solenoid pilot valves fails closed (common mode)	Dirty or contaminated air supply	Plant shutdown for repair	Common mode loss of normal scram	ARI for ATWS; air header dump valves for normal scram. Electric driven insertion for all drives. Viton-B solenoid valve seats.	Position indication
			Scram valves leak (common mode)	Slow drop in scram air header pressure	Plant shut down for repair	None. Excessive leakage will result in low HCU charging pressure alarm and scram before HCU accumulators are depleted.	Low charging water header pressure alarm and scram	Low scram air header pressure indication and alarm. Low charging water header pressure indication/ alarm and scram.

Table 15B-3 EMS Failure Mode and Effects Analysis

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection	Remarks
Remote mux unit (RMU)	Condition, format and transmit sensor and control signals	Loss of signal or false signal	Loss of electrical power, solid state device failure, loose connection, broken wire	Loss of sensor/ control signal or false signal rejected	Self-test feature and device annunciation in control room	Immediate detection of loss of signal, system test for false signal
Control room mux unit (CMU)	Condition, format and transmit sensor and control signals	Loss of signal or false signal	Loss of electrical power, solid state device failure, loose connection, broken wire	Loss of sensor/ control signal or false signal rejected	Self-test feature and device annunciation in control room	Immediate detection of loss of signal, system test for false signal
Multiplexer control units (MCU)	Convert digital to optical signals and vice versa	Loss of signal or false signal	Loss of electrical power, solid state device failure, loose connection, broken wire	Loss of sensor/ control signal or false signal rejected	Self-test feature and device annunciation in control room	Immediate detection of loss of signal, system test for false signal
Fiber optic cable	Transmit optical signals	Severed cable or misalignment of junctions	External force to break cable or bend junctions	Loss of signal on damaged cable only	Continuous, automatic system self-test	One cable in each loop must fail

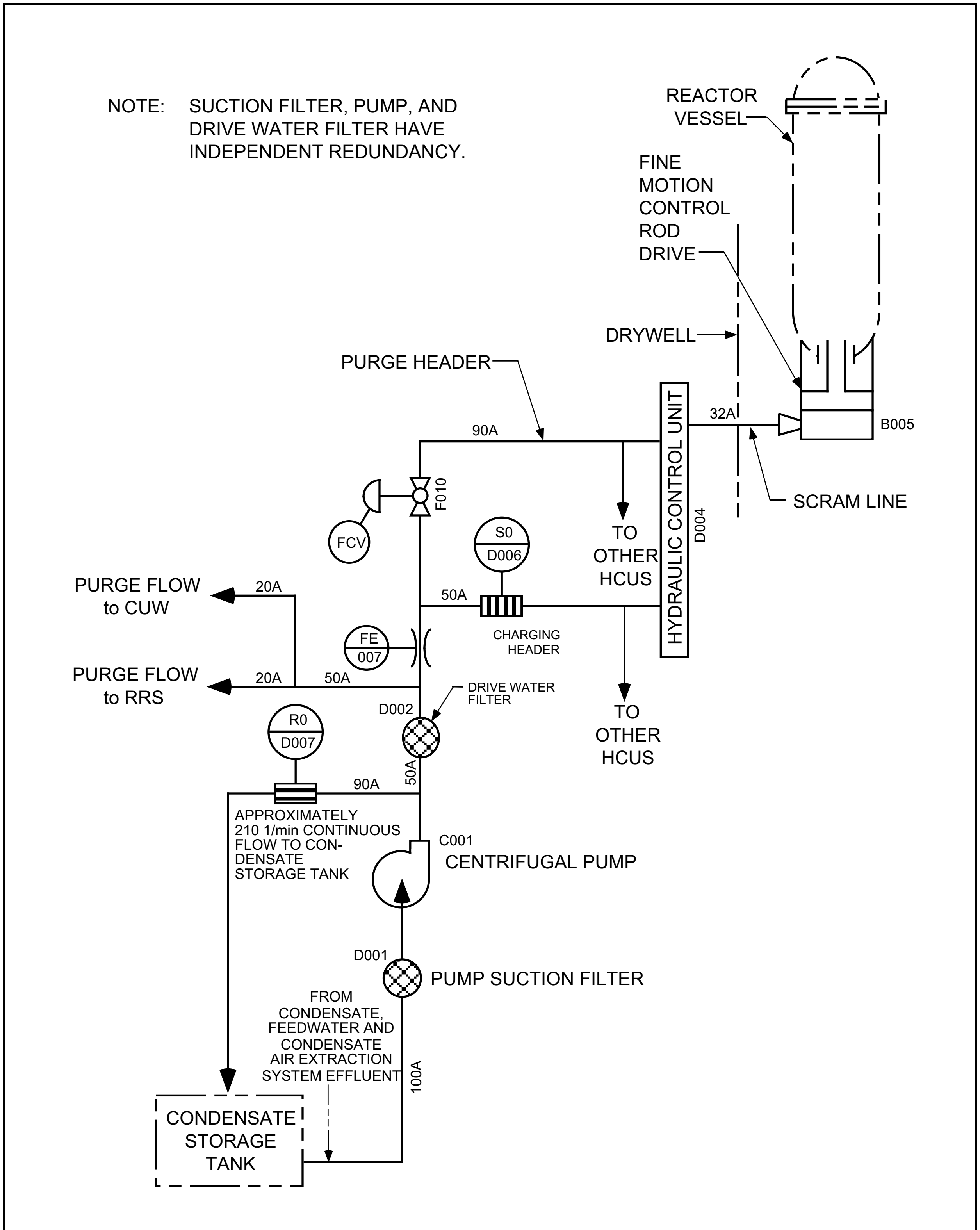


Figure 15B-1 Simplified CRD System Process Flow Diagram

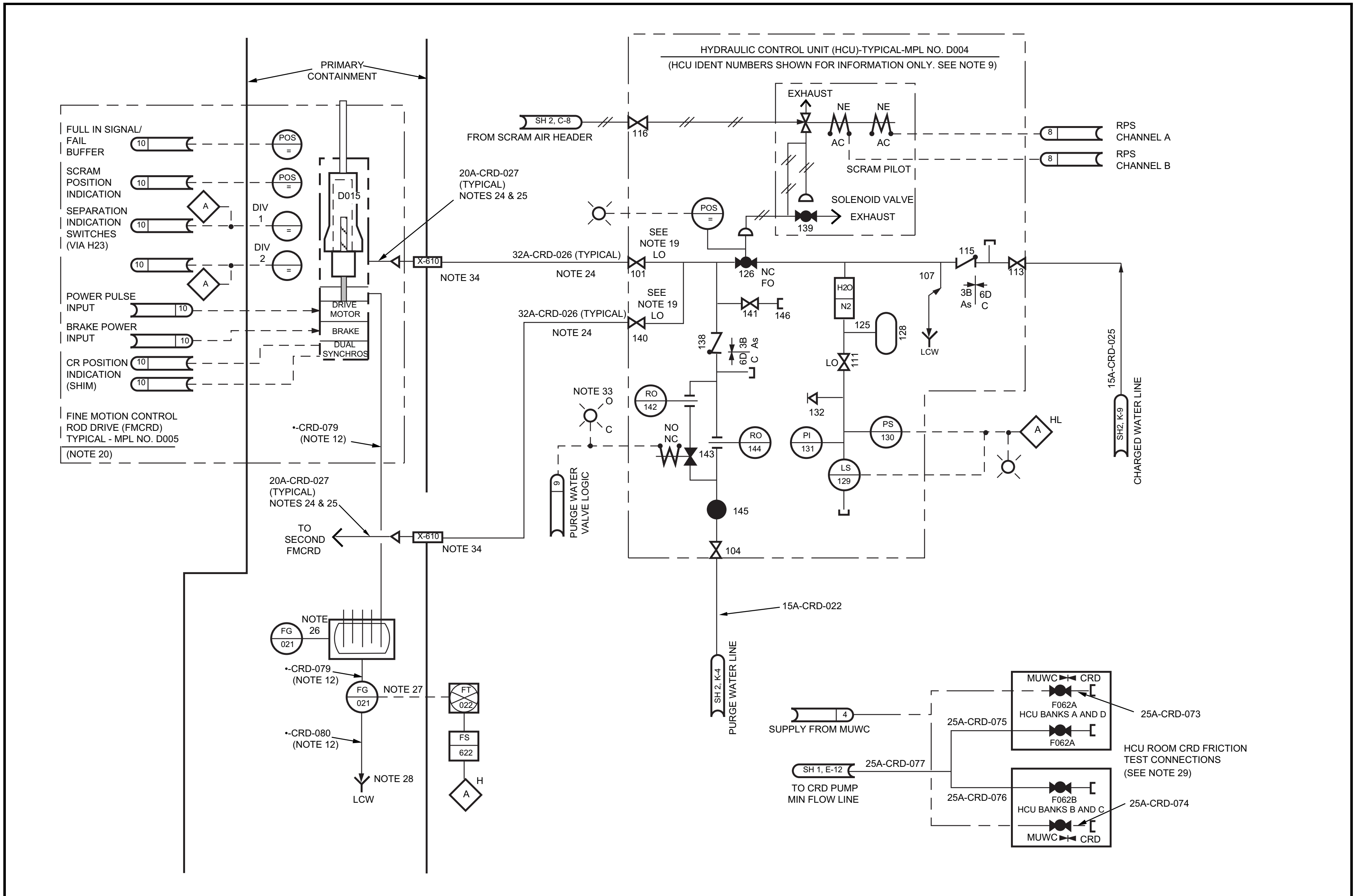


Figure 15B-2 Simplified Hydraulic Control Unit P&ID

The following figure is located in Chapter 21:

Figure 15.B-3 Fine Motion Control Rod Drive

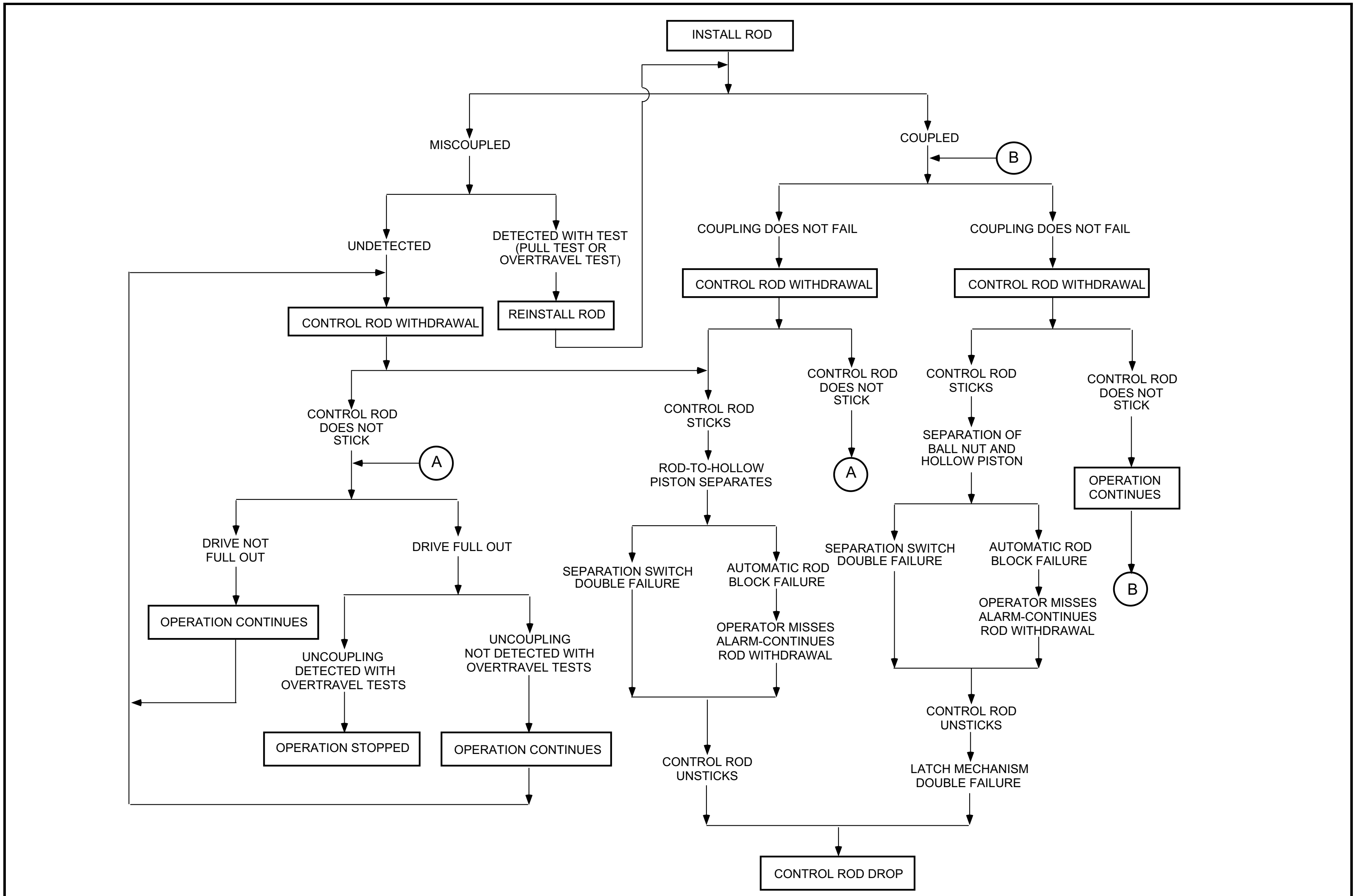


Figure 15B-4 Control Rod Drop Accident Scenario for FMCRD

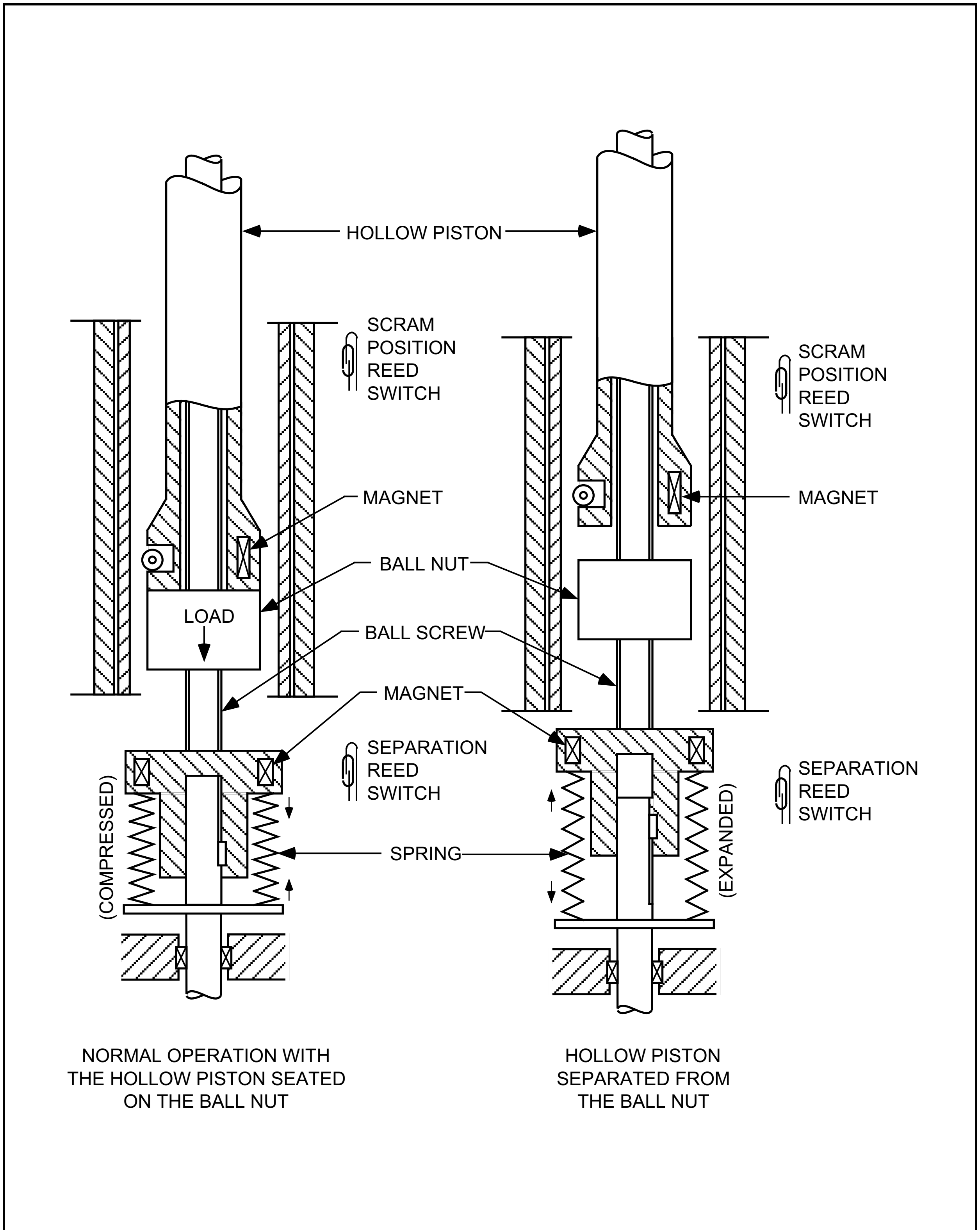


Figure 15B-5 Control Rod Separation Detection

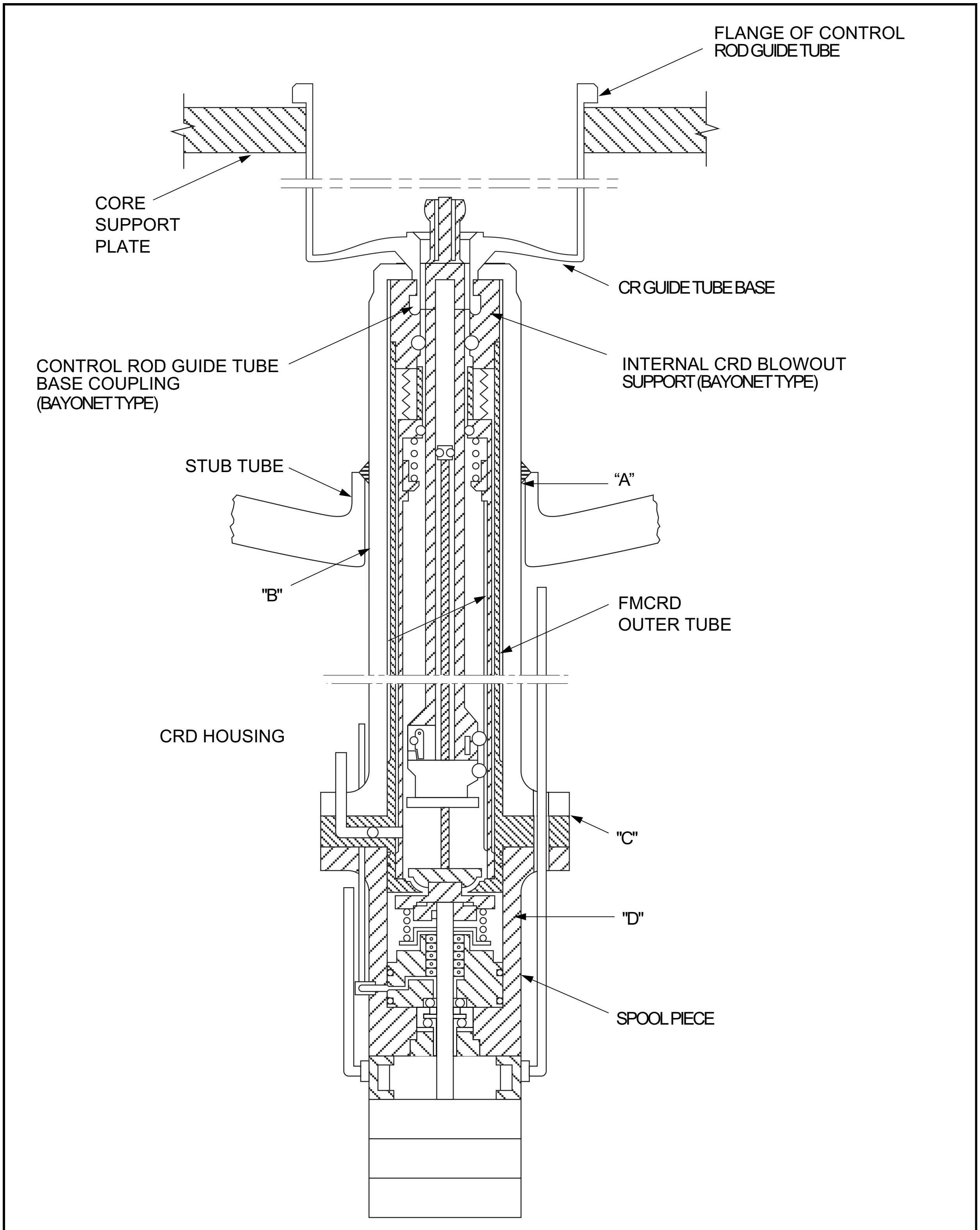


Figure 15B-6 Internal CRD Blowout Support Schematic

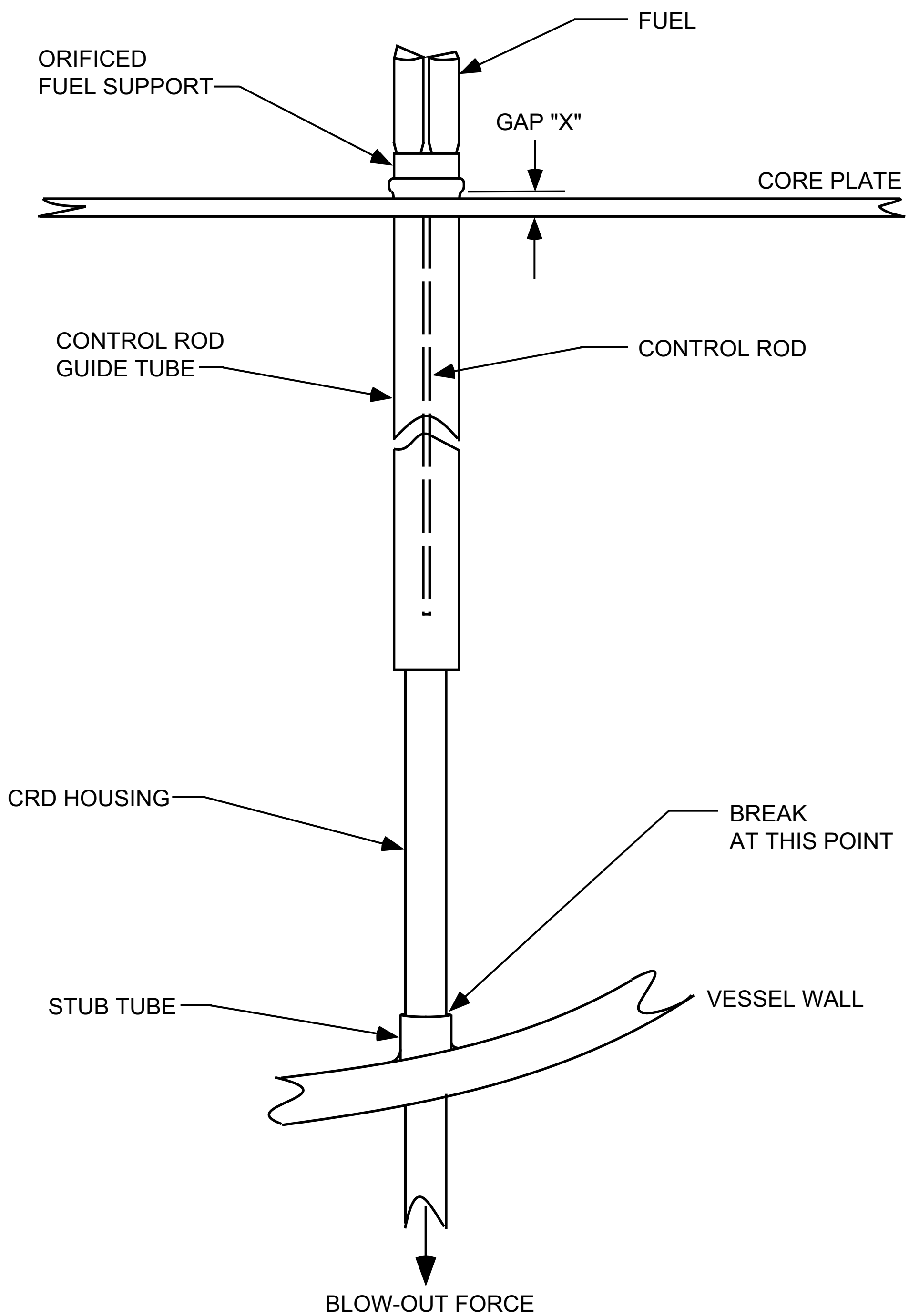


Figure 15B-7 FMCRD Internal Support

15C Not Used

15D Probability Analysis of Pressure Regulator Downscale Failure

15D.1 Introduction

A reliability analysis has been performed on the ABWR pressure regulator fault tolerant controller architecture. The purpose is to determine the frequency at which simultaneous closure of all four turbine control valves (TCVs) might be expected to occur, initiated by the control system due to failure of the pressure regulator.

Fail closure of all four turbine control valves initiated by failure of the pressure control system is defined as “Pressure Regulator Downscale Failure (PRDF)”.

15D.2 System Description

The elements of the control system are depicted in Figure 15D-1. The pressure sensor signals are subject to range limit checks to identify a fully failed sensor/signal. A selection logic is used to validate when all the three inputs are available and good. Upon failure detection of one signal, the validation logic automatically reverts to a high value gate (HVG) of the two remaining active signals.

The control system consists of three identical processing channels with necessary hardware and firmware. Means are provided to transfer data between processing channels. To avoid processing channel output divergence, the processing channels compare and vote on calculated integrator state variables. The signal voting and interprocessor communication is implemented to assure that no more than one sample period delay occurs between sampling the inputs and using them in the processor calculations.

Diagnostics are conducted by comparision of internal digital turbine control valve (TCV) position demand signals to determine the failure of any output signals to TCVs. The TCV demand output signal failure from the controller is considered as a channel failure. If two failures are detected, a turbine trip is initiated to avoid a PRDF.

15D.3 Analysis

15D.3.1 Analytical Conditions

(1) The assumption for this analysis is that TCV demand output signal failures from two channels having occured, one or more of these failures not detected by the detection diagnostics will result in closure of all four TCVs.

(2) It was assumed that the failure rates represent failures that result in loss of output.

(3) Mean-time-to-repair is 10 hours.

(4) Failure rates for electronic modules are estimated, based on anticipated complexity of the circuit functions. (processor failure rate = 10^{-5}/h).

15D.3.2 Approach

Using the system block diagram of Figure 15D-1, the event trees (Figures 15D-2 through 15D-4) were constructed to show the failure paths which could result in pressure regulator downscale failure. Basically, there are two ways that all four turbine control valves can be closed “simultaneously”: (1) failure of combinations of all four valves, or (2) failure of any two of the three channels, with the two possible combinations of at least one of the two failures not detected. The logic equations (Table 15D-1)were written from the event trees and after simplification were evaluated for the frequencies identified in the introduction.

15D.4 Results

The frequency of inadvertent closure of all 4 TCVs initiated by failure of the pressure regulator is found to be extremely low so that the event can be treated as a limiting fault (See Subsections 15.2.1.1.1 and 15.2.1.1.2.2).

Table 15D-1 Logic Equations

[Not part of DCD (Refer to SSAR)]

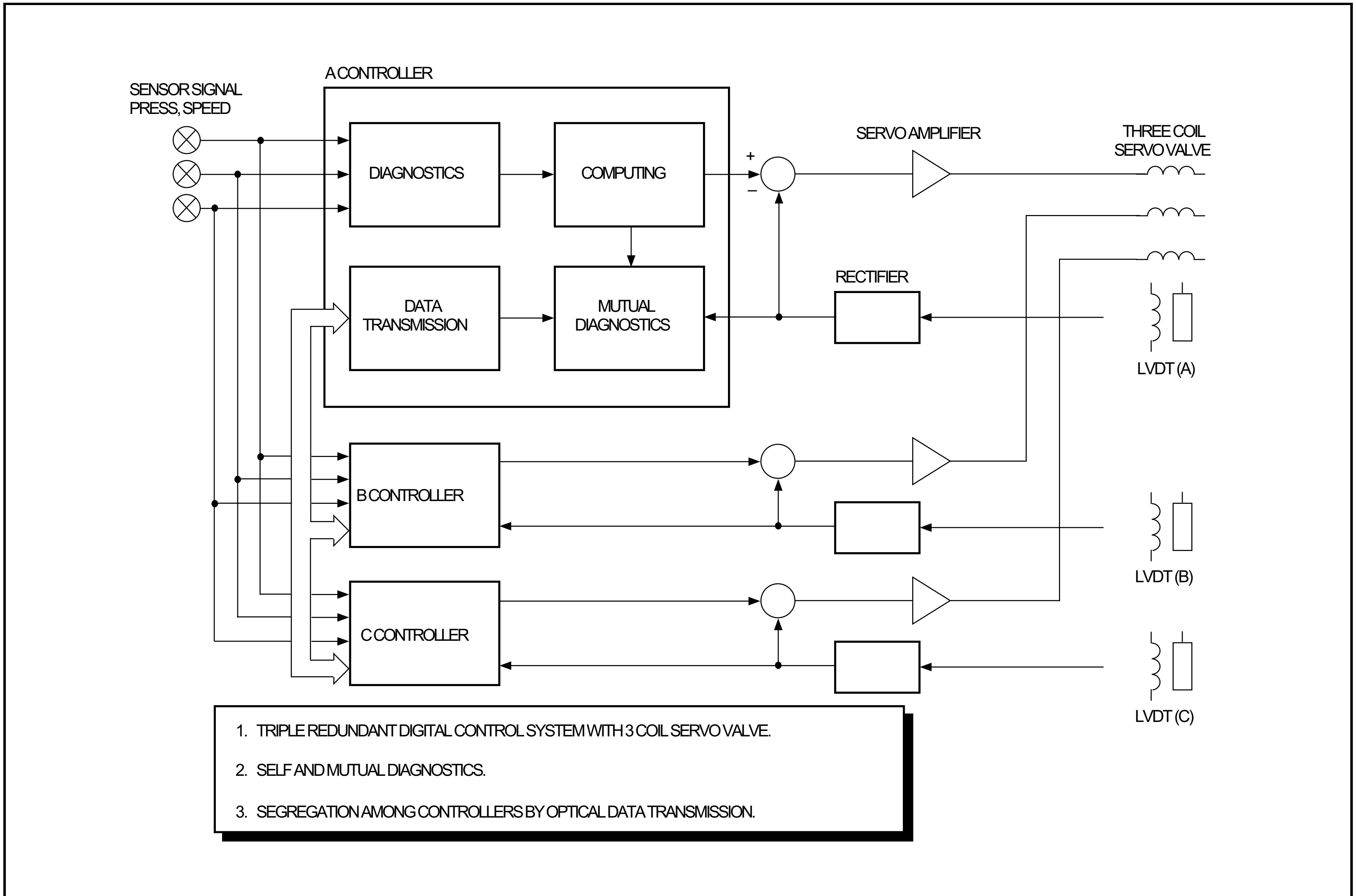


Figure 15D-1 Triple Redundant Control System

Figures 15D-2 through 15D-5 are not part of DCD (Refer to SSAR)

15E ATWS Performance Evaluation

15E.1 Introduction

Typical ATWS events are analyzed for ABWR to confirm the design for ABWR.

The procedure and assumptions used in this analysis are consistent with those used in the analyses for the operating plants as documented in Section 15E.8, Reference 15E-1.

15E.2 Performance Requirements

As identified in Section 15E.8, Reference 15E-1, the design should meet the following requirements:

(1) **Fuel Integrity—**The long-term core cooling capability shall be assured by meeting the cladding temperature and oxidation criteria of 10CFR50.46 (i.e., peak cladding temperature not exceeding 1204°C, and the local oxidation of the cladding not exceeding 17% of the total cladding thickness).

(2) **Containment Integrity—**The long-term containment capability shall be maintained. The maximum containment pressure shall not exceed the design pressure (3.16 kg/cm^{2}g) of the containment structure. The suppression pool temperature shall be limited to values shown in Table 15E-1.

(3) **Primary System—**The system transient pressure shall be limited such that the maximum primary stress within the reactor coolant pressure boundary (RCPB) does not exceed the emergency limits as defined in the ASME Code, Section III. If practical, the peak pressure should be limited to the upset limits in order to allow for more economical equipment design.

(4) **Long-Term Shutdown Cooling—**Subsequent to an ATWS event, the reactor shall be brought to a safe shutdown condition, and be cooled down and maintained in a cold shutdown condition.

These performance requirements are summarized in Table 15E-1.

15E.3 Analysis Conditions

Due to the extremely low probability of the occurrence of an ATWS, nominal parameters and initial conditions have been used in this analysis and also in Section 15E.8, Reference 15E-1. Tables 15E-2 and 15E-3 list the initial conditions and equipment performance characteristics, which are used in the analysis.

15E.4 ATWS Logic and Setpoints

The mitigation of ATWS events is accomplished by a multitude of equipment and procedures. These include ARI, FMCRD run-in, feedwater runback, RPT, recirculation runback, ADS inhibit, and SLCS. The logic of this ATWS mitigation is presented in Figures a, 15E-1b and 15E-1c. The following are the initiation signals and setpoints for the above response:

(1) ARI and FMCRD run-in

- High pressure (7.76 MPaG), or
- Level 2

(2) SLCS initiation

- High pressure (7.76 MPaG), and SRNM ATWS permissive for 3 minutes, or
- Level 2 and SRNM ATWS permissive for 3 minutes, or
- Manual ARI/FMCRD run-in signals and SRNM ATWS permissive for 3 minutes.

(3) RPT (RIPs not connected to M/G set)*

- High pressure (7.76 MPaG)

(4) RPT (RIPs connected to M/G set)

- Level 2

(5) Recirculation runback (10%/second)

- Any scram signals, or
- Any ARI/FMCRD run-in signals

(6) Feedwater runback

- High pressure (7.76 MPaG), and SRNM ATWS permissive for 2 minutes

(7) ADS inhibit

- Automated initiation of ADS is inhibited unless there is a coincident low reactor water level signal (level 1.5) and an APRM ATWS permissive signal

* Also tripped at Level 3, which is not a part of ATWS mitigation.

15E.5 Selection of Events

Based on conclusions from the evaluations for operating BWR plants as documented in Section 15E.8, Reference 15E-1, the following limiting events were selected to demonstrate the performance of the ATWS capabilities. They are grouped into three categories. The first category includes events which demonstrate ATWS mitigation on the most severe and limiting cases. The second category has events which are generally less severe for ATWS analysis but are analyzed to show the sensitivity of key ATWS parameters to these events. In each above case, the recirculation pump trip, ARI, electrical insertion of the control rod drives, boron injection and other ATWS mitigation actions are assumed to occur on the appropriate signals. No operator action is assumed, unless specifically mentioned. The third category covers the cases which have only minor impact to the reactor vessel containment. They are discussed briefly to support the assumption that they do not significantly influence the design of ATWS mitigation. No analysis was performed for events in the third category.

Category 1. Limiting Events

(1) Main Steam Isolation Valve (MSIV) Closure

Generic studies have shown that this transient produced high neutron flux, heat flux, vessel pressure, and suppression pool temperature. The maximum values from this event are, in most cases, bounding of all events considered.

(2) Loss of Normal AC Power

This transient is less severe than the MSIV closure in terms of vessel pressure, heat flux, neutron flux, and suppression pool temperature. However, because the loss of power to the condensate and feedwater pumps causes the feedwater flow to cease, very low vessel water levels are expected. Thus, the capability of the ECCS to recover the water level will be tested.

(3) Loss of Feedwater

This transient is less severe than the above two events. However, it is the only event which is mitigated by ARI or FMCRD run-in initiated from the low level signals. Thus, this event is analyzed to show that the low level trips are capable to mitigate the event.

(4) Loss of Feedwater Heater

This transient is very mild, as the increase of neutron flux never reaches the scram setpoint. The reactor shutdown is initiated by operator action. The main concern is that peak linear heat generation rate may exceed performance criteria when FMCRD run-in is initiated. The analysis is to show that the recirculation runback can mitigate this event.

Category 2. Moderate Impact Events

(5) Turbine Trip with Bypass Valves Open

This transient usually produces higher neutron flow heat flux and vessel pressure than those from MSIV closure event due to the fast closure of the turbine stop valves. However, the availability of the main condenser significantly reduces the amount of steam discharged into the suppression pool.

(6) Loss of Condenser Vacuum

The initial transient behavior of this event is similar to that of a turbine trip, as the reduction of vacuum in the main condenser initiates turbine stop valve closure. When the isolation setpoint is reached, the MSIVs start to close. The event follows the pattern of MSIV closure in suppression pool temperature and containment pressure.

(7) Feedwater Controller Failure at Maximum Demand

This transient produces peak values of key parameters similar to those of a turbine trip case. The availability of the main condenser significantly reduces the load of suppression pool from steam discharge from SRVS.

Category 3. Minimum Impact Events

(8) Recirculation Flow Controller Failure at Maximum Demand

This transient is not severe enough to trip any ATWS logic nor initiate HPCF or RCIC flow. It is considerably milder than the MSIV closure or turbine trip ATWS cases. This is a short-term transient with sudden power rise and relatively small pressure increase. The entire transient is over within 30 seconds, by which time the reactor settles out to a new equilibrium condition of less than 100% rated power. Since the peak pressure stays below the lowest SRV setpoint, steam flow discharge to the suppression pool does not take place.

The transient is not severe enough to trip the ATWS logic or initiate HPCF or RCIC flow, because the feedwater and level control is maintained. Manual ARI/FMCRD run-in has to be initiated by operator in case manual scram fails. The success of ARI or FMCRD run-in with recirculation runback can bring the reactor to hot shutdown just like normal scram. If control rods fail to insert after operator action, the boron injection would bring the reactor to hot shutdown.

(9) Startup of the Idle Recirculation Pump

The abnormal startup of an idle recirculation pump requires the inverter to provide electric current much higher than normal to counter the much higher reverse flow.

This overcurrent requirement activates the overcurrent protection logic of the electric bus which supplies the power to the idle RIP. This electric bus is tripped by the protection logic. Consequently, the other RIPs powered by this electric bus are also tripped. Therefore, this event is similar to the trip of three recirculation pump events. Since the scram is never initiated and there is no steam discharged into the suppression pool, there is no impact to the ATWS mitigation design. Therefore, further transient-specific analyses have not been done.

(10) Inadvertent Opening of All Bypass Valves

This event initiates a gradual decrease of the vessel pressure and power. It is followed by a rapid rise of pressure and power after the closure of MSIV on low steamline pressure. The characteristics of the remaining portion of this transient are very much the same as the MSIV closure event, except it starts at a much lower initial power level. The steam discharged into the containment is much less than that in the MSIV closure event. The same conclusion is also true for other key parameters.

(11) Shutdown Cooling (RHR) Malfunction–Decreasing Temperature

This event can only occur at very low pressure. The shutoff head of the shutdown cooling pumps is less than 2.07 MPaG. In this condition, the reactor has almost no voids in it and therefore little, if any, positive reactivity is increased. Hence, this event is not considered further.

All transient analyses, unless otherwise specified, were performed with the REDYA code. Other codes used in special analysis are ODYNA and PANACEA.

15E.6 Transient Responses

For every event selected for analysis, three cases were analyzed. The first one shows the ATWS performance with ARI. This case is intended to show the effectiveness of the ARI design. The second case, which uses FMCRD run-in, assuming a total failure of ARI, was performed to show the backup capability of FMCRD run-in. The third case was analyzed to show the in-depth ATWS mitigation capability of the ABWS. In this case, both ARI and FMCRD run-in are assumed to fail. Automatic boron injection with a 180-second delay is relied upon to mitigate the transient event.

If the ARI and FMCRD run-in fail at the same time, which has extremely low probability of occurrence, the peak reactor pressure would still be controlled by the recirculation runback and relief valves. However, the nuclear shutdown will then rely on the automatic SLCS injection. The boron would reach the core 60 seconds after the initiation. The operation of both SLCS pumps generates a 6.31 E-3 m^3/s volumetric flow rate of sodium pentaborate. The nuclear shutdown would begin when boron reaches the core.

Reference 15E-2 of Section 15E.8 provides results of an ATWS stability study performed with the 3-D TRACG code. The 3-D TRACG analyses covered the most limiting ATWS events and demonstrated that the mitigation design for ATWS is effective and it showed that the study of this appendix conducted with the REDYA code is bounding.

15E.6.1 Main Steam Isolation Valve Closure

This transient is considered an initiating event caused by either operator action or instrument failure. Scram signal paths that are assumed to fail include valve position, high neutron flux high vessel pressure, and all manual attempts. A short time after the MSIVs have closed completely, the ATWS high pressure setpoint is reached, which initiates four of the ten recirculation pumps to trip and the rest start to runback. The combined effect of the trip and runback reduces the core flow and increases core voids, thereby reducing power generation which limits pressure increase and steam discharge to the suppression pool. The ATWS high pressure signal causes the actuation of the ARI and the electric insertion of the FMCRD. The insertion of the control rods is successful in bringing the reactor to hot shutdown. Peak values of key parameters are shown in Table 15E-4 for the ARI case and Table 15E-5 for the FMCRD run-in case. In the case that control rods fail to insert, the reactor will be brought to hot shutdown by automatic boron injection in about 19.4 minutes from the beginning of the event. The transient behavior of this case is listed in Table 15E-6. The reactor system response is presented by Figure 15E-2 for ARI activated, Figure 15E-3 for FMCRD run-in case and Figure 15E-4 as SLCS operating, respectively. The normalized axial power shape change during FMCRD run-in is presented in Figure 15E-5. The increase of the local power is not expected to damage the fuel. Therefore, the performance criteria are met.

15E.6.2 Loss of AC Power

In this event, all scram signal paths, including valve position, high flux, high pressure, low level, and all manual attempts have been assumed to fail.

The loss of AC power has the following effects:

(1) An immediate load rejection will occur. This will cause the turbine control valves to close.

(2) As a result of the load rejection, four of the ten recirculation pumps will trip.

(3) Due to the loss of power to the condensate pumps, feedwater will be lost.

(4) The reactor will be isolated after loss of main condenser vacuum.

Figure 15E-6 shows the transient behavior under ARI activation, Figure 15E-7 for FMCRD run-in and Figure 15E-8 for automatic SLCS, respectively.

The fast closure of the turbine control valves causes a rapid increase of pressure, and the ATWS high pressure setpoint is reached shortly after the control valves have closed. Because the four pumps have already tripped at this time on the load rejection signal, only six remaining pumps will start runback. The ATWS high pressure signal initiates the rod insertion. The rod insertions are successful in bringing the reactor to hot shutdown. If both modes of rod insertion fail, the ATWS high pressure signal also initiates the timer for SLCS. After confirming the rod insertion failure by monitoring the high pressure and SRNM ATWS permissive signal for 3 minutes, the reactor is brought to hot shutdown when enough boron concentration is built up in the reactor core.

Tables 15E-7 to 15E-9 show the summary of peak values of key parameters for the three events.

15E.6.3 Loss of Feedwater

This event does not have rapid excursions, as in some of the other events, but is a long-term power reduction and depressurization. Since the pressure begins to fall at the onset of the transient, the need for relief valves does not arise until isolation occurs very late in the event and only single valve cycling is expected to handle decay heat. The containment limits are not approached.

In this event all feedwater flow is assumed to be lost in about five seconds.
Figure 15E-9 shows the transient behavior for ARI activated. Figure 15E-10 represents FMCRD run-in event. The mitigation of this event by SLCS is illustrated in Figure 15E-11.

After the loss of feedwater has taken place, the pressure, water level and neutron flux begin to fall. At around 6.5 seconds, low water (L3) is reached. This trips four recirculation pumps. At about 22 seconds, low water (L2) is also reached. This trips remaining recirculation pumps, activates ARI, FMCRD run-in, starts SLCS clock, and initiates RCIC. Successful insertion of control rods brings the reactor to hot shutdown. Failure of rod insertion will initiate SLCS upon the timer run-up while the SRNM ATWS permissive signal is present. At about 16.9 minutes, the reactor becomes hot shutdown as the boron concentration reaches sufficient value.
Tables 15E-10 to 15E-12 show the summary of peak values of key parameters for the three cases.

15E.6.4 Loss of Feedwater Heating

This transient does not trip any automatic ATWS logic. ARI, FMCRD run-in, and SLCS timer are assumed to be initiated by operator at about 10 minutes after the beginning of this event. At this time, the reactor has settled in a new steady state at a higher power level. There is no steam discharge to the suppression pool because of the relatively low vessel pressure. Figure 15E-12 shows the transient behavior for ARI, Figure 15E-13 for FMCRD run-in and Figure 15E-14 for SLCS case, respectively. Upon the failure of rod insertion, the SLCS can bring the reactor to hot shutdown at about 33.3 minutes.

The mild nature of this transient forestalls any significant peak values for the key parameters normally associated with ATWS study. However, the slow insertion rate of FMCRD run-in allows the reactor to reestablish quasi-steady axial power shape. The peak value of these new profiles, which were calculated by the PANACEA code, are shown in Figure 15E-15. The peak cladding temperature does not exceed the coolable geometry criteria. Figure 15E-16 presents the normalized axial power shape change during the event. Table 15E-13 shows the peak values of the key parameters for FMCRD run-in case. The same values apply to ARI and SLCS cases as well.

15E.6.5 Turbine Trip with Bypass

The initial characteristics of this transient are much like the MSIV closure described in Section 15E.6.1 with rapid steam shutoff. Pressure and power increases are limited by the action of the relief valves and RPT/recirculation runback. As this event progresses, however, the availability of the main condenser makes it possible for the relief valves to be closed after about 48 seconds. This terminates the steam discharge to the suppression pool. Figure 15E-17 shows the transient behavior for ARI, Figure 15E-18 for FMCRD run-in and Figure 15E-19 for SLCS cases, respectively.

The closure of the turbine stop valves causes a rapid increase of pressure; the ATWS high pressure setpoint is reached shortly after the closure. The high pressure initiates four of the recirculation pumps to trip and the rest to start runback, initiates ARI, FMCRD run-in and SLCS timer. Upon successful insertion of the control rods, the reactor achieves hot shutdown. If the rods fail to insert into the core, the SLCS will be initiated by the SRNM ATWS permissive signal and the high pressure signal when the timer runs up. In this case, the hot shutdown is reached at about 19 minutes. Tables 15E-14 to 15E-16 show the summary of peak values of key parameters for these events.

15E.6.6 Loss of Condenser Vacuum

This transient starts with a turbine trip because of the low condenser vacuum; therefore, the beginning is the same as the turbine trip event (Section 15E.6.5). However, the MSIVs and turbine bypass valves also close after the condenser vacuum has further dropped to their closure setpoints, and relief valve cycling increases considerably compared to the original turbine trip case. Hence, this event is similar to the turbine trip event as far as the peak power and pressure characteristics are concerned and similar to the MSIV closure case with respect to suppression pool temperature and pressure. Figure 15E-20 shows the transient behavior for ARI event, Figures 15E-21 for FMCRD run-in case and Figure 15E-22f or SLCS condition, respectively. The high pressure ATWS setpoint is reached shortly after the closure of turbine stop valves. The high pressure initiates trip for four of the ten RIPs and runback of the other six. It starts ARI, FMCRD run-in and SLCS timer. A successful insertion of control rods brings the reactor to hot shutdown. Otherwise, the injection of boron is initiated upon SRNM ATWS permissive and high pressure signals. As the poison reaches sufficient concentration in the core, the reactor achieves hot shutdown in about 19 minutes. Tables 15E-17 to 15E-19 show the summary of peak values of key parameters for these events.

15E.6.7 Feedwater Controller Failure

The initial portion of this transient results in a gradual power increase, then a sharp pressure rise and power peak as the turbine stop valves close at high water level. The long-term segment of this transient is similar to that of turbine trip with bypass valves operating. The discharge of steam into the suppression pool is minimized by the availability of the main condenser and turbine bypass valves. Figure 15E-23 shows the transient behavior for ARI, Figure 15E-24 in FMCRD run-in and Figures 15E-25 for SLCS case, respectively.

The closure of the turbine stop valves starts a rapid increase of pressure. The ATWS high pressure setpoint is reached shortly after the valve closure. The high pressure trips four of the ten recirculation pumps and starts runback of the other six, and initiates ARI, FMCRD run-in, and SLCS timer. The reactor reaches hot shutdown once the control rods complete the insertion into the core. If the rod insertion fails, the initiation of SLCS is confirmed by the SRNM ATWS permissive signal and the hot shutdown is achieved at about 20 minutes. Tables 15E-20 to 15E-22 show the summary of peak values of key parameters for these events.

