GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

April 6, 1992

MFN No. 081-92
Docket No. STN 52-001
SLK-9249

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Robert C. Pierson, Director
Standardization and Non-Power Reactor Project Directorate

Subject: **GE Responses to the Resolution of Issues Related to Chapter 15 of ABWR DSER SECY-91-355**

Reference: GE Responses Agenda Items 1,5,9 and 16 Discussed During the GE\NRC Reactor Systems Branch Meeting on November 20-21, 1991, MFN No. 010-92 dated January 10, 1992

Enclosed are thirty-four (34) copies of the GE responses to NRC staff comments and to Outstanding Issues 136 and 139. In addition, the test has been revised to incorporate the response to Outstanding Issue 135 provided in the referenced letter.

It is intended that GE will amend the SSAR with this response in a future amendment.

Sincerely,

P.W. Marriott

P.W. Marriott, Manager
Regulatory and Analysis Services
M/C 444, (408) 925-6948

cc: F.A. Ross (DOE)
N.D. Fletcher (DOE)
C. Poslusny, Jr. (NRC)
G. Thomas (NRC)
R. C. Berglund (GE)
J. F. Quirk (GE)

100113

D034
1/1

documentation. The limiting events which establish CPR operating limit:

(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:** ~~Runout of Two Feedwater Pumps.~~ 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 utility applicant referencing the ABWR design to the USNRC for information, see Subsection 15.0.5.2 for interface 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 Tables 15.0-4 will be provided by the utility applicant referencing the ABWR design to the USNRC for information. See 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 uncetainties meet the criteria in Appendix 4B.

In Table 15.0-3, a summary of applicable accidents is provided. This table compare GE calculated amount of failed fuel to that u. in worst-case radiological calculatiuons 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 interface 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 (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

In this chapter, the consequences of radioactivity rel for the core loading in Figure 4.3-1 during the three types of events: (a) incidents of moderate frequency (anticipated operational occurences); (b) infrequent incidents (abnormal operational occurences); and (c) limiting faults (design basis accidents), are given. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitive evaluation is presented or results are referenced from a more limiting or enveloping case or event.

15.0.5 Interface Requirements

15.0.5.1 Anticipated Operational Occurences (AOO)

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

Table 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR SYSTEM RESPONSE ANALYSIS TRANSIENTS (Continued)

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 (kg/cm^2g)	77.7
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 (sec)	7
34.	Reactor Internal Pump Trip Delay (sec)	0.16
35.	Recirculation Pump Trip Inertia Time Constant for Analysis (sec) ***	0.62
36.	Total Steamline Volume (m^3)	113.2
37.	Set pressure of Recirculation pump trip (kg/cm^2g)	79.1

* For transients simulated on the ODYN model, this input is calculated by ODYN.
** EOEC = End of Equilibrium Cycle.
*** The inertia time constant is defined by the expression:

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

- t = inertia time constant (sec);
- J_o = pump motor inertia (kg-m);
- n = pump speed (rps);
- g = gravitational constant (m/sec^2); and
- T_o = pump shaft torque (kg-m)

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 (Kg/Cm^2g)	Max. Vessel Bottom Pressure (Kg/Cm^2g)	Max. Steam Line Pressure (Kg/Cm^2g)	Max Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blowdown	Duration of Blowdown (seconds)
15.1		Decrease in core coolant temperature									
15.1.1		Loss of Feedwater heating	112.8	73.1	75.9	71.6	112.8	0.07	a	0	0
15.1.2	15.1-2	Runout of one feedwater pump	104.5	73.2	75.8	71.7	101.8	0.06	a	0	0
15.1.2	15.1-3	~~Runout of two feedwater pumps~~ Feedwater Controller Failure - Maximum Demand	139.0	83.3	84.9	82.8	105.9	0.10	a+	10	6
15.1.3	15.1-4	Opening of one Bypass Valve	102.1	73.1	75.6	71.6	100.0	**	a	0	0
15.1.3	15.1-5	Opening of all Control and Bypass Valves	102.0	80.4	81.8	80.1	100.0	**	a+	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	Closure of One Turbine Control Valve	~~120.8~~ 129.4	~~[illegible]~~ 75.1	~~[illegible]~~ 77.6	~~74.5~~ 73.7	~~103.1~~ 103.6	0.10	a	0	0
15.2.1	15.2-2	Pres. Regulator Downscale Fail.	154.8	85.8	87.4	85.1	103.0	N/A	c	18	6
15.2.2	15.2-3	Generator Load Rejection, Bypass on	148.1	83.2	84.7	82.7	100.2	0.06	a	10	5

* *Frequency definition is discussed in Subsection 15.0.4.1*
** *Not limiting (See Subsection 15.0.4.5.)*
a *Moderate Frequency*
b *Infrequent*
c *Limiting Fault*
N/A *Not applicable*
\+ *This event should be classified as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.*