15E.7 Conclusion

Based upon the results of this analysis, the proposed ATWS design for the ABWR is satisfactory in mitigating the consequences of an ATWS. All performance criteria specified in Section 15E.2 are met.

It is also concluded from results of the above analysis that automatic boron injection could mitigate the most limiting ATWS event with margin (at least 0.108 MPa margin in peak containment pressure). Therefore, an automatic SLCS injection as a backup for ATWS mitigation is acceptable.

15E.8 Reference

15E-1 “Assessment of BWR Mitigation of ATWS”, NEDE-24222, September 1979.

15E-2 Letter, J.N. Fox (GE) to C. Poslusny (NRC), “ATWS Stability Study”, February 22, 1993.

Table 15E-1 Performance Requirements

	RPV Peak Pressure	Maximum Pool Temperature	Fuel Integrity	Maximum Containment Pressure
ARI/RPT	10.35 MPaG	97.2°C*	Coolable Geometry	0.310 MPaG
FMCRD/RPT	10.35 MPaG	97.2°C*	Coolable Geometry	0.310 MPaG
Boron/RPT	10.35 MPaG	Containment Design Pressure	Coolable Geometry	0.310 MPaG

* 97.2°C pool temperature should not be reached before the reactor reaches the hot shutdown condition.

Table 15E-2 Initial Operating Conditions

Parameters	Value
Dome Pressure (MPaG)	7.07
Core Flow (Mkg/hr)/(% NBR)	52.2/100
Vessel Diameter (m)	7.06
Numbers of Fuel Bundles	872
Power (MWt)/(% NBR)	3926/100
Steam/Feed Flow (kg/sec)/(% NBR)	2123/100
Feedwater Temperature (°C)	215.6
Void Reactivity Coefficient (¢/%)	–9.7
Doppler Coefficient (¢/°C)	–0.504
ARI/FMCRD Reactivity Curve	D Curve
Suppression Pool Volume (m^3)/ (Full NBR FW Flow-Min)	3580/28.1
Initial Suppression Pool Temperature (°C)	37.7
Condensate Storage Temperature (°C)	48.9

Table 15E-3 Equipment Performance Characteristics

Parameters	Value
Closure Time of MSIV (s)	3.0
Relief Valve System Capacity (% NBR Steam Flow)/No. of Valves	91.3 at 1st setpoint/18
Relief Valve Setpoint Range (MPaG)	7.89/8.24
Relief Valve Opening Time (s)	0.15
Pressure Drop Below Setpoint for Relief Valve Closure (MPaG)	0.520
Relief Valve Closure Time Delay (s)	0.9
Relief Valve Closure Time Constant (s)	0.2
RCIC Low Water Level Initiation Setpoint	Level 2
HPCF Low Water Level Initiation Setpoint	Level 1.5
HPCF Start Time (s)	20
HPCF/RCIC High Water Level Shutoff Setpoint*	Level 8
Number of HPCF Pumps	2
HPCF Flow Rate per Pump† (kg/s)/(% NBR Steam Flow)	50.4/2.37
RCIC Start Time (s)	≥ 29
RCIC Flow Rate (kg/s)/(% NBR Steam Flow)	50.4/2.37
ATWS Dome Pressure Sensor Time Constant (s)	0.5
ATWS Logic Time Delay (s)	0.3
Recirculation Pump System Inertia (kg-m^2)	21.5
Delay Before Start of Electrohydraulic Rod Insertion (with/without offsite power)(s)	1.0/39.0
Electrohydraulic Control Rod Insertion Time (s)	135
ARI Rod Insertion Time (s)	25
RHR Pool Cooling Capacity (MJ/s/°C)/(NBR at 38°C ΔT)	1.11/1.57
Water Level Setpoint Above Which RHR Pool Cooling Is Allowed	Level 1
Setpoint for Low Water Level Closure of MSIV	Level 1.5
Setpoint for Low Steamline Pressure Closure of MSIV (MPaG)	5.17

* HPCF and RCIC high level shutoff is independent of drywell pressure for ATWS mitigation. Automatic reset is required so restart will automatically occur if level returns below the level setpoint. Manual action to control level in the normal range is preferred rather than automatic cycling between L8/L2 during the post-hot shutdown phase of any ATWS event.

† The nominal flow versus pressure head curve is used. The value given for ABWR is at 82.7 kg/cm^2g.

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Table 15E-4 MSIV Closure Summary (ARI)

	Value	time
Maximum Neutron Flux (%)	451	1.7 s
Maximum Vessel Bottom Pressure (MPaG)	8.95	4.6 s
Maximum Average Heat Flux (%)	131	3.0 s
Maximum Bulk Suppression Pool Temperature, (°C)	59.9	303 min
Associated Containment Pressure (MPaG)	0.024	303 min
Peak Cladding Temperature (°C)	613	17.9 s

Table 15E-5 MSIV Closure Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	451	1.7 s
Maximum Vessel Bottom Pressure (MPaG)	8.95	4.6 s
Maximum Average Heat Flux (%)	131	3.0 s
Maximum Bulk Suppression Pool Temperature, (°C)	65.8	148 min
Associated Containment Pressure (MPaG)	0.031	148 min
Peak Cladding Temperature (°C)	536	8.5 s

Table 15E-6 MSIV Closure Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	451	1.7 s
Maximum Vessel Bottom Pressure (MPaG)	8.95	4.6 s
Maximum Average Heat Flux (%)	131	3.0 s
Maximum Bulk Suppression Pool Temperature, (°C)	81.6	33.4 min
Associated Containment Pressure (MPaG)	0.061	33.4 min
Peak Cladding Temperature (°C)	697	140.0 s

Table 15E-7 Loss of AC Power Summary (ARI)

	Value	time
Maximum Neutron Flux (%)	170	0.69 s
Maximum Vessel Bottom Pressure (MPaG)	8.33	3.0 s
Maximum Average Heat Flux (%)	102	0.89 s
Maximum Bulk Suppression Pool Temperature, (°C)	58.5	351 min
Associated Containment Pressure (MPaG)	0.022	351 min

Table 15E-8 Loss of AC Power Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	170	0.69 s
Maximum Vessel Bottom Pressure (MPaG)	8.33	3.0 s
Maximum Average Heat Flux (%)	102	0.89 s
Maximum Bulk Suppression Pool Temperature, (°C)	59.2	325 min
Associated Containment Pressure (MPaG)	0.023	325 min

Table 15E-9 Loss of AC Power Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	453	371 s
Maximum Vessel Bottom Pressure (MPaG)	8.33	3.0 s
Maximum Average Heat Flux (%)	102	0.89 s
Maximum Bulk Suppression Pool Temperature, (°C)	65	163 min
Associated Containment Pressure (MPaG)	0.030	163 min

Table 15E-10 Loss of Feedwater Summary (ARI)

	Value	time
Maximum Neutron Flux (%)	116	424 s
Maximum Vessel Bottom Pressure (MPaG)	7.39	430 s
Maximum Average Heat Flux (%)	116	430 s
Maximum Bulk Suppression Pool Temperature, (°C)	58.0	384 min
Associated Containment Pressure (MPaGg)	0.022	384 min

Table 15E-11 Loss of Feedwater Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	116	424 s
Maximum Vessel Bottom Pressure (MPaG)	7.39	430 s
Maximum Average Heat Flux (%)	116	430 s
Maximum Bulk Suppression Pool Temperature, (°C)	57.9	383 min
Associated Containment Pressure (MPaG)	0.022	383 min

Table 15E-12 Loss of Feedwater Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	116	424 s
Maximum Vessel Bottom Pressure (MPaG)	7.39	430 s
Maximum Average Heat Flux (%)	116	430 s
Maximum Bulk Suppression Pool Temperature, (°C)	63.1	212 min
Associated Containment Pressure (MPaG)	0.028	212 min

Table 15E-13 Loss of Feedwater Heating Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	116	424 s
Maximum Vessel Bottom Pressure (MPaG)	7.39	430 s
Maximum Average Heat Flux (%)	116	430 s
Maximum Bulk Suppression Pool Temperature, (°C)	*	*
Associated Containment Pressure (MPaG)	*	*

* Initial values

Table 15E-14 Turbine Trip with Bypass Summary (ARI)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.54	2.46 s
Maximum Average Heat Flux (%)	126.5	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	34.4	33 s
Associated Containment Pressure (MPaG)	0.001	33 s

Table 15E-15 Turbine Trip with Bypass Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.54	2.46 s
Maximum Average Heat Flux (%)	126.5	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	34.7	90 s
Associated Containment Pressure (MPaG)	0.002	90 s

Table 15E-16 Turbine Trip with Bypass Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.55	2.46 s
Maximum Average Heat Flux (%)	126.5	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	42.1	12 min
Associated Containment Pressure (MPaG)	0.007	12 min

Table 15E-17 Loss of Condenser Vacuum Summary (ARI)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.55	2.46 s
Maximum Average Heat Flux (%)	127	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	59.4	316 min
Associated Containment Pressure (MPaG)	0.024	316 min

Table 15E-18 Loss of Condenser Vacuum (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.55	2.46 s
Maximum Average Heat Flux (%)	127.0	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	60.7	282 min
Associated Containment Pressure (MPaG)	0.025	282 min

Table 15E-19 Loss of Condenser Vacuum Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	757	0.79 s
Maximum Vessel Bottom Pressure (MPaG)	8.55	2.46 s
Maximum Average Heat Flux (%)	127	1.10 s
Maximum Bulk Suppression Pool Temperature, (°C)	80.1	49.5 min
Associated Containment Pressure (MPaG)	0.059	49.5 min

Table 15E-20 Feedwater Controller Failure Summary (ARI)

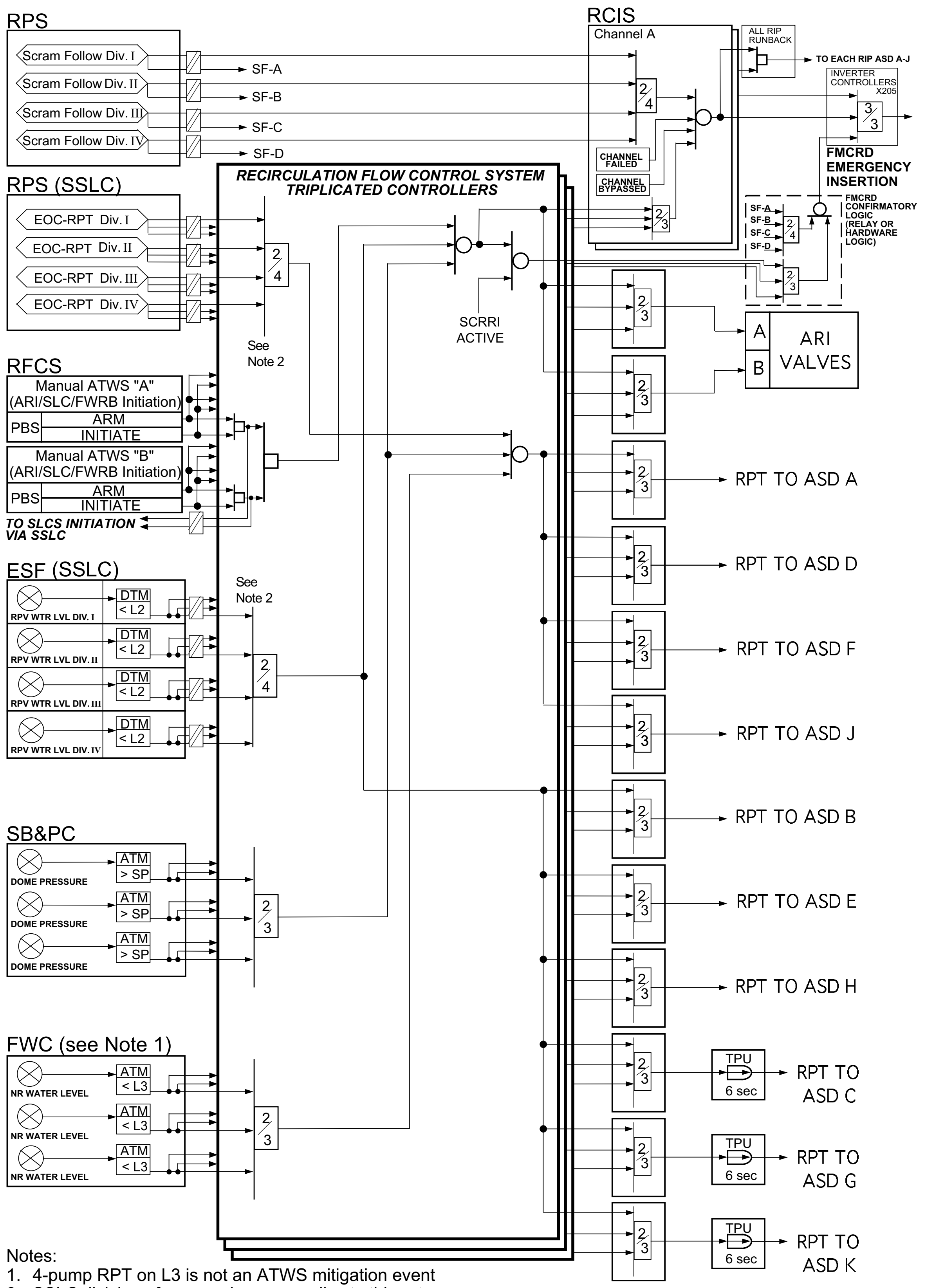
	Value	time
Maximum Neutron Flux (%)	647	19.9 s
Maximum Vessel Bottom Pressure (MPaG)	8.53	21.6 s
Maximum Average Heat Flux (%)	127.7	20.3 s
Maximum Bulk Suppression Pool Temperature, (°C)	34.6	48 min
Associated Containment Pressure (MPaG)	0.00098	48 min

Table 15E-21 Feedwater Controller Failure Summary (FMCRD Run-In)

	Value	time
Maximum Neutron Flux (%)	647	19.9 s
Maximum Vessel Bottom Pressure (MPaG)	8.53	21.6 s
Maximum Average Heat Flux (%)	127.7	20.3 s
Maximum Bulk Suppression Pool Temperature, (°C)	34.6	60 s
Associated Containment Pressure (MPaG)	0.00098	60 s

Table 15E-22 Feedwater Controller Failure Summary (Boron Injection)

	Value	time
Maximum Neutron Flux (%)	647	19.9 s
Maximum Vessel Bottom Pressure (MPaG)	8.53	21.6 s
Maximum Average Heat Flux (%)	127.7	20.3 s
Maximum Bulk Suppression Pool Temperature, (°C)	34.8	48 sec
Associated Containment Pressure (MPaG)	1.96E-3	48 sec



Notes:
1. 4-pump RPT on L3 is not an ATWS mitigation event
2. SSLC division-of-sensors bypass applies to this voter

Figure 15E-1a ATWS Mitigation Logic (ARI, FMCRD Run-In, RPT, Manual Initiation)

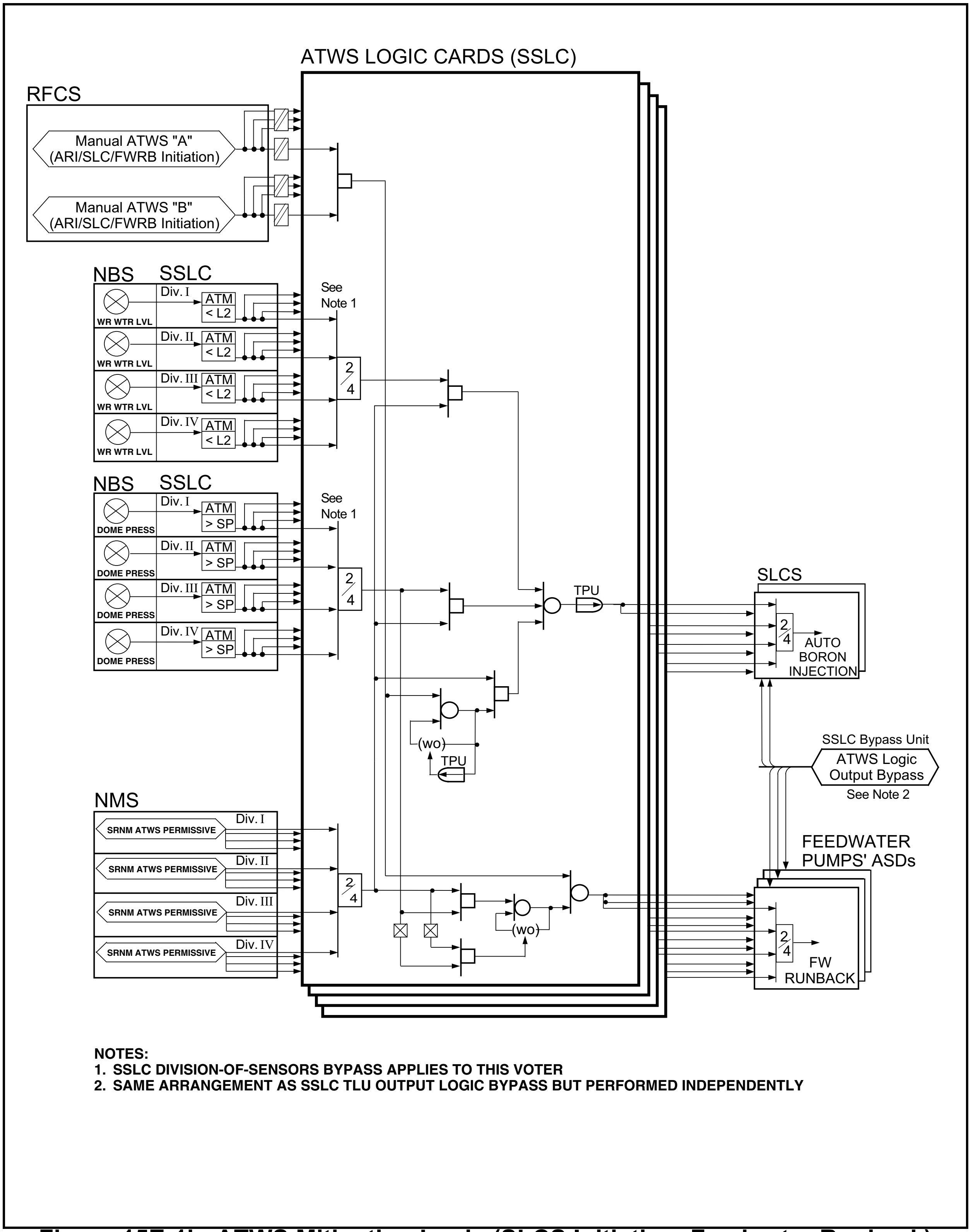
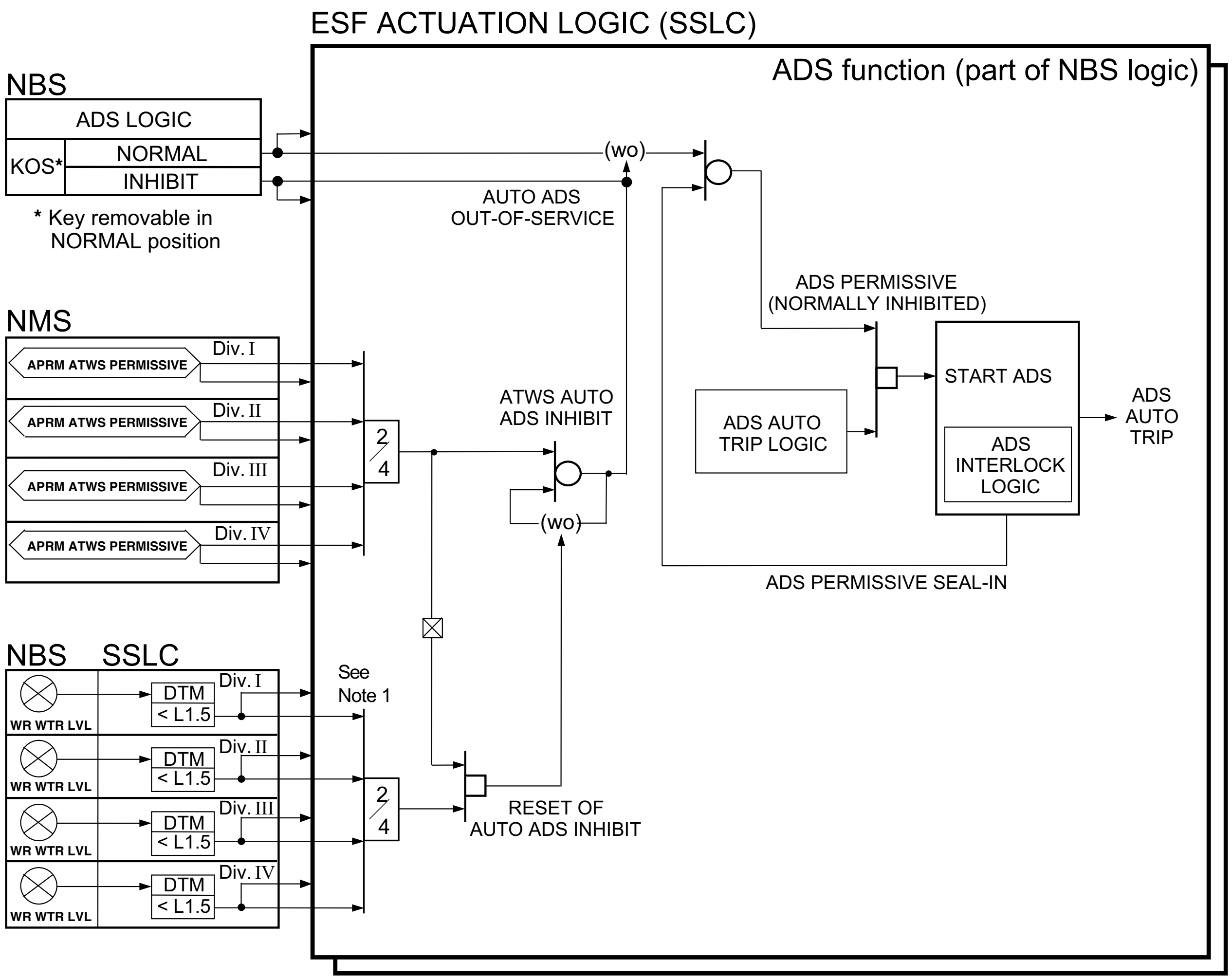


Figure 15E-1b ATWS Mitigation Logic (SLCS Initiation, Feedwater Runback)



NOTES:

1. SSLC DIVISION-OF-SENSORS BYPASS APPLIES TO THIS VOTER

Figure 15E-1c ATWS Mitigation Logic (ADS Inhibit)

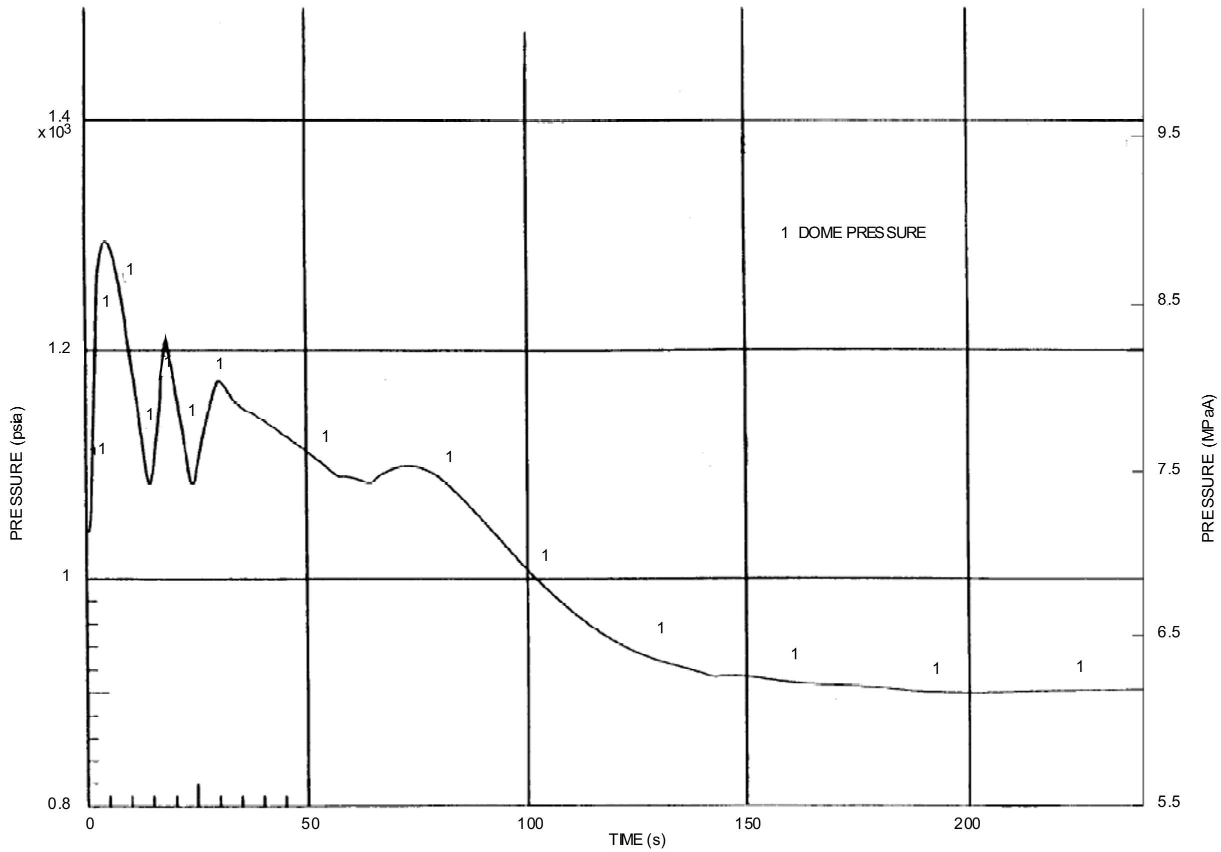


Figure 15E-2 ABWR MSIV Closure, ARI

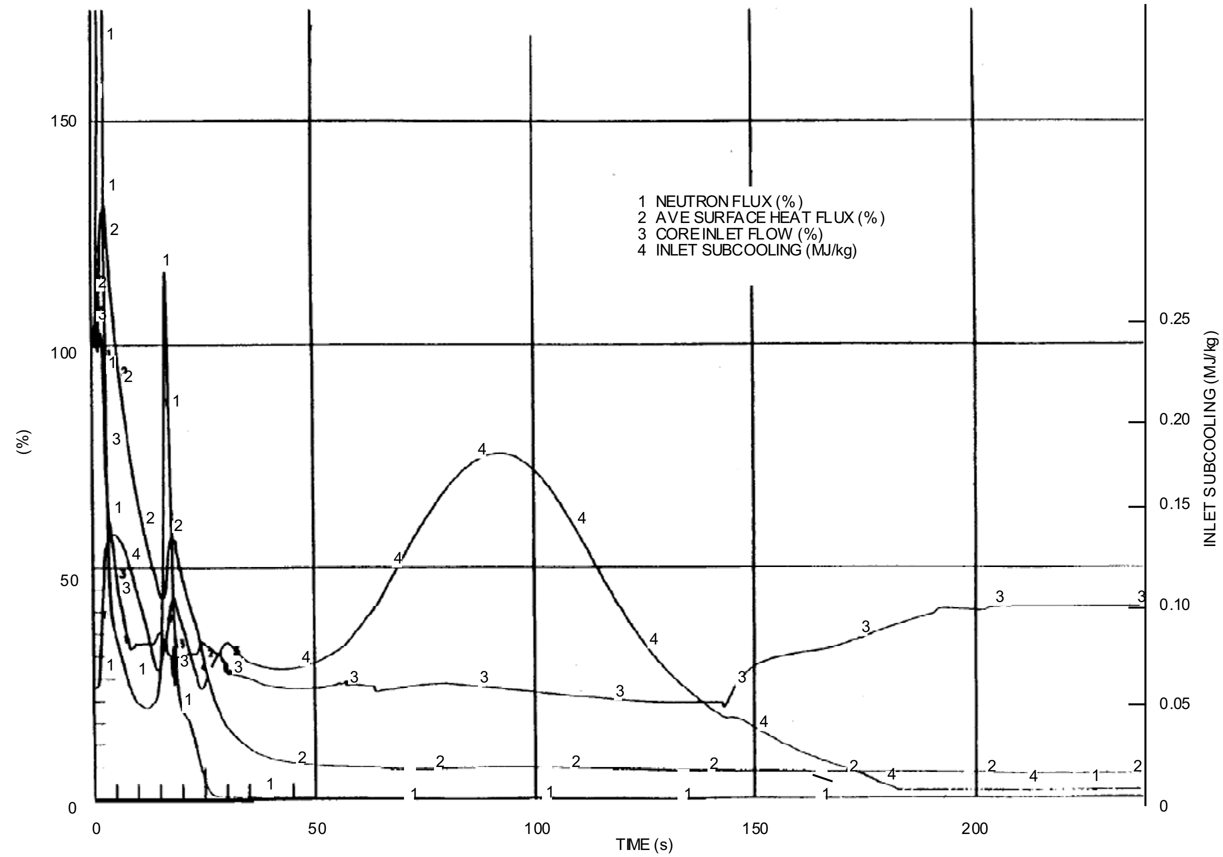


Figure 15E-2 ABWR MSIV Closure, ARI (Continued)

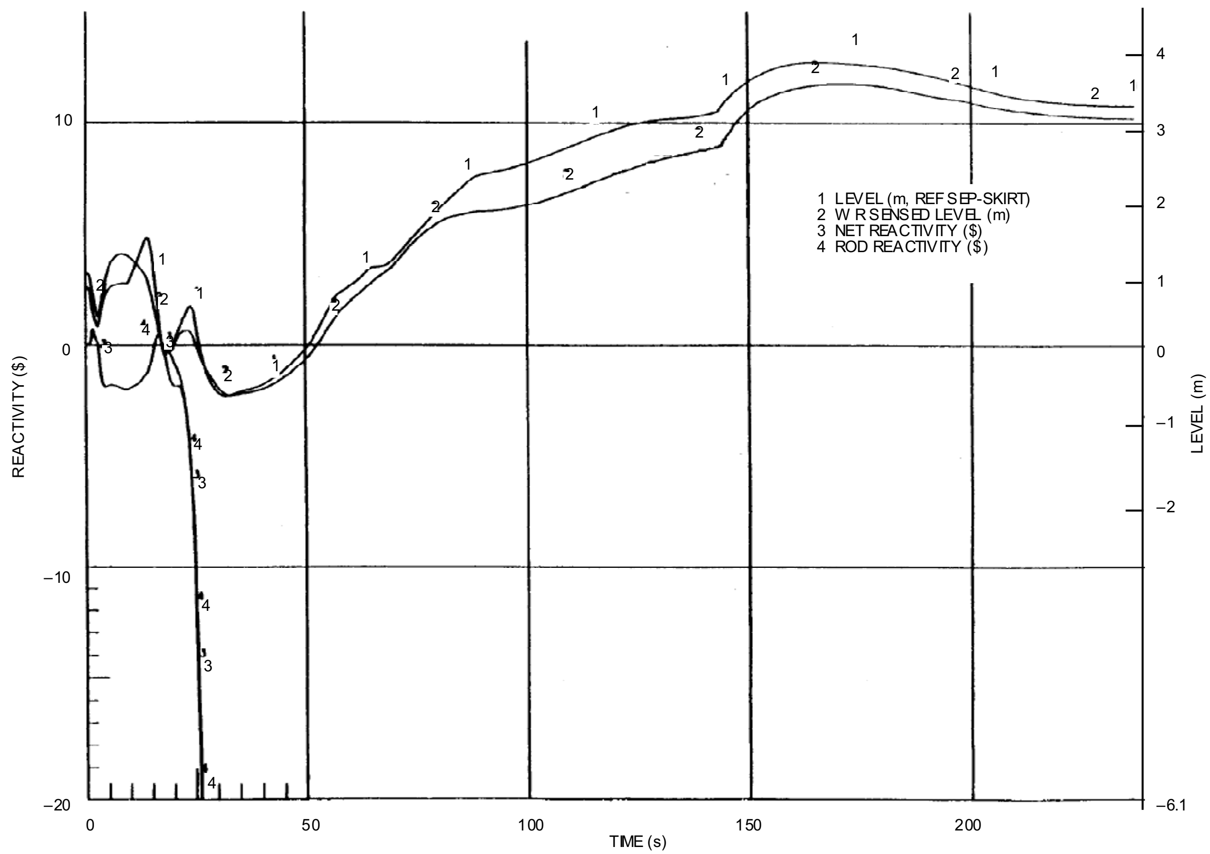


Figure 15E-2 ABWR MSV Closure, ARI (Continued)

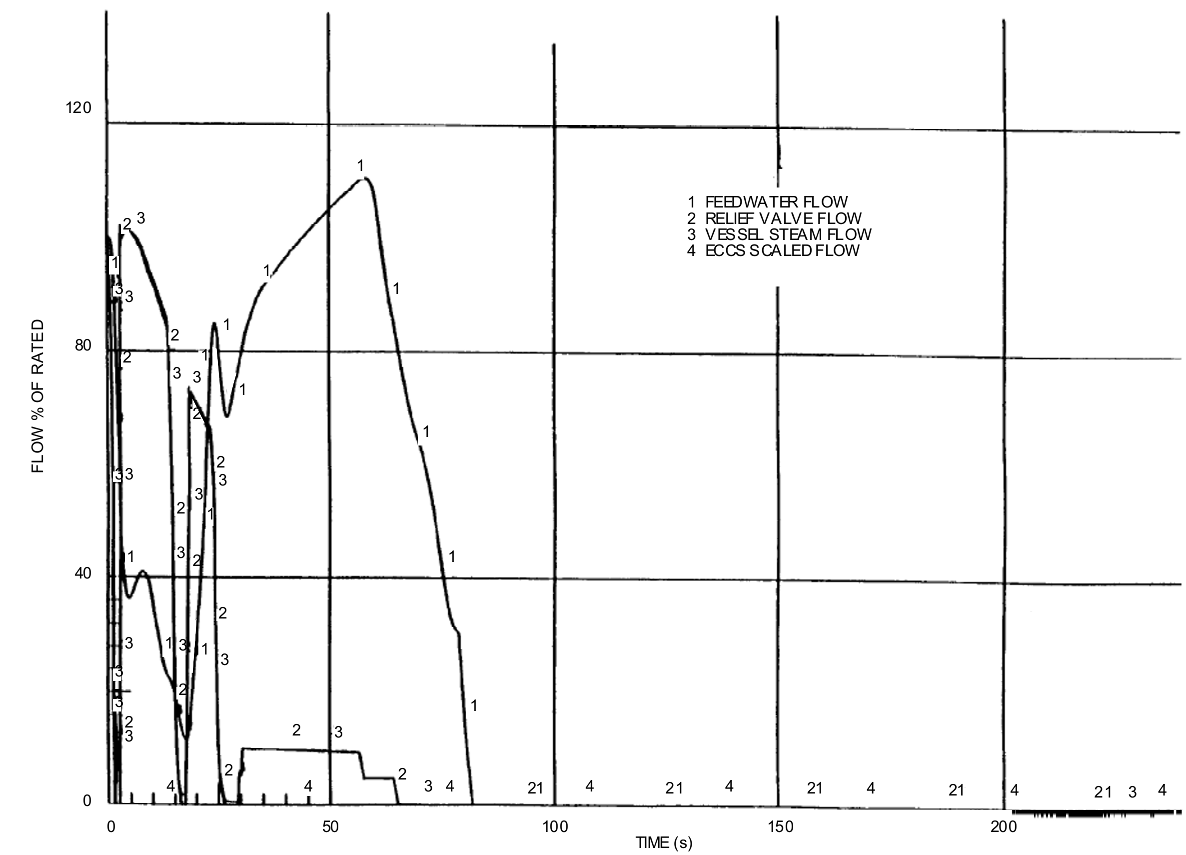


Figure 15E-2 ABWR MSIV Closure, ARI (Continued)

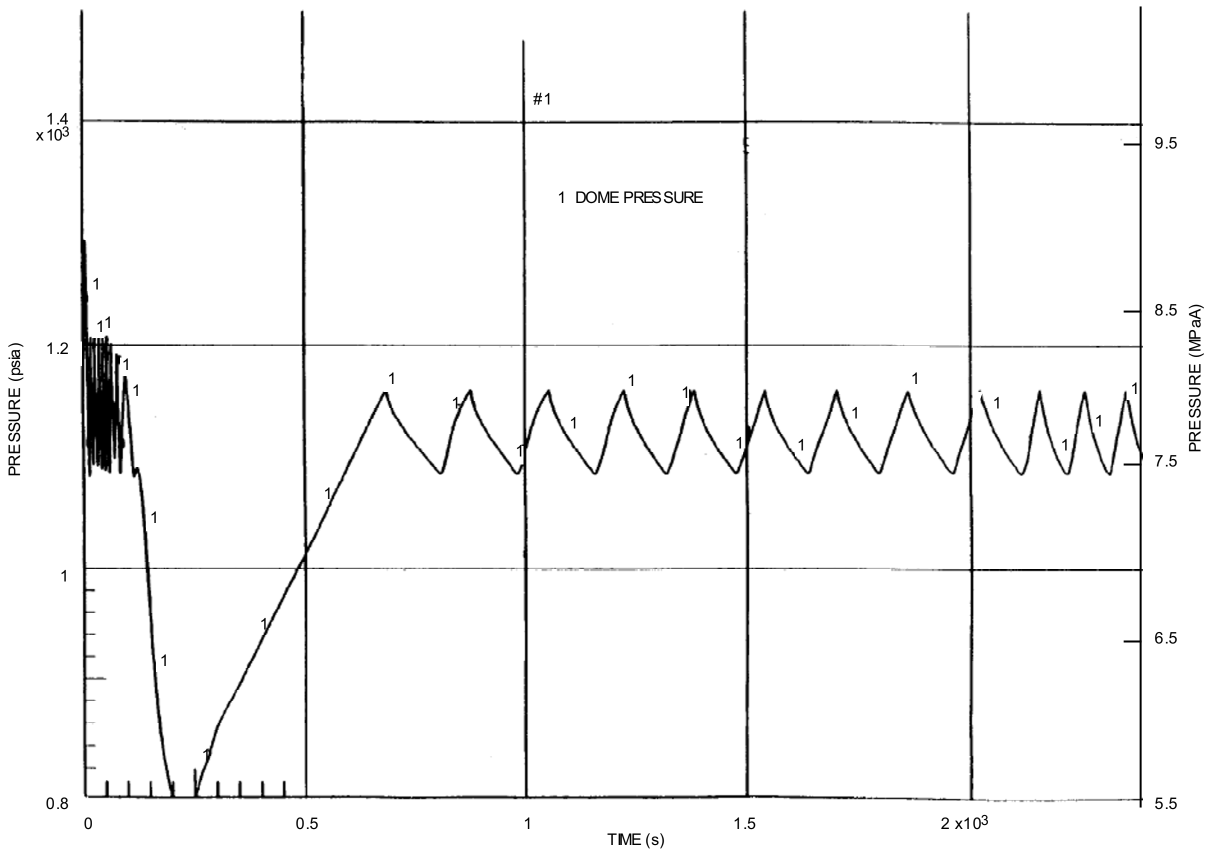


Figure 15E-3 ABWR MSIV Closure, FMCRD Run–in

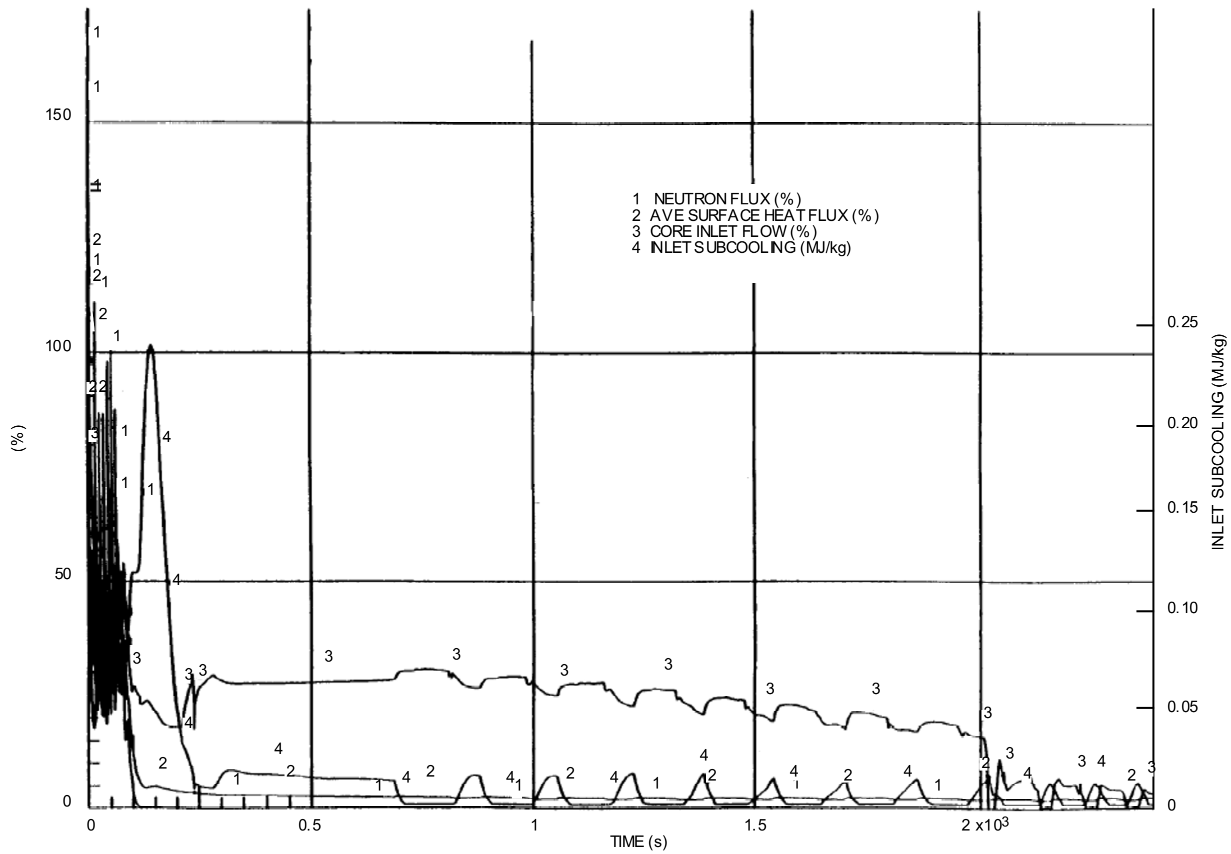


Figure 15E-3 ABWR MSIV Closure, FMCRD Run-in (Continued)

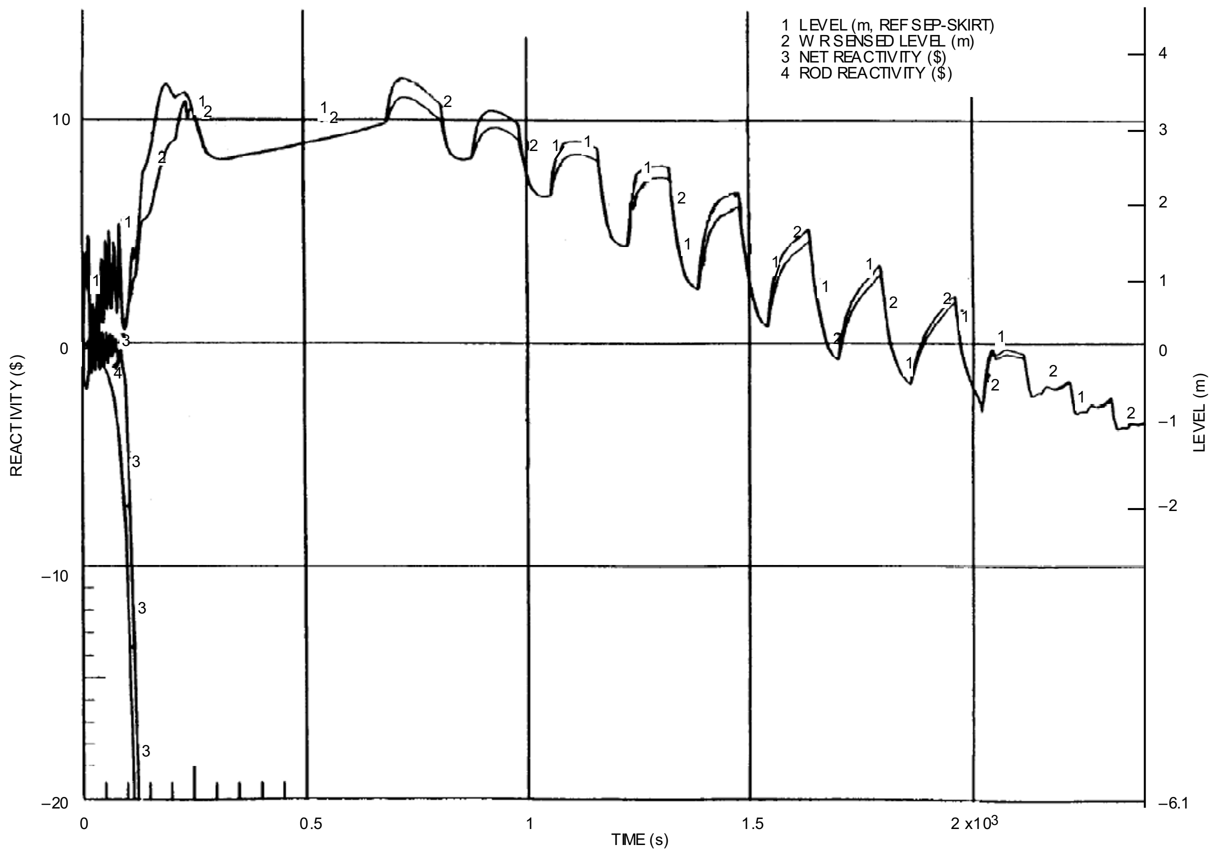


Figure 15E-3 ABWR MSIV Closure, FMCRD Run-in (Continued)

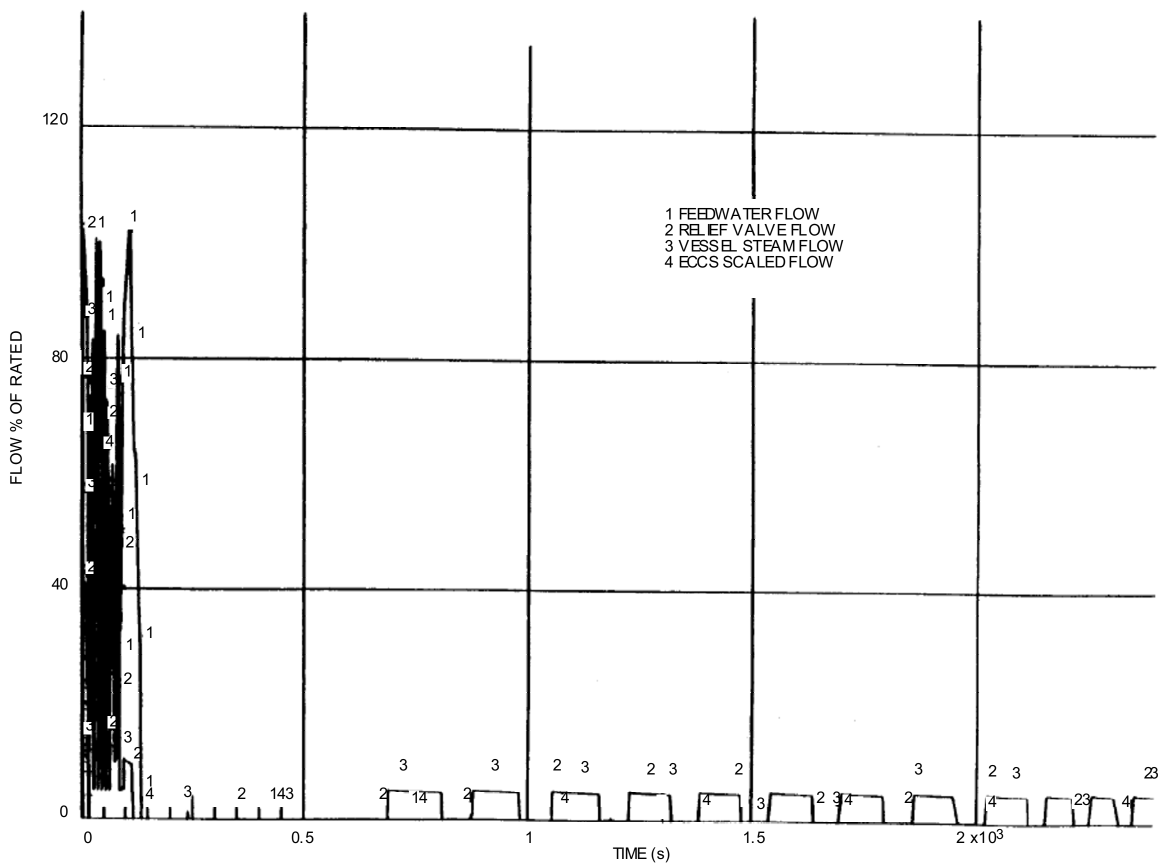


Figure 15E-3 ABWR MSIV Closure, FMCRD Run-in (Continued)

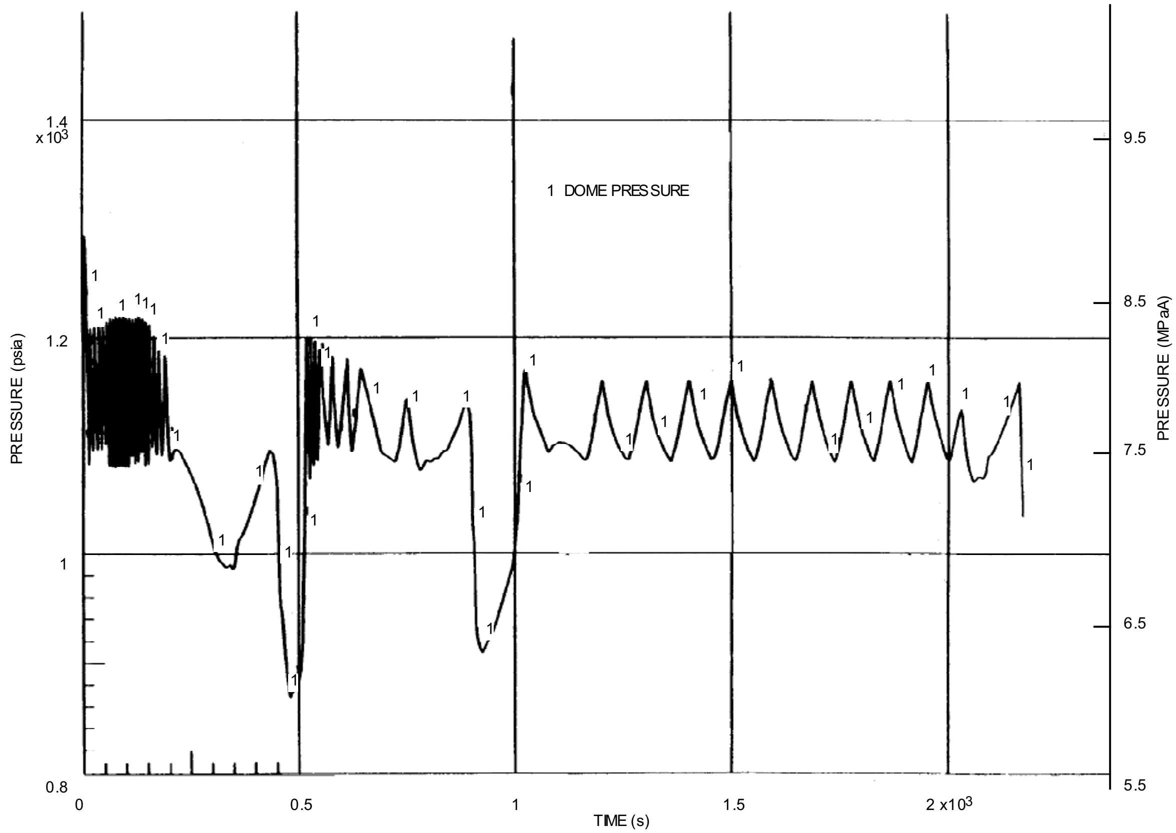


Figure 15E-4 ABWR MSIV Closure, SLCS

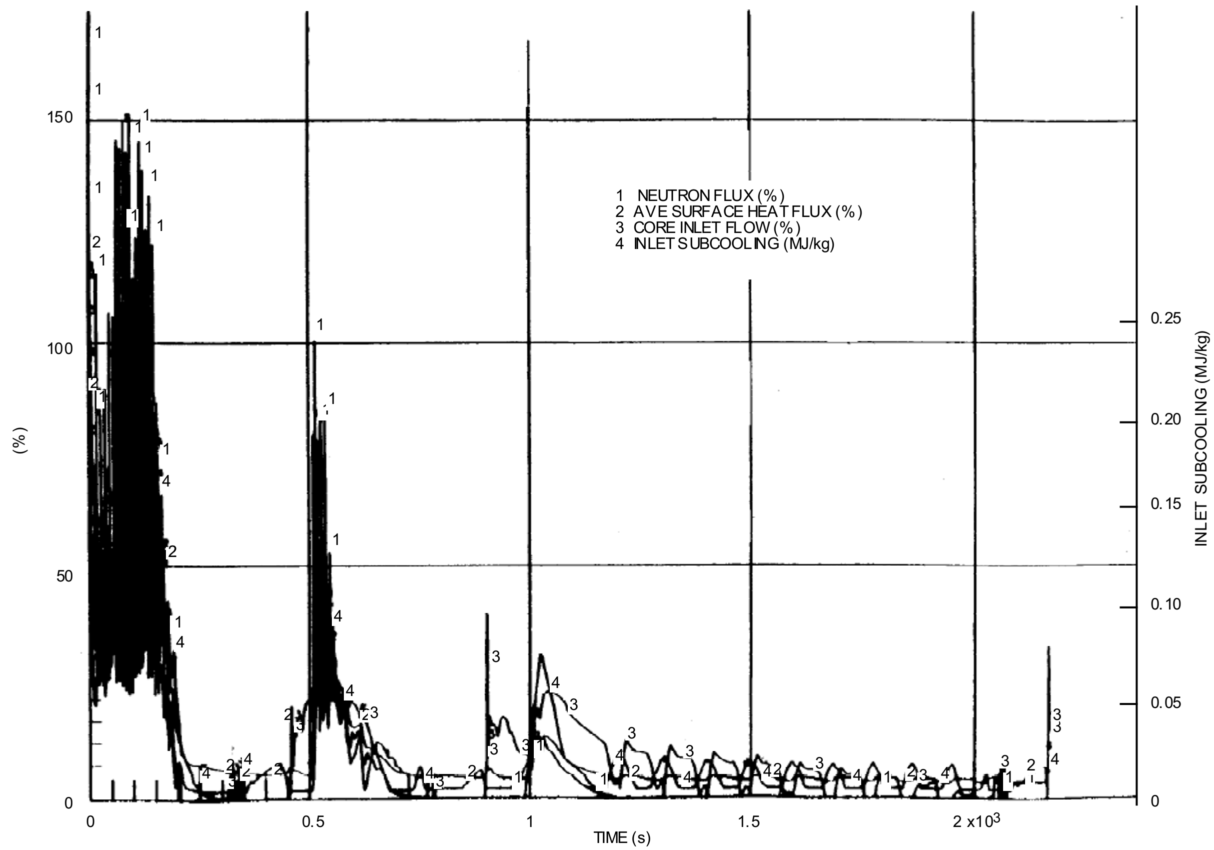


Figure 15E-4 ABWR MSIV Closure, SLCS (Continued)

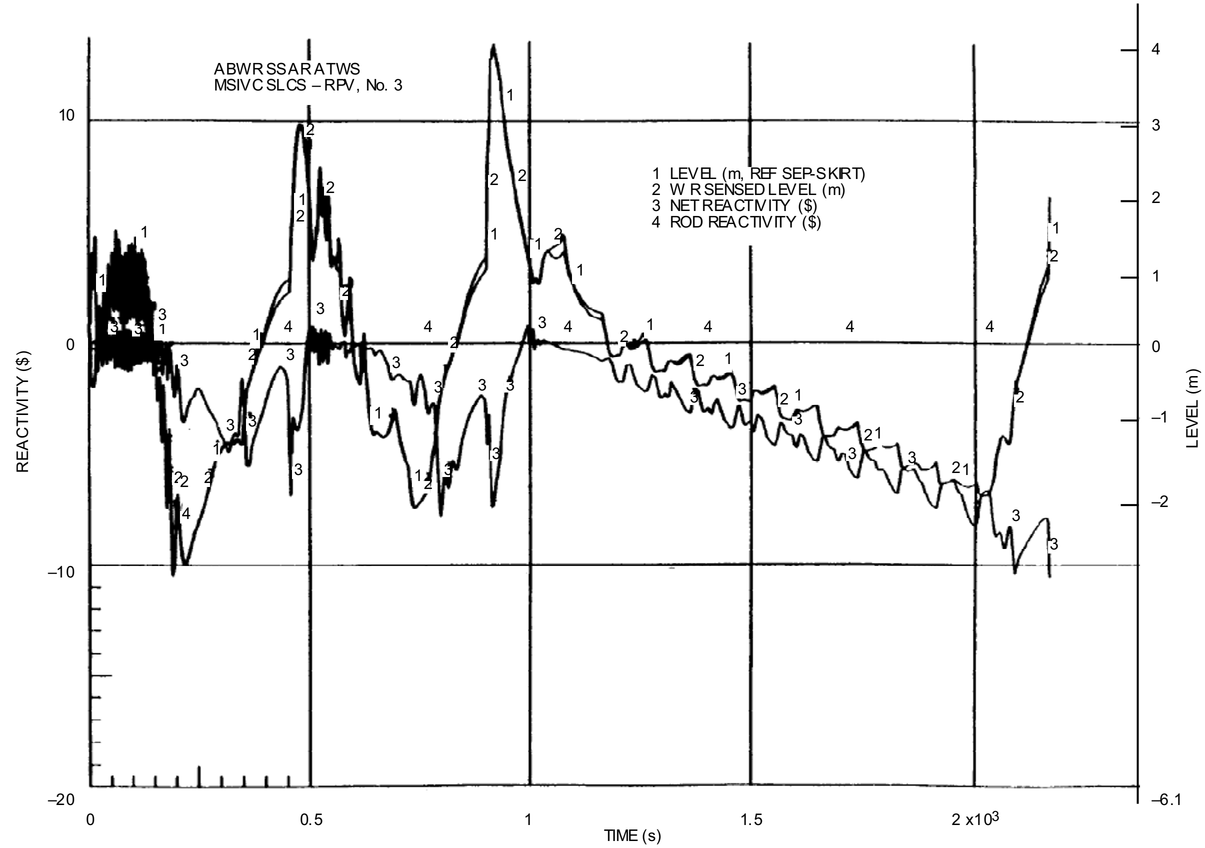


Figure 15E-4 ABWR MSIV Closure, SLCS (Continued)

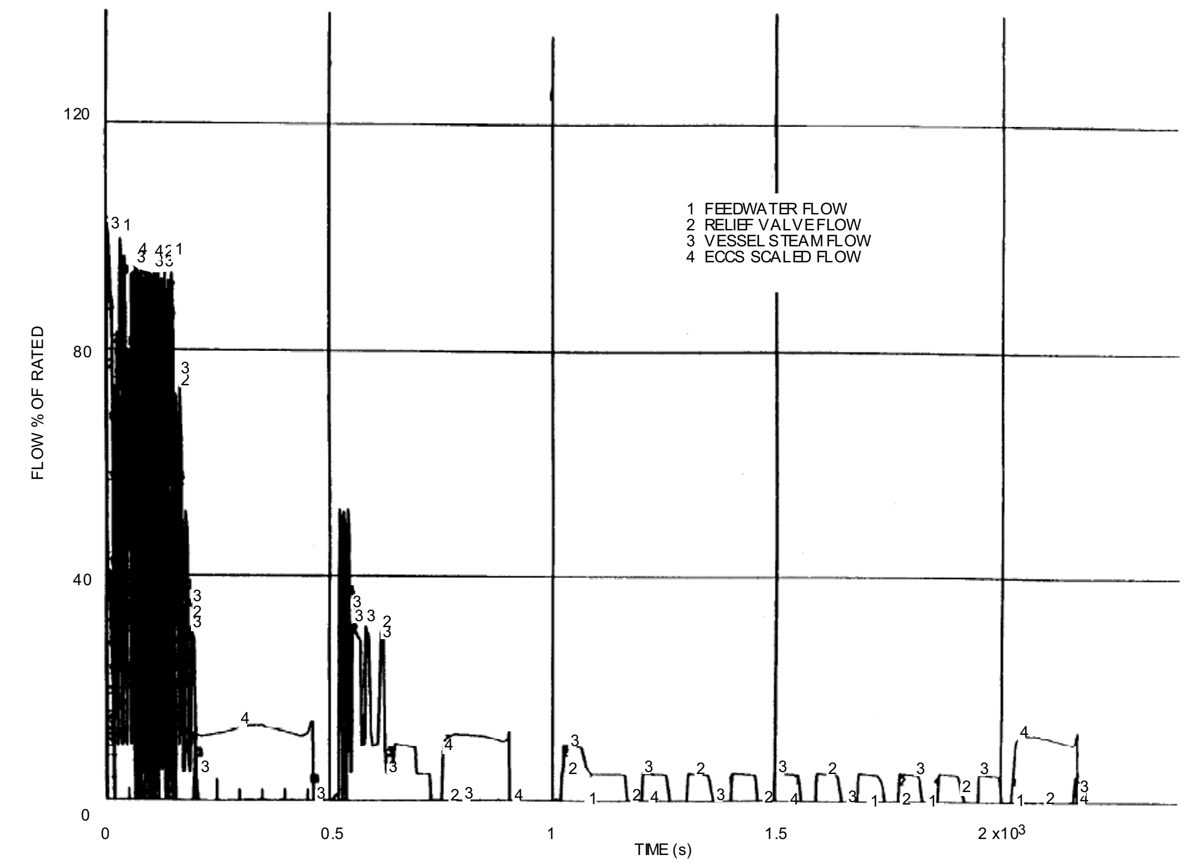


Figure 15E-4 ABWR MSIV Closure, SLCS (Continued)

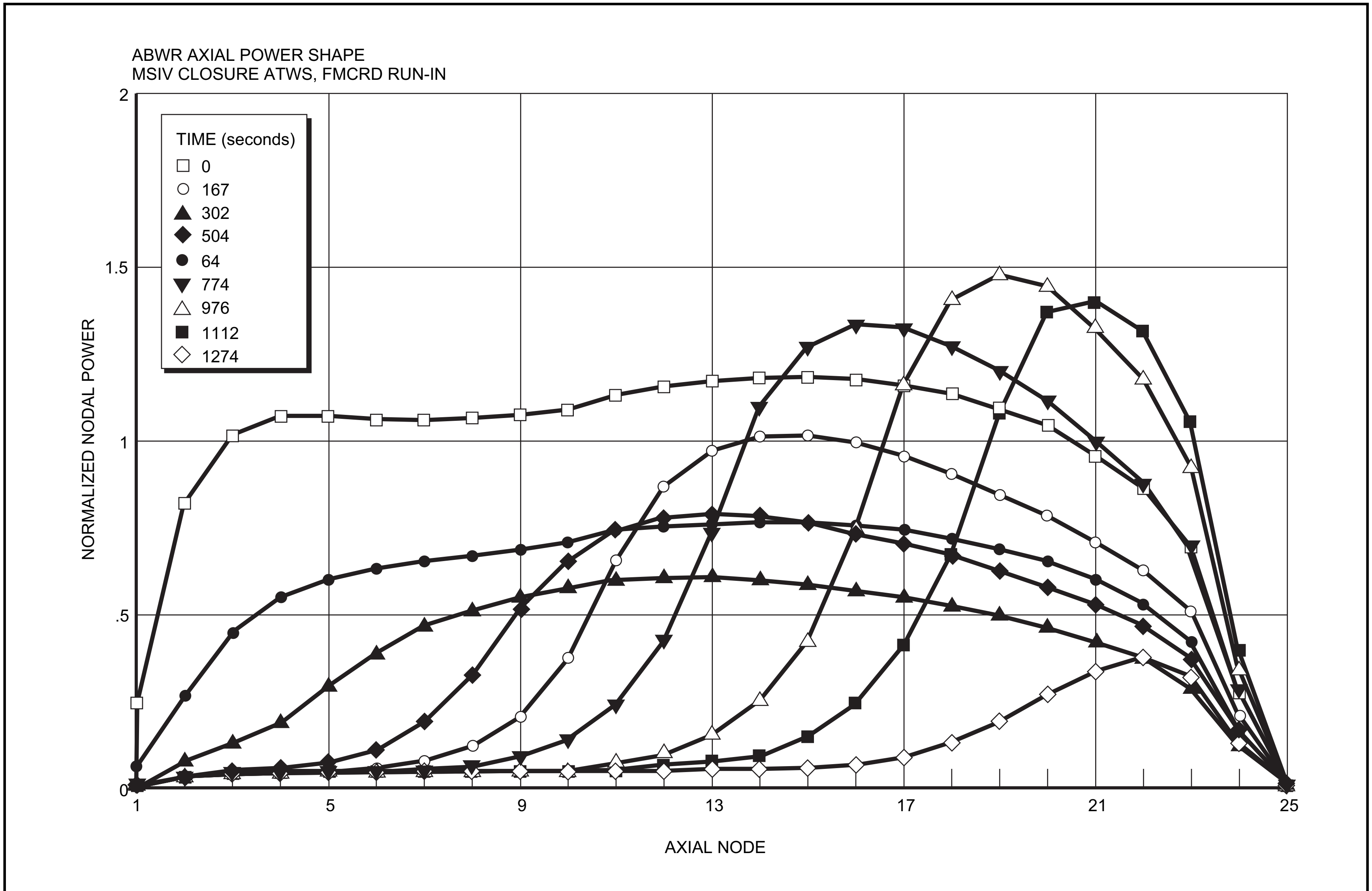


Figure 15E-5 ABWR MSIV Closure, FMCRD Run–in

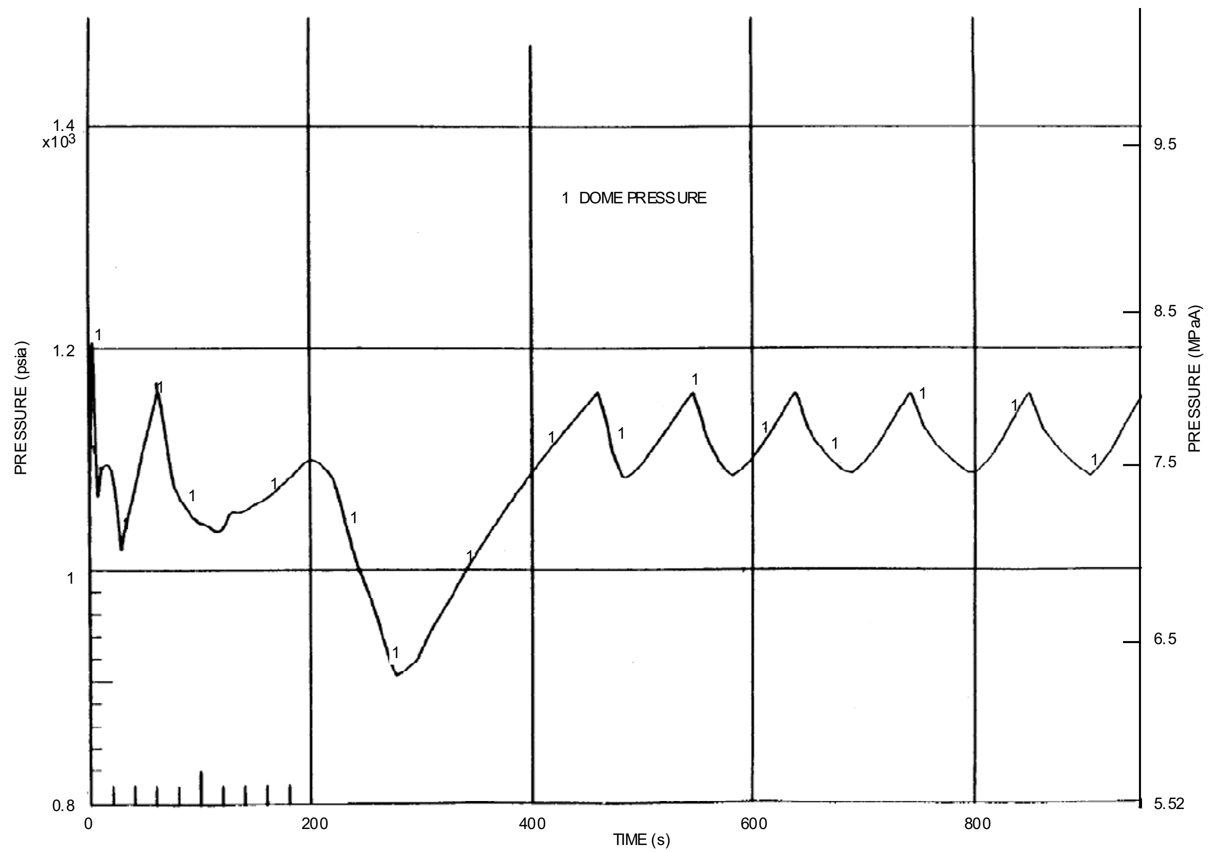


Figure 15E-6 ABWR Loss of AC Power, ARI

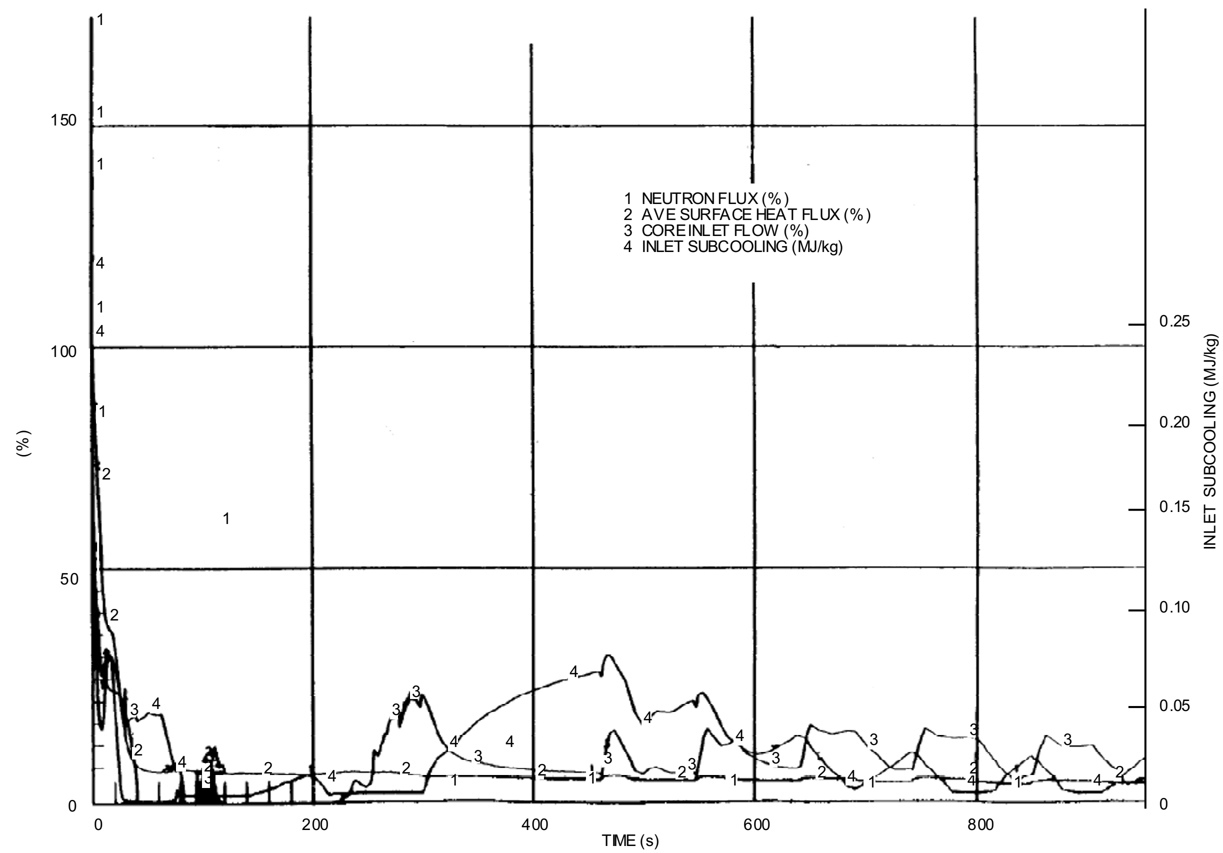


Figure 15E-6 ABWR Loss of AC Power, ARI (Continued)

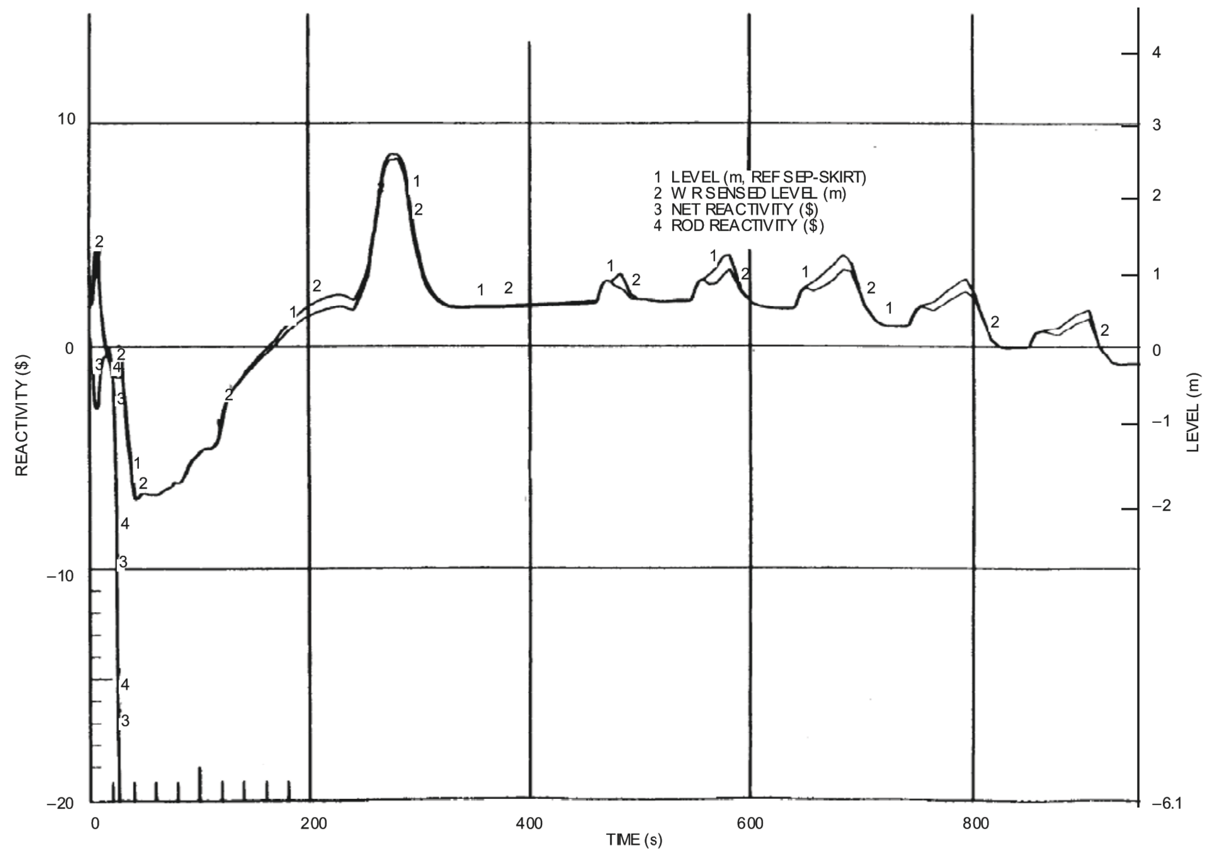


Figure 15E-6 ABWR Loss of AC Power, ARI (Continued)

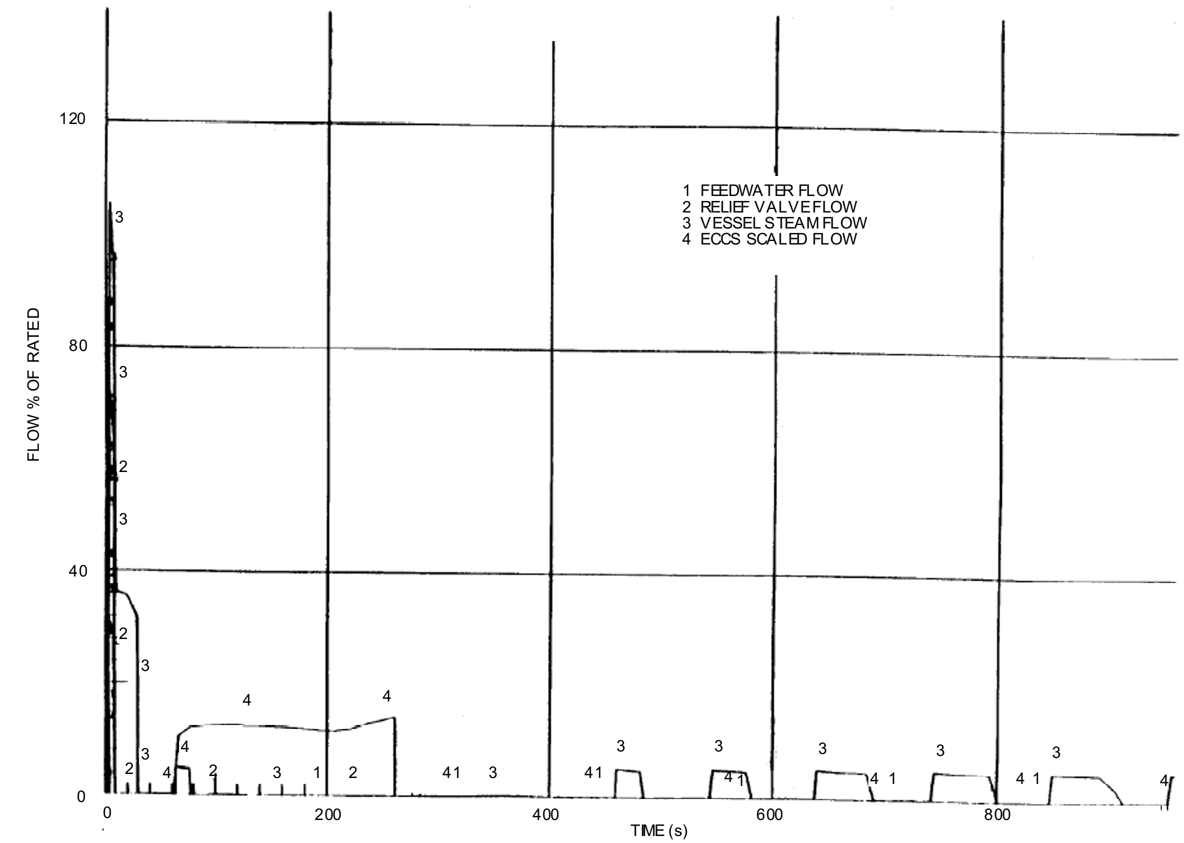


Figure 15E-6 ABWR Loss of AC Power, ARI (Continued)

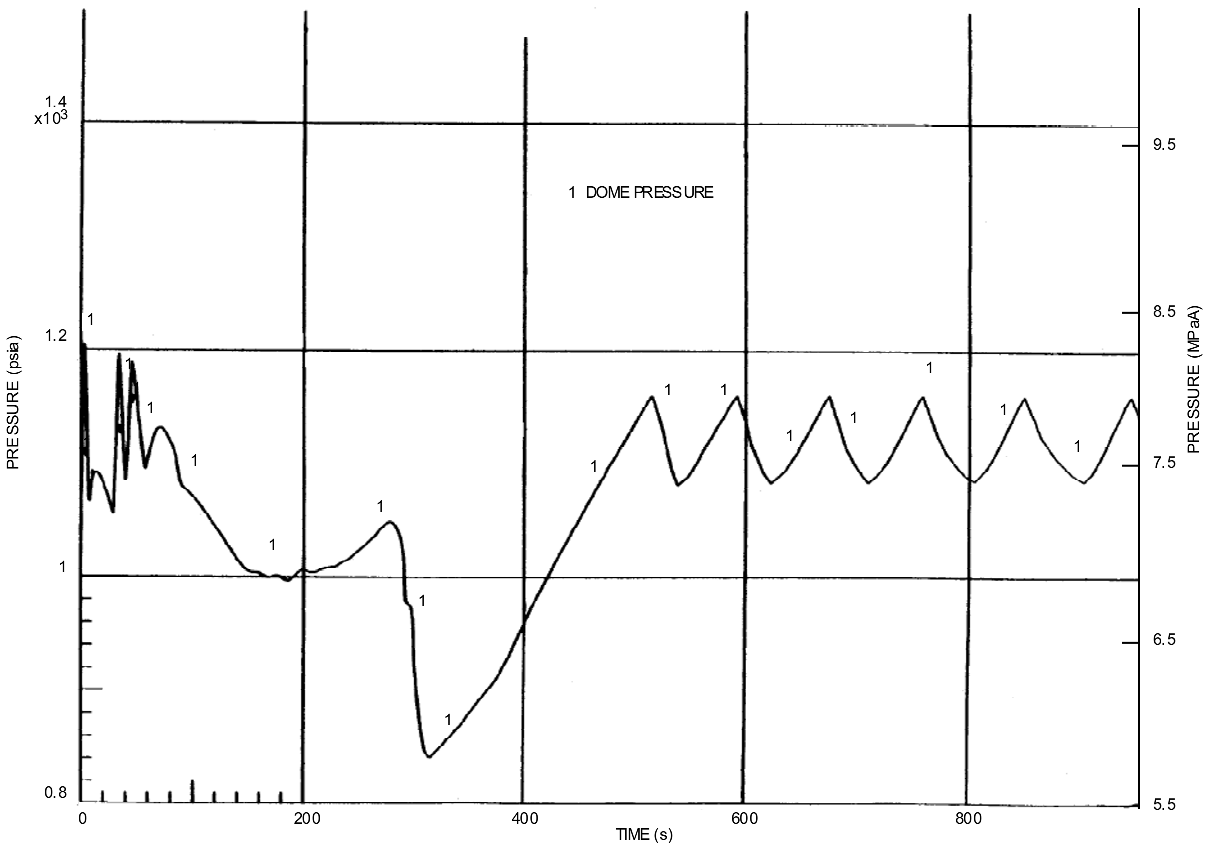


Figure 15E-7 ABWR Loss of AC Power, FMCRD Run–in

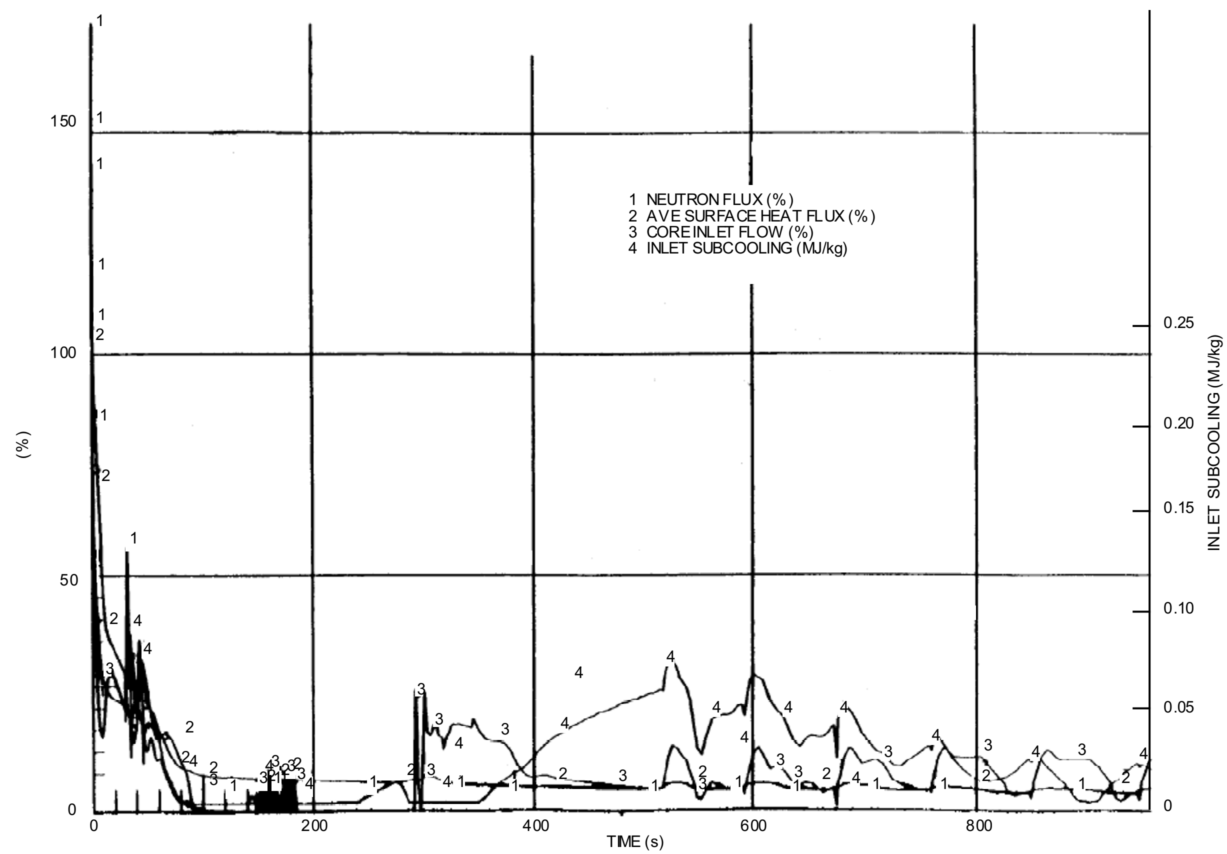


Figure 15E-7 ABWR Loss of AC Power, FMCRD Run–in (Continued)

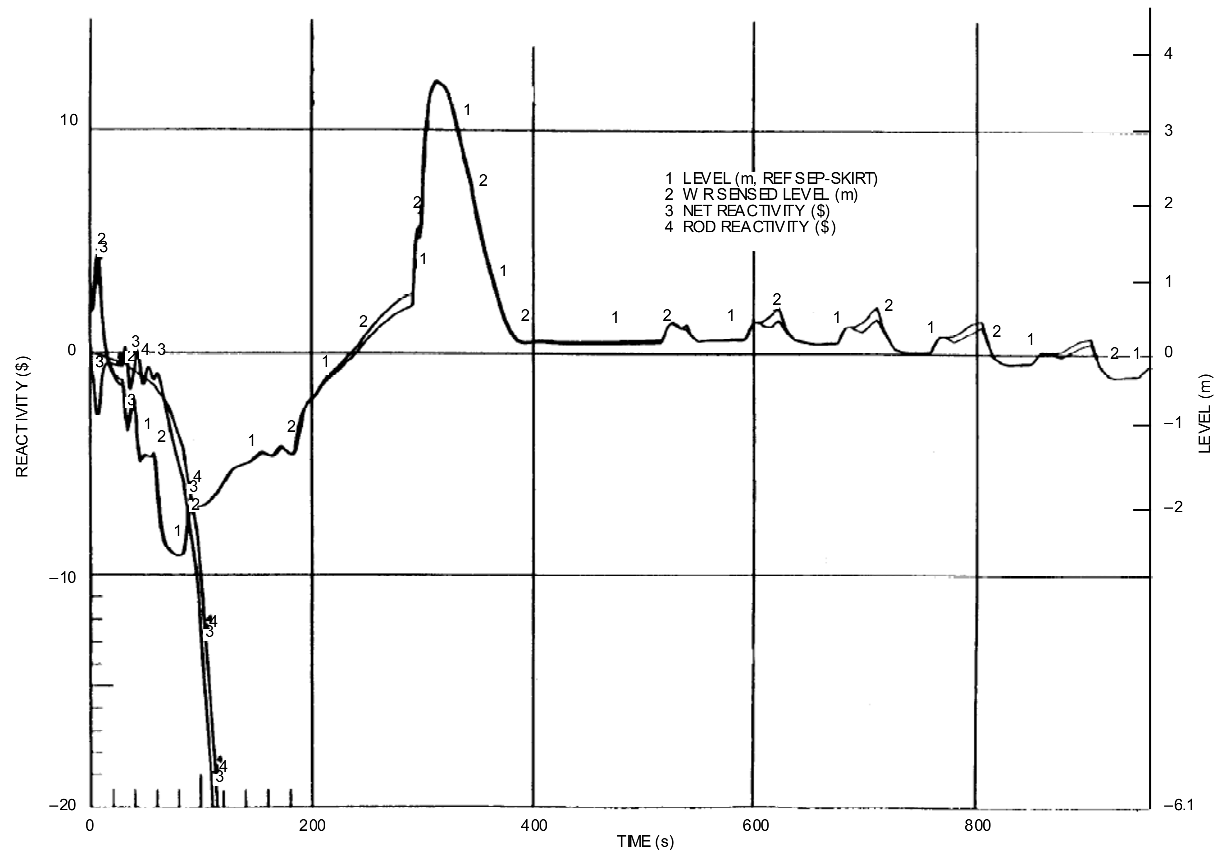


Figure 15E-7 ABWR Loss of AC Power, FMCRD Run–in (Continued)

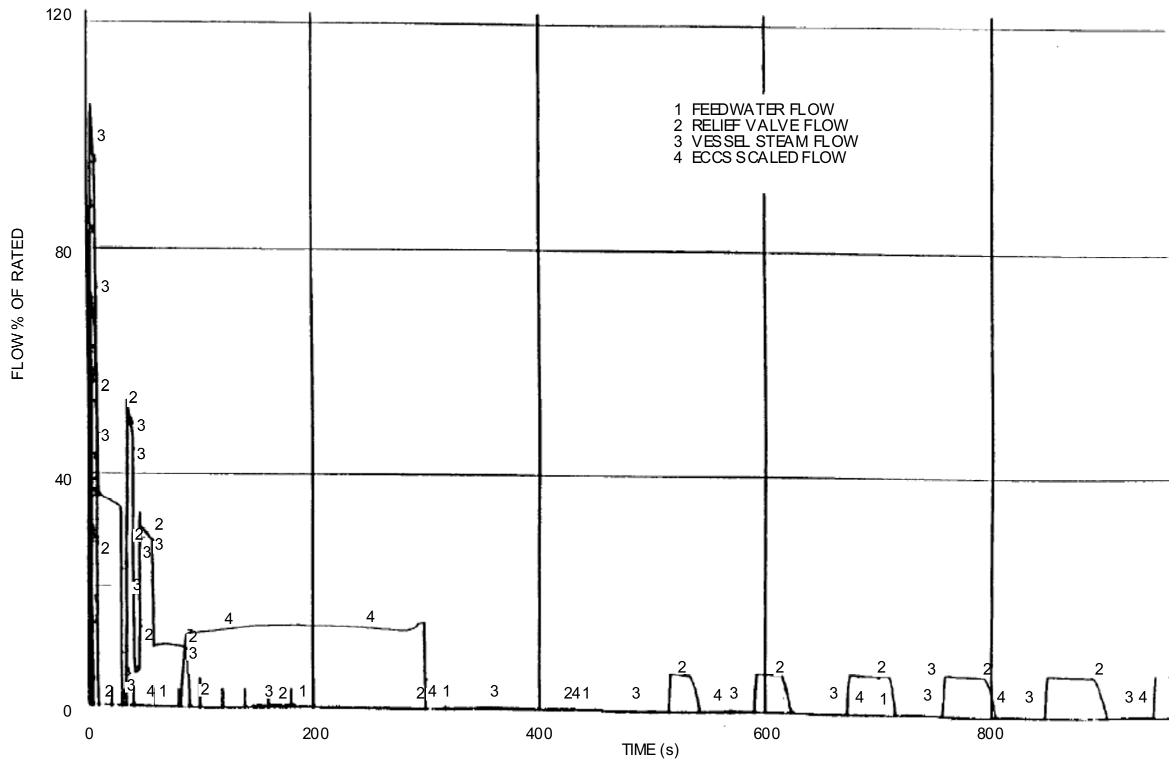


Figure 15E-7 ABWR Loss of AC Power, FMCRD Run–in (Continued)

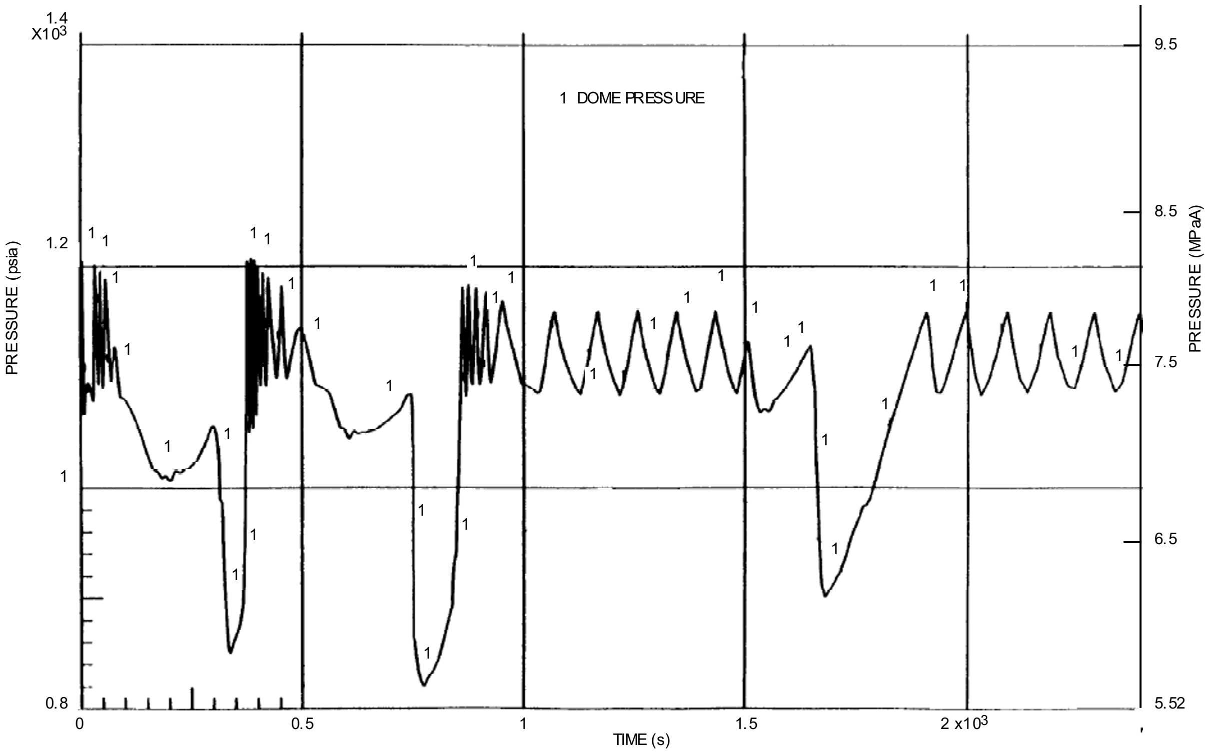


Figure 15E-8 ABWR Loss of AC Power, SLCS

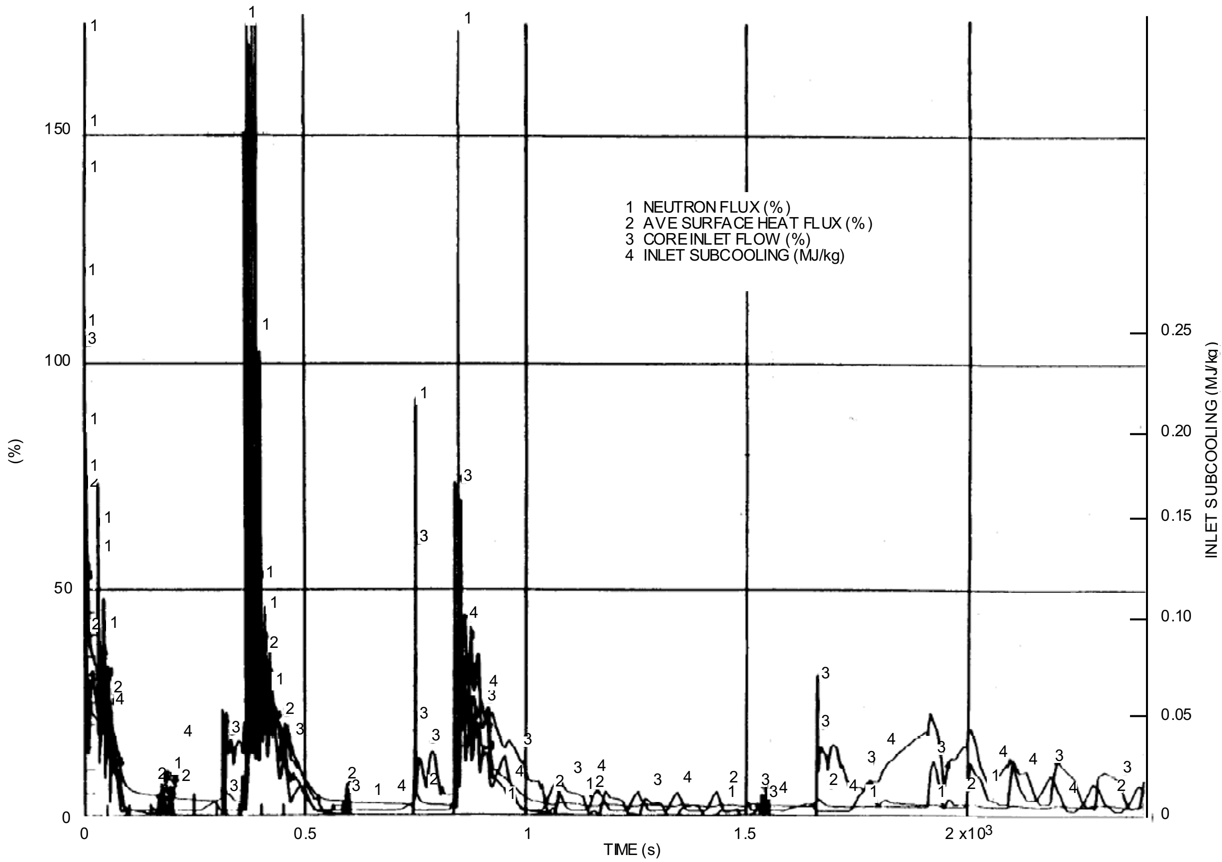


Figure 15E-8 ABWR Loss of AC Power, SLCS (Continued)