Table 15.0-2

RESULTS SUMMARY OF SYSTEM RESPONSE ANALYSIS TRANSIENT EVENTS (Cont.)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (Kg/Cm^2g)	Max. Vessel Bottom Pressure (Kg/Cm^2g)	Max. Steam Line Pressure (Kg/Cm^2g)	Max Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blowdown	Duration of Blowdown (seconds)
15.2.2	15.2-4	Generator Load Rejection, Failure of One Bypass Valve	155.3	84.2	85.8	83.6	100.5	0.07	a+	14	5
15.2.2	15.2-5	Generator Load Rejection with failure of all Bypass Valves	184.6	86.1	87.7	85.6	102.3	0.10	a+	18	6
15.2.3	15.2-6	Turbine Trip Bypass--On	122.1	83.0	84.6	82.6	100.0	0.05	a	10	5
15.2.3	15.2-7	Turbine Trip w/Failure of One Bypass Valve	131.9	84.1	85.6	83.4	100.0	0.05	a+	14	5
15.2.3	15.2-8	Turbine Trip with failure of all Bypass Valves	158.6	86.1	87.7	85.4	100.6	0.08	a+	18	6
15.2.4	15.2-9	Inadvertent MSIV Closure	102.1	84.6	86.4	84.1	100.1	**	a	18	5
15.2.5	15.2-10	Loss of Condenser Vacuum	122.3	83.0	84.6	82.6	100.0	**	a	10	5
15.[illegible]	15.2-11	Loss of AC Power	113.2	82.9	84.4	82.7	100.0	0.05	a	10	5
15.2.7	15.2-12	Loss of All Feedwater Flow	102.0	73.1	75.7	71.6	100.1	**	a	0	0
15.2.8		Feedwater Piping Break			SEE	TEXT					

* *Frequency definition is discussed in Subsection 15.0.4.1*
** *Not limiting (See Subsection 15.0.4.5.)*
a *Moderate Frequency*
b *Infrequent*
c *Limiting Fault*
N/A *Not applicable*
\+ *This event should be classified as an infrequent event or a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.*

Table 15.0-2

RESULTS SUMMARY OF SYSTEM RESPONSE ANALYSIS TRANSIENT EVENTS (Cont.)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (Kg/Cm^2g)	Max. Vessel Bottom Pressure (Kg/Cm^2g)	Max. Steam Line Pressure (Kg/Cm^2g)	Max Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blowdown	Duration of Blowdown (seconds)
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	73.3	76.0	71.7	100.1	0.04	[illegible]	0	0
15.3.1	15.3-2	Trip of All Reactor Internal Pumps	102.0	83.2	84.1	82.7	100.2	***	c		
15.3.2	15.3-3	Fast Runback of One Reactor Internal Pump	102.0	73.0	75.9	71.6	100.0	**	a	0	0
15.3.2	15.3-4	Fast Runback of All Reactor Internal Pumps	102.0	73.1	76.0	71.6	100.0	**	a+	0	0
15.3.3	15.3-5	Seizure of One Reactor Internal Pump	102.0	73.1	75.9	71.6	100.0	**	c	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					

* *Frequency definition is discussed in Subsection 15.0.4.1*
** *Not limiting (See Subsection 15.0.4.5.)*
*** *CPR criterion does not apply. PCT < 593.3°C*
a *Moderate Frequency*
b *Infrequent*
c *Limiting Fault*
\+ *This event should be classified as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.*

Table 15.0-2

RESULTS SUMMARY OF SYSTEM RESPONSE ANALYSIS TRANSIENT EVENTS (Cont.)

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max. Dome Pressure (Kg/Cm²g)	Max. Vessel Bottom Pressure (Kg/Cm²g)	Max. Steam Line Pressure (Kg/Cm²g)	Max Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No of Valves First Blowdown	Duration of Blowdown (seconds)
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	71.1	72.3	70.6	116.1	****	a	0	0
15.4.5	15.4-3	Fast Runout of All Reactor Internal Pumps	135.0	72.5	74.7	71.5	168.5	****	a +	0	0
15.4.7		Misplaced Bundle Accident			SEE	TEXT					
15.4.8		Rod Ejection Accident			SEE	TEXT					
15.4.9		Control Rod Drop Accident			SEE	TEXT					
15.5		Increase in Reactor Coolant Inventory									
15.5.1	15.5-1	Inadvertent HPCF Startup	102.0	73.1	75.6	71.6	100.0	**	a	0	0

* *Frequency definition is discussed in Subsection 15.0.4.1*
** *Not limiting (See Subsection 15.0.4.5.)*
**** *Transients initiated from low power.*
a *Moderate Frequency*
b *Infrequent*
c *Limiting Fault*
\+ *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

SUBSECTION I.D.	TITLE	FAILED FUEL RODS GE CALCULATED VALUE	FAILED FUEL RODS NRC WORST-CASE ASSUMPTION
15.2.1	Pressure Regulator Downscale Failure	None	< 0.2 %
15.3.1	Trip of All Reactor Internal Pumps	None	~~60%~~ < 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)	DELTA CPR	FIGURE
Closure of One Turbine Control Valve	X	X	X	X
Load Rejection with all Bypass Valves Failure	X	X	X	X
~~Runout of 2 Feedwater Pumps~~ Feedwater Controller Failure - Maximum Demand	X	X	X	X