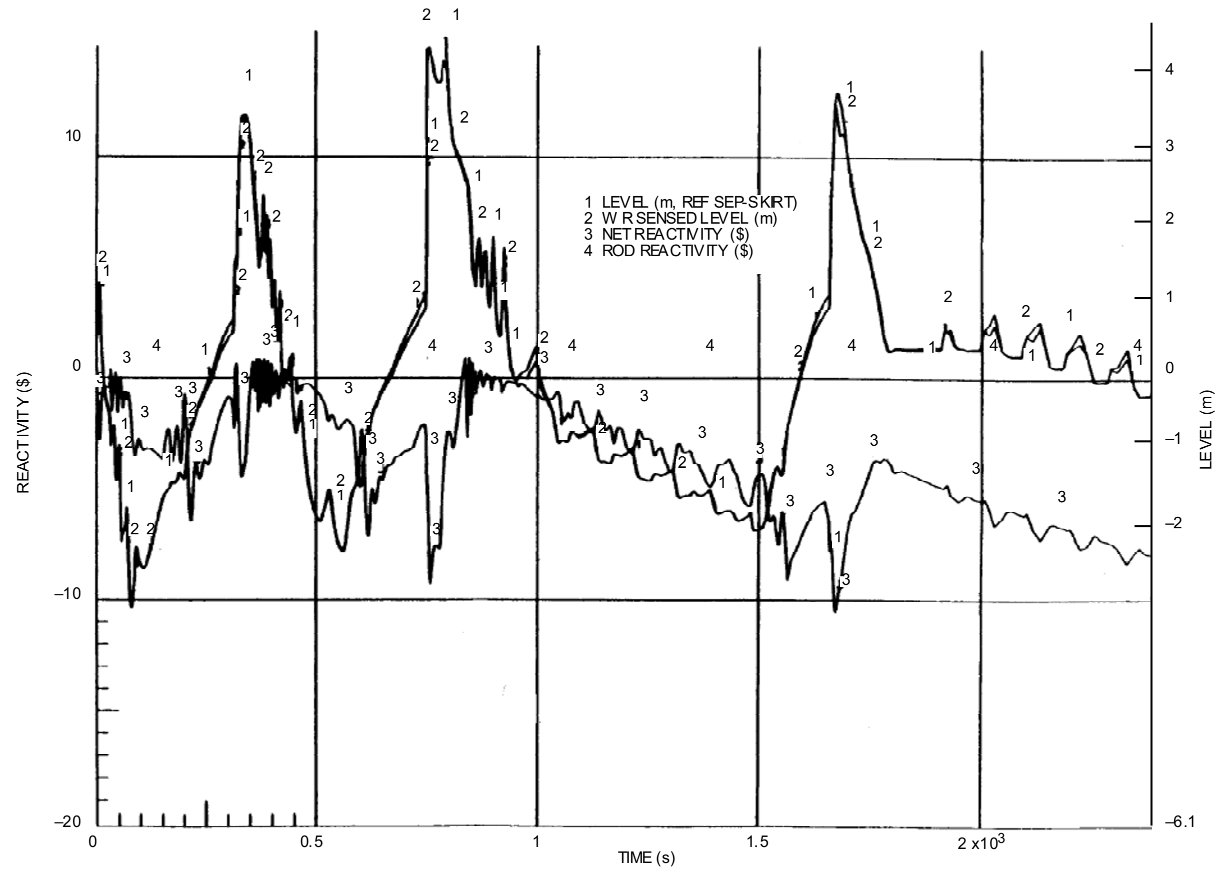


Figure 15E-8 ABWR Loss of AC Power, SLCS (Continued)

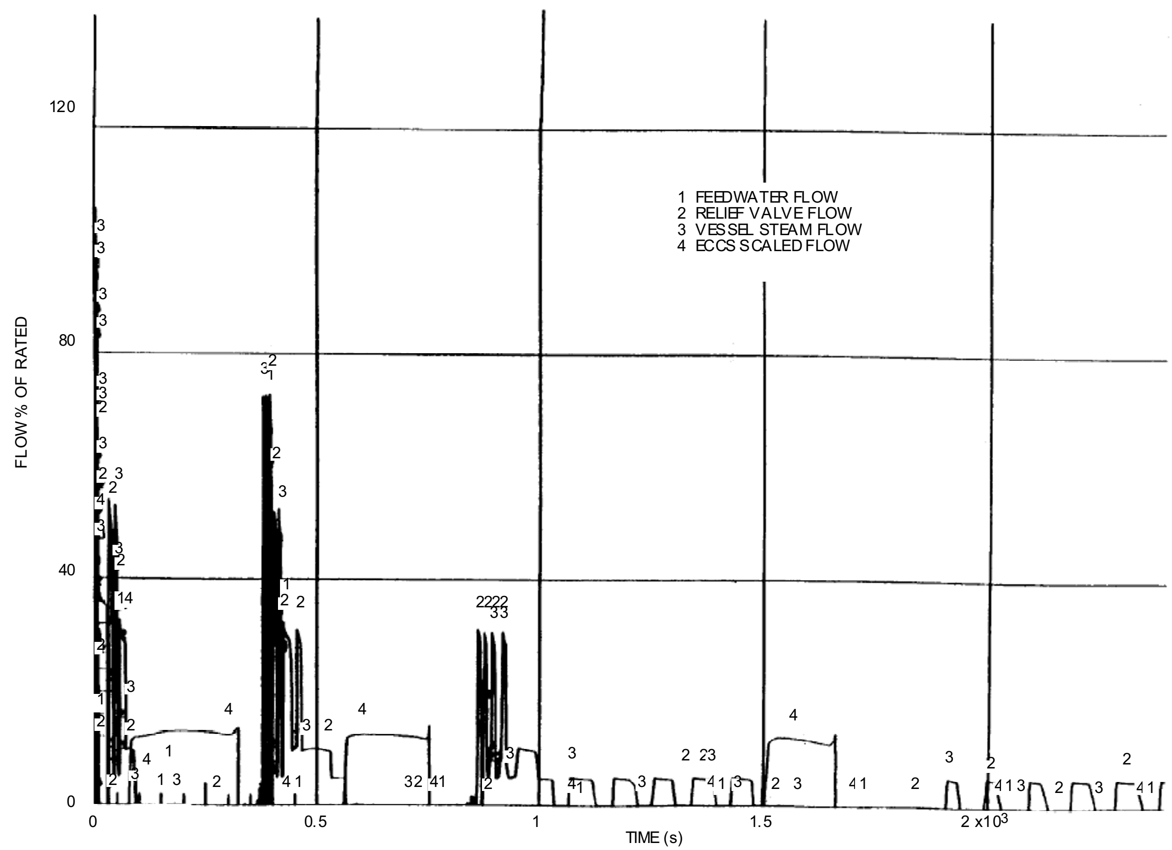


Figure 15E-8 ABWR Loss of AC Power, SLCS (Continued)

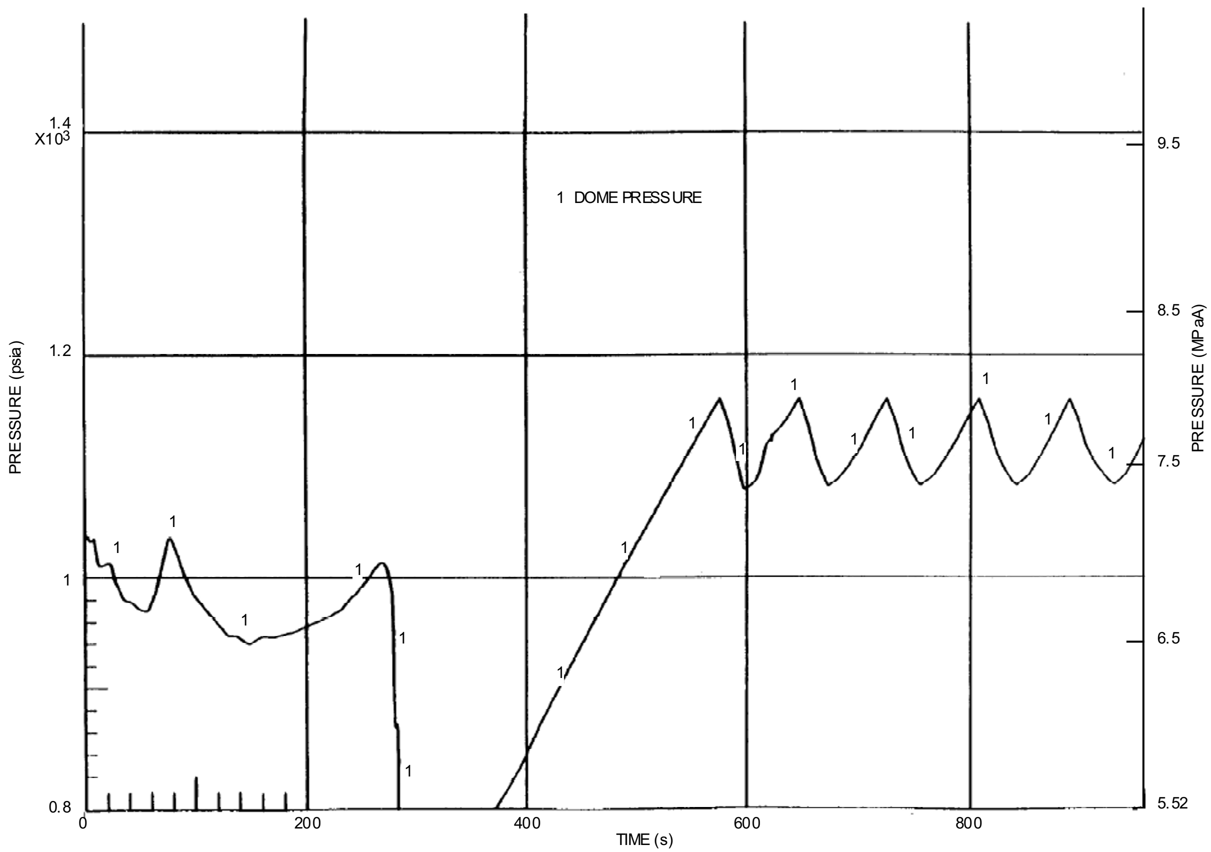


Figure 15E-9 ABWR Loss of Feedwater Flow, ARI

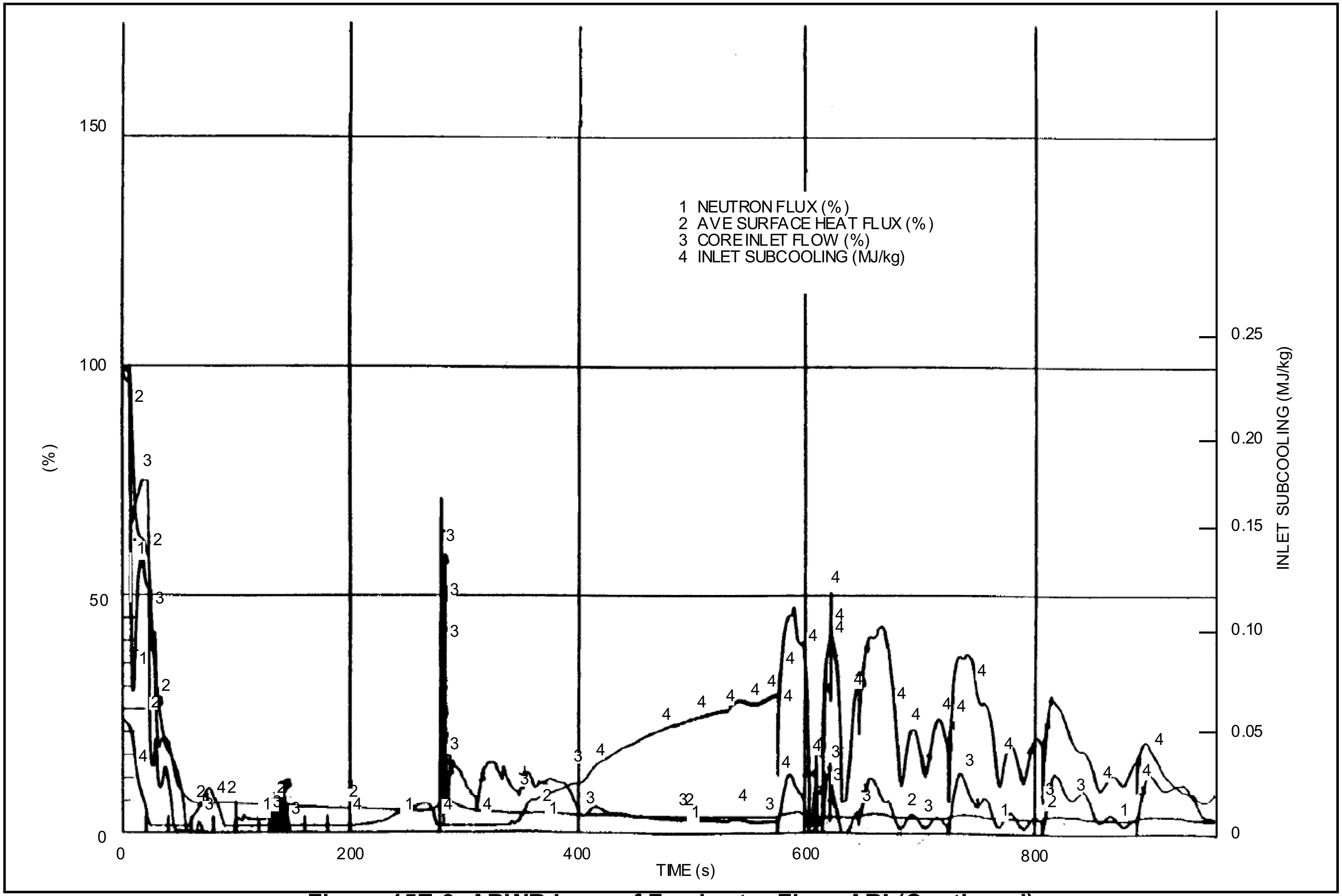


Figure 15E-9 ABWR Loss of Feedwater Flow, ARI (Continued)

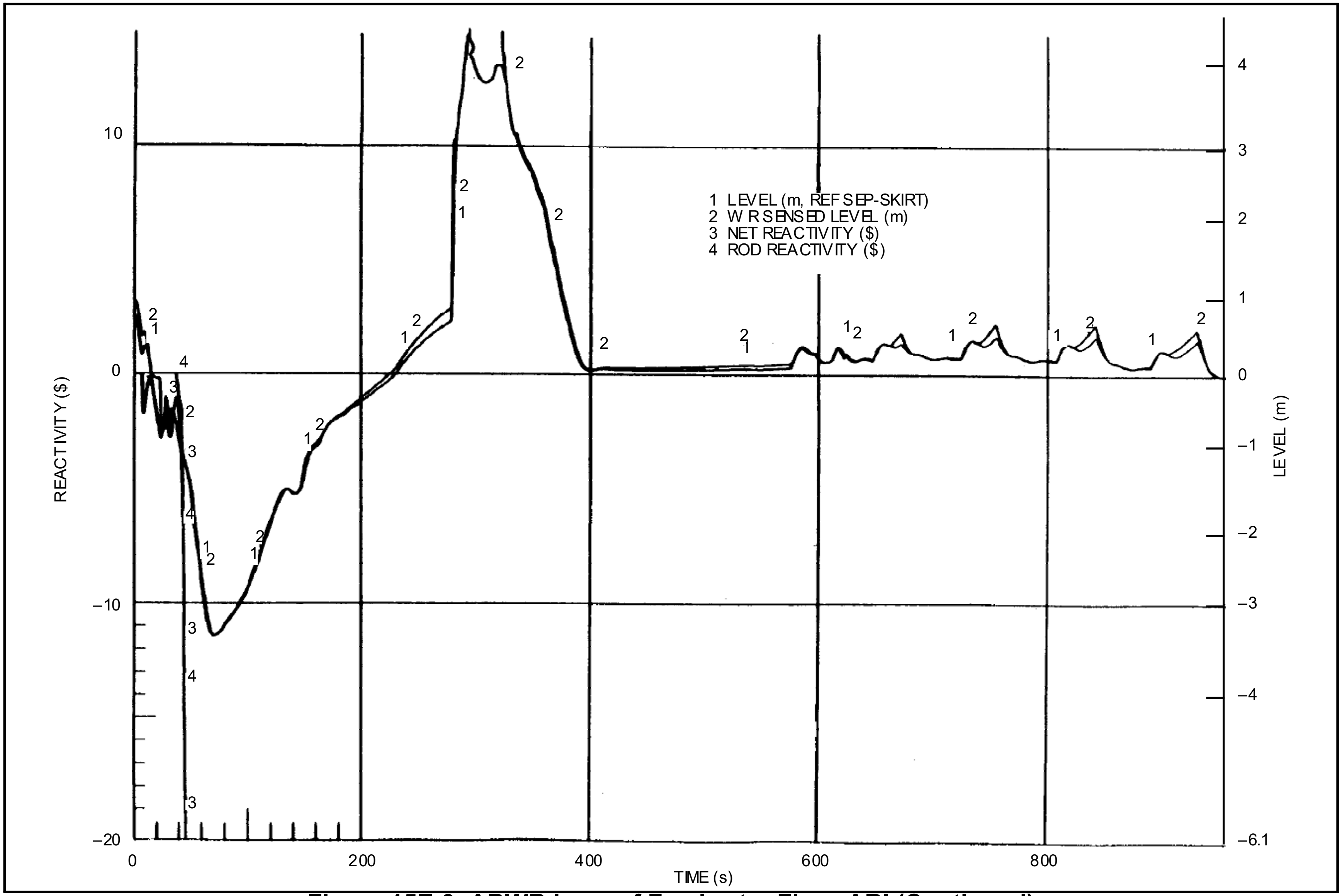


Figure 15E-9 ABWR Loss of Feedwater Flow, ARI (Continued)

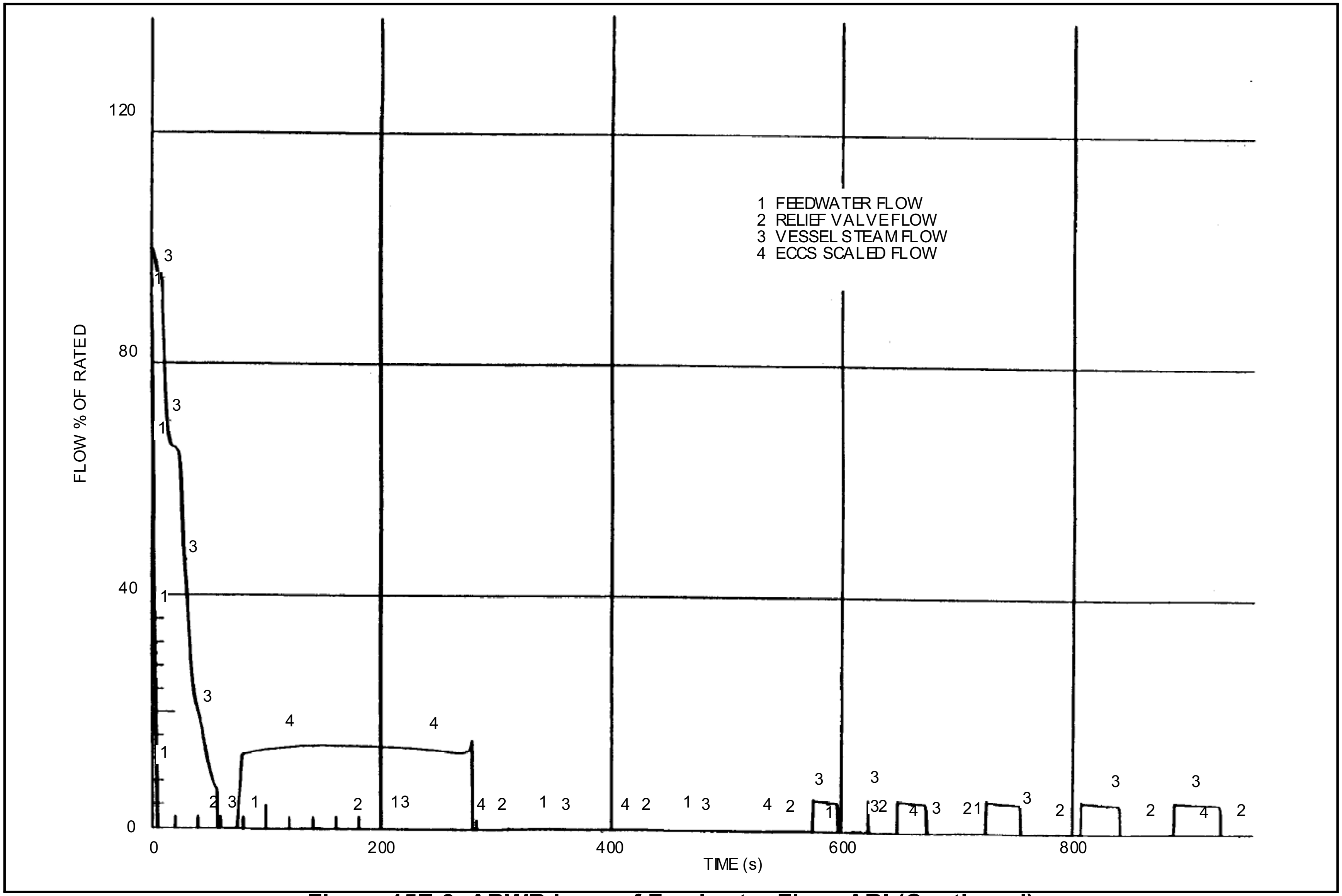


Figure 15E-9 ABWR Loss of Feedwater Flow, ARI (Continued)

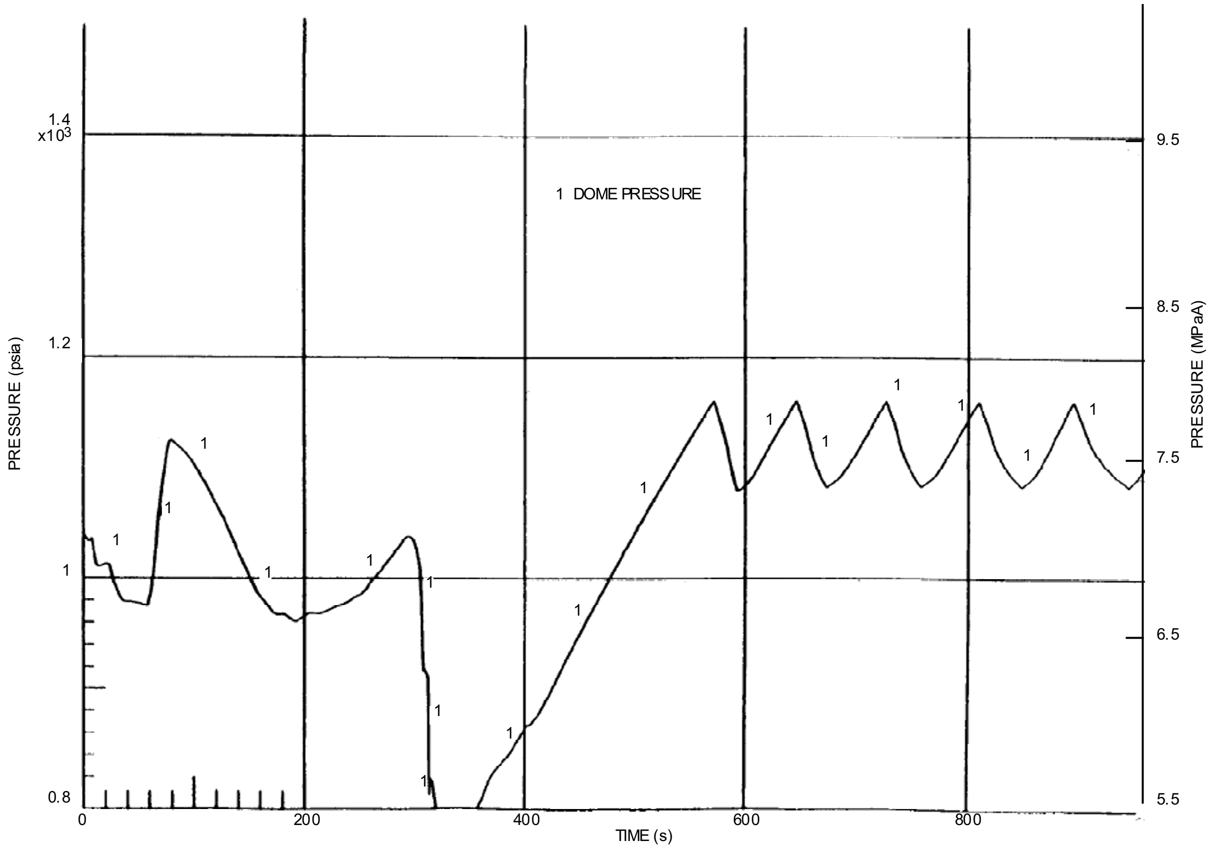


Figure 15E-10 ABWR Loss of Feedwater Flow, FMCRD Run–in

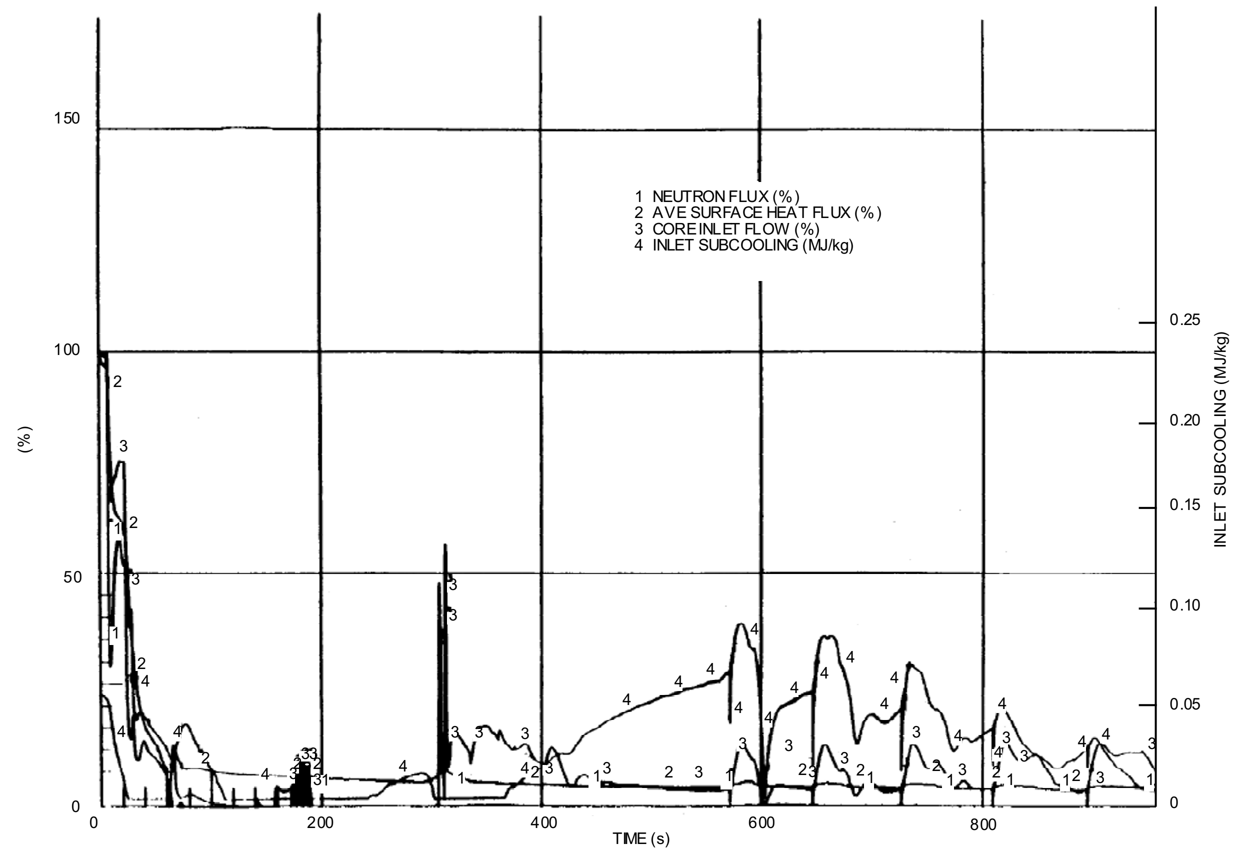


Figure 15E-10 ABWR Loss of Feedwater Flow, FMCRD Run–in (Continued)

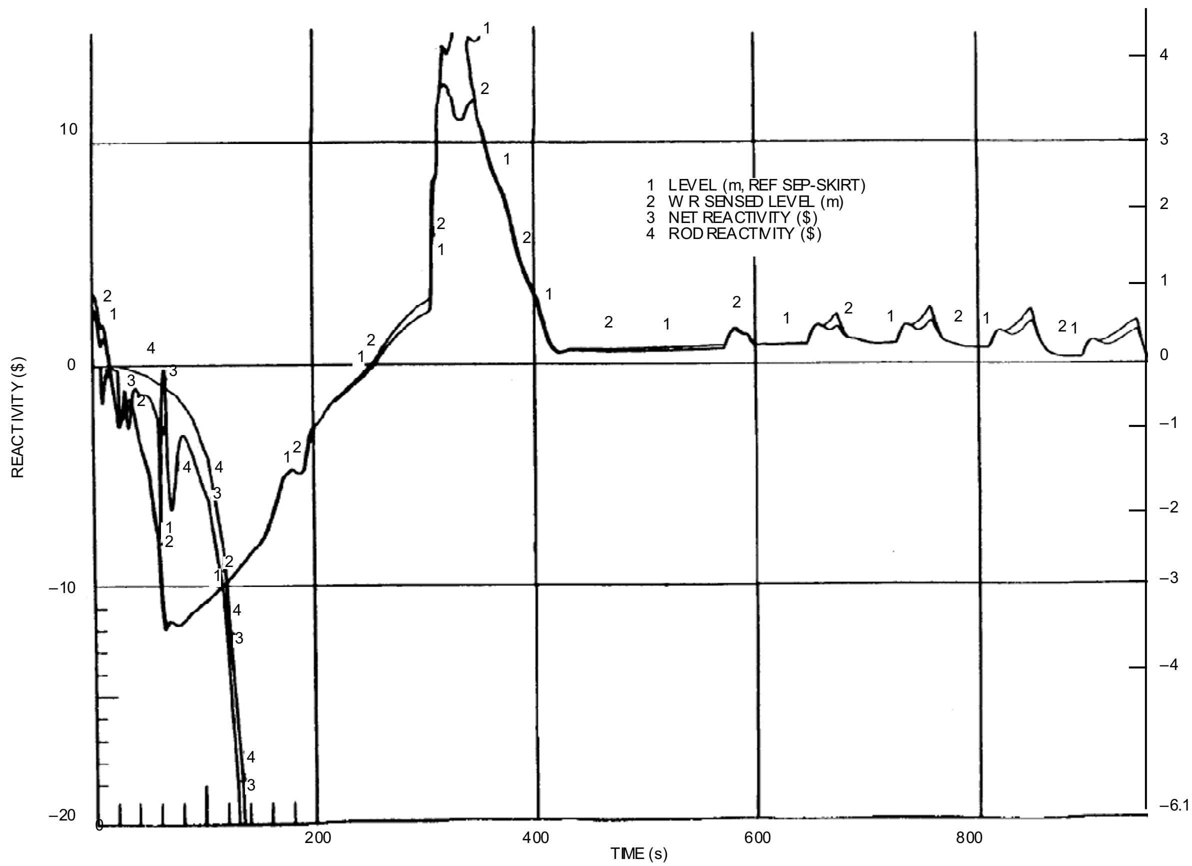


Figure 15E-10 ABWR Loss of Feedwater Flow, FMCRD Run–in (Continued)

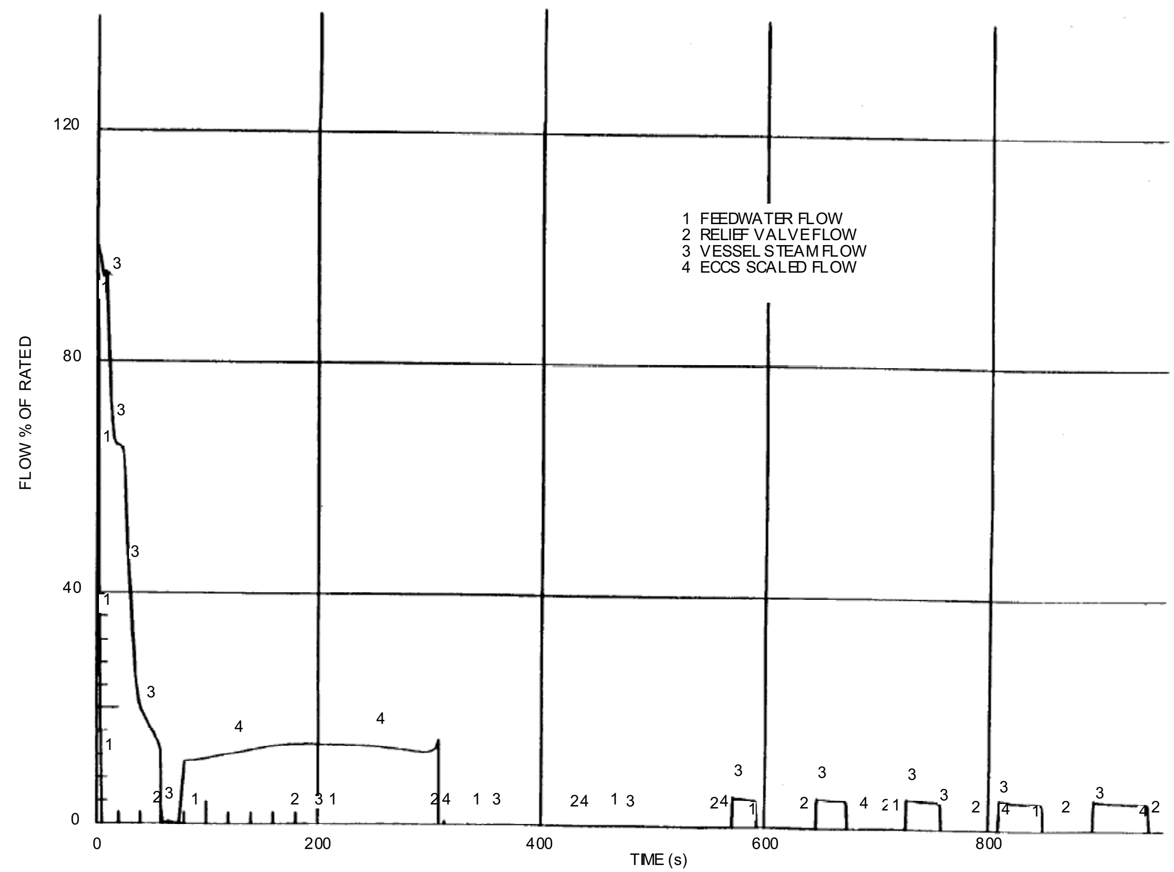


Figure 15E-10 ABWR Loss of Feedwater Flow, FMCRD Run–in (Continued)

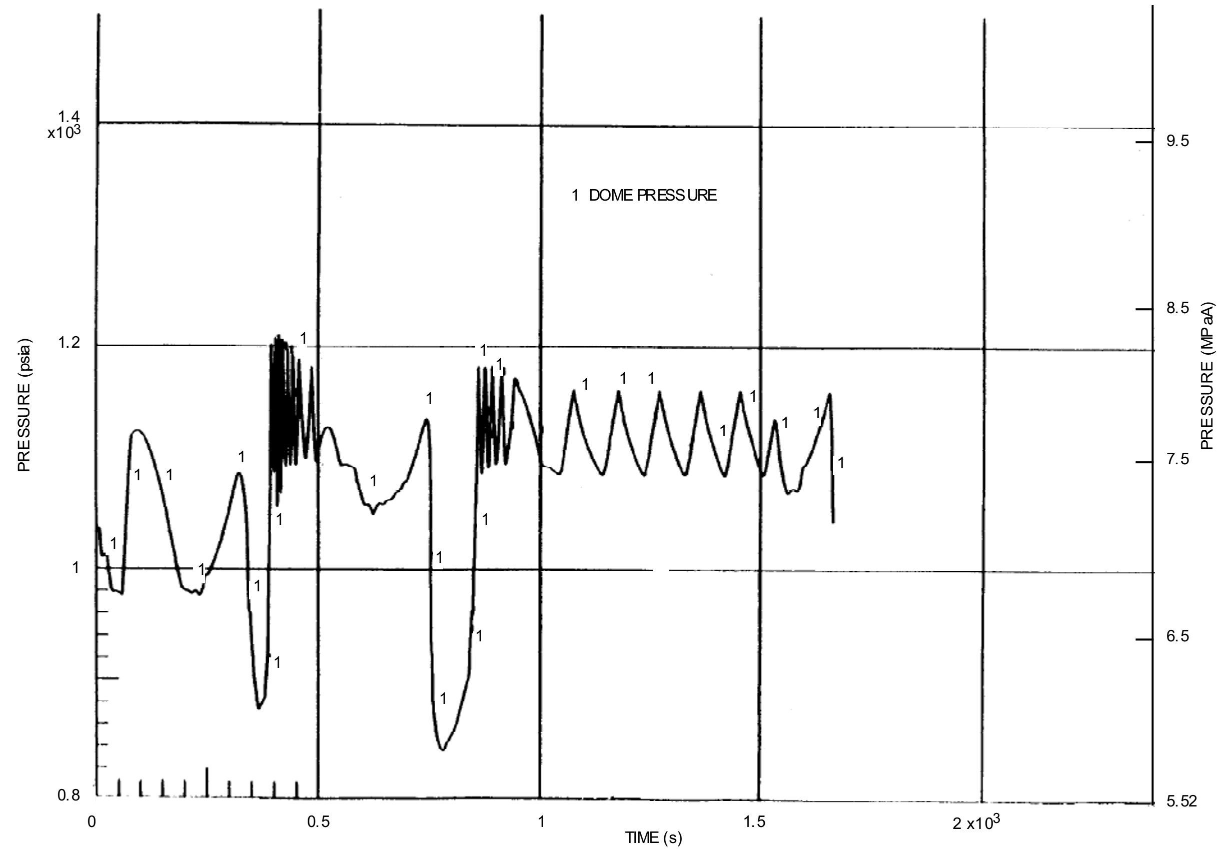


Figure 15E-11 ABWR Loss of Feedwater Flow, SLCS

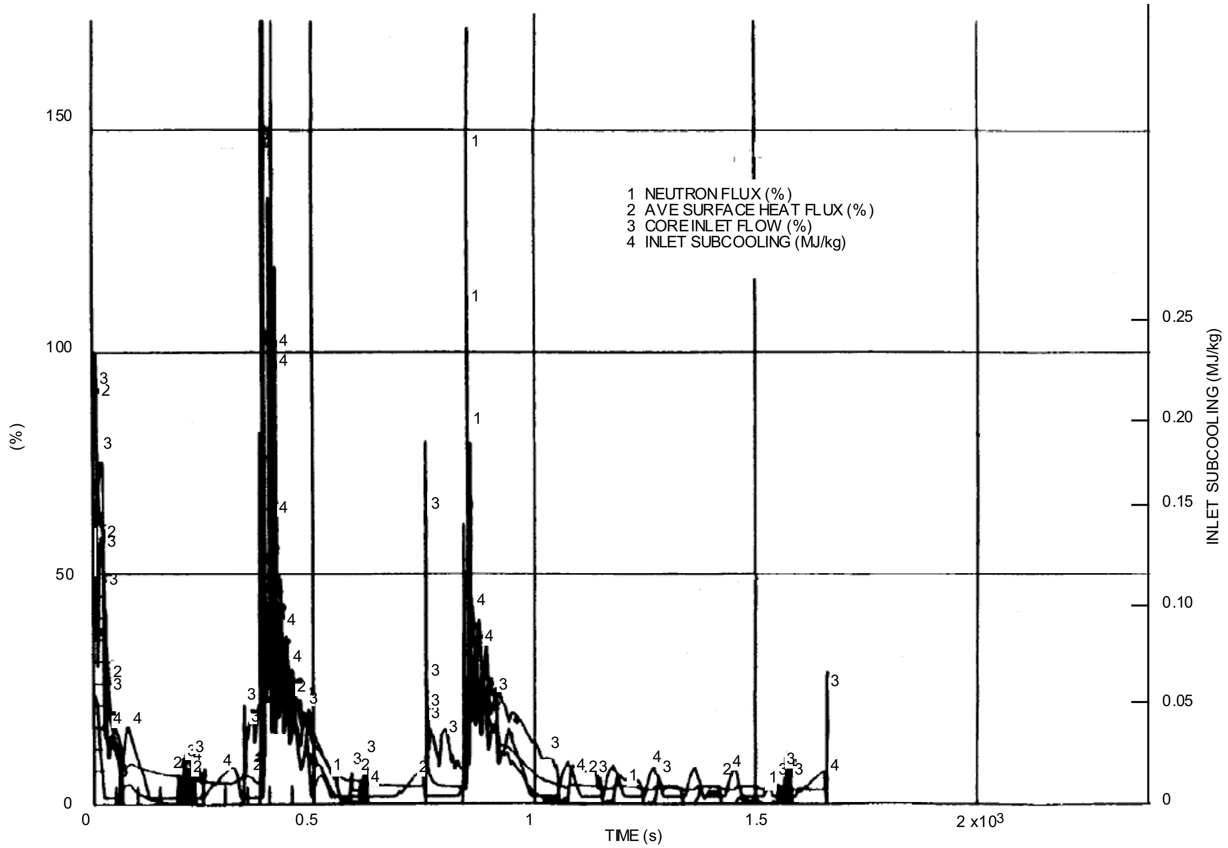


Figure 15E-11 ABWR Loss of Feedwater Flow, SLCS (Continued)

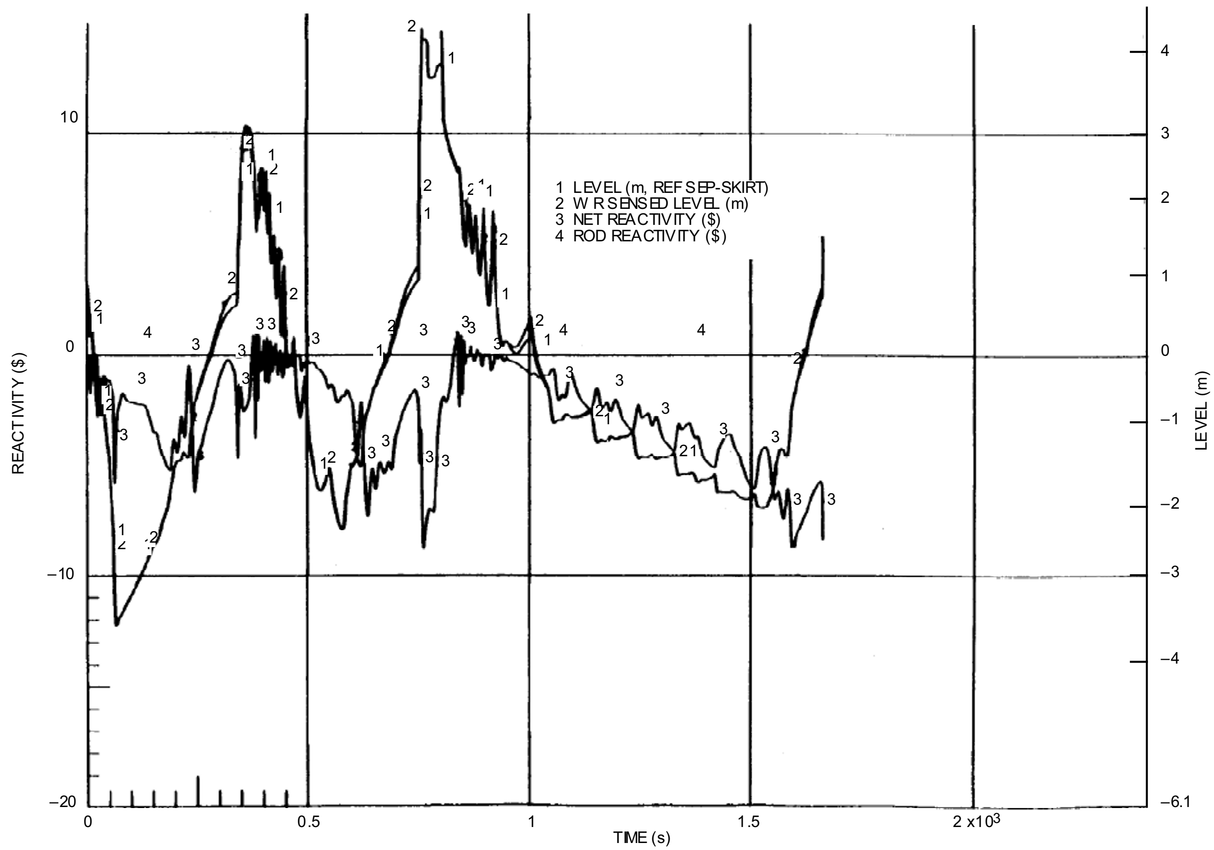


Figure 15E-11 ABWR Loss of Feedwater Flow, SLCS (Continued)

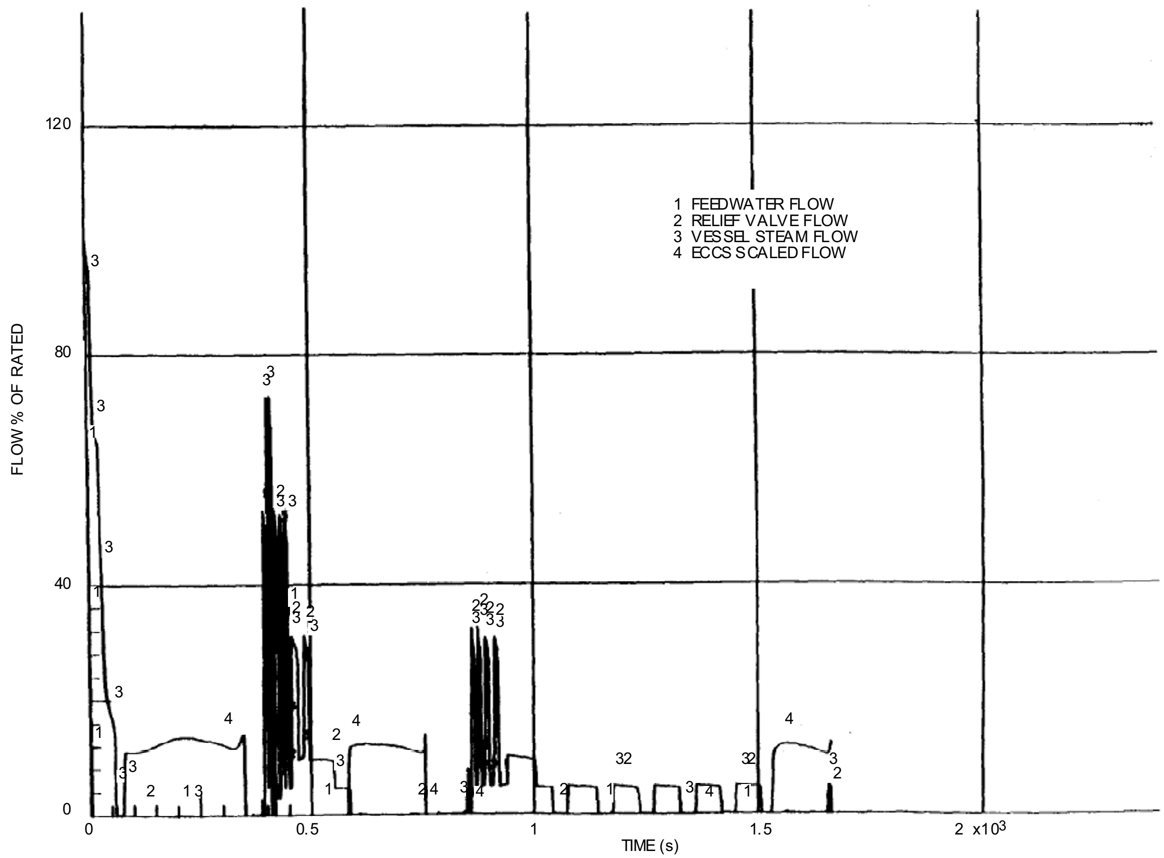


Figure 15E-11 ABWR Loss of Feedwater Flow, SLCS (Continued)

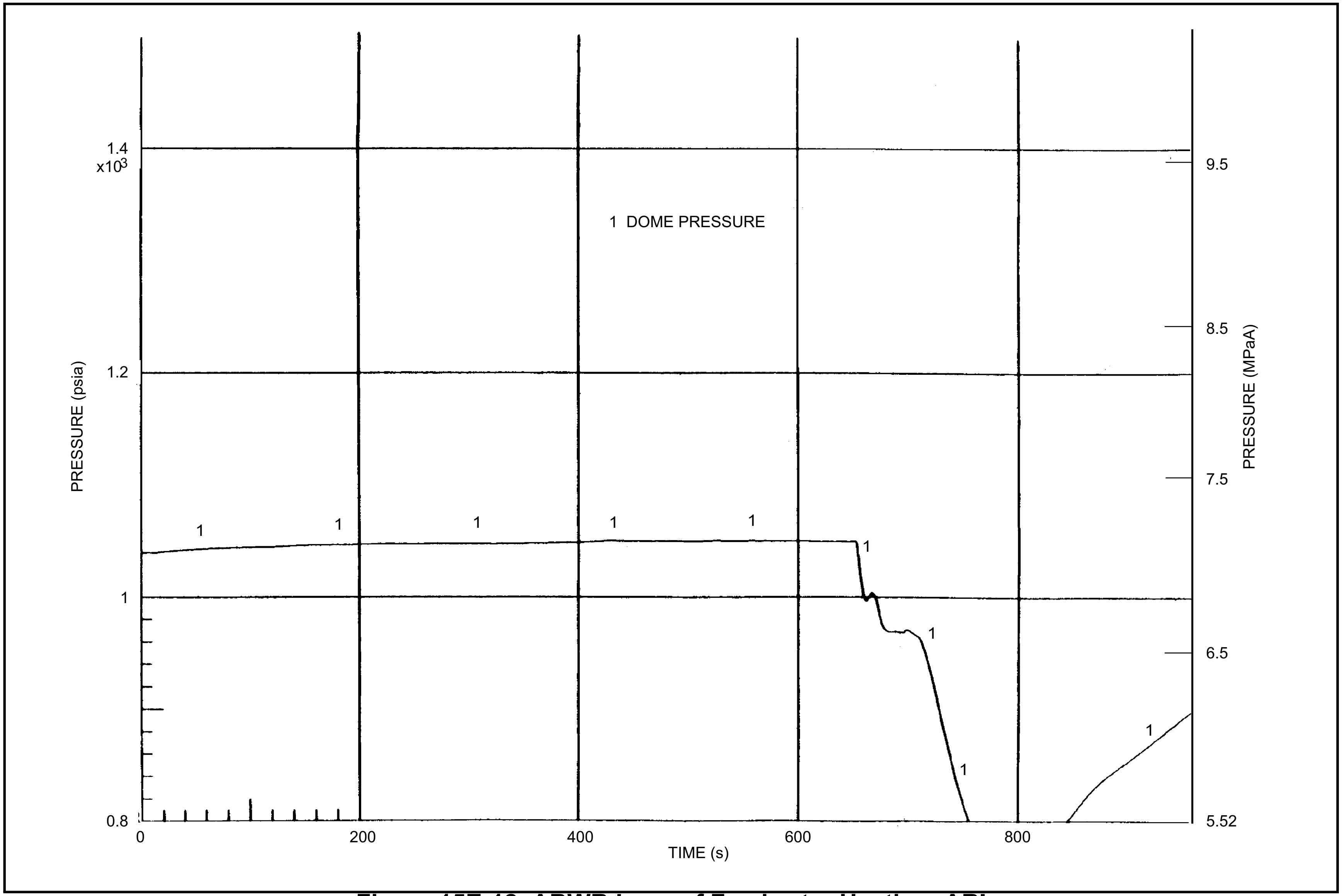


Figure 15E-12 ABWR Loss of Feedwater Heating, ARI

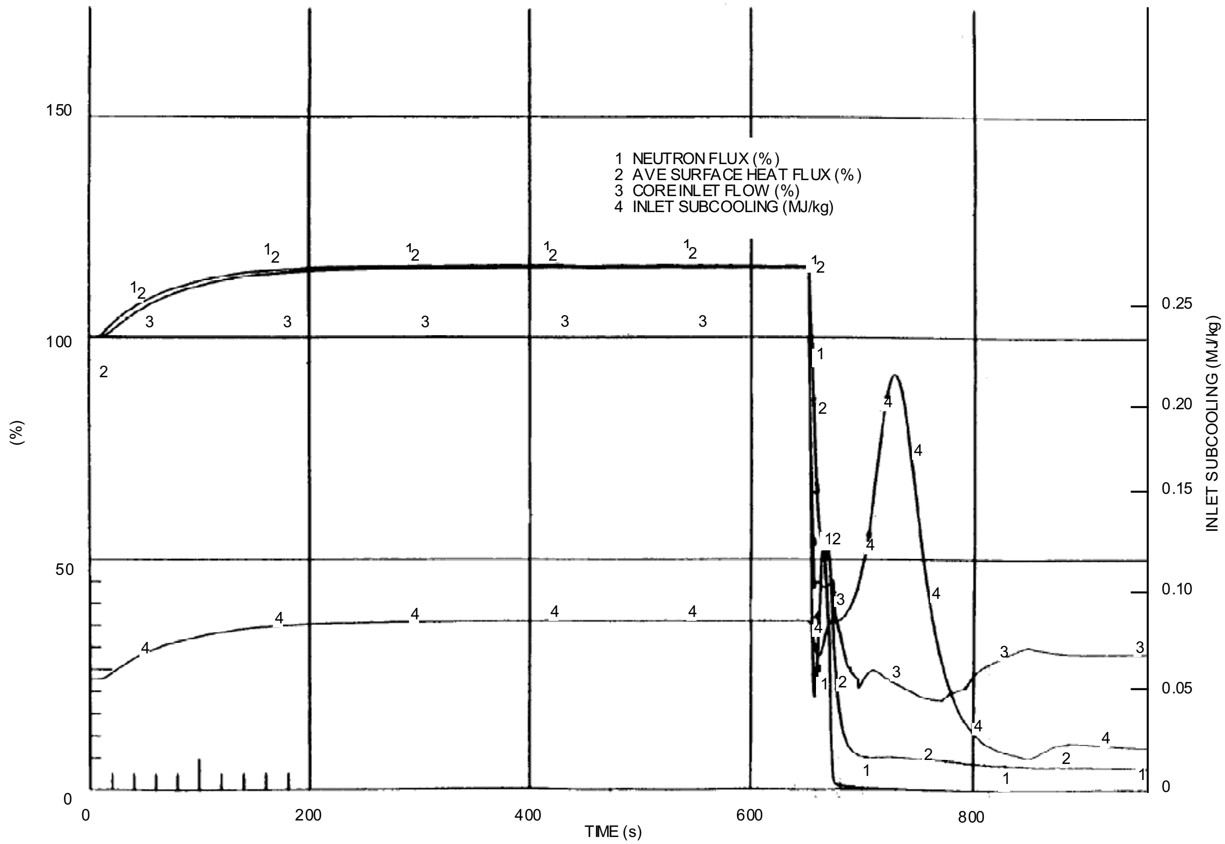


Figure 15E-12 ABWR Loss of Feedwater Heating, ARI (Continued)

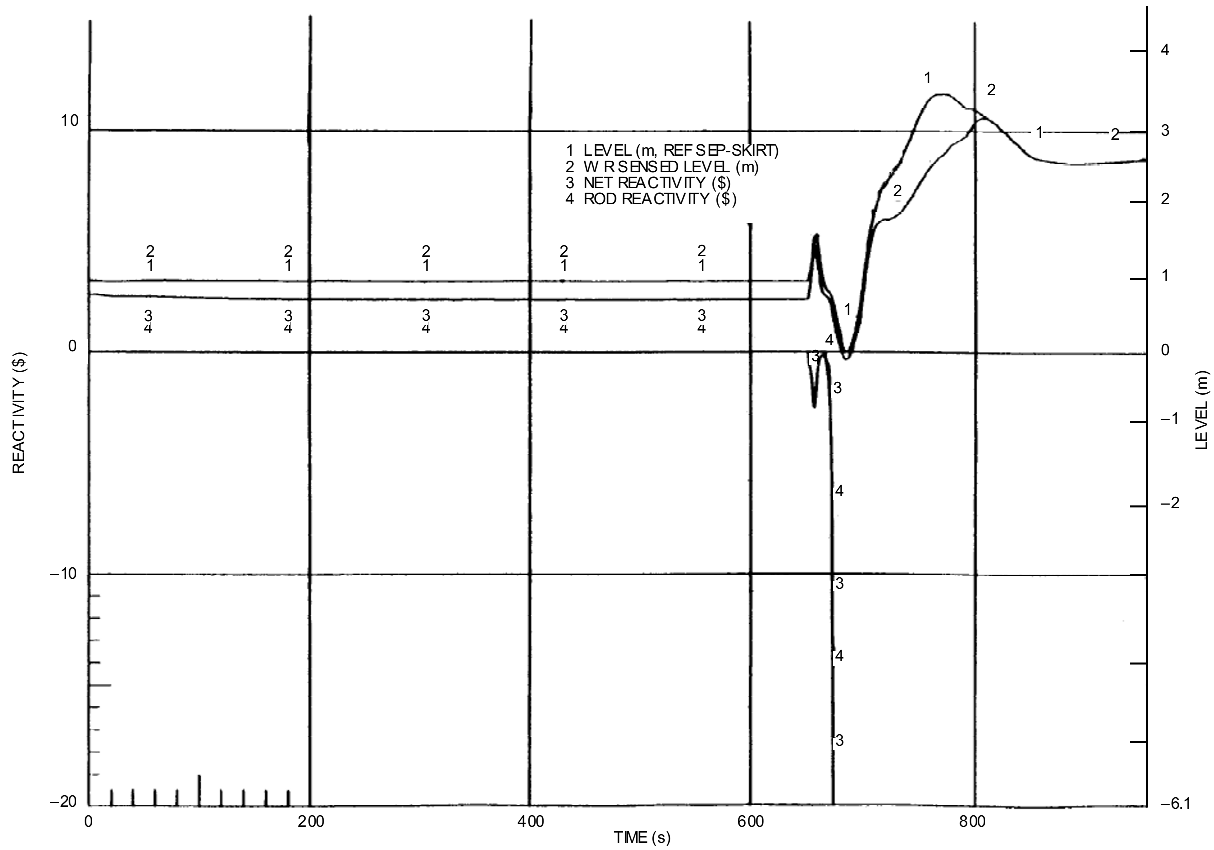


Figure 15E-12 ABWR Loss of Feedwater Heating, ARI (Continued)

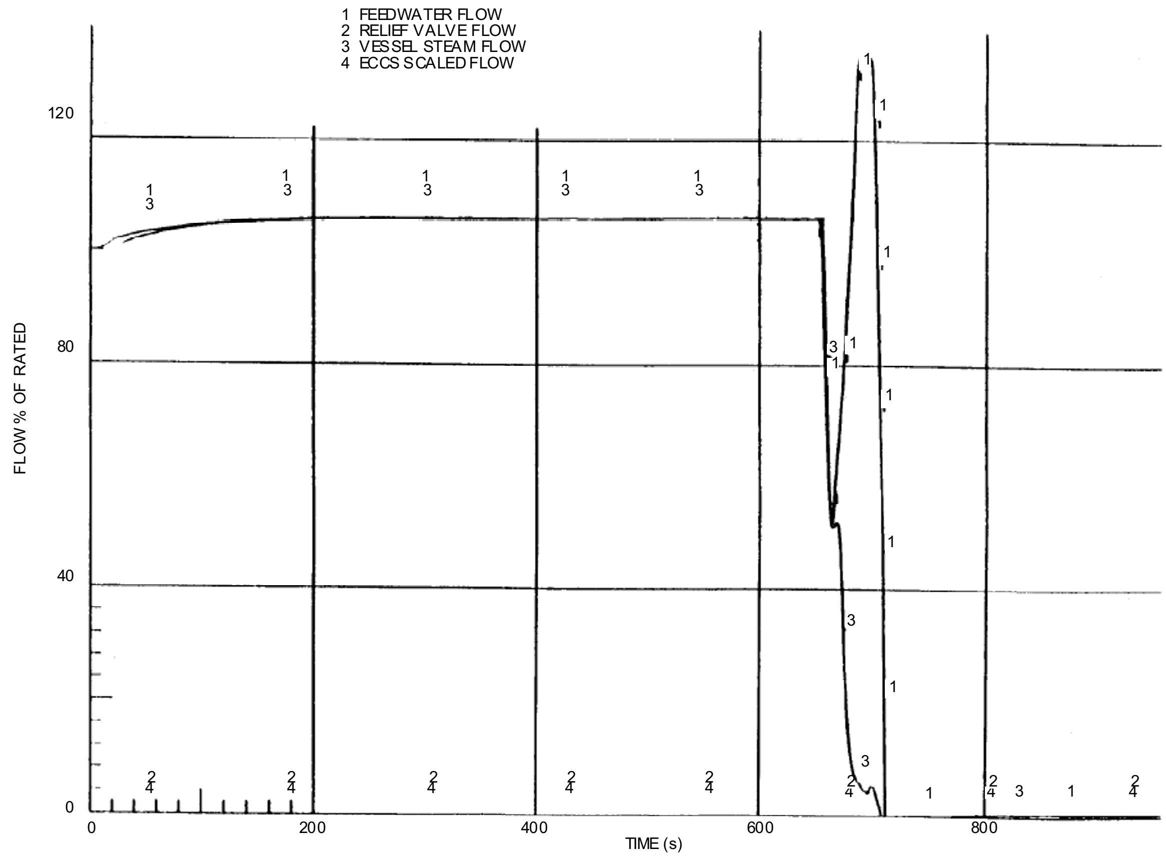


Figure 15E-12 ABWR Loss of Feedwater Heating, ARI (Continued)

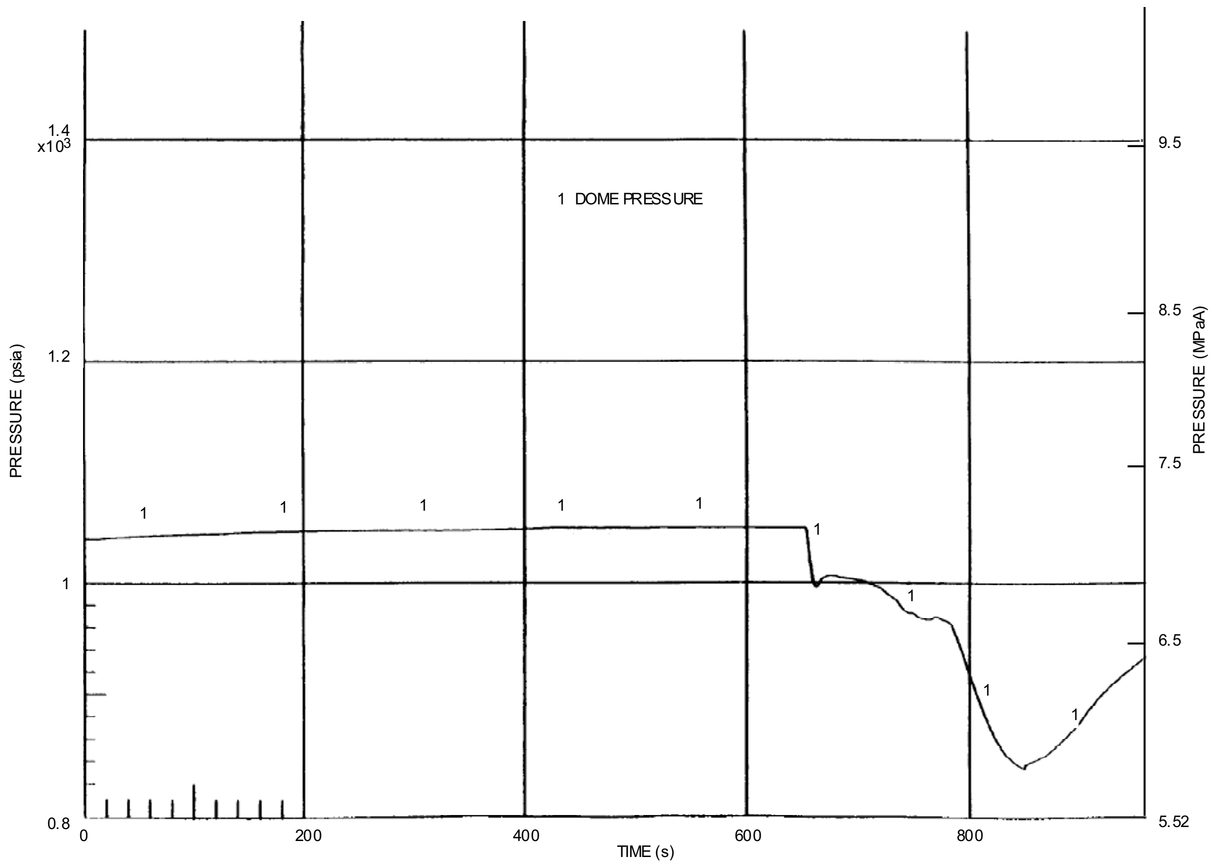


Figure 15E-13 ABWR Loss of Feedwater Heating, FMCRD Run–in

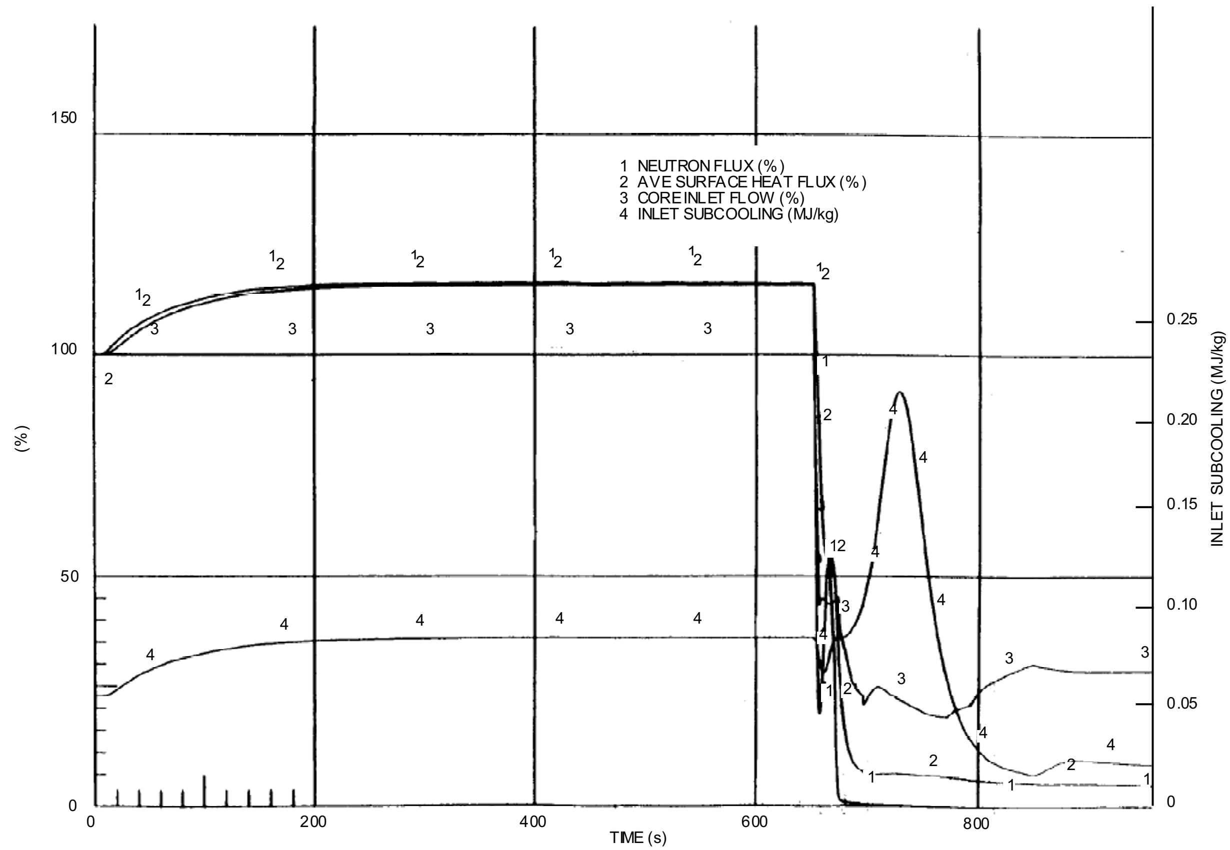


Figure 15E-13 ABWR Loss of Feedwater Heating, FMCRD Run–in (Continued)

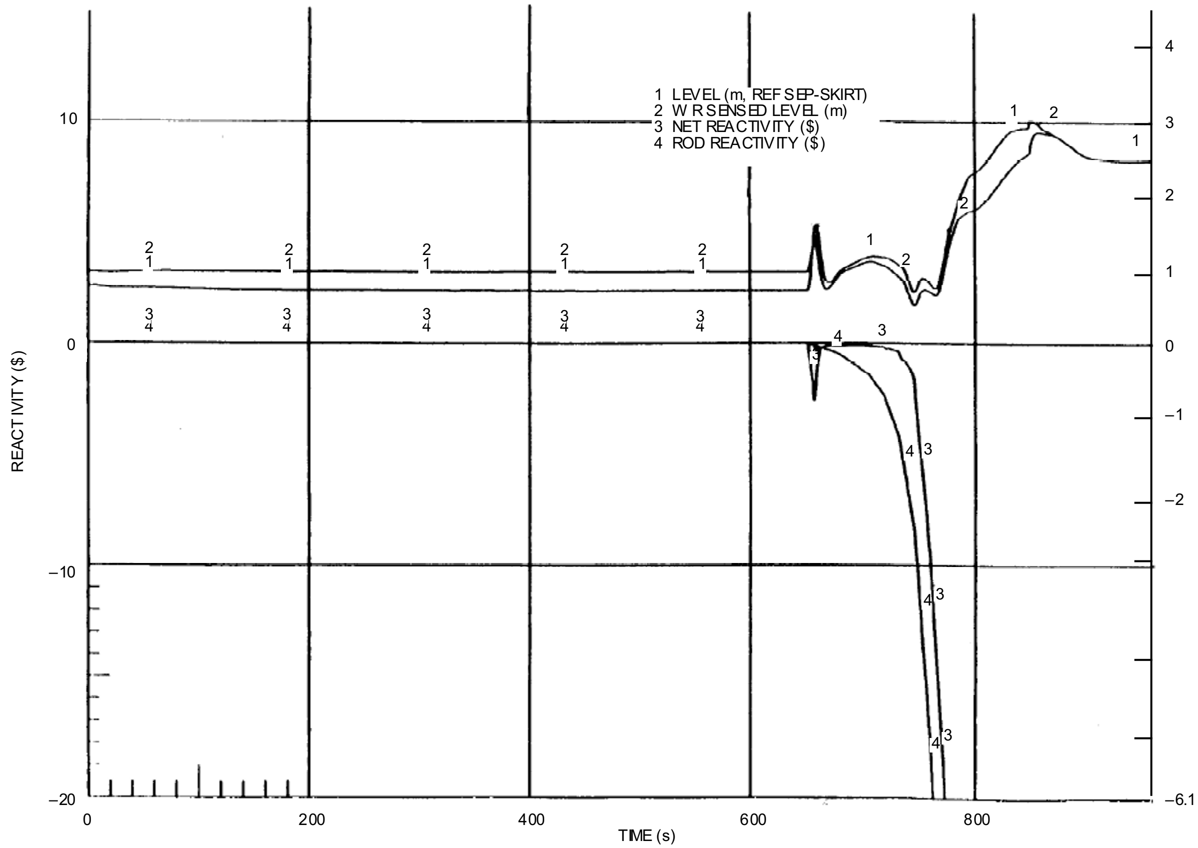


Figure 15E-13 ABWR Loss of Feedwater Heating, FMCRD Run–in (Continued)

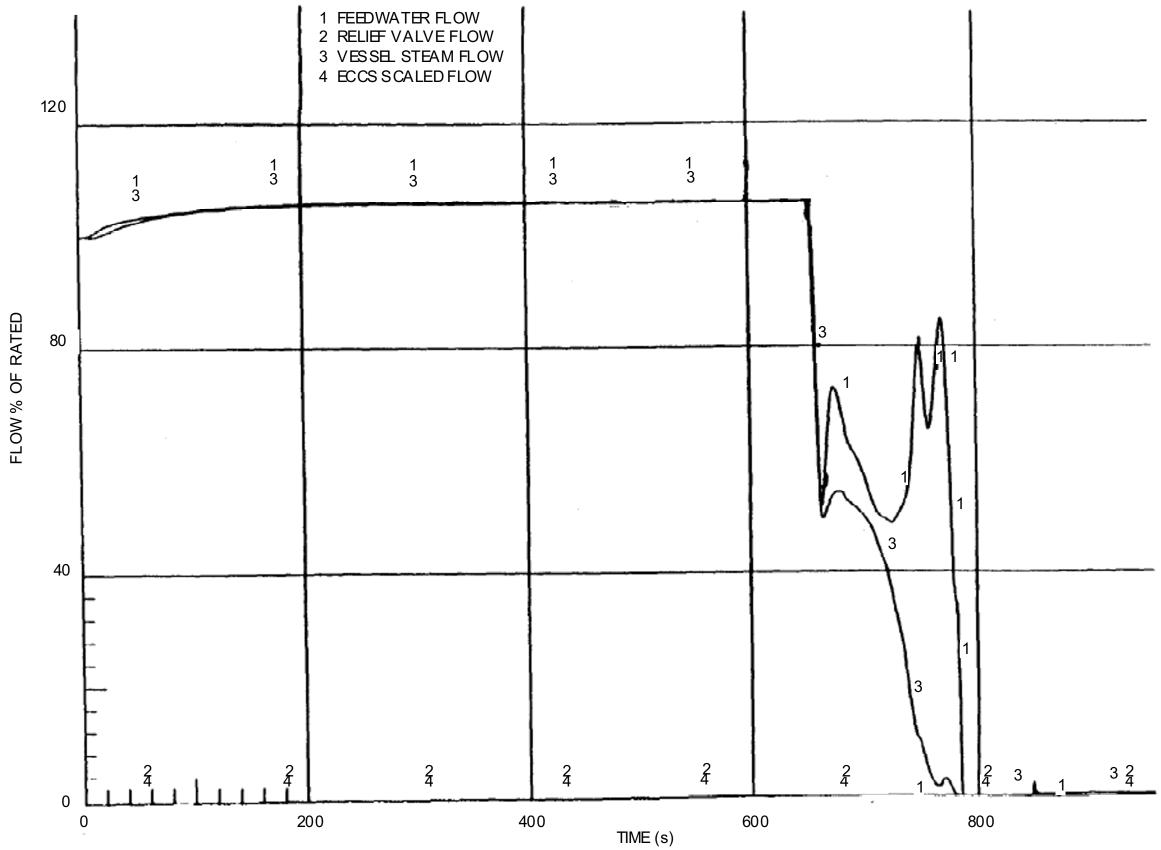


Figure 15E-13 ABWR Loss of Feedwater Heating, FMCRD Run–in (Continued)

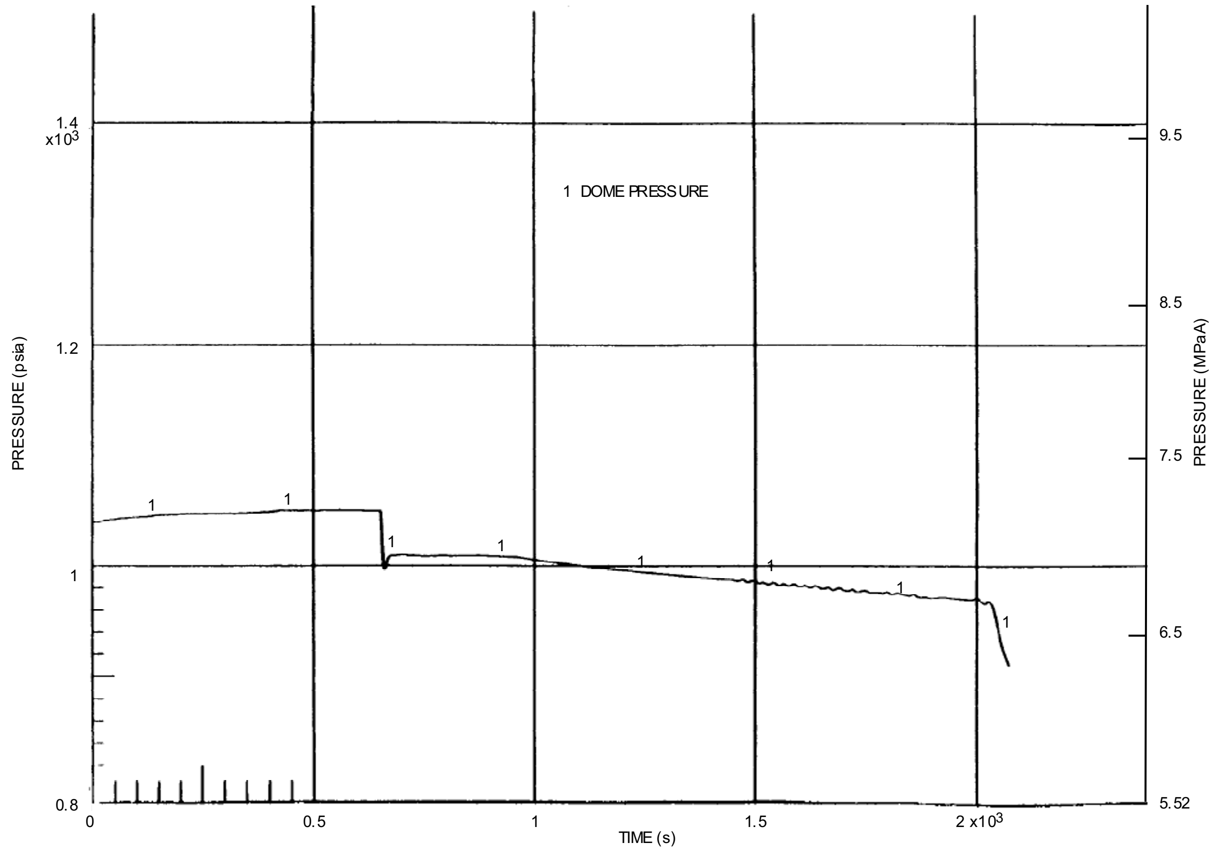


Figure 15E-14 ABWR Loss of Feedwater Heating, SLCS

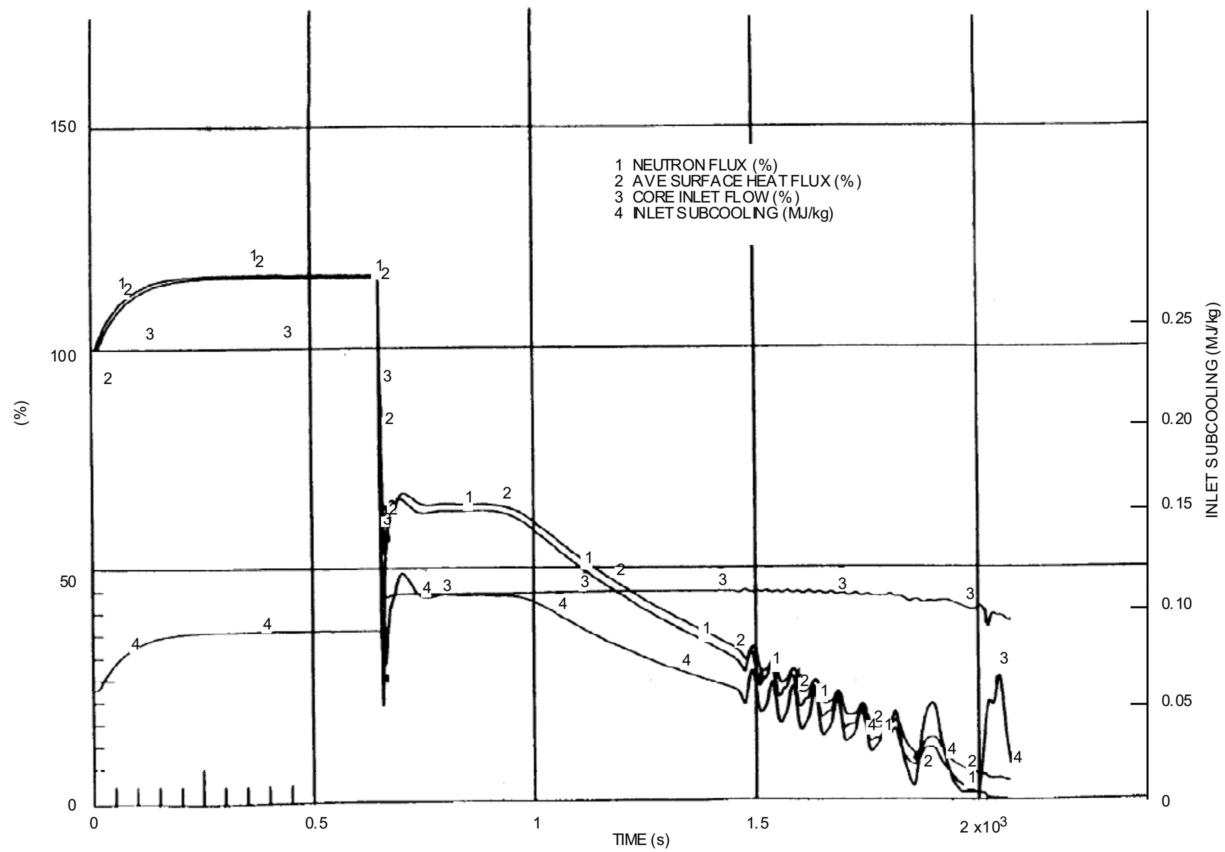


Figure 15E-14 ABWR Loss of Feedwater Heating, SLCS (Continued)

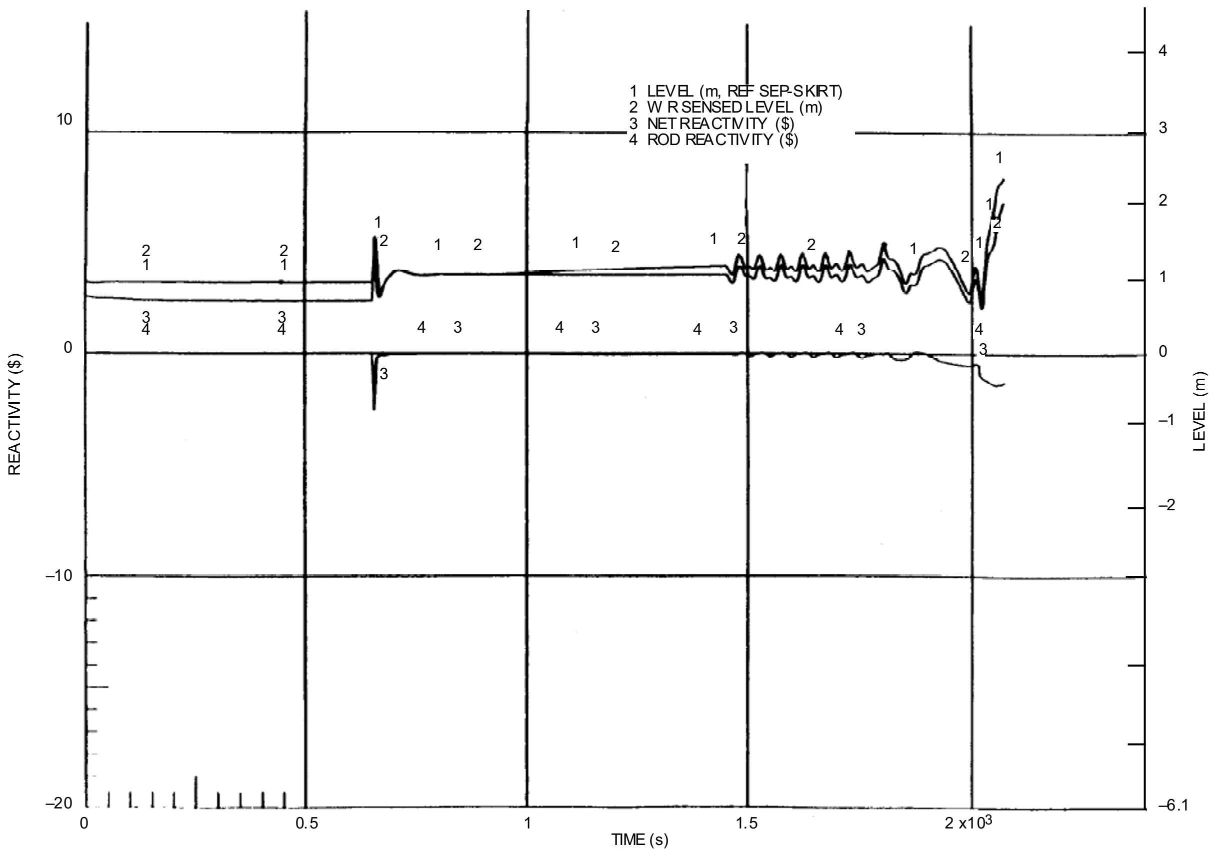


Figure 15E-14 ABWR Loss of Feedwater Heating, SLCS (Continued)

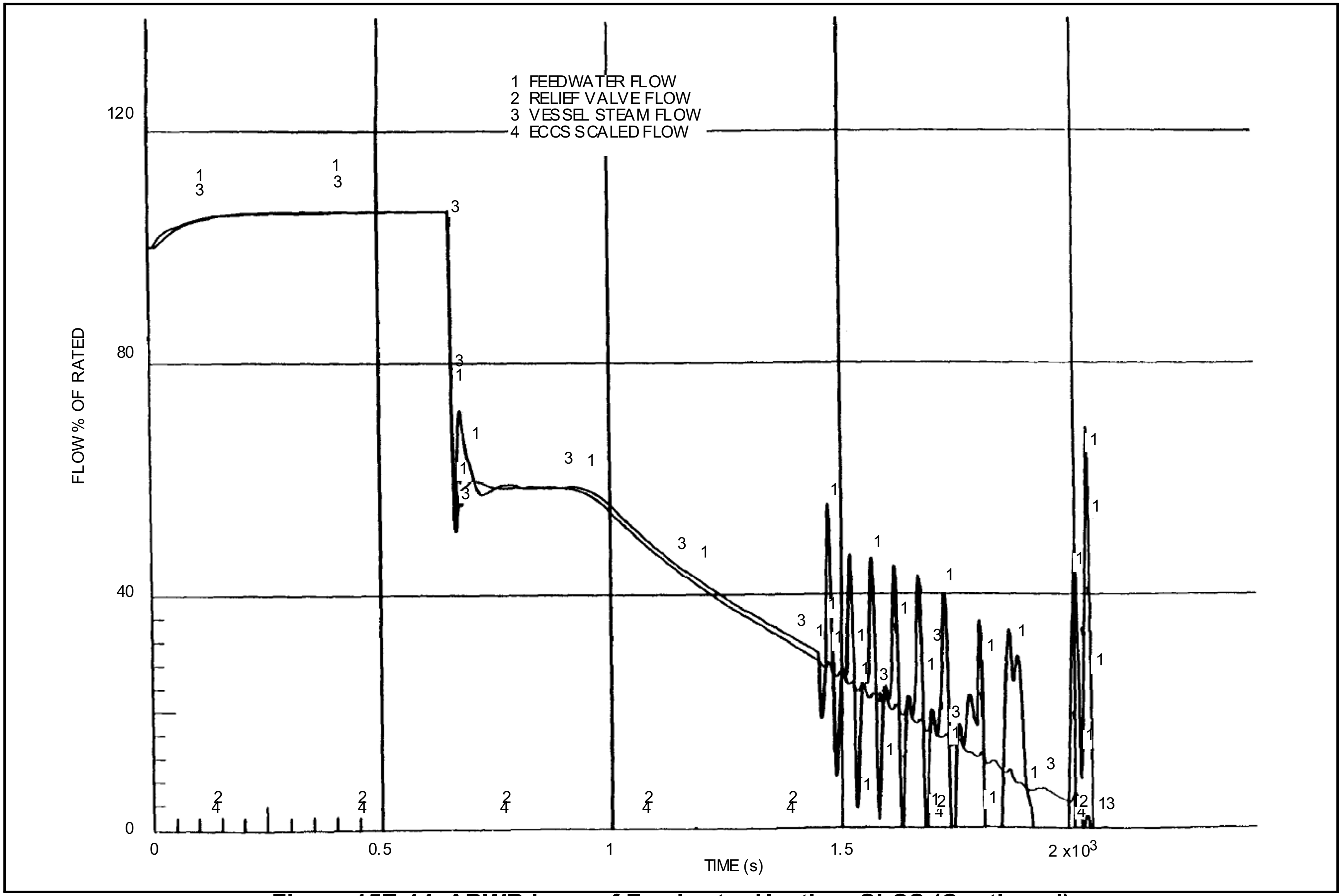


Figure 15E-14 ABWR Loss of Feedwater Heating, SLCS (Continued)