SECTION 15.1

CONTENTS

Section	Title	Page
15.1.1	**Loss of Feedwater Heating**	15.1-1
15.1.1.1	Identification of Causes and Frequency Classification	15.1-1
15.1.1.1.1	Identification of Causes	15.1-1
15.1.1.1.2	Frequency Classification	15.1-1
15.1.1.2	Sequence of Events and Systems Operations	15.1-1
15.1.1.2.1	Sequence of Events	15.1-1
15.1.1.2.2	Systems Operation	15.1-1
15.1.1.3	Core and System Performance	15.1-2
15.1.1.3.1	Input Parameters and Initial Conditions	15.1-2
15.1.1.3.2	Results	15.1-2
15.1.1.4	Barrier Performance	15.1-2
15.1.1.5	Radiological Consequences	15.1-2
15.1.2	**Feedwater Controller Failure--Maximum Demand**	15.1-2
15.1.2.1	Identification of Causes and Frequency Classification	15.1-2
15.1.2.1.1	Identification of Causes	15.1-2
15.1.2.1.2	Frequency Classification	15.1-3
15.1.2.1.2.1	Runout of One Feedwater Pump	15.1-3
15.1.2.1.2.2	Feedwater Controller Failure - Maximum Demand ~~Runout of Two Feedwater Pump~~	15.1-3
15.1.2.2	Sequence of Events and Systems Operations	15.1-3

SECTION 15.1

CONTENTS (Continued)

Feedwater Controller Failure - Maximum Demand

Section	Title	Page
15.1.2.2.1	Sequence of Events	15.1-3
15.1.2.2.1.1	Runout of One Feedwater Pump	15.1-3
15.1.2.2.1.2	~~Runout of Two Feedwater Pumps~~	15.1-3
15.1.2.2.1.3	Identification of Operator Actions	15.1-3
15.1.2.2.1.3.1	Runout of One Feedwater Pump	15.1-3
15.1.2.2.1.3.2	~~Runout of Two Feedwater Pumps~~	15.1-4
15.1.2.2.2	Systems Operation	15.1-4
15.1.2.2.2.1	Runout of One Feedwater Pump	15.1-4
15.1.2.2.2.2	~~Runout of Two Feedwater Pumps~~	15.1-4
15.1.2.3	Core and System Performance	15.1-4
15.1.2.3.1	Input Parameters and Initial Conditions	15.1-4
15.1.2.3.2	Results	15.1-4
15.1.2.3.2.1	Runout of One Feedwater Pump	15.1-4
15.1.2.3.2.2	~~Runout of Two Feedwater Pumps~~	15.1-4
15.1.2.4	Barrier Performance	15.1-4
15.1.2.5	Radiological Consequences	15.1-5
15.1.3	**Pressure Regulator Failure--Open**	15.1-5
15.1.3.1	Identification of Causes and Frequency Classification	15.1-5
15.1.3.1.1	Identification of Causes	15.1-5
15.1.3.1.2	Frequency Classification	15.1-5

SECTION 15.1

CONTENTS (Continued)

Section	Title	Page
15.1.6.2.2	Systems Operation	15.1-8
15.1.6.3	Core and System Performance	15.1-9
15.1.6.4	Barrier Performance	15.1-9
15.1.6.5	Radiological Consequences	15.1-9

TABLES

Table	Title	Page
15.1-1	Sequence of Events for Loss of Feedwater Heating	15.1-10
15.1-2	Loss of 55.6°C Feedwater Heating	15.1-10
15.1-2a	Loss of 16.7°C Feedwater Heating	15.1-10
15.1-3	Single Failure Modes for Digital Controls	15.1-11
15.1-4	Sequence of Events for Figure 15.1-2	15.1-12
15.1-5	Sequence of Events for Figure 15.1-3	15.1-12
15.1-6	Sequence of Events for Figure 15.1-4	15.1-13
15.1-7	Sequence of Events for Figure 15.1-5	15.1-13
15.1-8	Sequence of Events for Inadvertent Safety/Relief Valve Opening	15.1-14
15.1-9	Sequence of Events for Inadvertent RHR Shutdown Cooling Operations	15.1-14

SECTION 15.1

ILLUSTRATIONS

Figure	Title	Page
15.1-1	Simplified Block Diagram of Fault-Tolerant Digital Controller System	15.1-15
15.1-2	Runout of One Feedwater Pump	15.1-16
15.1-3	Feedwater Controller Failure - Maximum Demand ~~Runout of Two Feedwater Pumps~~	15.1-17
15.1-4	Inadvertent Opening of One Bypass Valve	15.1-18
15.1-5	Inadvertent Opening of All Control and Bypass Valves	15.1-19

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; or

(2) steam is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the steam 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. Insert (A) 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 ΔT 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 Δ CPR during the event. The same signal is also sent to the RC&IS to initiate the SCRRI (Selected control rods run-in) to automatically reduce the reactor power and avoid a scram. ~~and~~ 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 ΔT 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 data base, 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 safeguard features (ESF) is not

Insertion Ⓐ

The ABWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) 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 FW temperature drop based on the reference heat balance shown in Figure 10.1-1 is less than 30°C (53°F). Therefore, the use of 55.6°C (100°F) temperature drop in the transient analysis is conservative.

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 is feedwater heating. Another case with the ΔT 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, has been used to evaluate this event for the equilibrium cycle. The results are summarized in Table 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 Δ 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 0.4 Kg/Cm^2) and consequently, the reactor coolant pressure boundary 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 feedwater control system 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. (See 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 on-line. 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 will be considered 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 failed, the feedwater control system 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 affect on the plant are Modes 6 and 7, a failure of the output voter and a 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 lock-up 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 (less than 0.0088 failures per reactor year). 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 feedwater control system causes a run-out 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 ~~both~~ all feedwater pumps to run out to their maximum capacity. However the probability of this to occur is extremely low (less than 7×10^{-5} failu per reactor year).

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 less than once per 100 reactor years, this event is conservatively evaluated as an incident of moderate frequency.

15.1.2.1.2.2 ~~Runout of Two Feedwater Pumps~~ Feedwater Controller Failure - Maximum Demand

The frequency of occurrence for this event is estimated to be less than once per 10000 years. It should 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.

15.1.2.2 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 ~~Runout of Two Feedwater Pumps~~ 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.

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.

Feedwater Controller Failure -
15.1.2.2.1.3.2 ~~Runout of Two Feedwater Pumps~~
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; and

(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.

Feedwater Controller Failure -
15.1.2.2.2.2 ~~Runout of Two Feedwater Pumps~~
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 are high level tripping of the main turbine and feedwater pumps, scram and recirculation pump trip (RPT) due to turbine trip, and low water level initiation of the reactor core isolation cooling (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 74.9 kg/cm^{2}g. The total feedwater flow for ~~both~~ all pumps runout is assumed to be 130% of rated at the design pressure of 74.9 kg/cm^{2}g.

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 6 inches) and then settles back to its normal level. The vessel pressures only increase about 0.1 kg/cm^2. MCPR remains above the safety limit.

Feedwater Controller Failure - Maximum
15.1.2.3.2.2 ~~Runout of Two Feedwater Pumps~~
Demand

The simulated runout of ~~two~~ all feedwater pumps ~~accident~~ 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 82.8kg/cm^{2}g and the pressure at the bottom of the vessel to about 84.9 kg/cm^{2}g.

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

The 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

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.

Insert (R)

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 should be no single failure could cause this event, it ~~is~~ 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,

Insertion (a)

The discharge of steam : the suppression pool causes the temperature of the suppression pool to increase. When the pool temperature reaches the setpoint of 43.3°C (110°F), the suppression pool cooling function of 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 48.9°C (120°F), a reactor Scram is automatically initiated.

Table 15.1-1

SEQUENCE OF EVENTS FOR LOSS OF FEEDWATER HEATING

TIME (sec)	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 pov :r 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

Table 15.1-4

SEQUENCE OF EVENTS FOR FIGURE 15.1-2

TIME (sec)	EVENTS
0	Initiate simulated runout of one feedwater pump (at system design pressure of 74.9 kg/cm^2g 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 valve 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 (sec)	EVENT
0	Initiate simulated runout of All feedwater pumps (130% at system design pressure of 74.9 kg/cm^2g on feedwater flow)
18.35	L8 vessel level setpoint initiates trips 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 (sec)	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 (sec)	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 ~~turbine~~ pumps 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

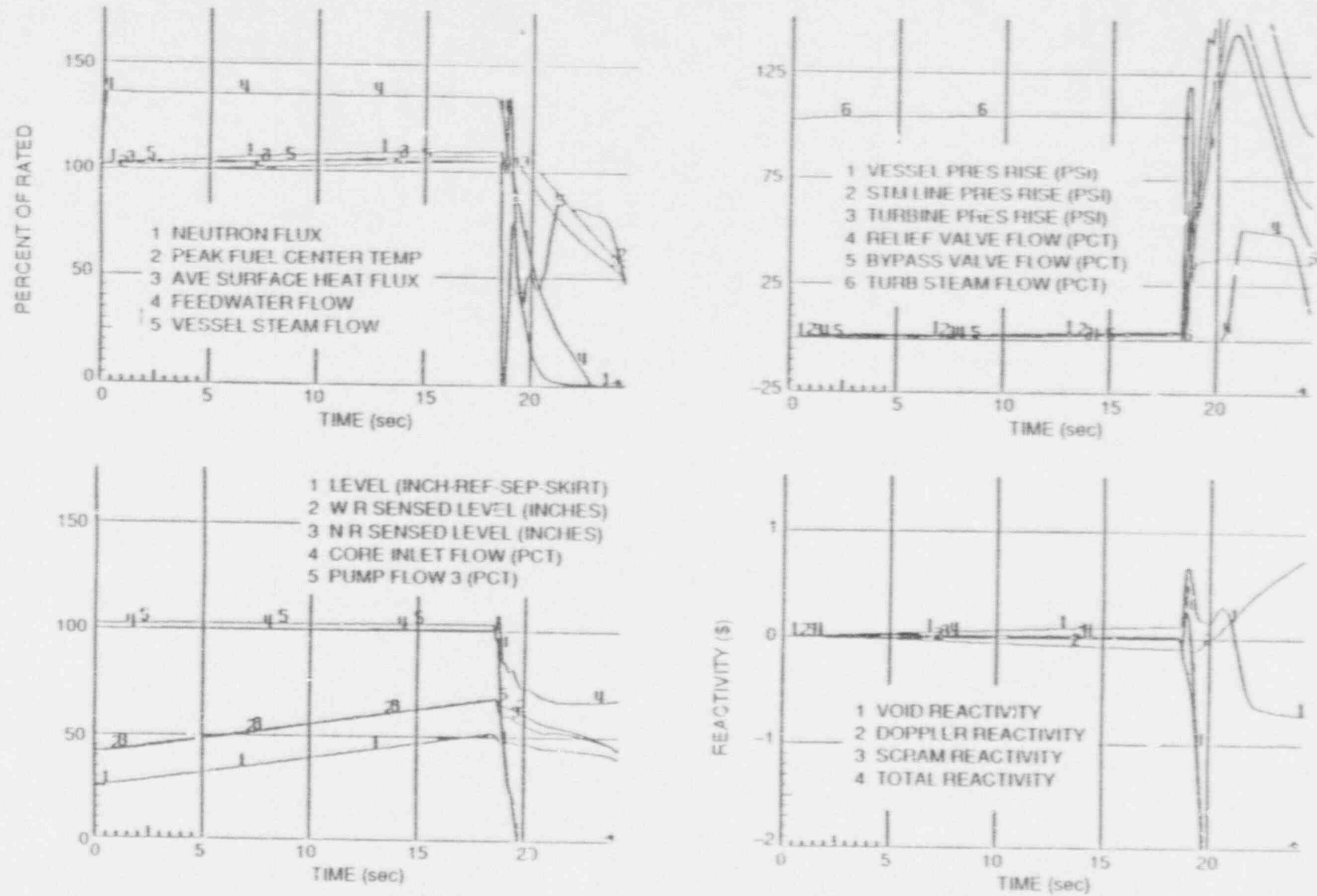
SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