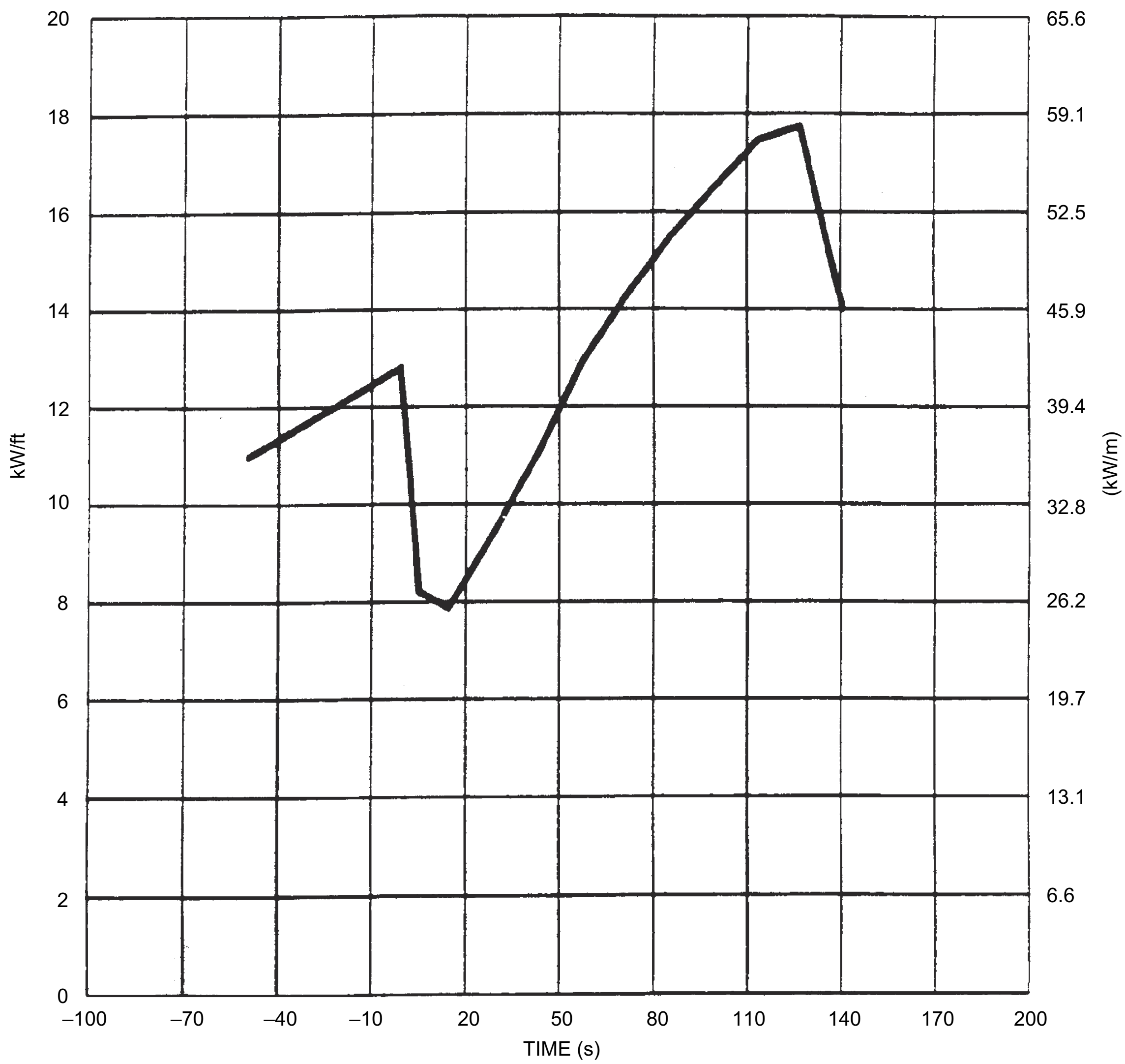


Figure 15E-15 ABWR Loss of Feedwater Heating, Max. LHGR

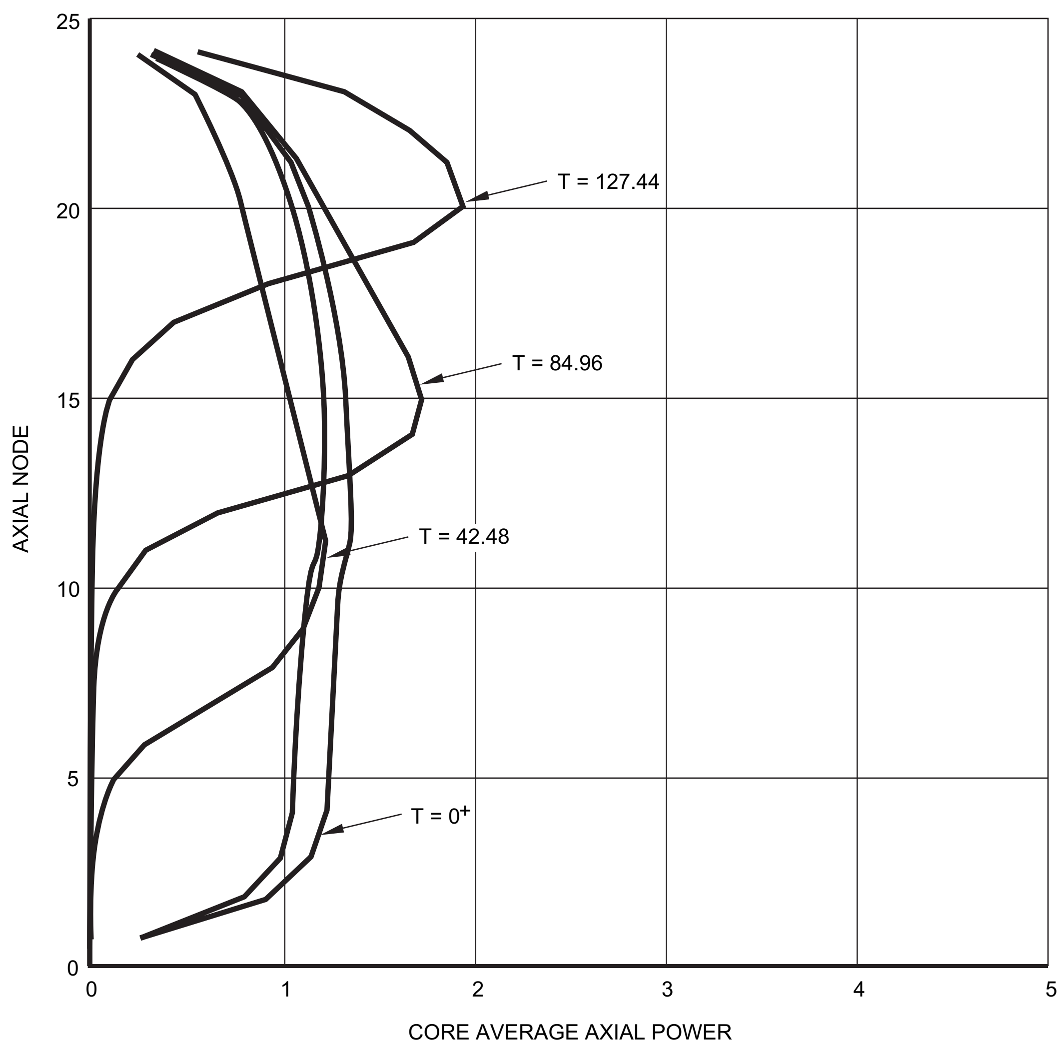


Figure 15E-16 ABWR Loss of Feedwater Heating, FMCRD Run–in

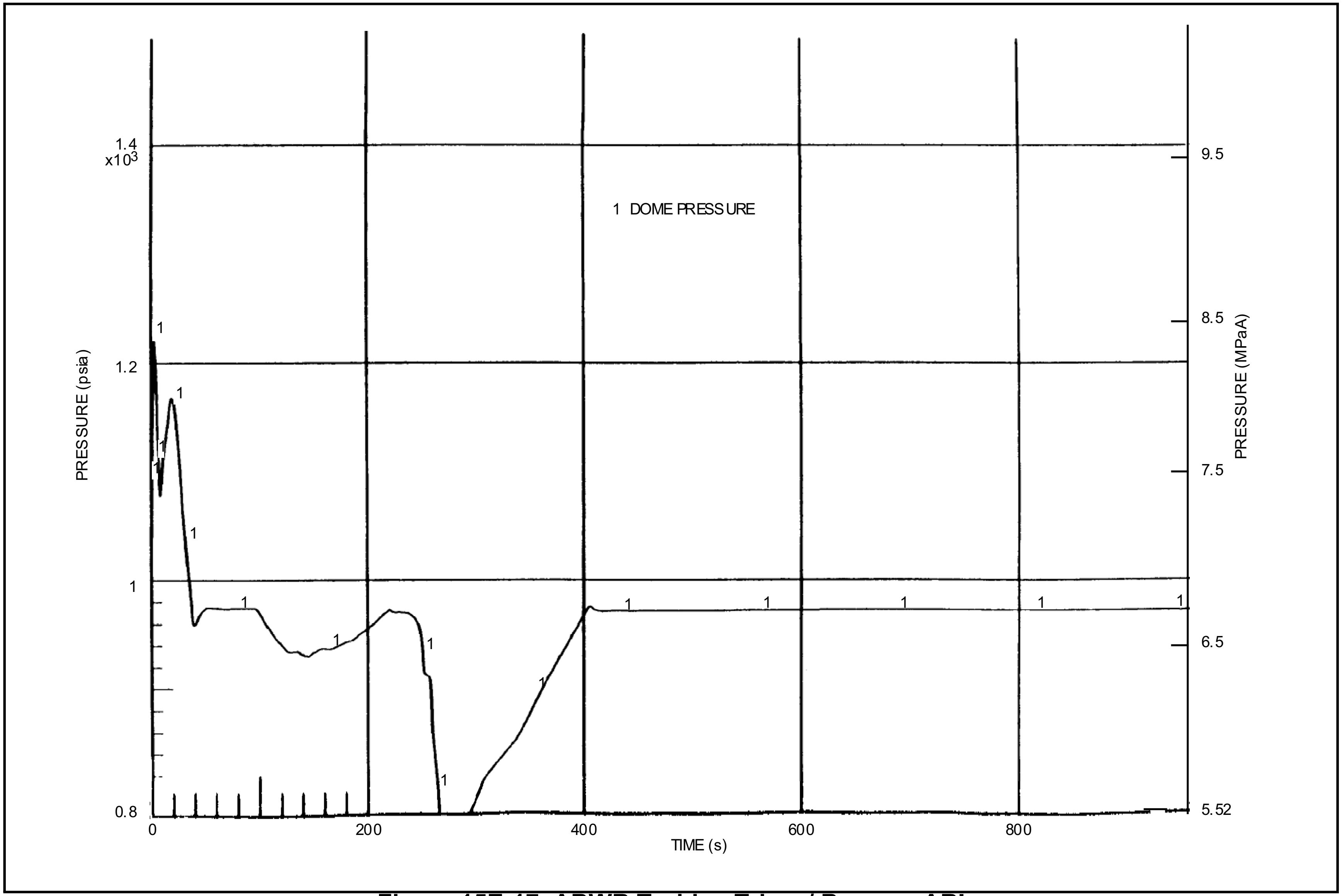


Figure 15E-17 ABWR Turbine Trip w/ Bypass, ARI

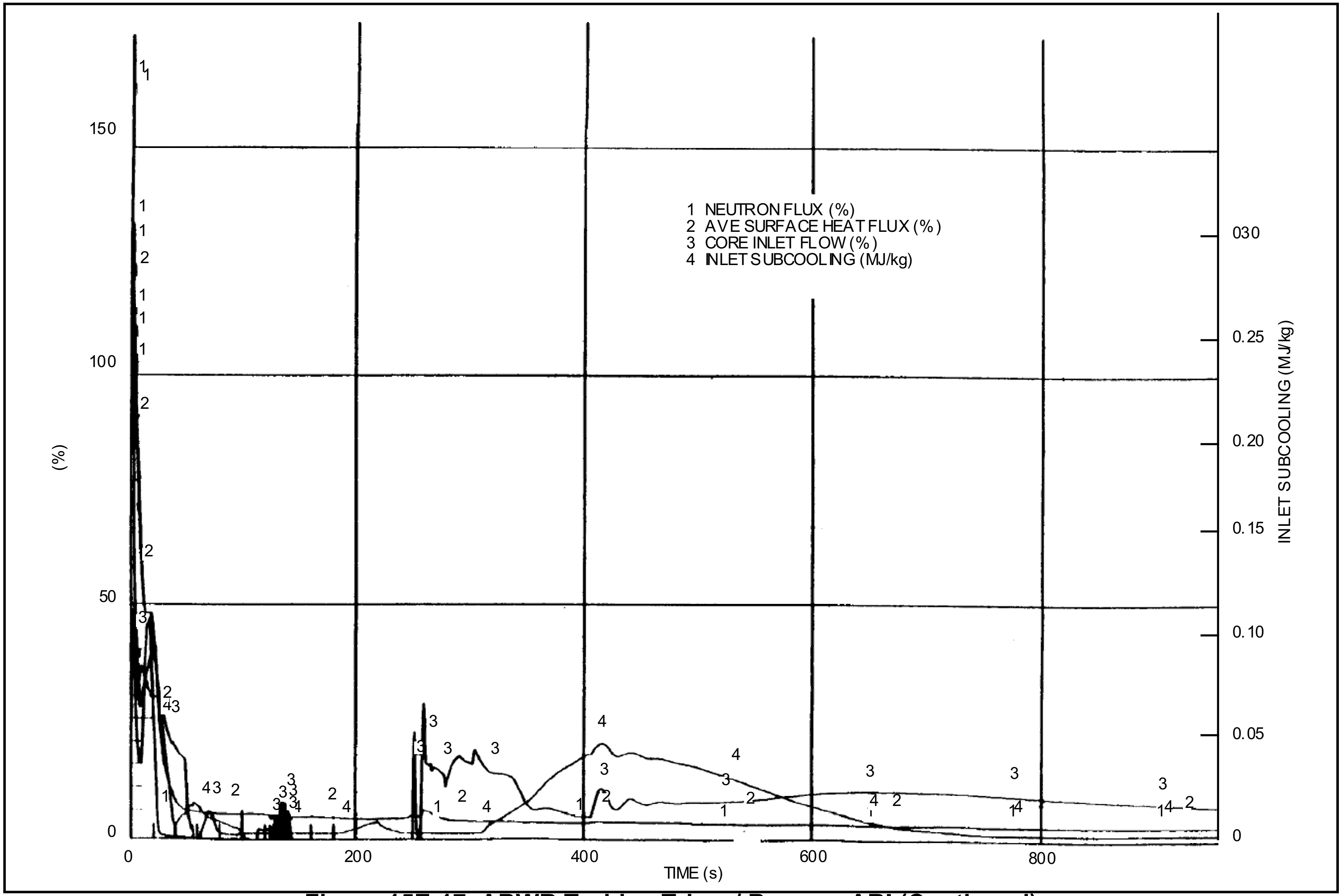


Figure 15E-17 ABWR Turbine Trip w/ Bypass, ARI (Continued)

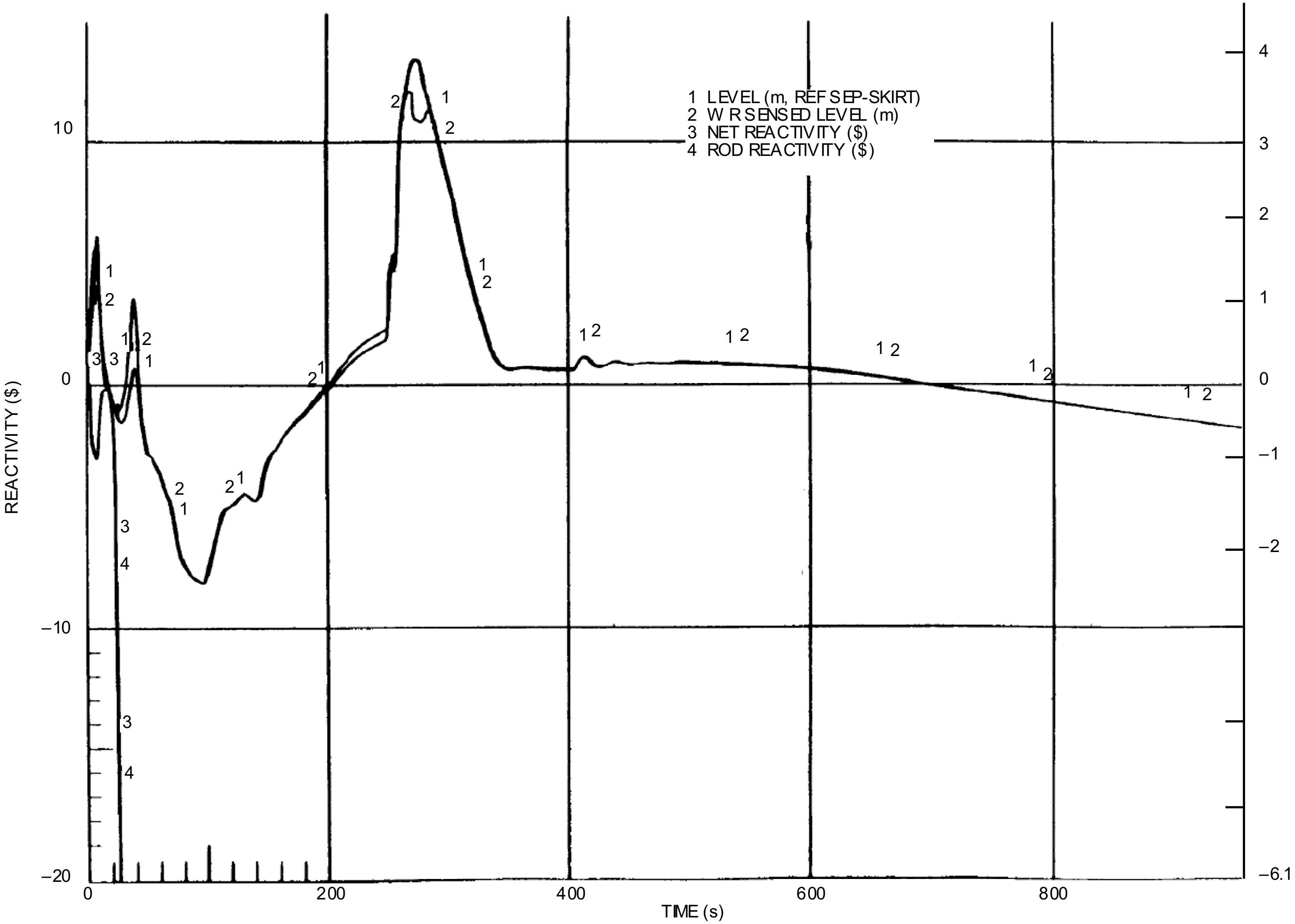


Figure 15E-17 ABWR Turbine Trip w/ Bypass, ARI (Continued)

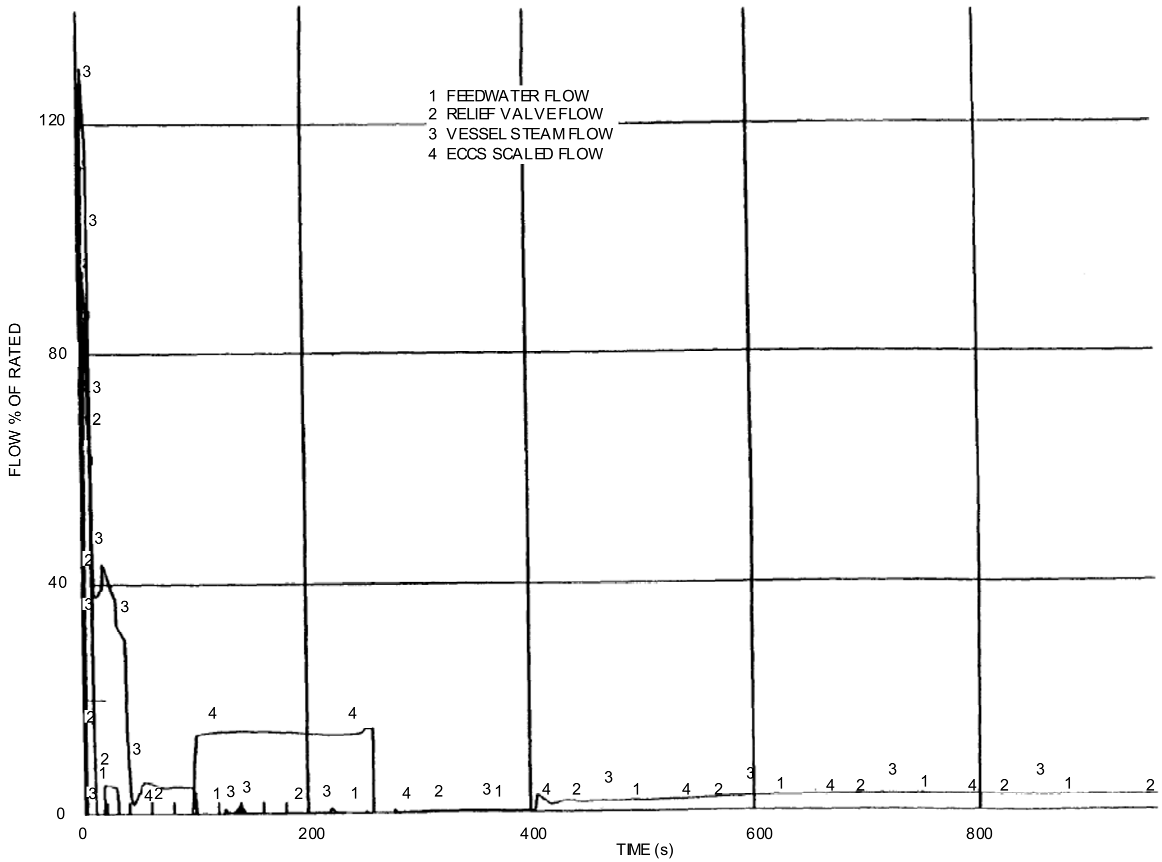


Figure 15E-17 ABWR Turbine Trip w/ Bypass, ARI (Continued)

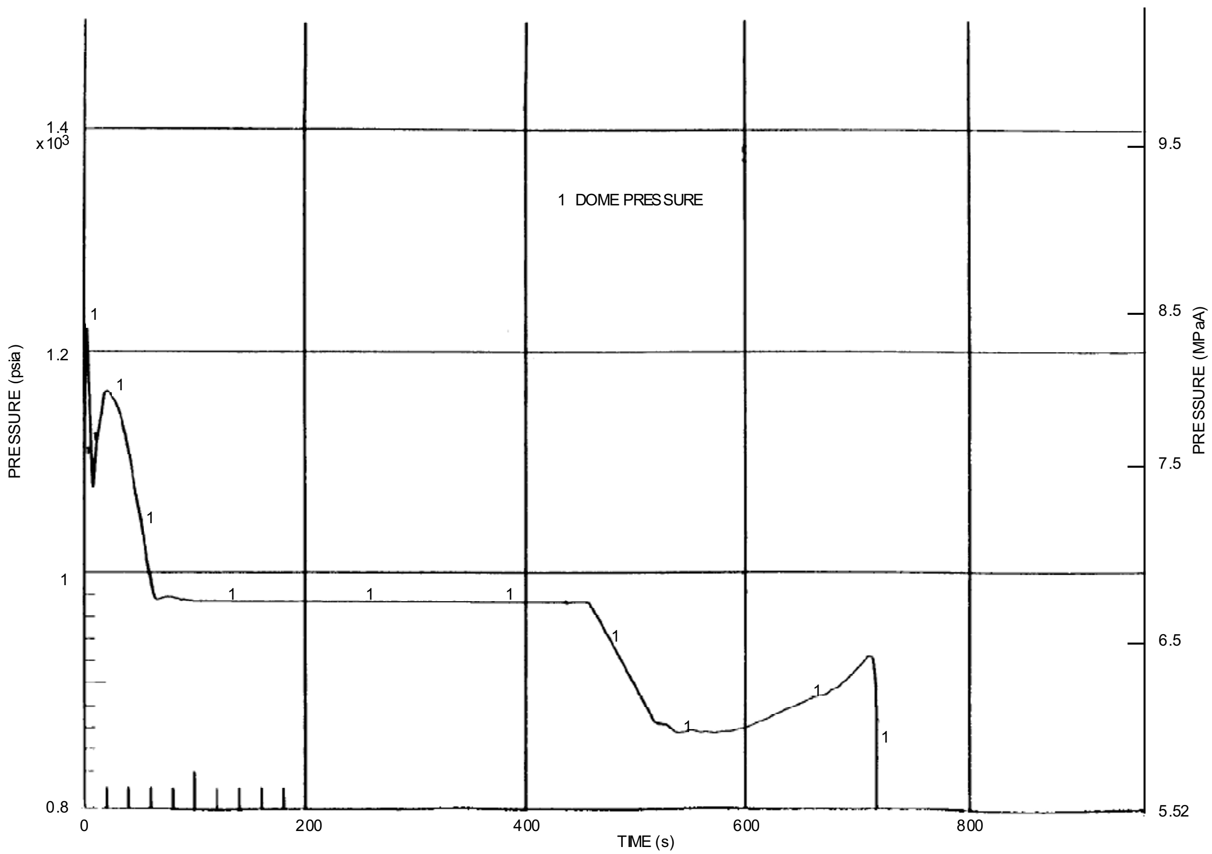


Figure 15E-18 ABWR Turbine Trip w/ Bypass, FMCRD Run–in

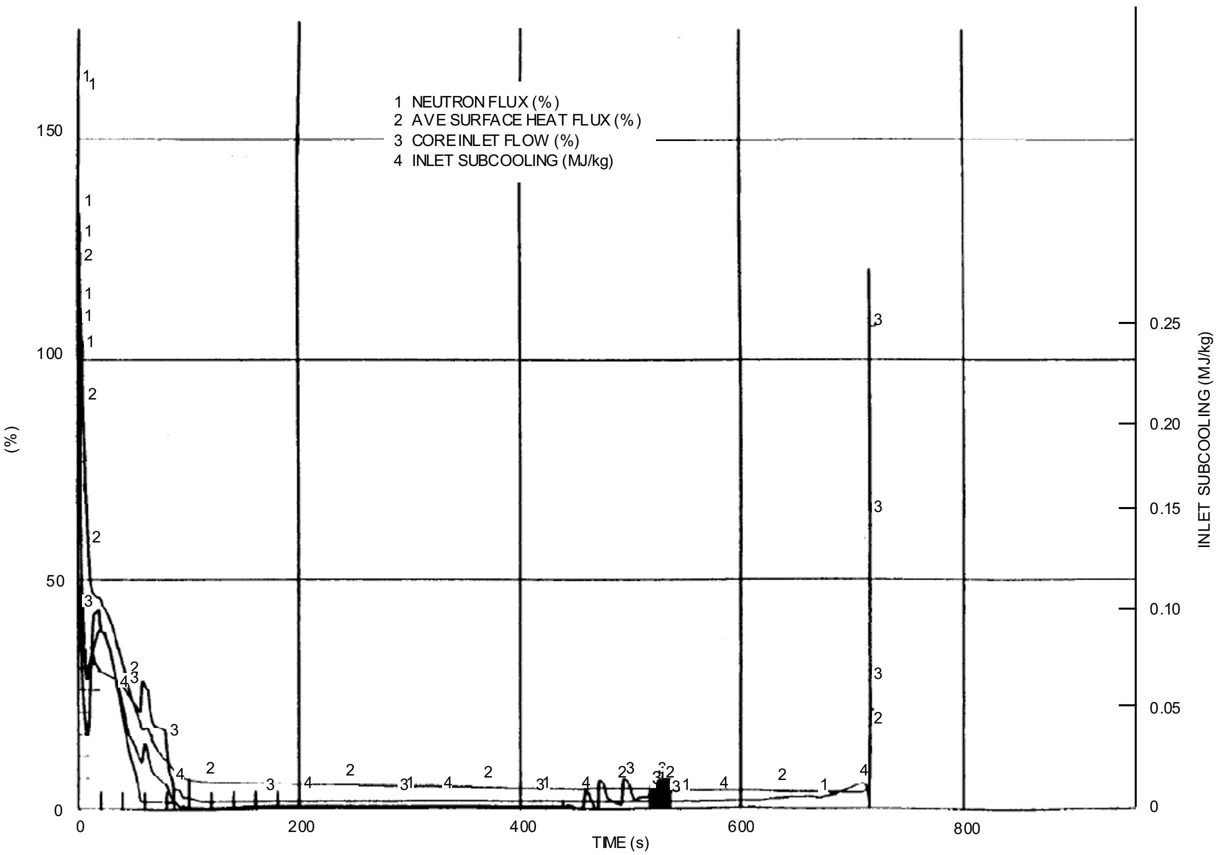


Figure 15E-18 ABWR Turbine Trip w/ Bypass, FMCRD Run–in (Continued)

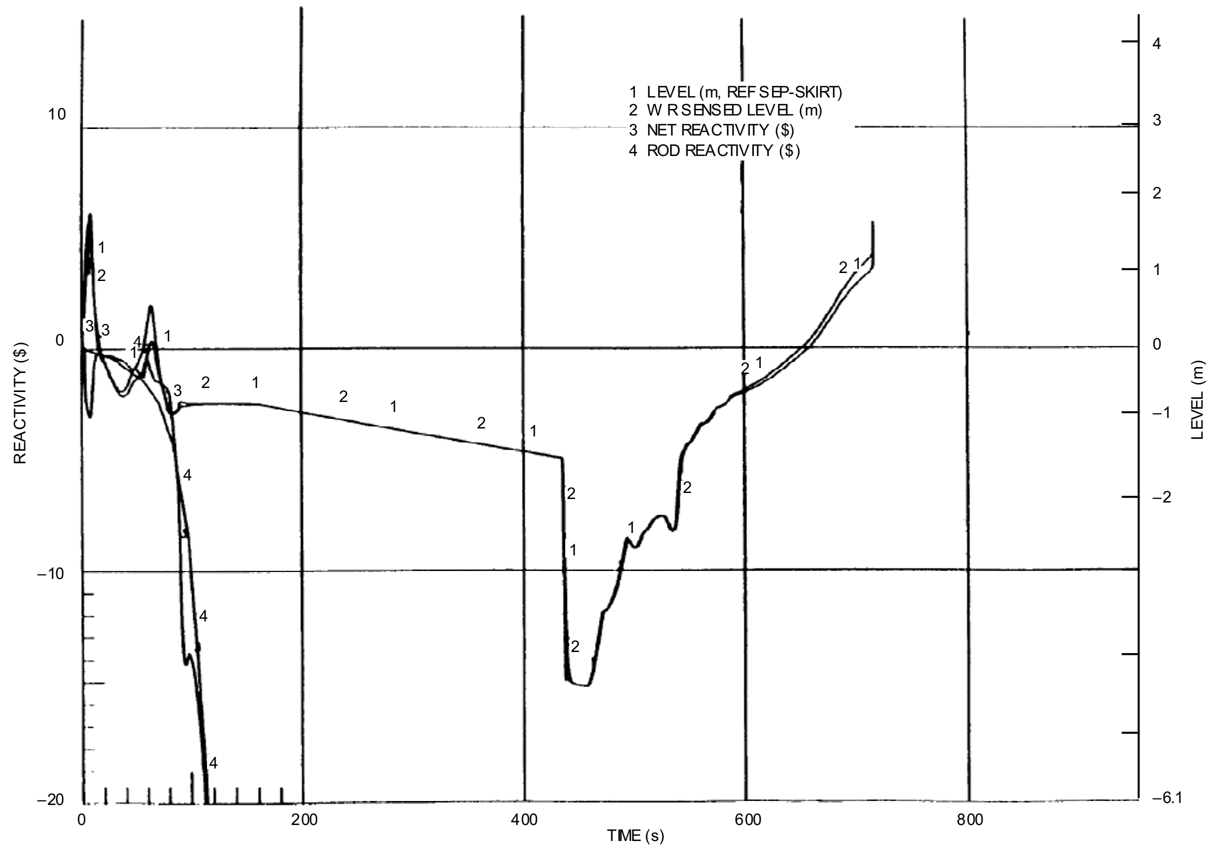


Figure 15E-18 ABWR Turbine Trip w/ Bypass, FMCRD Run–in (Continued)

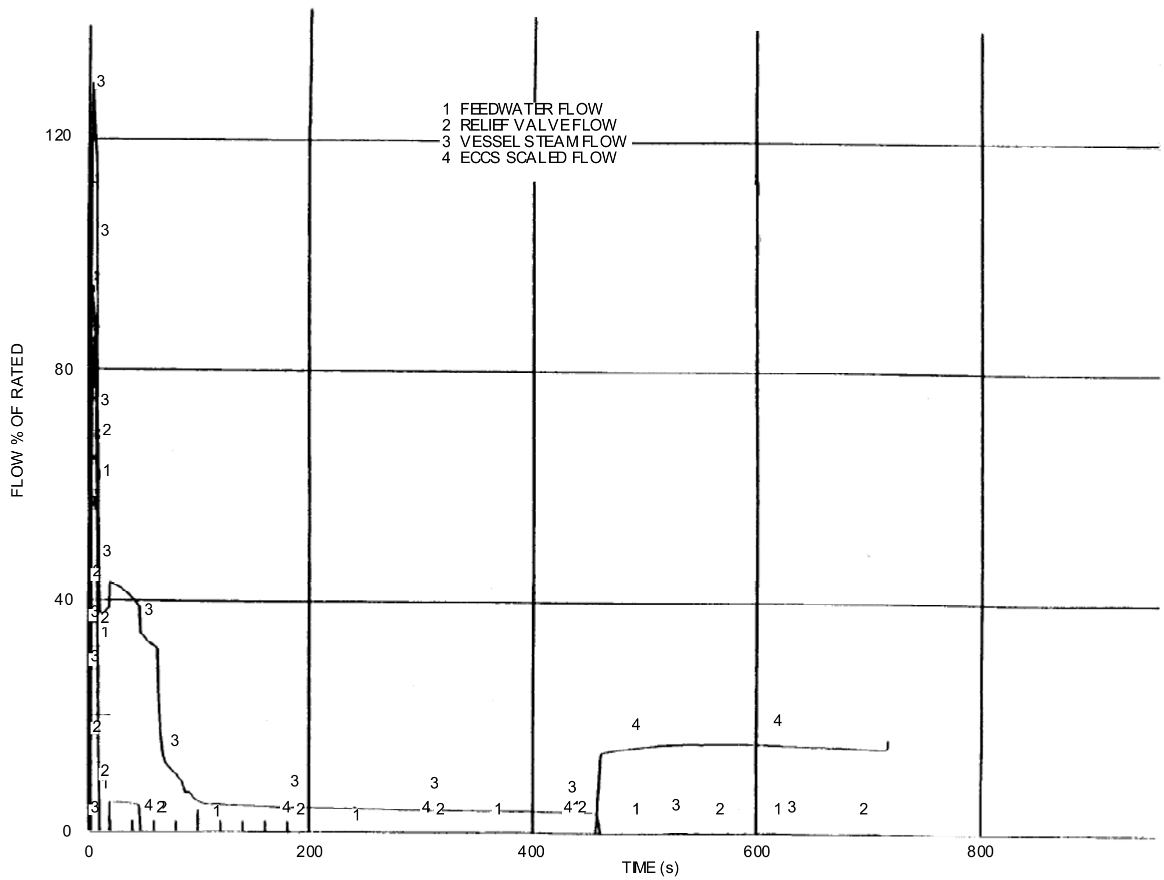


Figure 15E-18 ABWR Turbine Trip w/ Bypass, FMCRD Run–in (Continued)

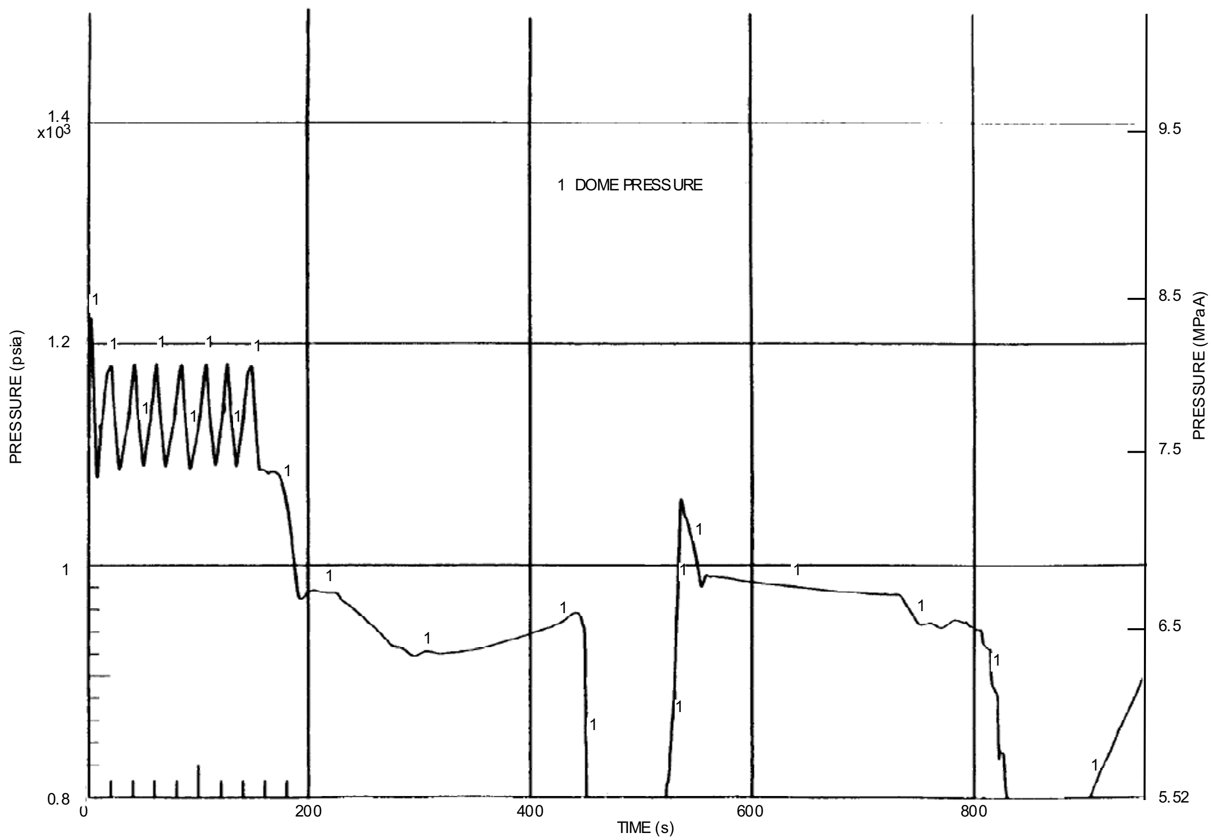


Figure 15E-19 ABWR Turbine Trip w/ Bypass, SLCS

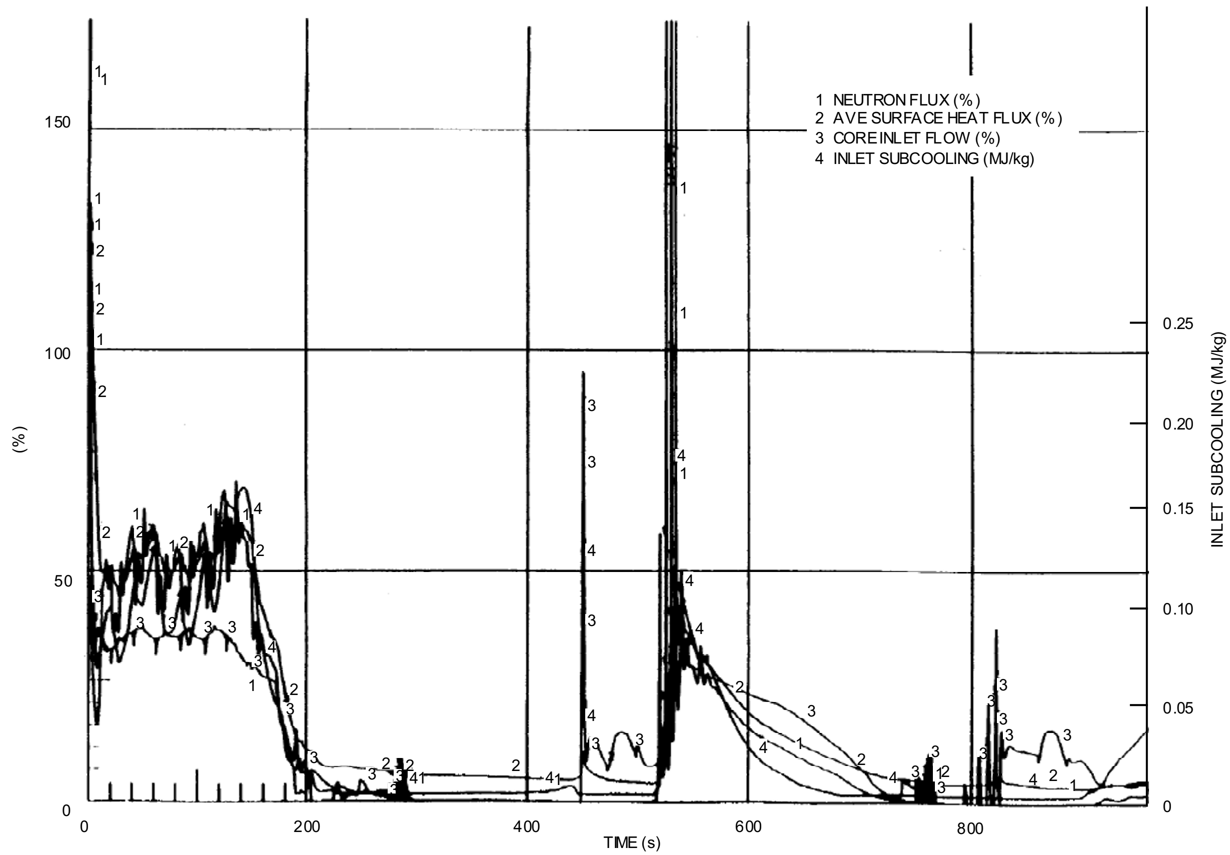


Figure 15E-19 ABWR Turbine Trip w/ Bypass, SLCS (Continued)

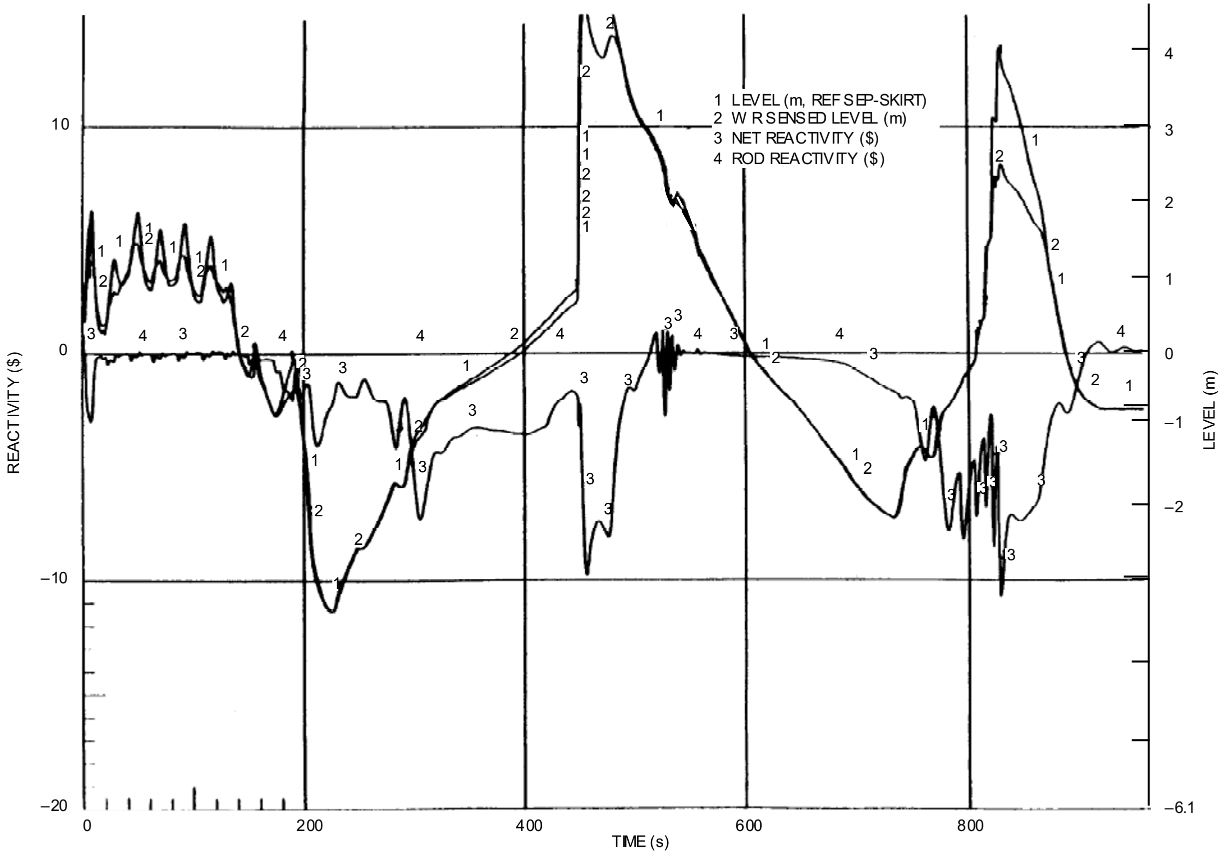


Figure 15E-19 ABWR Turbine Trip w/ Bypass, SLCS (Continued)

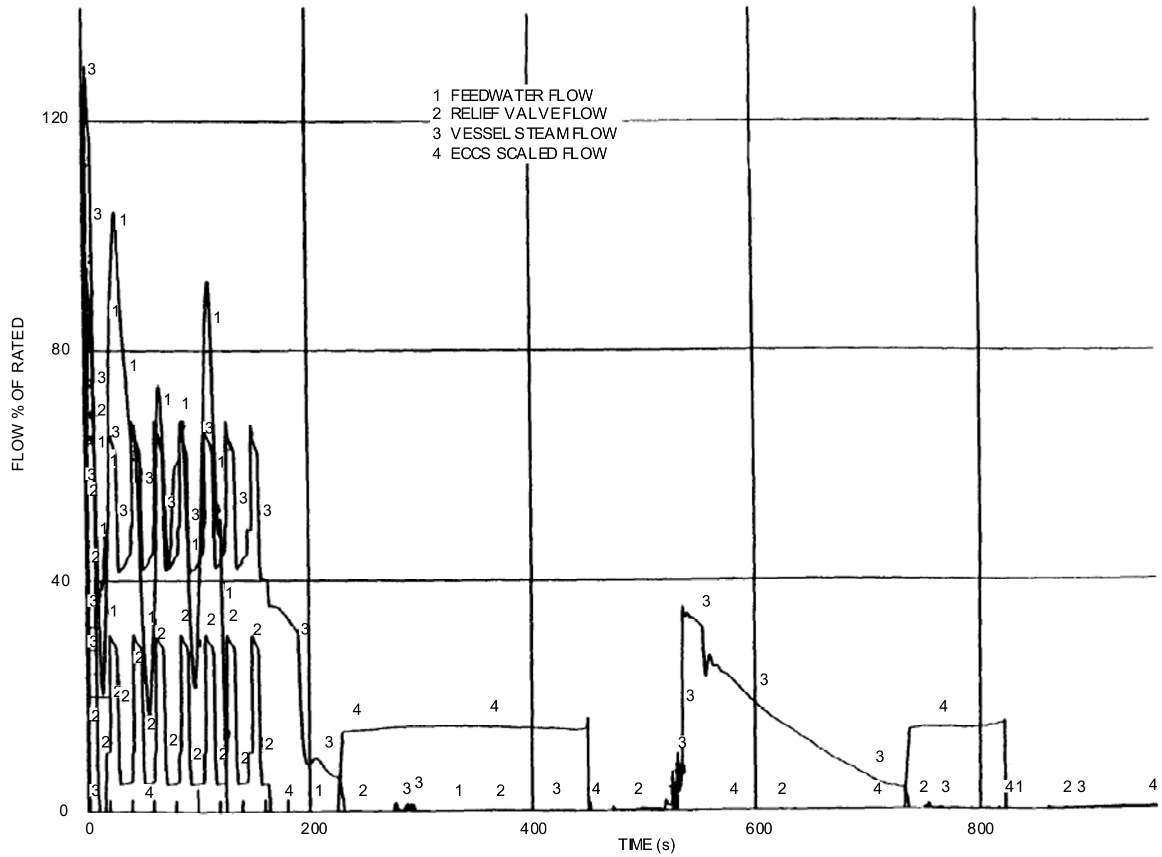


Figure 15E-19 ABWR Turbine Trip w/ Bypass, SLCS (Continued)

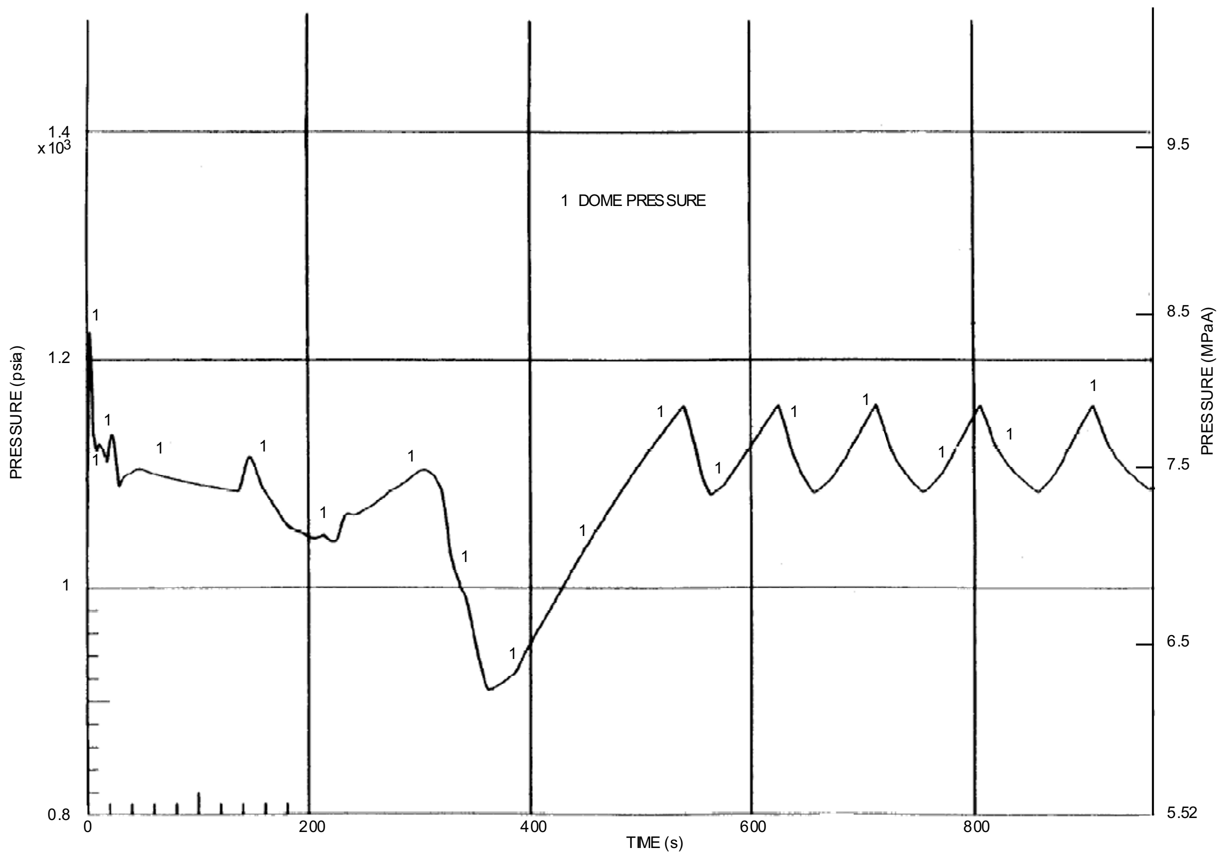


Figure 15E-20 ABWR Loss of Condenser Vacuum, ARI

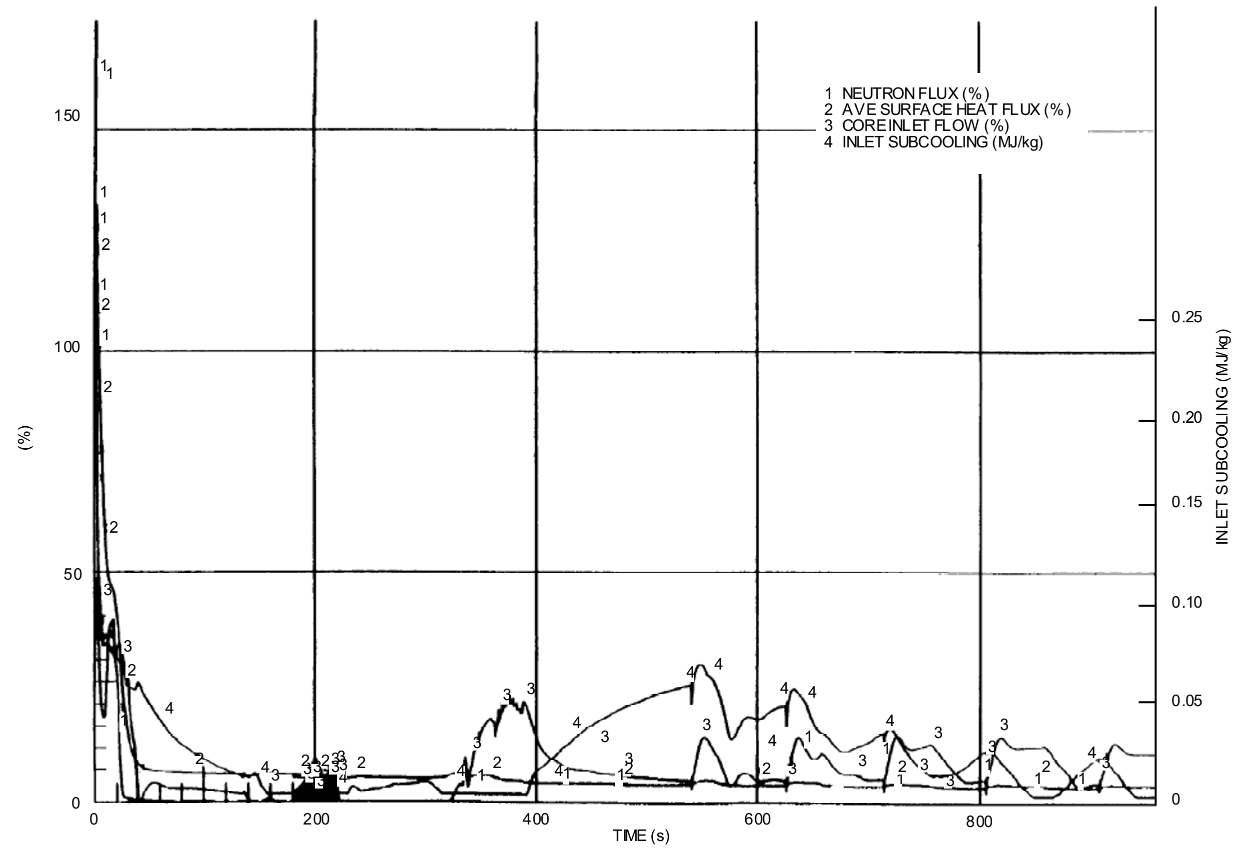


Figure 15E-20 ABWR Loss of Condenser Vacuum, ARI (Continued)