TIME (sec)	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 take.



Feedwater Controller Failure - Maximum Demand

Figure 15.1-3 ~~RUNOUT OF TWO FEEDWATER PUMPS~~

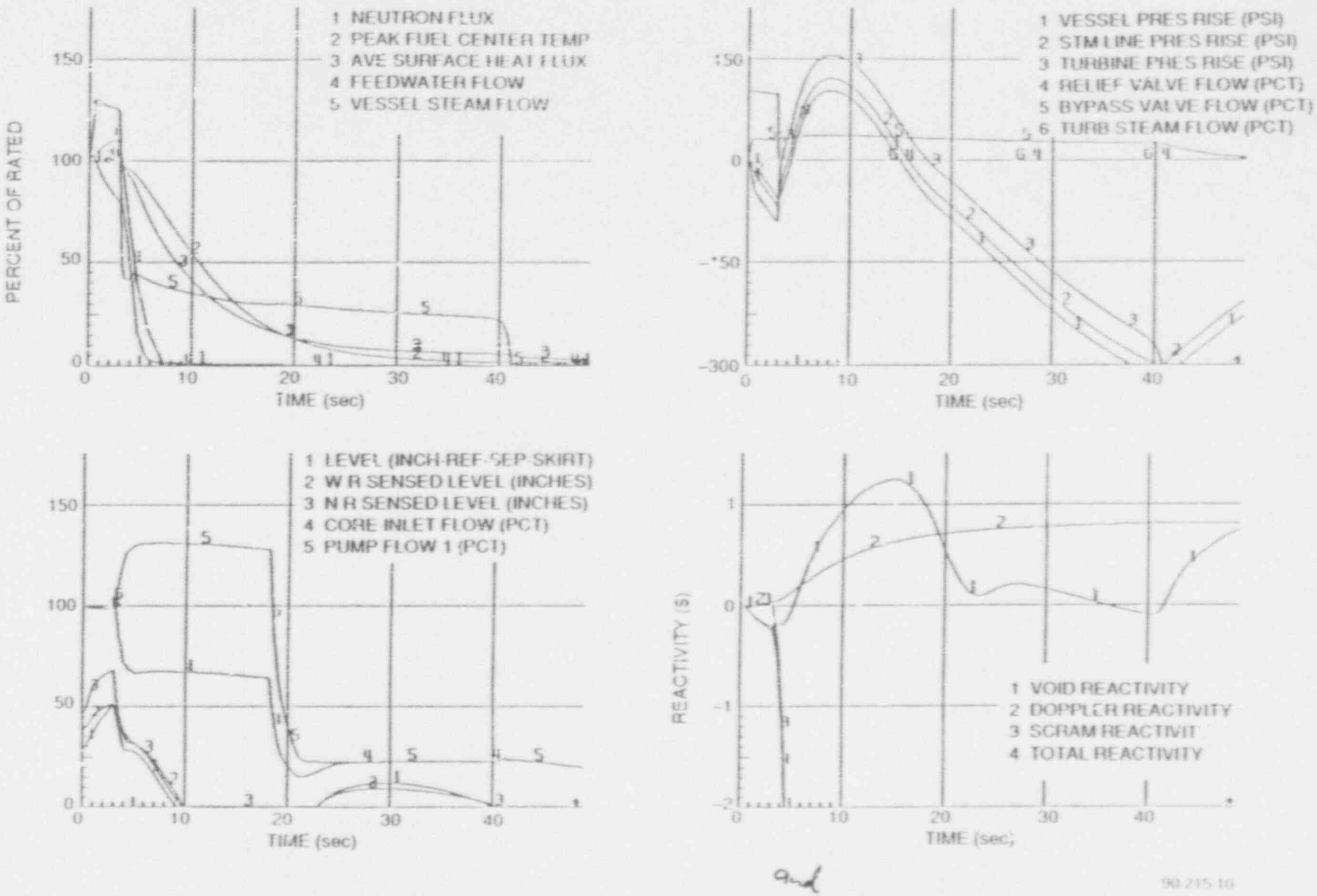


Figure 15.1-5 OPENING OF ALL CONTROL and BYPASS VALVES

SECTION 15.2

CONTENTS (Continued)

Section	Title	Page
15.2.1.5.2	Pressure Regulator Downscale Failure	
15.2.1.5	Radiological Consequences	15.2-2
15.2.1.5.1	Inadvertent Closure of one Turbine Control Valve	
15.2.2	**Generator Load Rejection**	15.2-2
15.2.2.1	Identification of Causes and Frequency Classification	15.2-3
15.2.2.1.1	Identification of Causes	15.2-3
15.2.2.1.2	Frequency Classification	15.2-3
15.2.2.1.2.1	Generator Load Rejection	15.2-3
15.2.2.1.2.2	Generator Load Rejection with Failure of One Bypass Valve	15.2-3
15.2.2.1.2.3	Generator Load Rejection with Failure of All Bypass Valves	15.2-3
15.2.2.2	Sequence of Events and Systems Operations	15.2-3
15.2.2.2.1	Sequence of Events	15.2-3
15.2.2.2.1.1	Generator Load Rejection--Turbine Control Valve Fast Closure	15.2-3
15.2.2.2.1.2	Generator Load Rejection with Failure of One Bypass Valve	15.2-3
15.2.2.2.1.3	Generator Load Rejection with Failure of All Bypass Valves	15.2-3
15.2.2.2.1.4	Identification of Operator Actions	15.2-3
15.2.2.2.2	Systems Operation	15.2-4
15.2.2.2.2.1	Generator Load Rejection with Bypass	15.2-4
15.2.2.2.2.2	Generator Load Rejection with Failure of One Bypass Valve	15.2-4

SECTION 15.2

CONTENTS (Continued)

Section	Title	Page
15.2.7.2.2	Systems Operation	15.2-16
15.2.7.3	Core and System Performance	15.2-16
15.2.7.4	Barrier Performance	15.2-16
15.2.7.5	Radiological Consequences	15.2-16
15.2.8	**Feedwater Line Break**	15.2-16
15.2.9	**Failure of RHR Shutdown Cooling**	15.2-16
15.2.10	**References**	15.2-17

TABLES

Table	Title	Page
15.2-1a	Sequence of Events for Figure 15.2-1a	15.2-18
15.2-1b	Sequence of Events for Figure 15.2-1b	
15.2-2	Sequence of Events for Figure 15.2-2	15.2-18
15.2-3	Sequence of Events for Figure 15.2-3	15.2-19
15.2-4	Sequence of Events for Figure 15.2-4	15.2-19
15.2-5	Sequence of Events for Figure 15.2-5	15.2-20
15.2-6	Sequence of Events for Figure 15.2-6	15.2-20
15.2-7	Sequence of Events for Figure 15.2-7	15.2-21
15.2-8	Sequence of Events for Figure 15.2-8	15.2-21
15.2-9	Sequence of Events for Figure 15.2-9	15.2-22
15.2-10	Post-Transient Release Rate to the Containment with Suppression Pool Cleanup	15.2-23
15.2-11	Activity Released to the Environment	15.2-24

SECTION 15.2

TABLES (Continued)

Table	Title	Page
15.2-12	Estimated Doses and Atmospheric Dispersion Factors	15.2-24
15.2-13	Typical Rates of Decay for Condenser Vacuum	15.2-25
15.2-14	Sequence of Events for Figure 15.2-10	15.2-25
15.2-15	Trip Signals Associated with the Loss of Condenser Vacuum	15.2-26
15.2-16	Sequence of Events for Figure 15.2-11	15.2-27
15.2-17	Sequence of Events for Figure 15.2-12	15.2-28

ILLUSTRATIONS

Figure	Title	Page
15.2-1a	Fast Closure of One Turbine Control Valve	15.2-30
15.2-1b	Slow Closure of One Turbine Control Valve	
15.2-2	Pressure Regulator Downscale Failure	15.2-31
15.2-3	Generator Load Rejection with Bypass	15.2-32
15.2-4	Load Rejection with One Bypass Valve Failure	15.2-33
15.2-5	Load Rejection with All Bypass Failure	15.2-34
15.2-6	Turbine Trip with Bypass	15.2-35
15.2-7	Turbine Trip with One Bypass Valve Failure	15.2-36
15.2-8	Turbine Trip with All Bypass Failure	15.2-37
15.2-9	MSIV Closure Direct Scram	15.2-38
15.2-10	Loss of Condenser Vacuum	15.2-39
15.2-11	Loss of AC Power	15.2-40

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 controls valves as well as an inhibit of 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 to occur is extremely low (less than $7x10^{-5}$ failure per reactor year), 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 (0.0088 failure per reactor year).

15.2.1.1.2.2 Pressure Regulator Downscale Failure

The probability of occurence of this event is calculated to be less than $7x10^{-5}$ per year 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 a ~~voter/~~ 6 actuator failure of the SB&PCS as presented in 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 15.2-1a for a fast closure and in Table 15.2-1b for Figure 15.2-1b for a slow closure.

15.2.1.2.1.2 Pressure Regulator Downscale Failure

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

15.2.1.2.1.3 Identification of Operator Actions

The operator should:

(1) monitor that all rods are in;

(2) monitor reactor water level and pressure;

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; and

(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.

steam

After a closure of one turbine control valve, the ~~team~~ 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, ~~the~~ cases with full arc and partial-arc turbine admission ~~is~~ are analyzed to cover all ~~plants.~~ potential operating conditions.

← Insert (X)

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 seconds) is pesented in Figure 15.2-1a. 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 ~~130~~ 129 % NBR by the reactor scram. Peak fuel surface heat flux does not exceed ~~103.4~~ 102.6% of its initial value. MCPR for this transient is still above the safety MCPR limit (ΔCPR = 0.10). Therefore, the design basis is satisfied. ~~See Subsection 15.8.4.5 for additional evaluation of results.~~

← Insert (Y)

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 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 74.5 kg/cm^2g. The peak vessel bottom pressure reaches 78.2 kg/cm^2g, below the transient pressure limit of 96.7 kg/cm^2g.