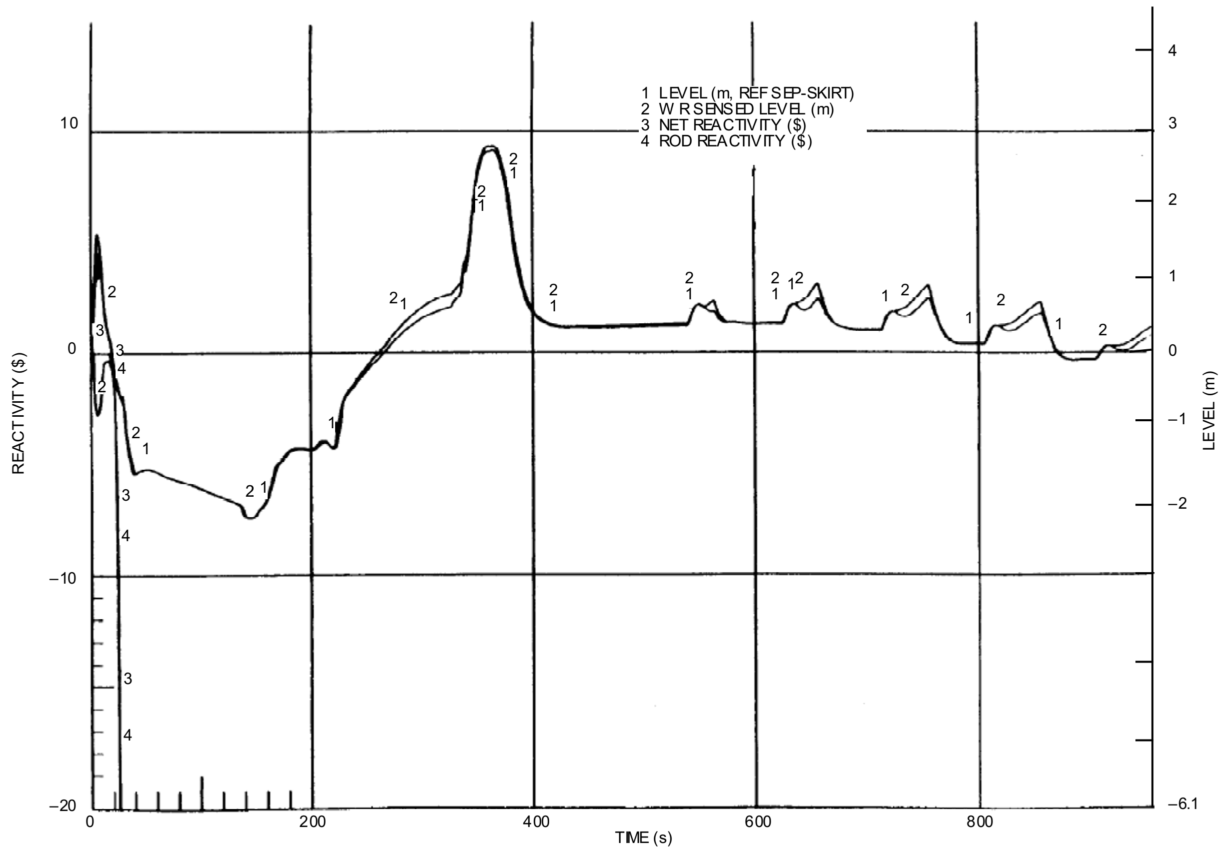


Figure 15E-20 ABWR Loss of Condenser Vacuum, ARI (Continued)

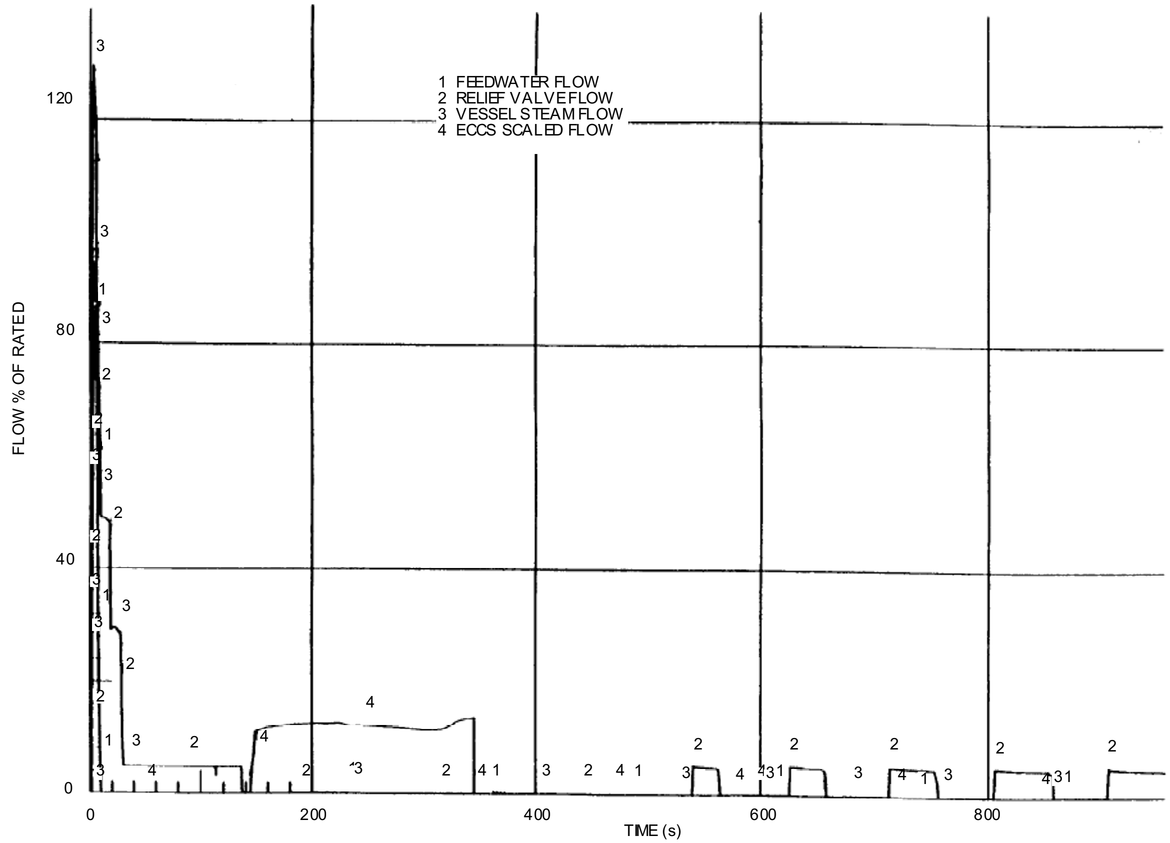


Figure 15E-20 ABWR Loss of Condenser Vacuum, ARI (Continued)

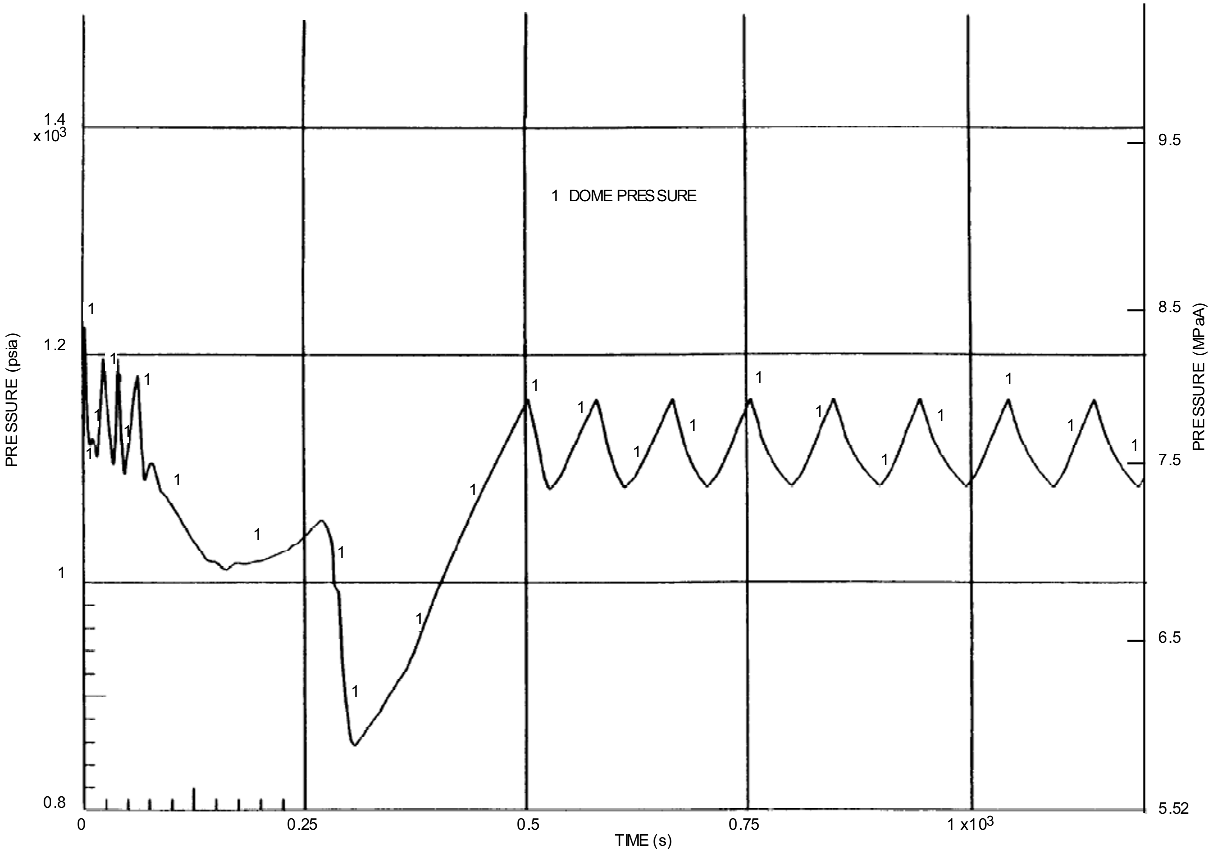


Figure 15E-21 ABWR Loss of Condenser Vacuum, FMCRD Run–in

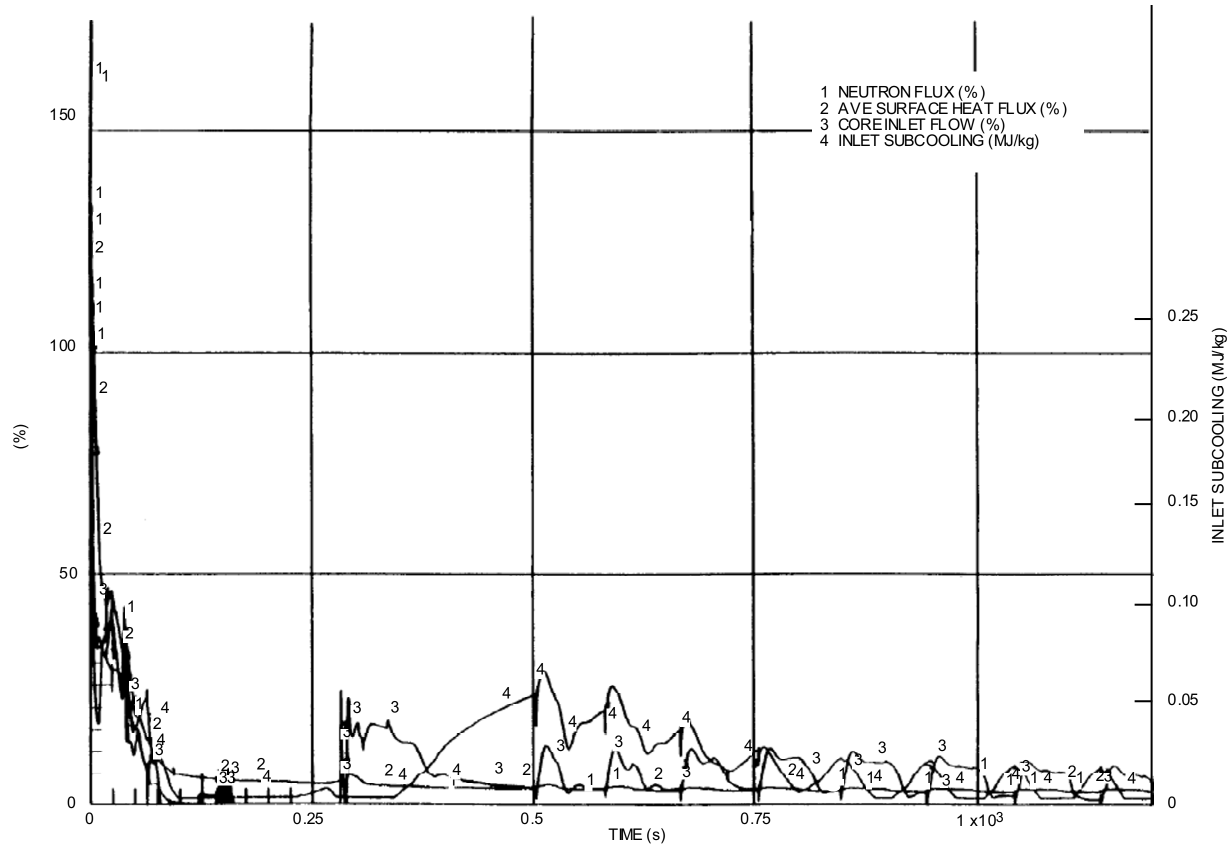


Figure 15E-21 ABWR Loss of Condenser Vacuum, FMCRD Run–in (Continued)

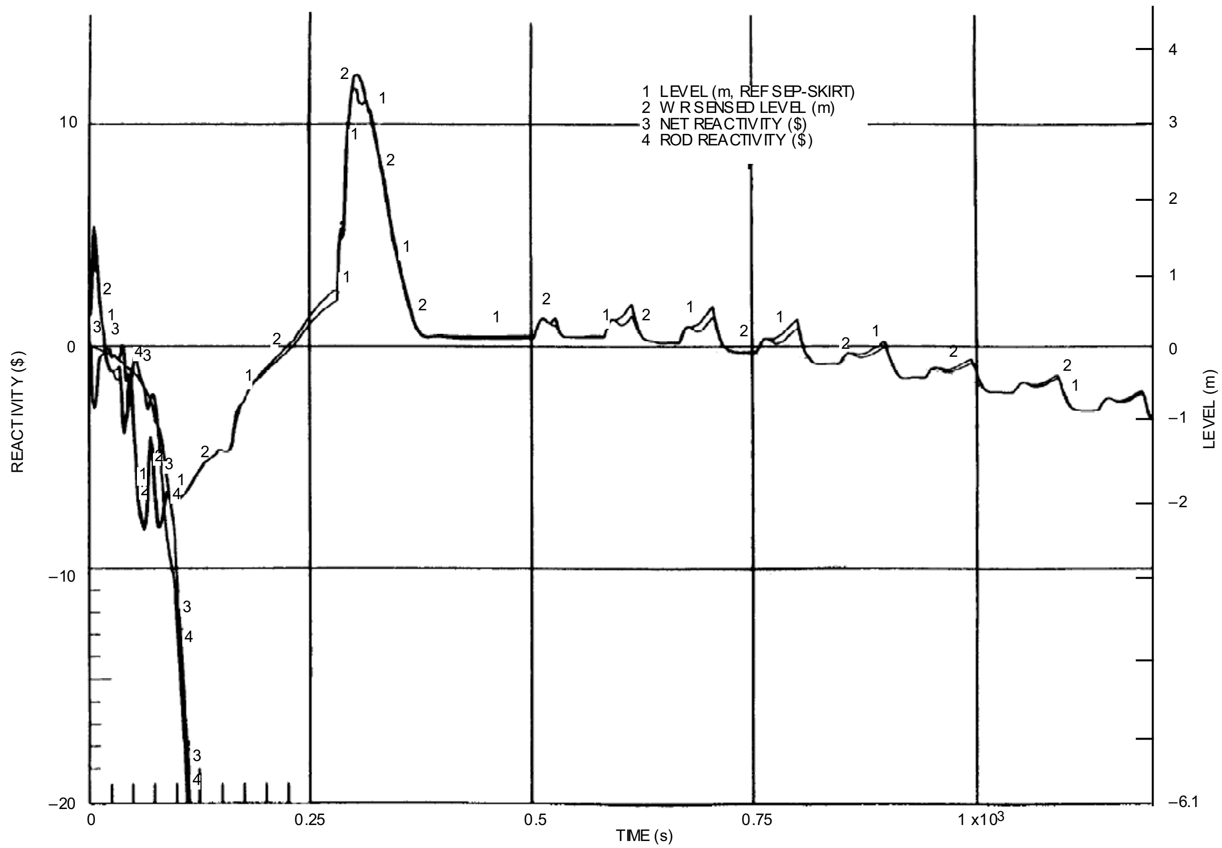


Figure 15E-21 ABWR Loss of Condenser Vacuum, FMCRD Run–in (Continued)

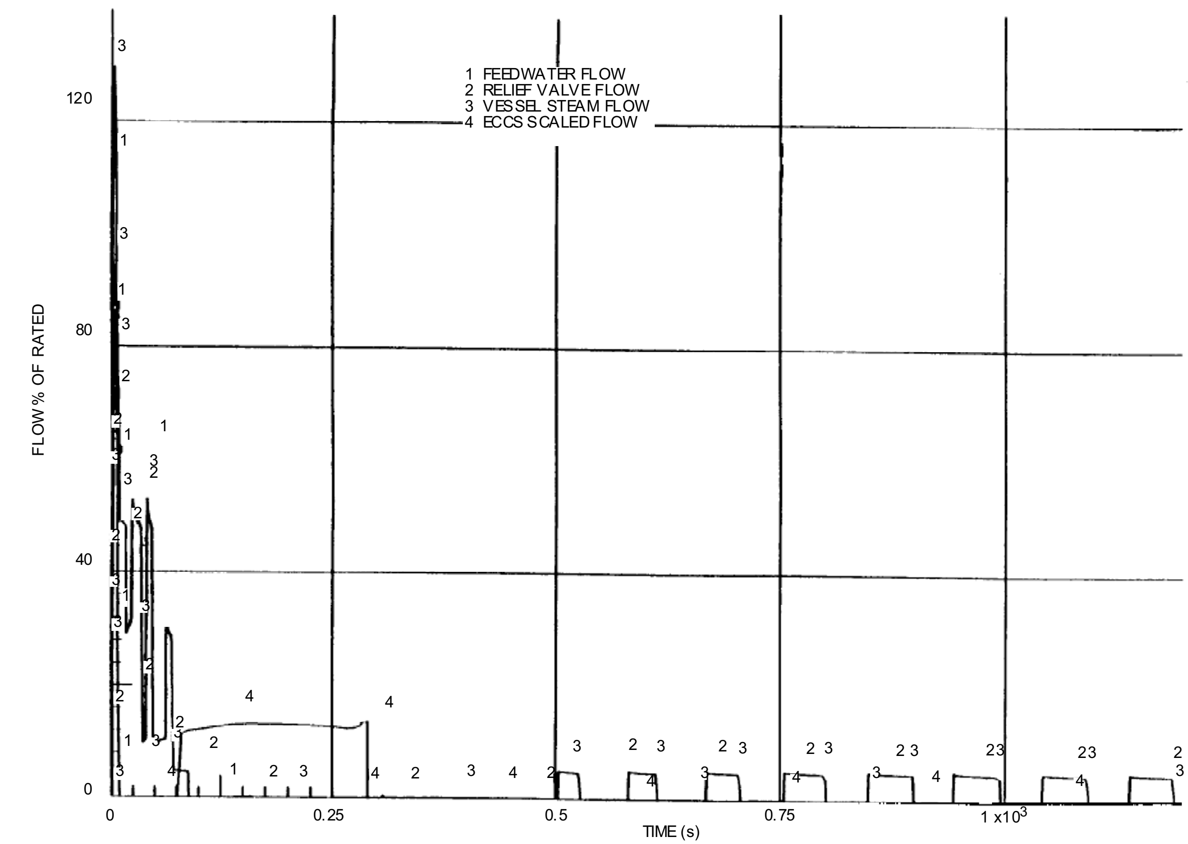


Figure 15E-21 ABWR Loss of Condenser Vacuum, FMCRD Run–in (Continued)

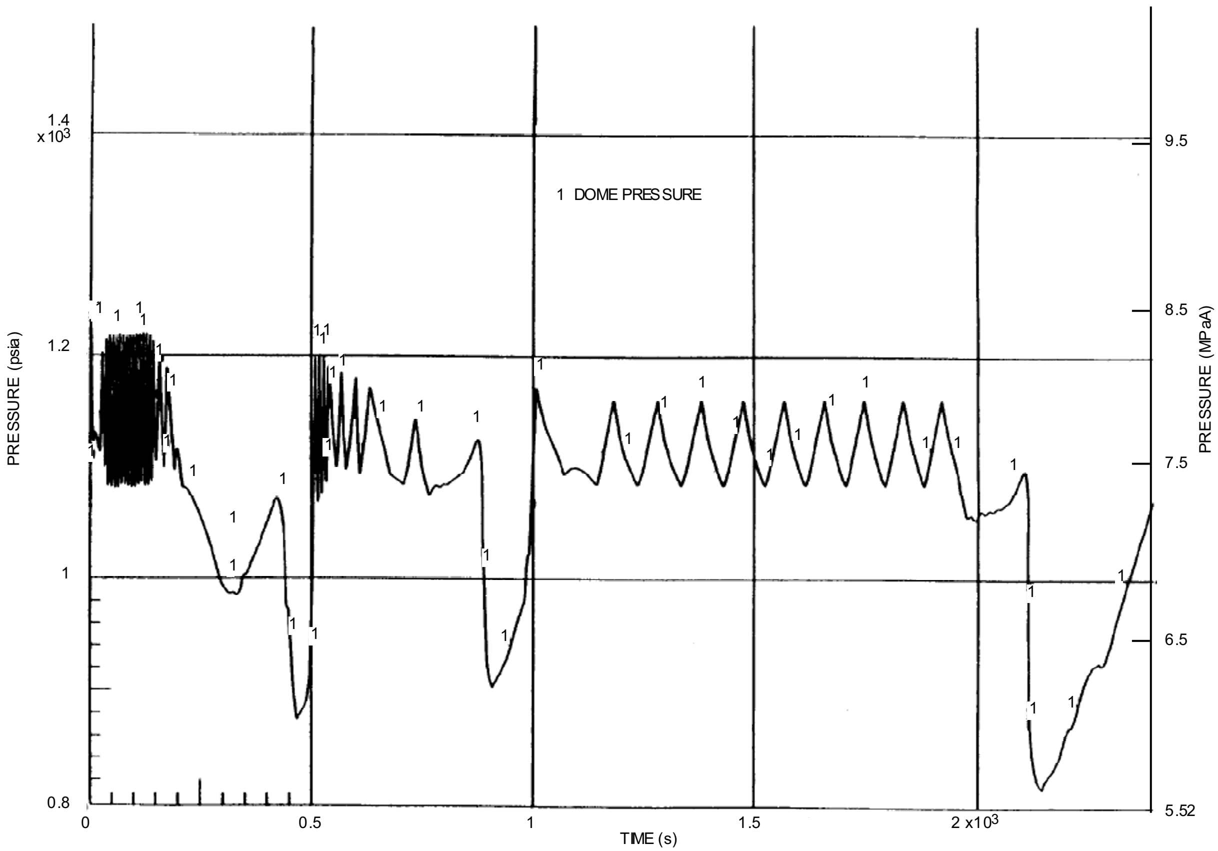


Figure 15E-22 ABWR Loss of Condenser Vacuum, SLCS

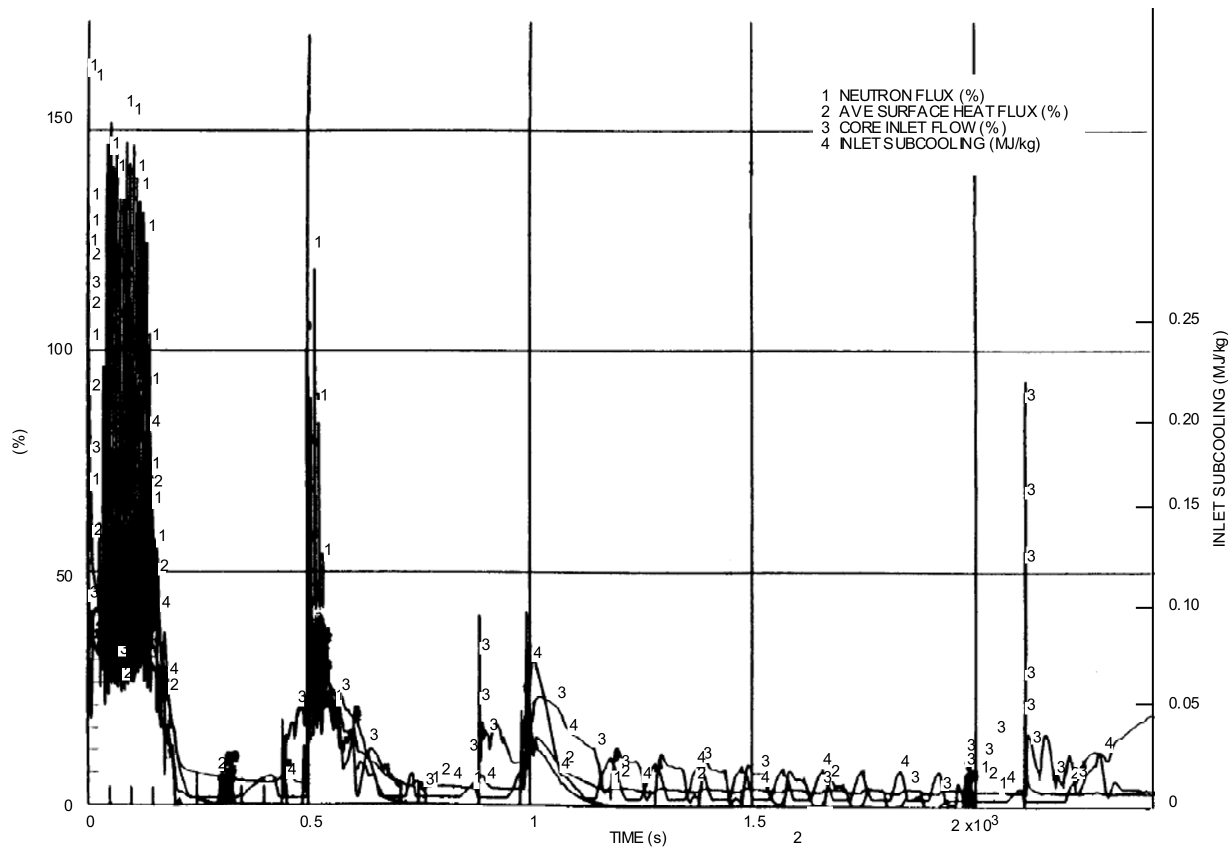


Figure 15E-22 ABWR Loss of Condenser Vacuum, SLCS (Continued)

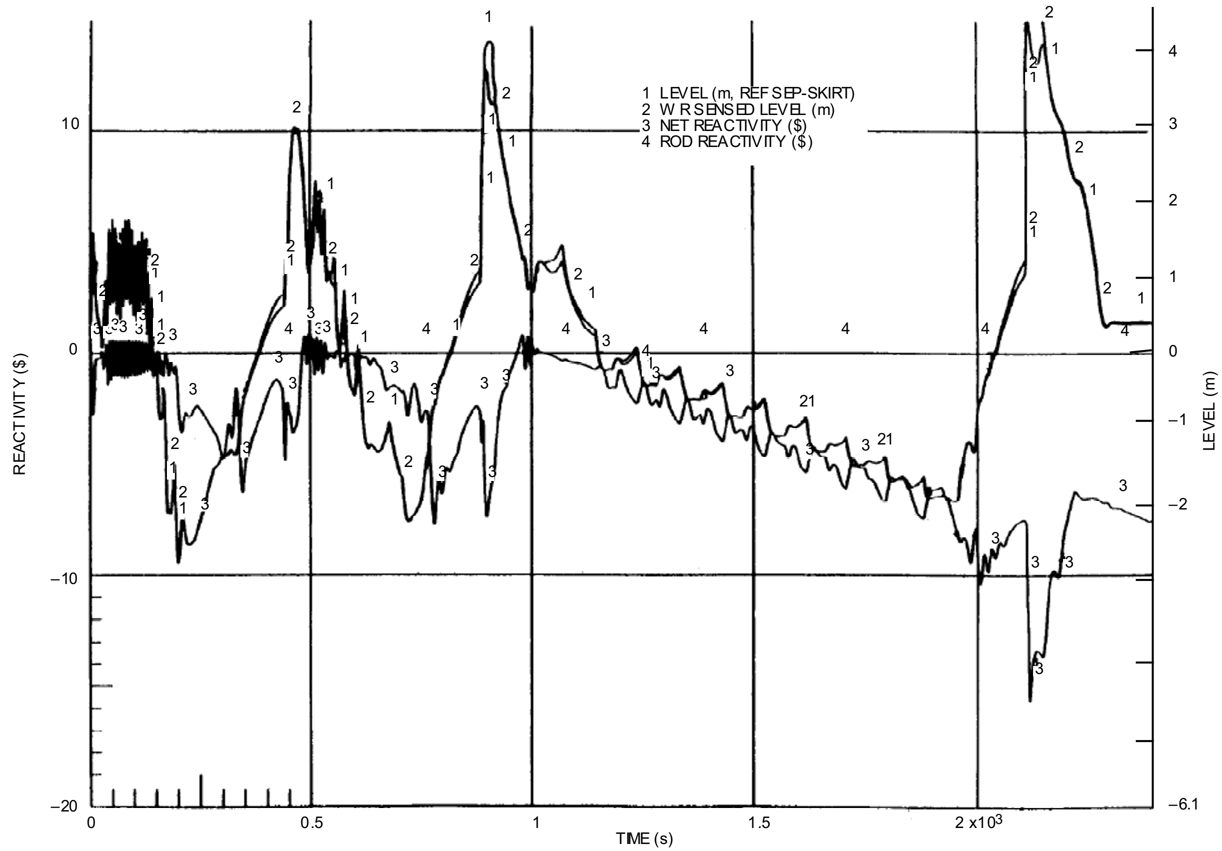


Figure 15E-22 ABWR Loss of Condenser Vacuum, SLCS (Continued)

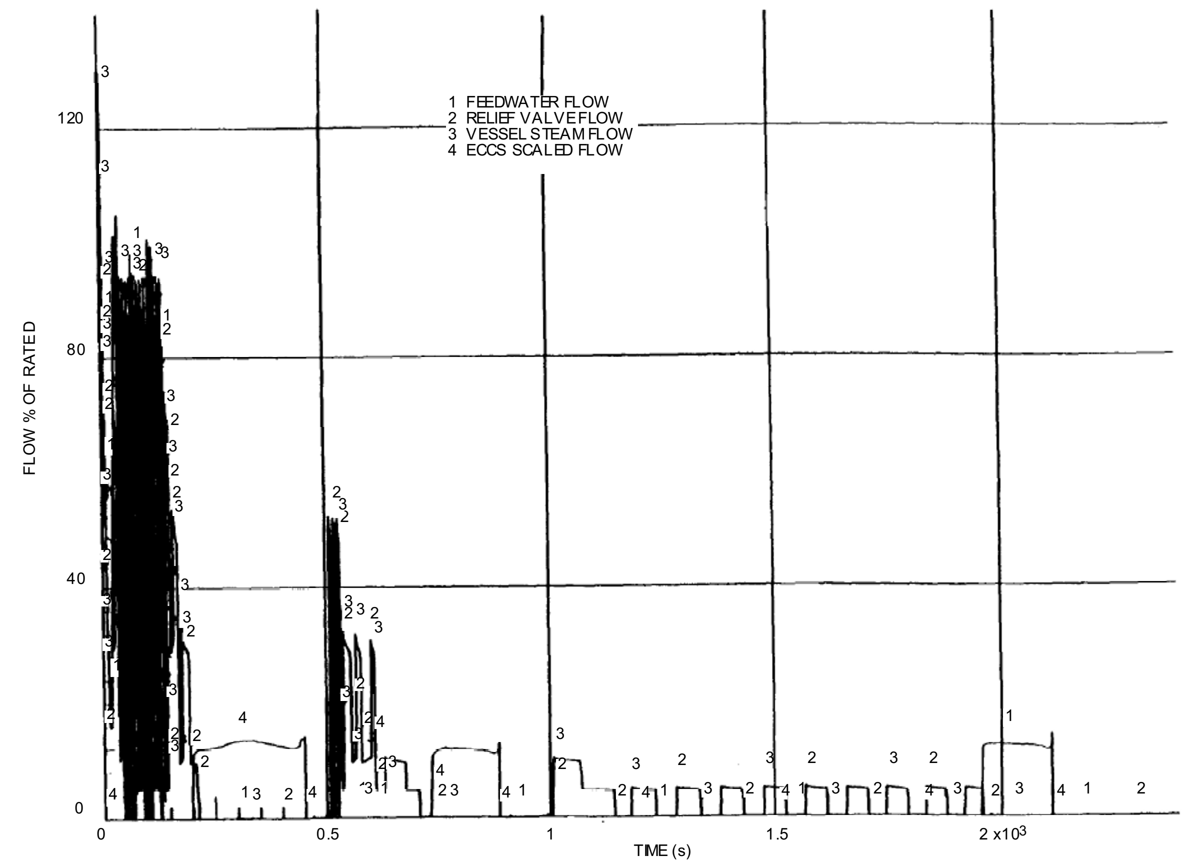


Figure 15E-22 ABWR Loss of Condenser Vacuum, SLCS (Continued)

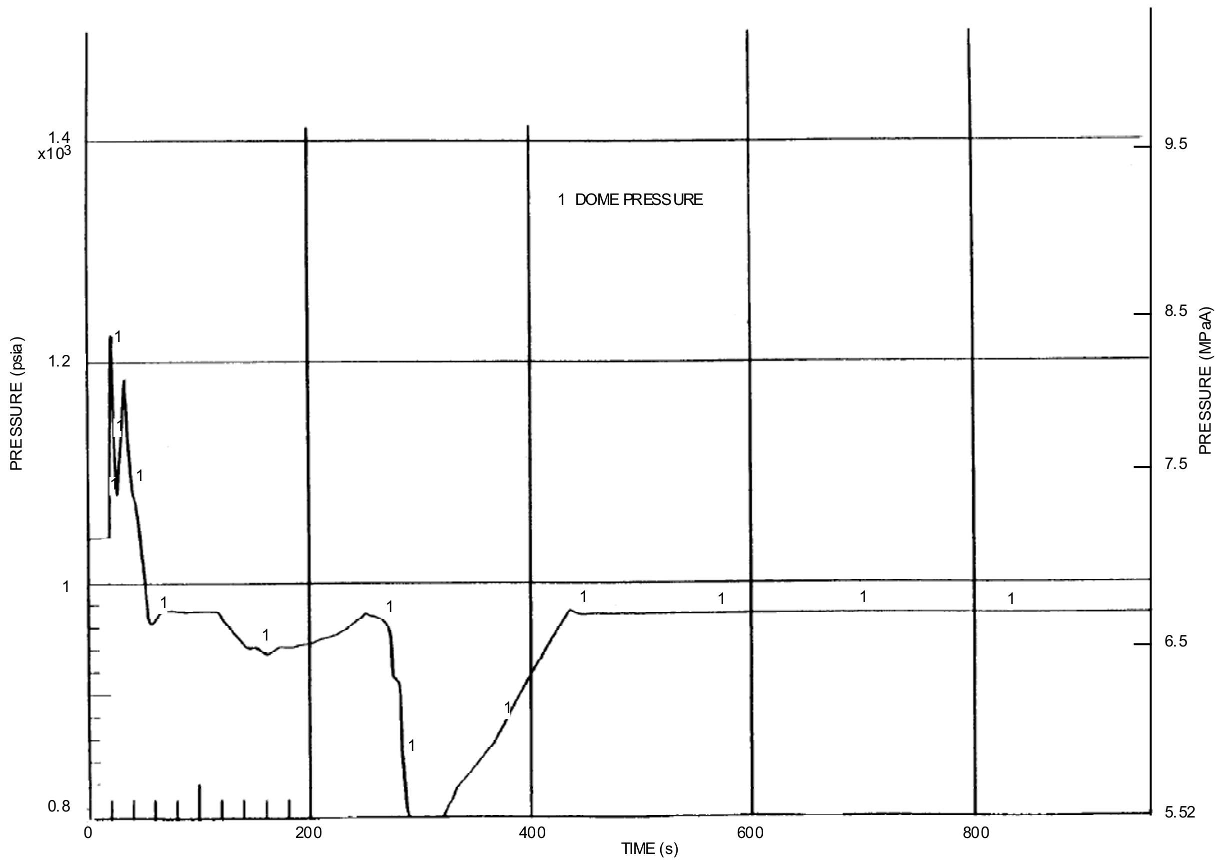


Figure 15E-23 ABWR Feedwater Controller Failure Maximum Demand, ARI

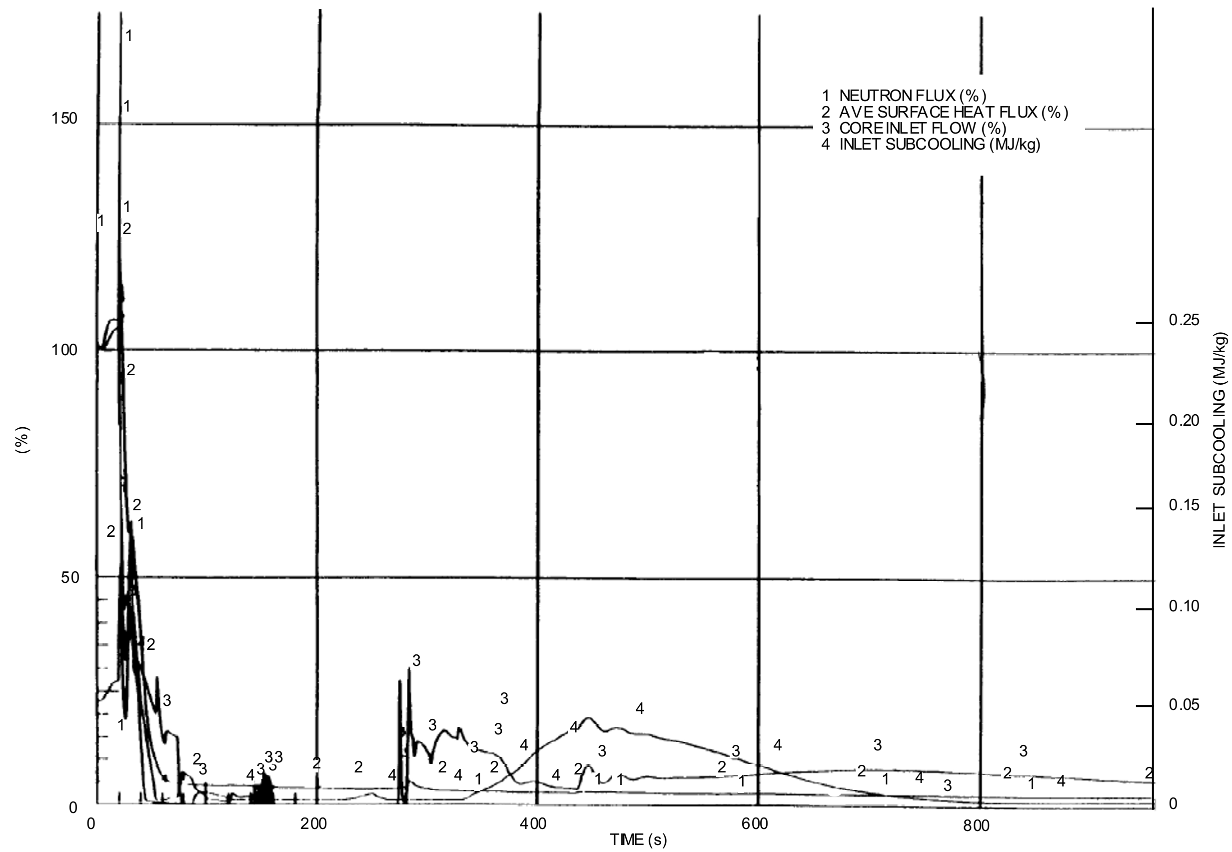


Figure 15E-23 ABWR Feedwater Controller Failure Maximum Demand, ARI (Continued)

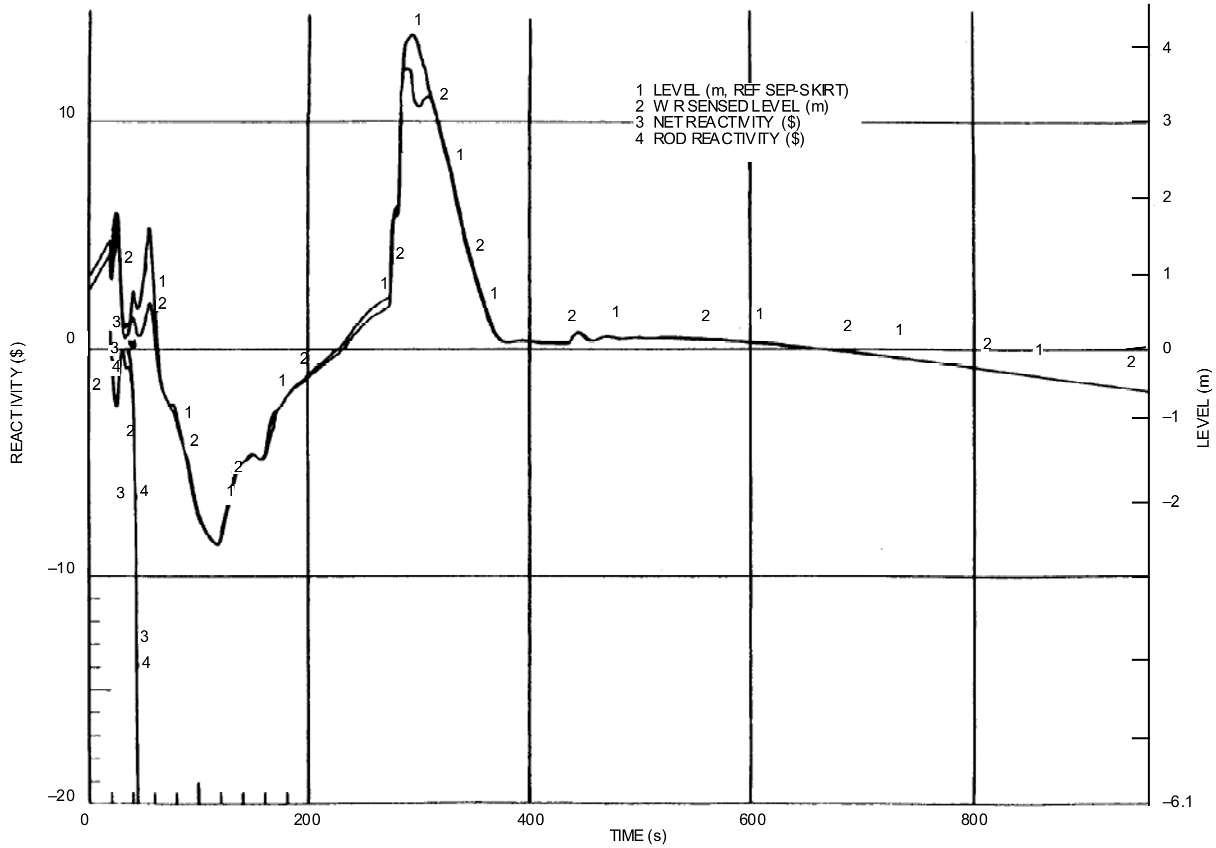


Figure 15E-23 ABWR Feedwater Controller Failure Maximum Demand, ARI (Continued)

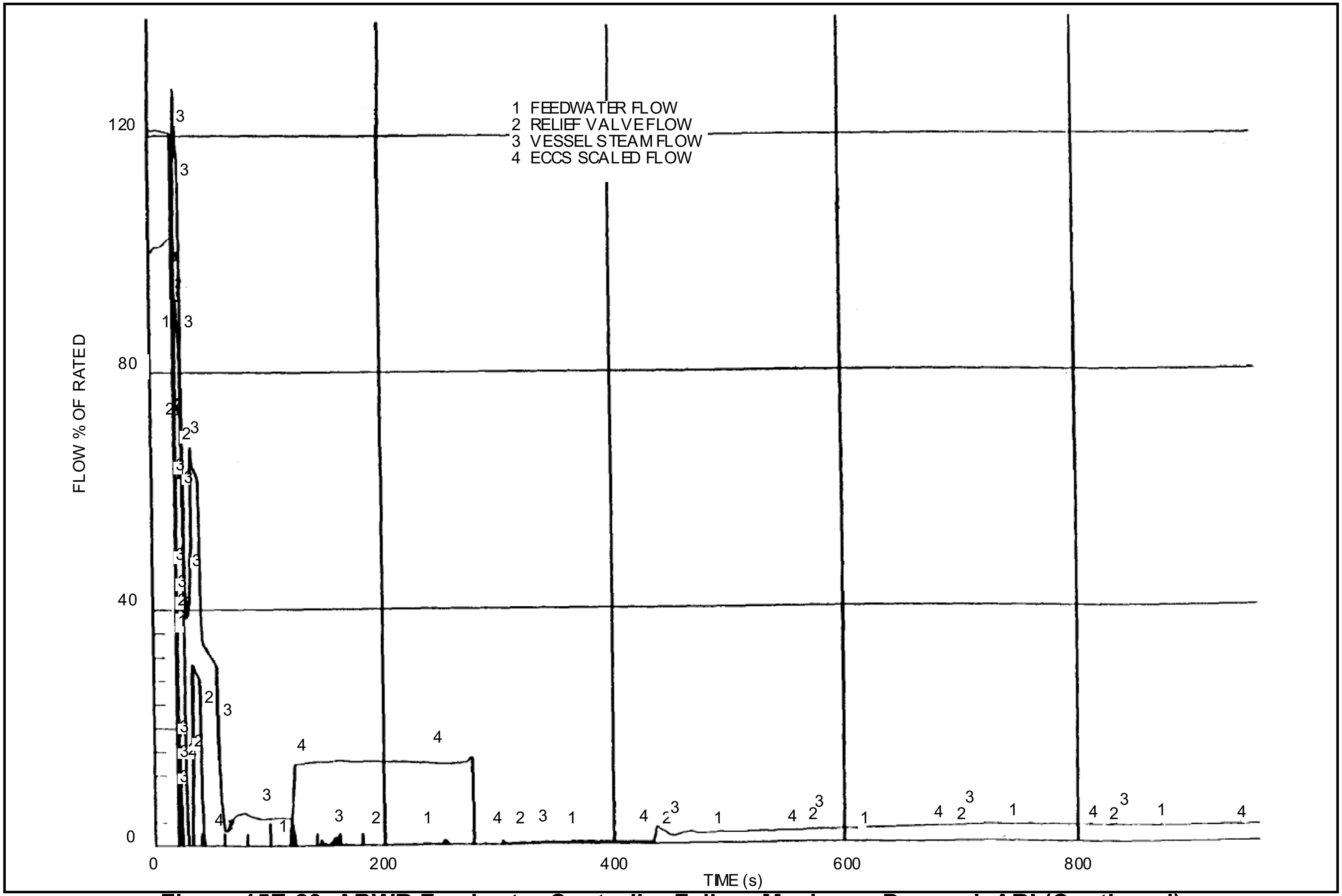


Figure 15E-23 ABWR Feedwater Controller Failure Maximum Demand, ARI (Continued)