15.2.1.4.2 Pressure Regulator Downscale Failure

Peak pressure at the SRVs reaches 85.1 kg/cm^2g. The peak nuclear system pressure reaches 87.4 kg/cm^2g at the bottom of the vessel, below the nuclear barrier pressure limit.

Insertion (X) -1

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 limiter 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 analyses that the nominal setpoint

Insertion (X) - 2

for the maximum combined flow limiter should be set as 115% for plants with full arc turbine admission, and as 125% for plants with partial-arc turbine admission.

Insertion (Y)

A slow closure of one turbine control valve is also analyzed as shown in Figure 15.2.-16. 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 remaining three turbine control valves, the reactor power settles back 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 (ΔCPR = 0.09). Therefore, the design basis is satisfied.

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

15.2.1.5 Radiological Consequences

15.2.1.5.1 Inadvertent Closure of One Turbine Control Valve

~~While~~ The consequences of this event do not result in any fuel failures, ~~radioactivity is nevertheless~~ not any 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 15.2.6.5 (for a Type 2 event).~~ Therefore, the radiological exposures noted in

Subsection 15.2.4.5 cover the consequences of this event.

Insert ©

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 (TCV) is initiated whenever electrical grid disturbances occur which result 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 turbine-generator (T-G) rotor. Closure of the main turbine control valves 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 turbine control valves are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the Steam Bypass and Pressure Control System (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 fail to open on demand. The worst single failure can only cause one turbine bypass valve fail to open on demand. Therefore, the probability of this to occur is very low (less than one failure every 11 year). Therefore, generator load rejection with failure of one turbine bypass valve is considered an infrequent event; while generator load rejection with failure of all turbine bypass valves 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: $< 3.6\times10^{-5}$/plant year

Frequency Basis: Thorough search of domestic plant operating records have revealed three instances of bypass failure during 628 bypass system operations. This gives a probability of bypass failure of 0.0048. 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 of 0.0036 event/plant year. With the triplicated fault-tolerant design used in ABWR, this failure frequency is lowered by at least a factor of 100. Therefore, this event should be classified as a limiting fault, however, criteria for moderate frequent incidents are conservatively 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;

Insertion ©

15.2.1.5.2 Pressure Regulator Downscale Failure

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

Table 15.2-1a

SEQUENCE OF EVENTS FOR FIGURE 15.2-1a

TIME (sec)	EVENT
0	Simulate one main turbine control valve to fast close
0	Failed turbine control valve starts to close
~~1.3~~ 3.0	Neutron flux reaches high flux scram setpoint and initiates a reactor scram
~~1.5~~ 2.8	Turbine bypass valves start to open.
~~6.5~~ 5.1	Water level reaches level 3 setpoint. Four RIPs are tripped.