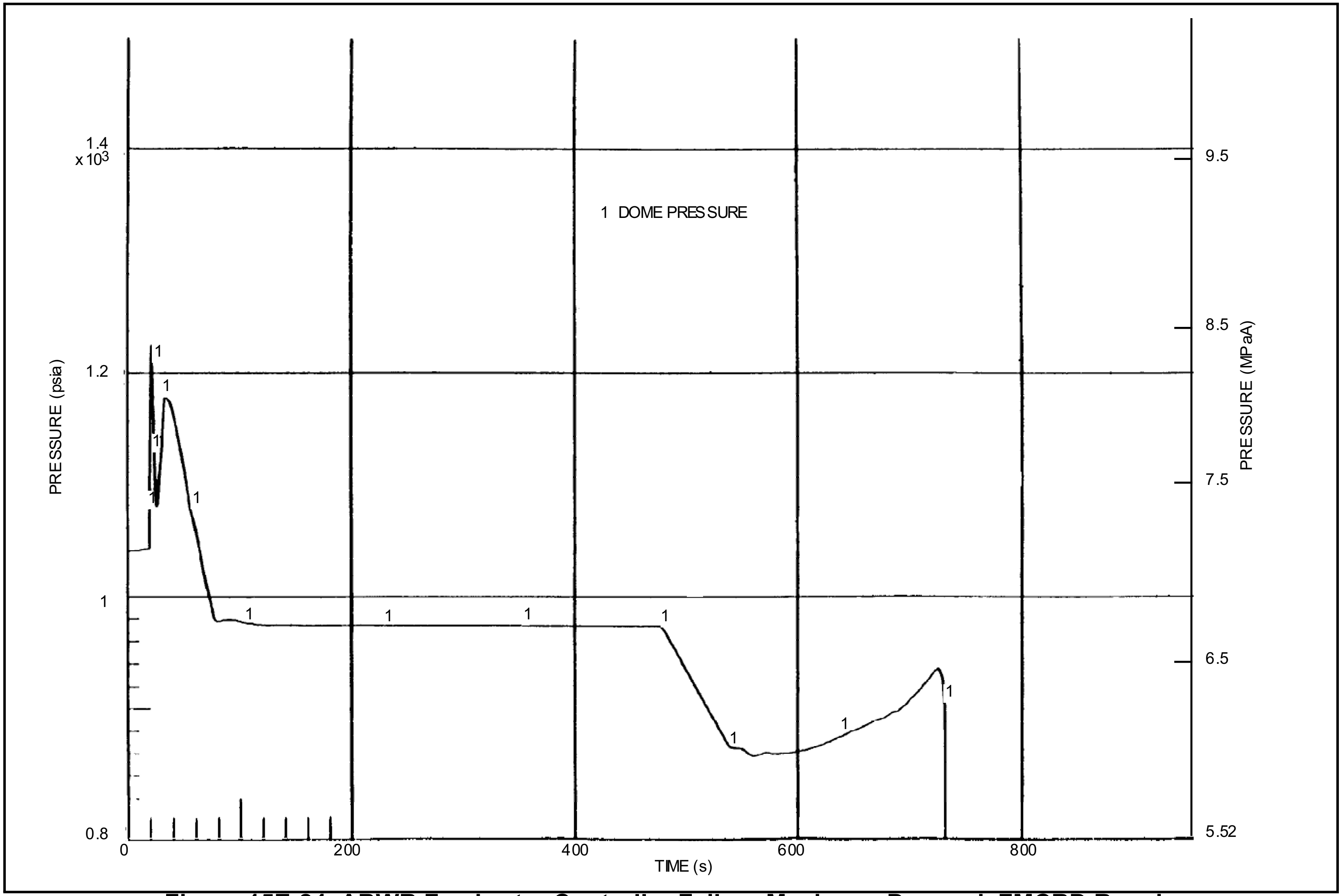


Figure 15E-24 ABWR Feedwater Controller Failure Maximum Demand, FMCRD Run–in

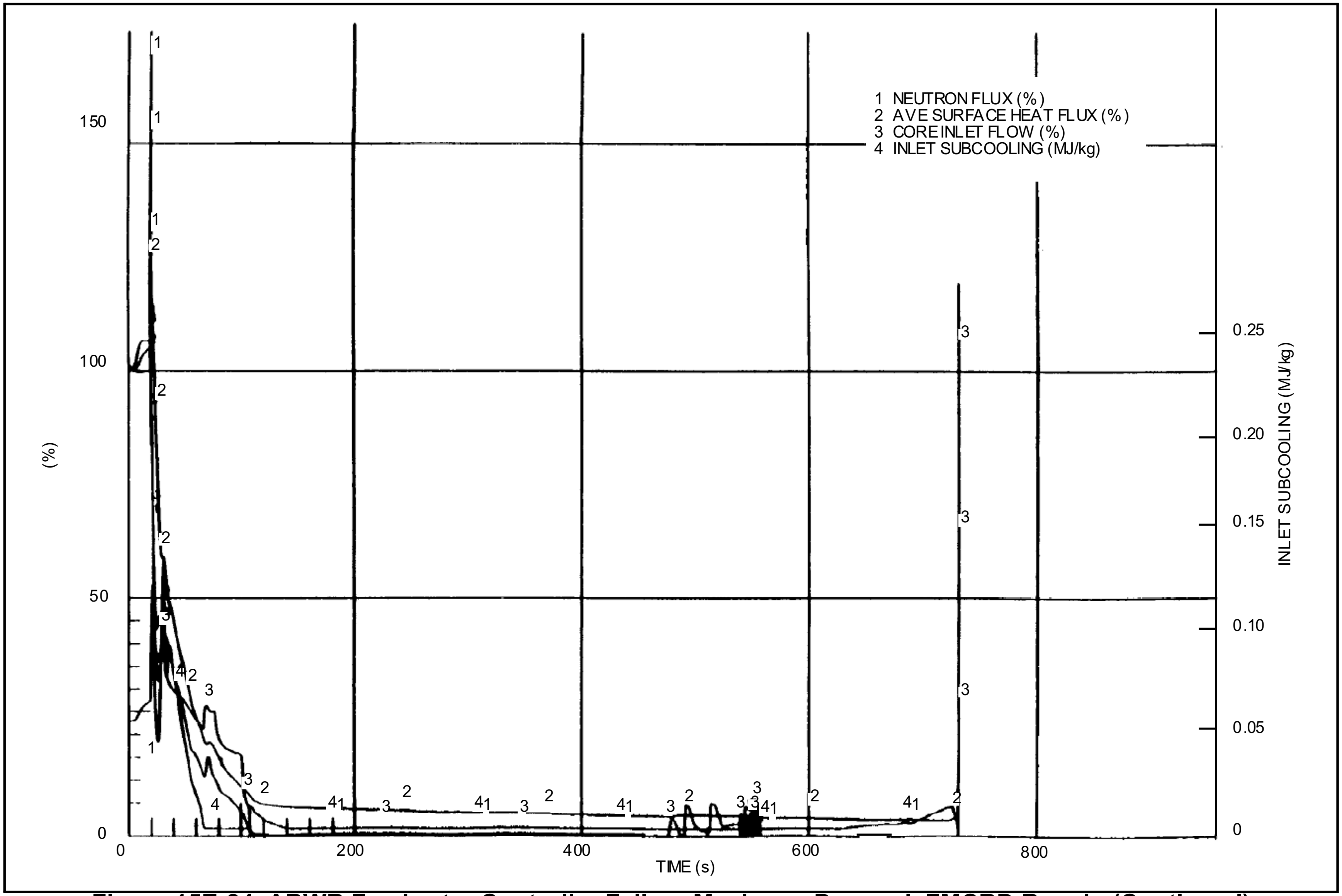


Figure 15E-24 ABWR Feedwater Controller Failure Maximum Demand, FMCRD Run–in (Continued)

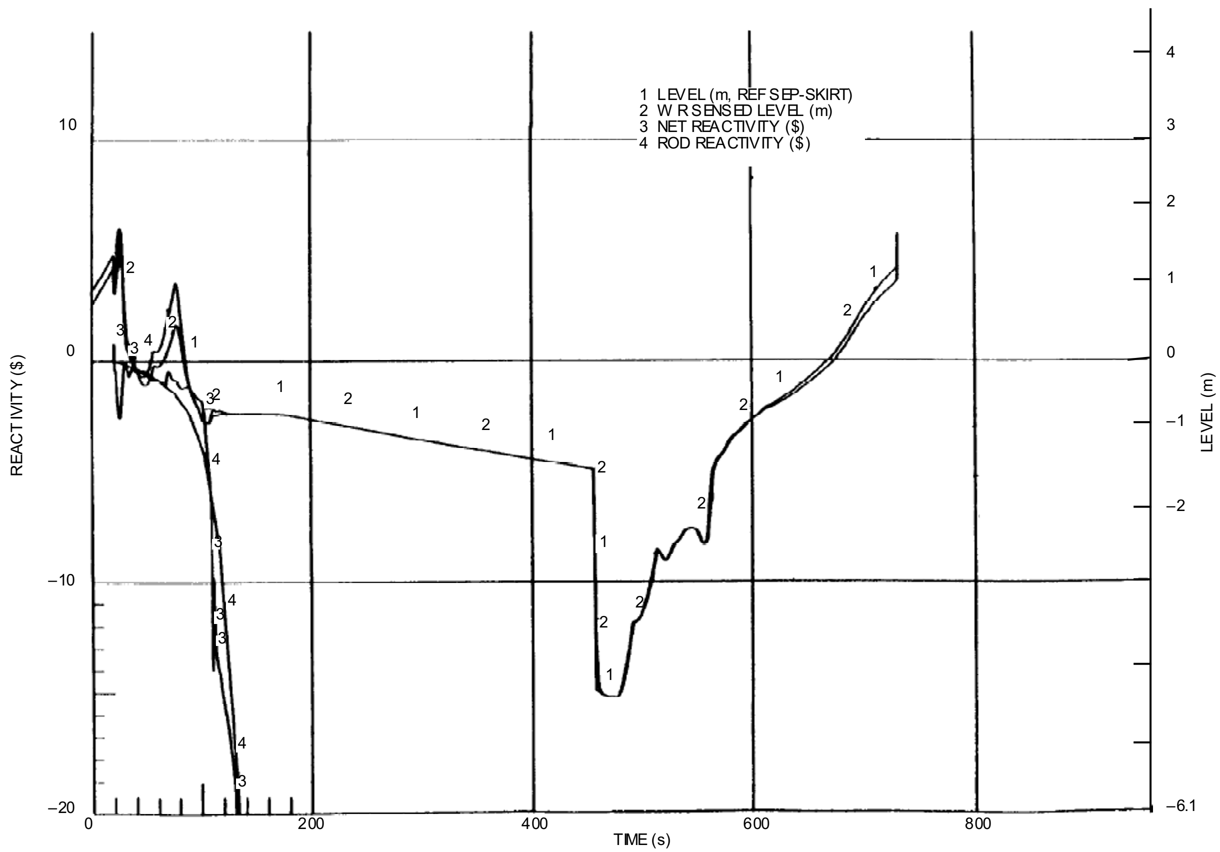


Figure 15E-24 ABWR Feedwater Controller Failure Maximum Demand, FMCRD Run–in (Continued)

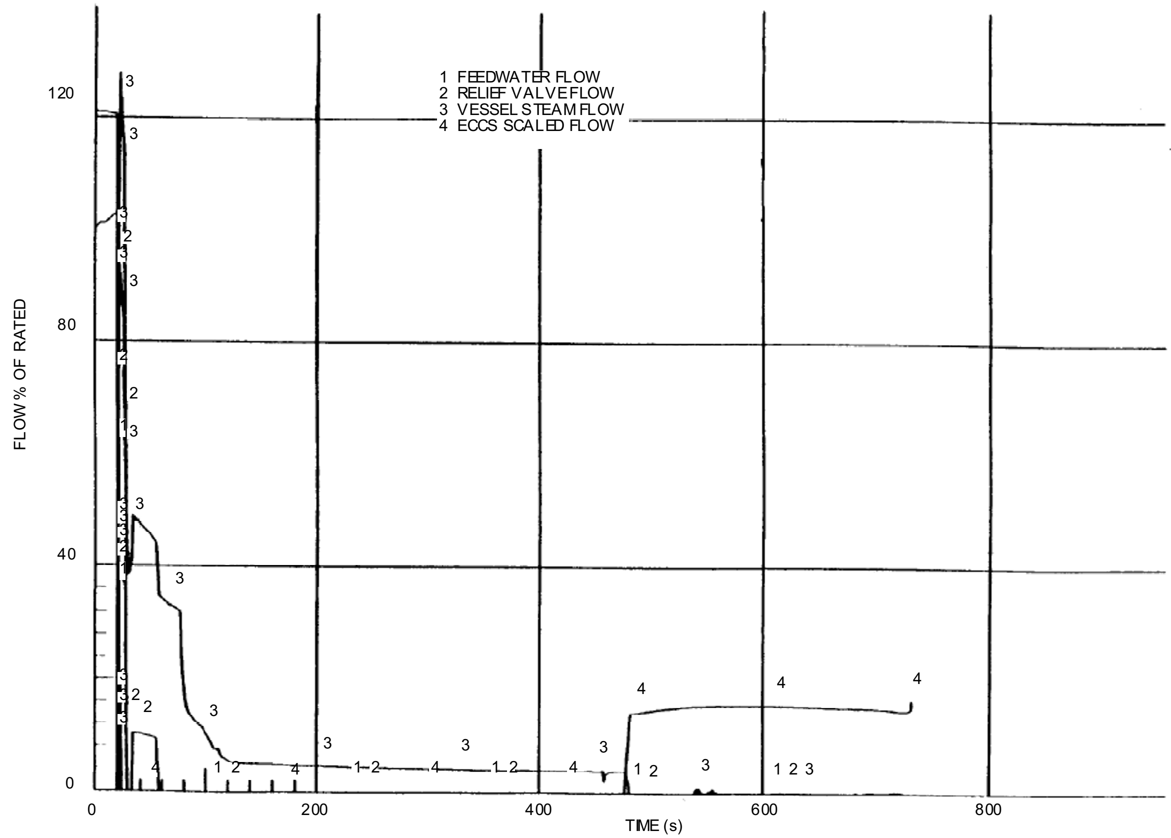


Figure 15E-24 ABWR Feedwater Controller Failure Maximum Demand, FMCRD Run–in (Continued)

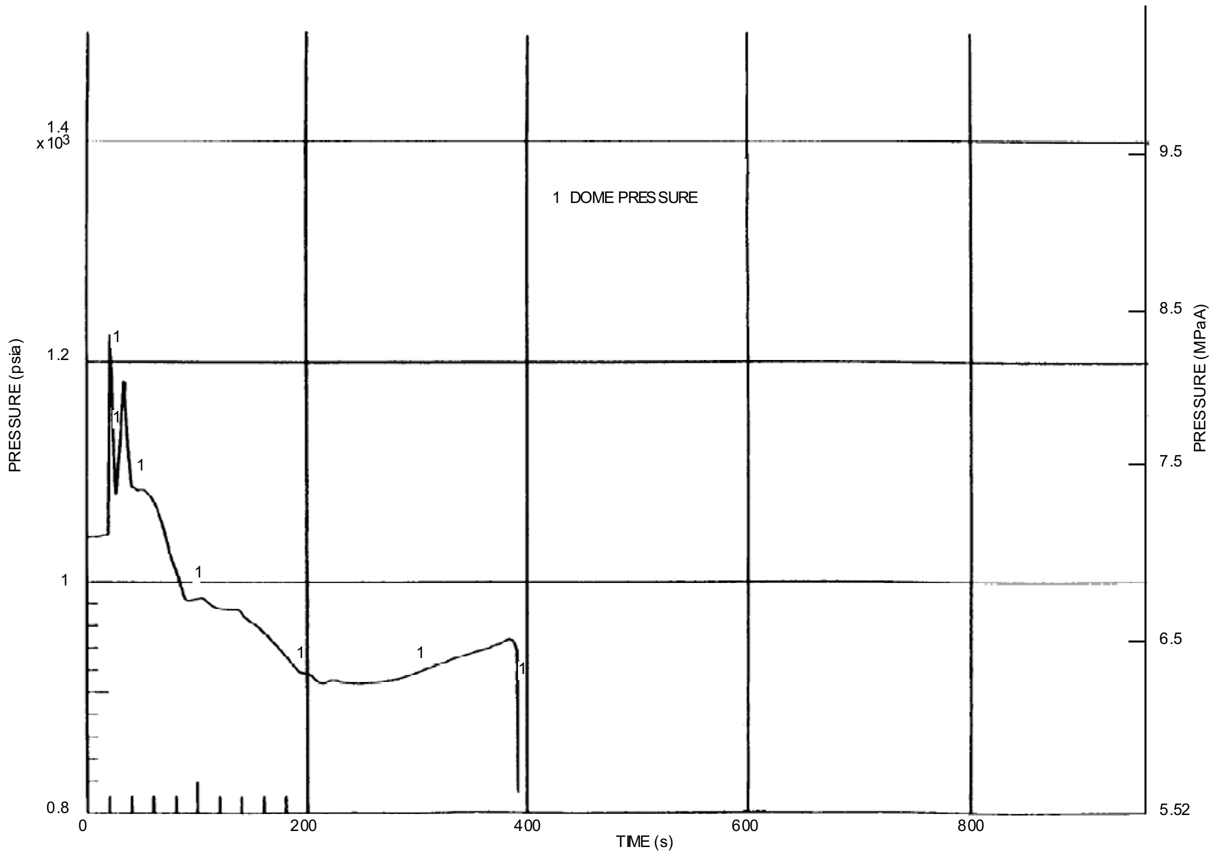


Figure 15E-25 ABWR Feedwater Contoller Failure Maximum Demand, SLCS

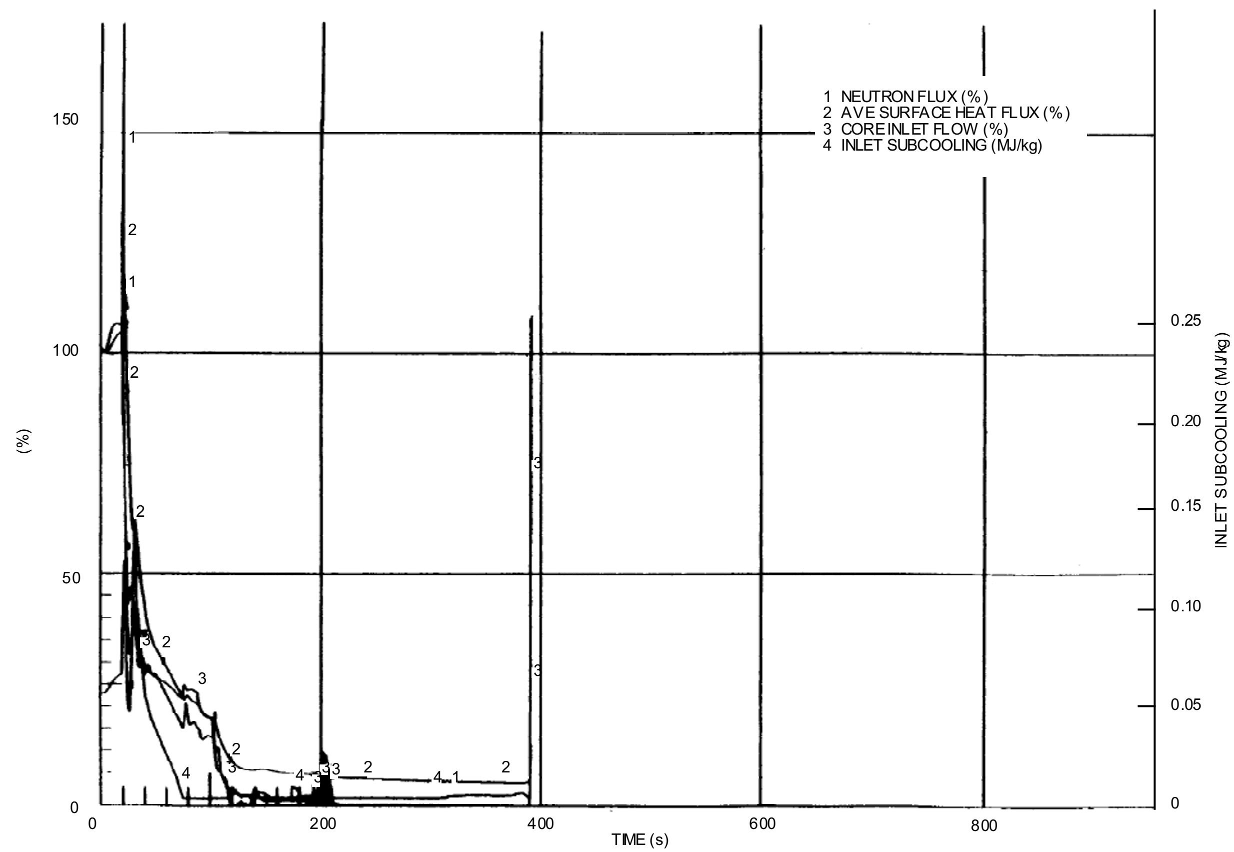


Figure 15E-25 ABWR Feedwater Contoller Failure Maximum Demand, SLCS (Continued)

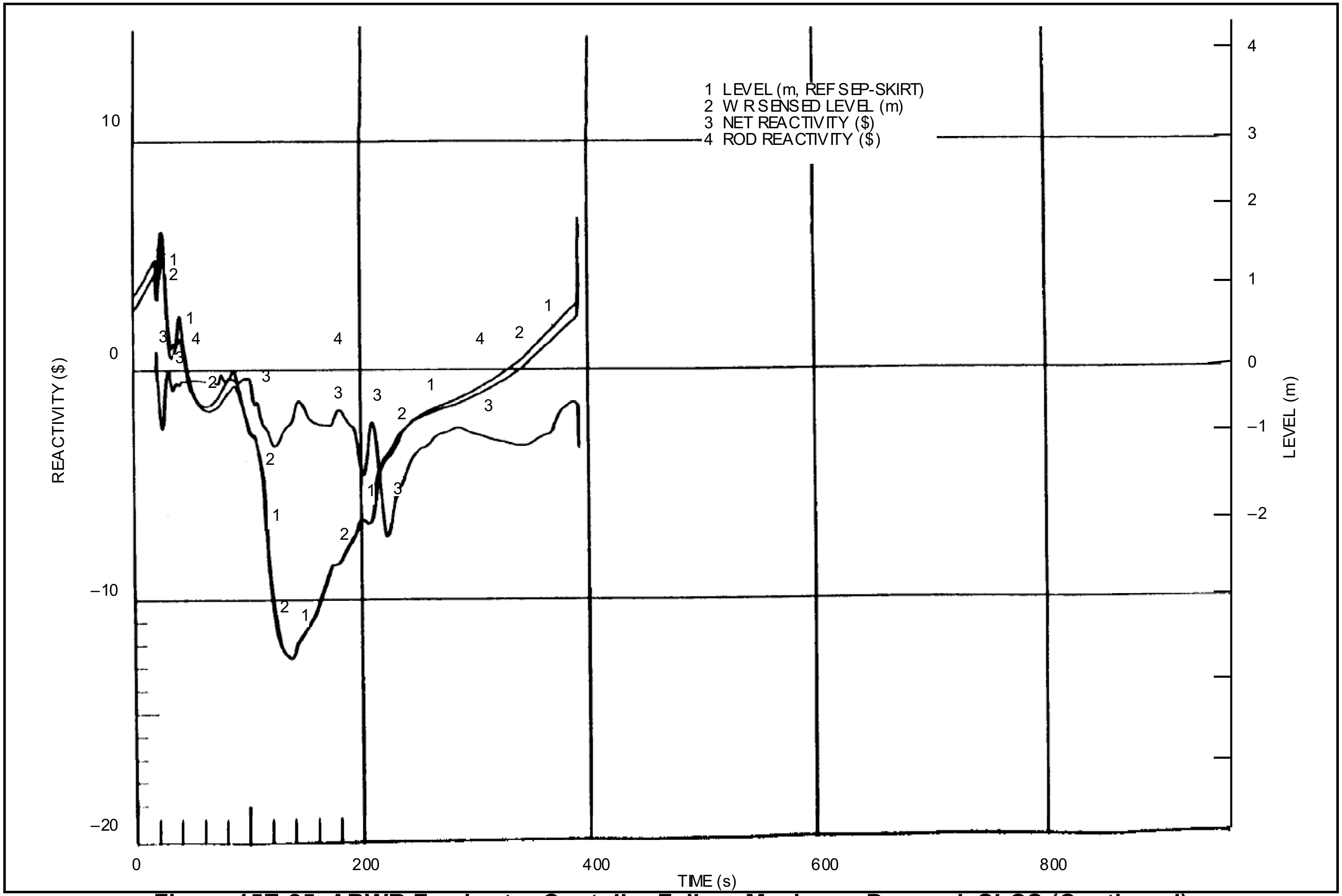


Figure 15E-25 ABWR Feedwater Contoller Failure Maximum Demand, SLCS (Continued)

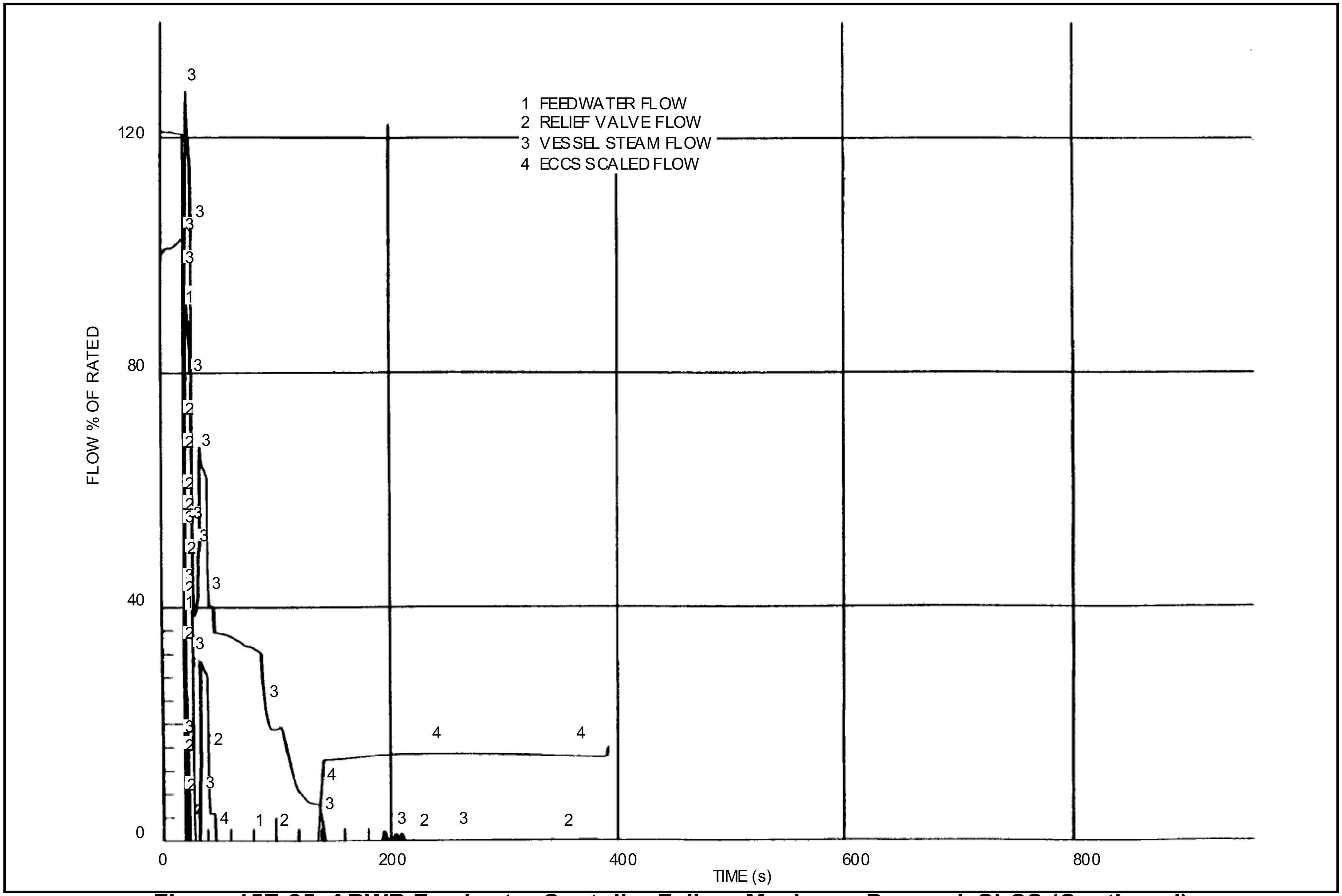


Figure 15E-25 ABWR Feedwater Contoller Failure Maximum Demand, SLCS (Continued)

15F LOCA Inventory Curves

15F.1 Introduction

This appendix provides additional detail on the distribution of iodine isotopes for the design basis LOCA analysis found in Subsection 15.6.5. The information is in the form of a series of curves as is explained below.

Curves Explanation

15F-1 Provides the total airborne fraction of iodine in the primary containment as a function of time.

15F-2 Provides the total airborne fraction of iodine in the reactor building as a function of time.

15F-3 Provides the distribution of elemental (including elemental and particulate) and organic iodine in the condenser which originated in the primary containment as a function of time.

15F-4 Provides the distribution of elemental and particulate iodine which originated in the primary containment in the main steamline and drain line piping. Shown is the:

- Fraction of total core inventory on the pipe surfaces as a function of time noted as FRACTION IN PIPES.
- Fraction of total core inventory converted to organic iodine which was originally fixed to the pipes and resuspended as a function of:
  - Time integrated release to the condenser.
  - Time integrated release from condenser.

15F-5 Provides fraction of core inventory released to the environment.

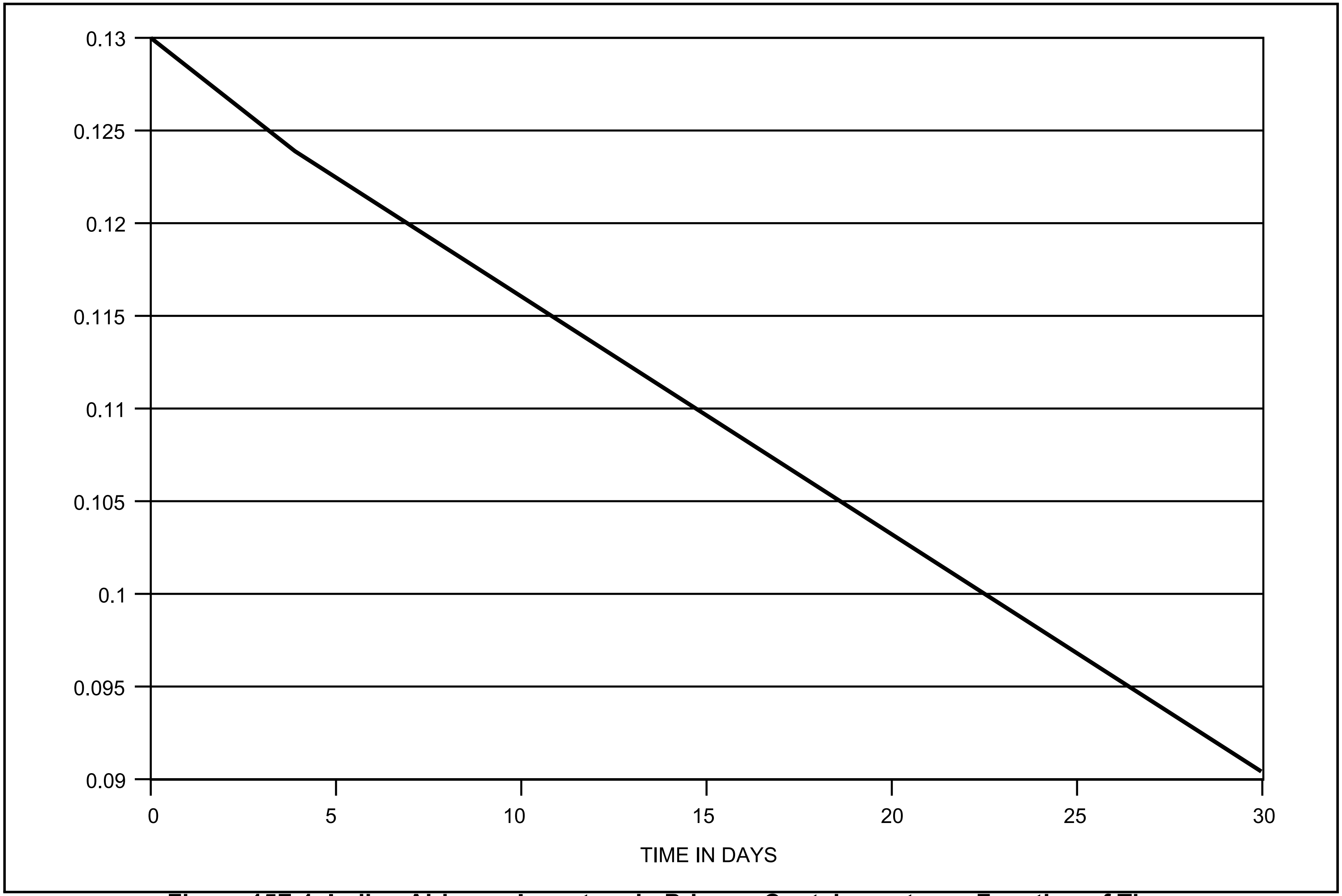


Figure 15F-1 Iodine Airborne Inventory in Primary Containment as a Function of Time

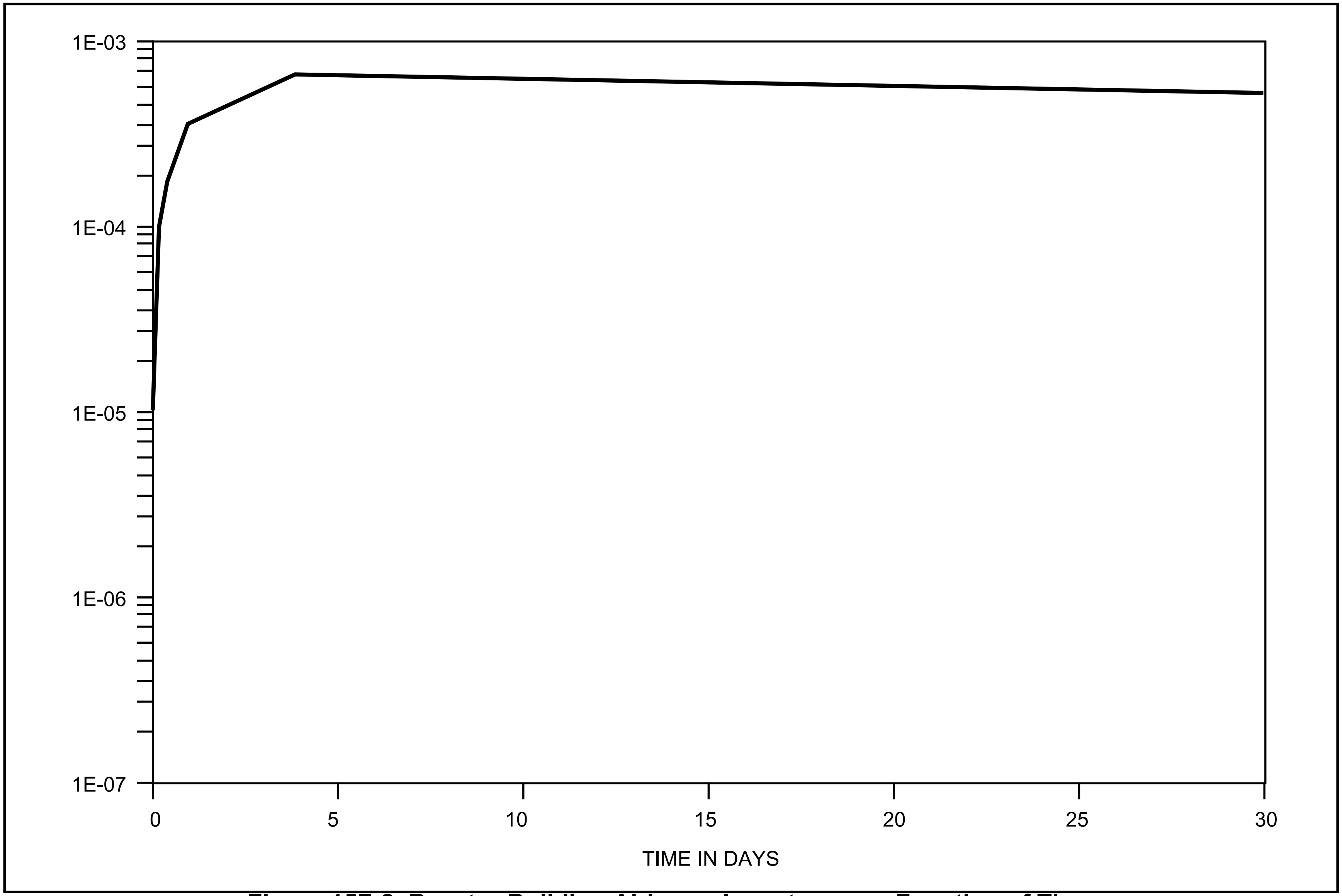


Figure 15F-2 Reactor Building Airborne Inventory as a Function of Time

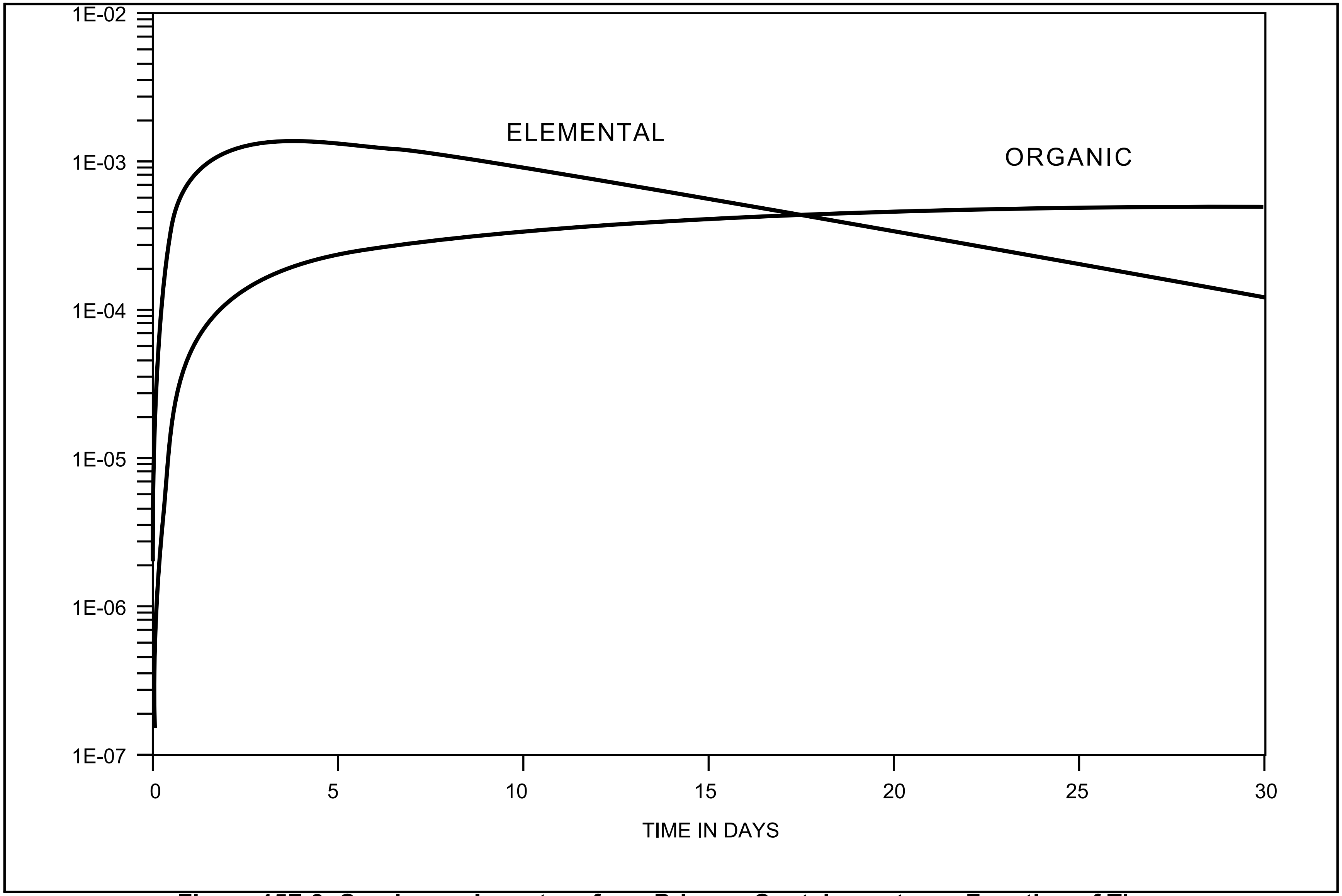


Figure 15F-3 Condenser Inventory from Primary Containment as a Function of Time

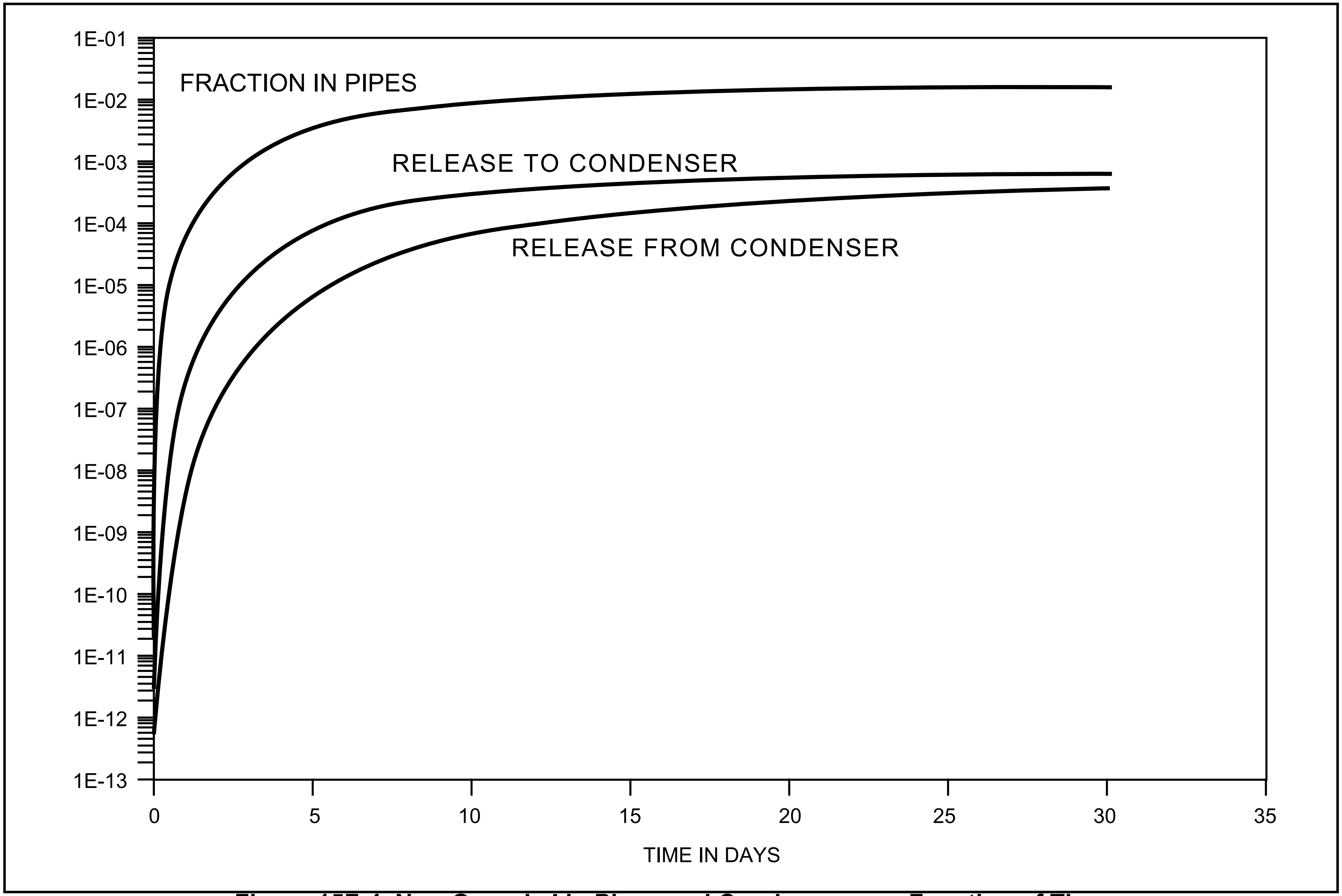


Figure 15F-4 Non-Organic I in Pipes and Condenser as a Function of Time

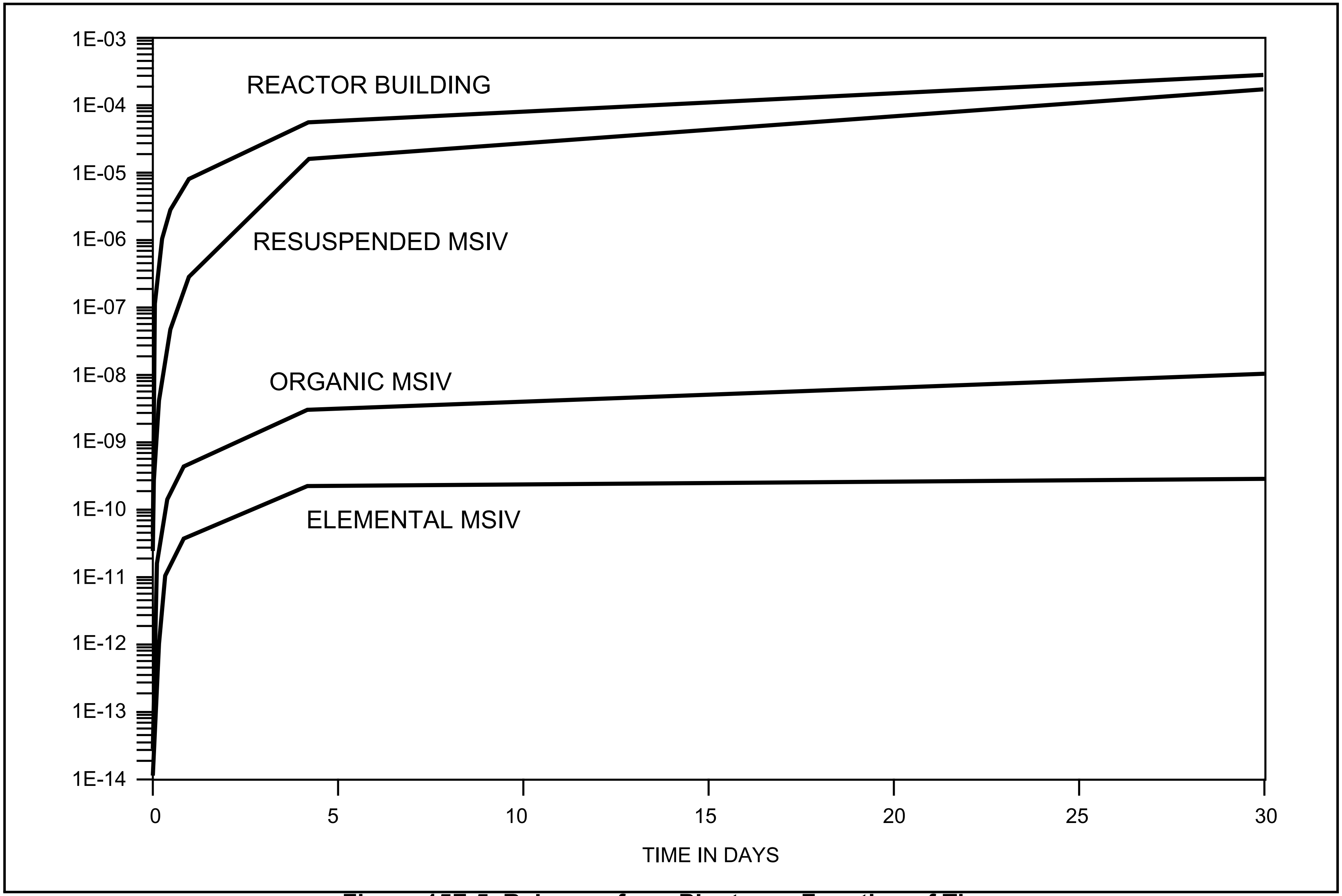


Figure 15F-5 Releases from Plant as a Function of Time