Table 15.2-2

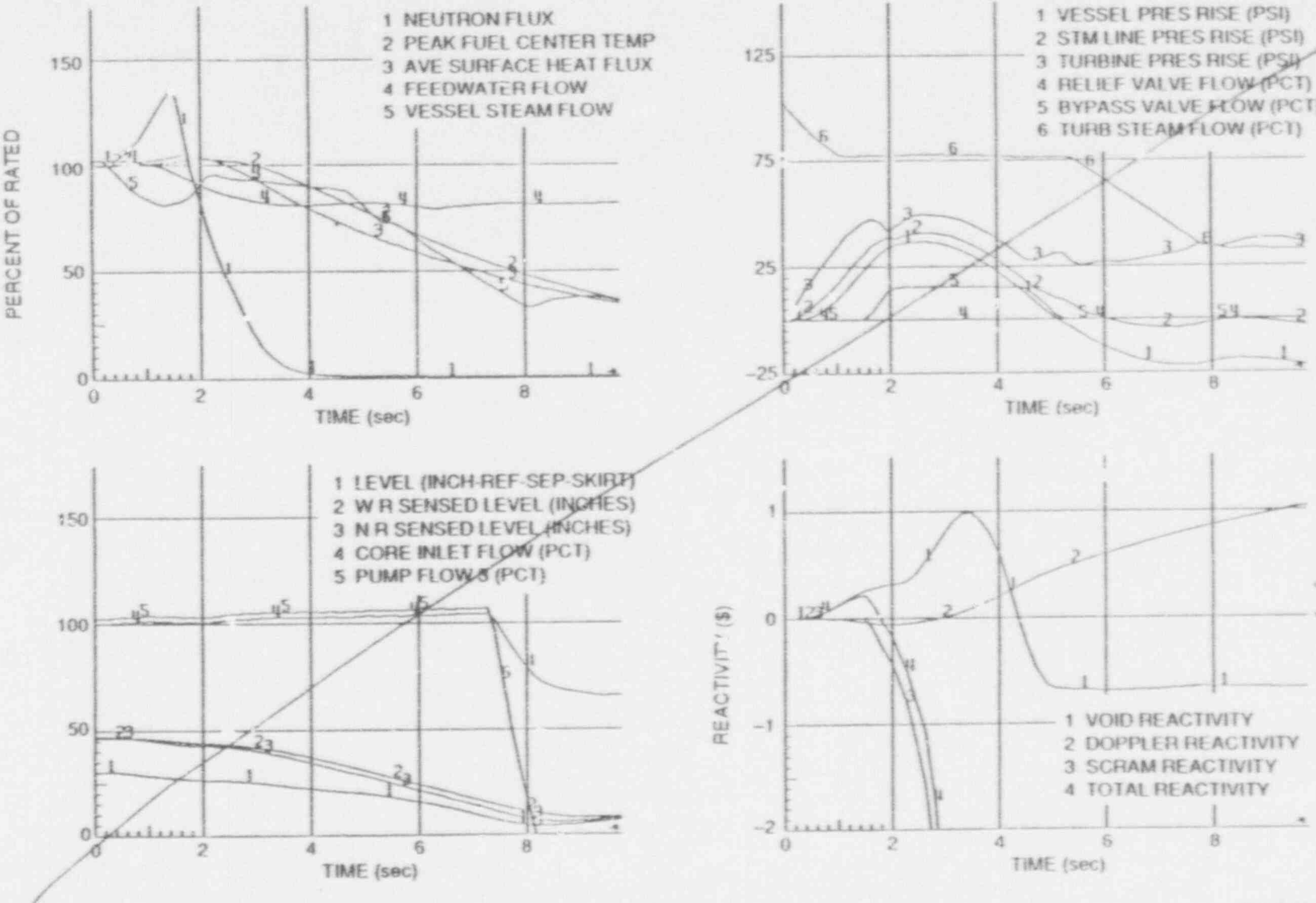
SEQUENCE OF EVENTS FOR FIGURE 15.2-2

TIME (sec)	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 15.2-1b

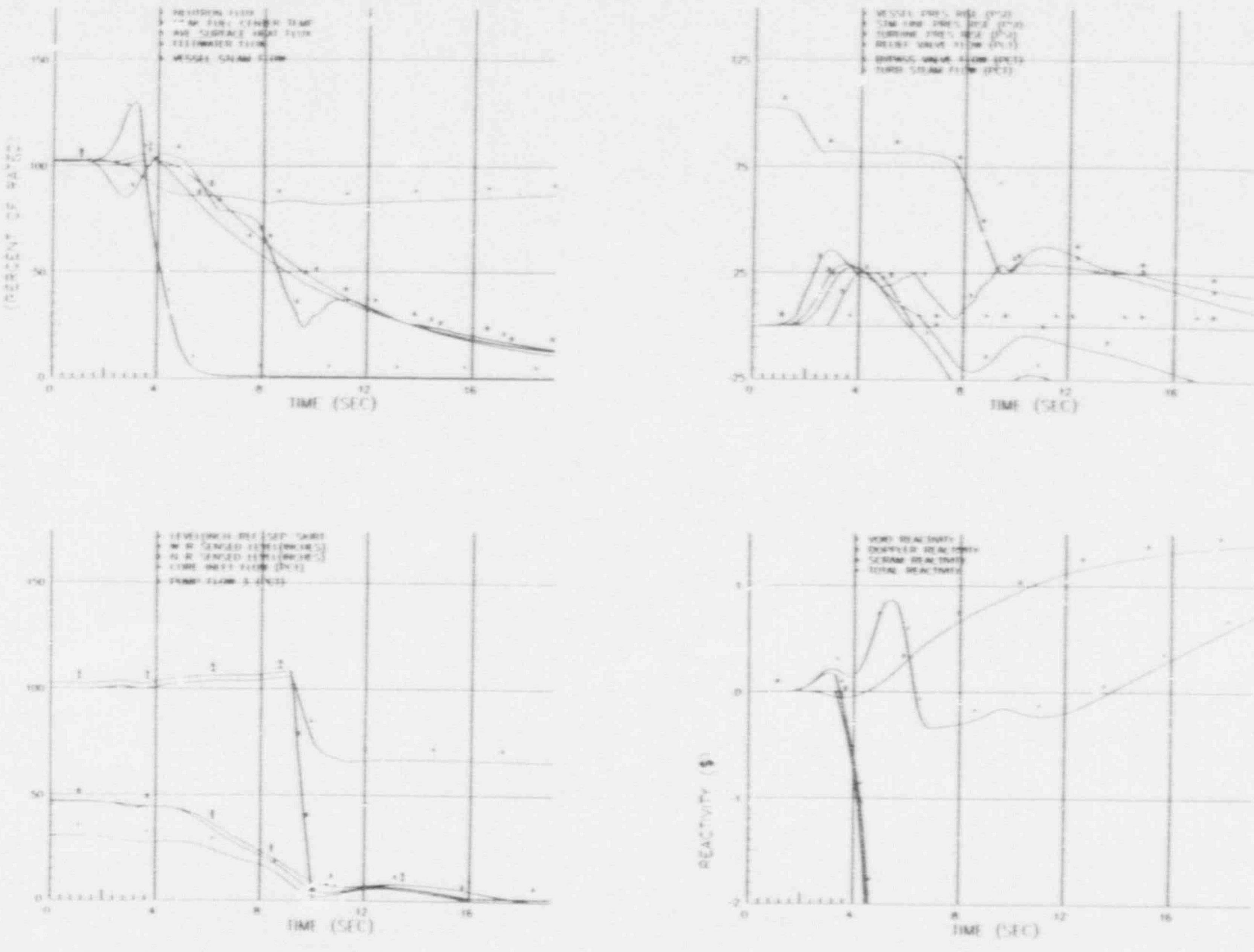
SEQUENCE OF EVENTS FOR FIGURE 15.2-1b

Time (sec)	Event
0	Simulate one main turbine control valve to slow closure
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.



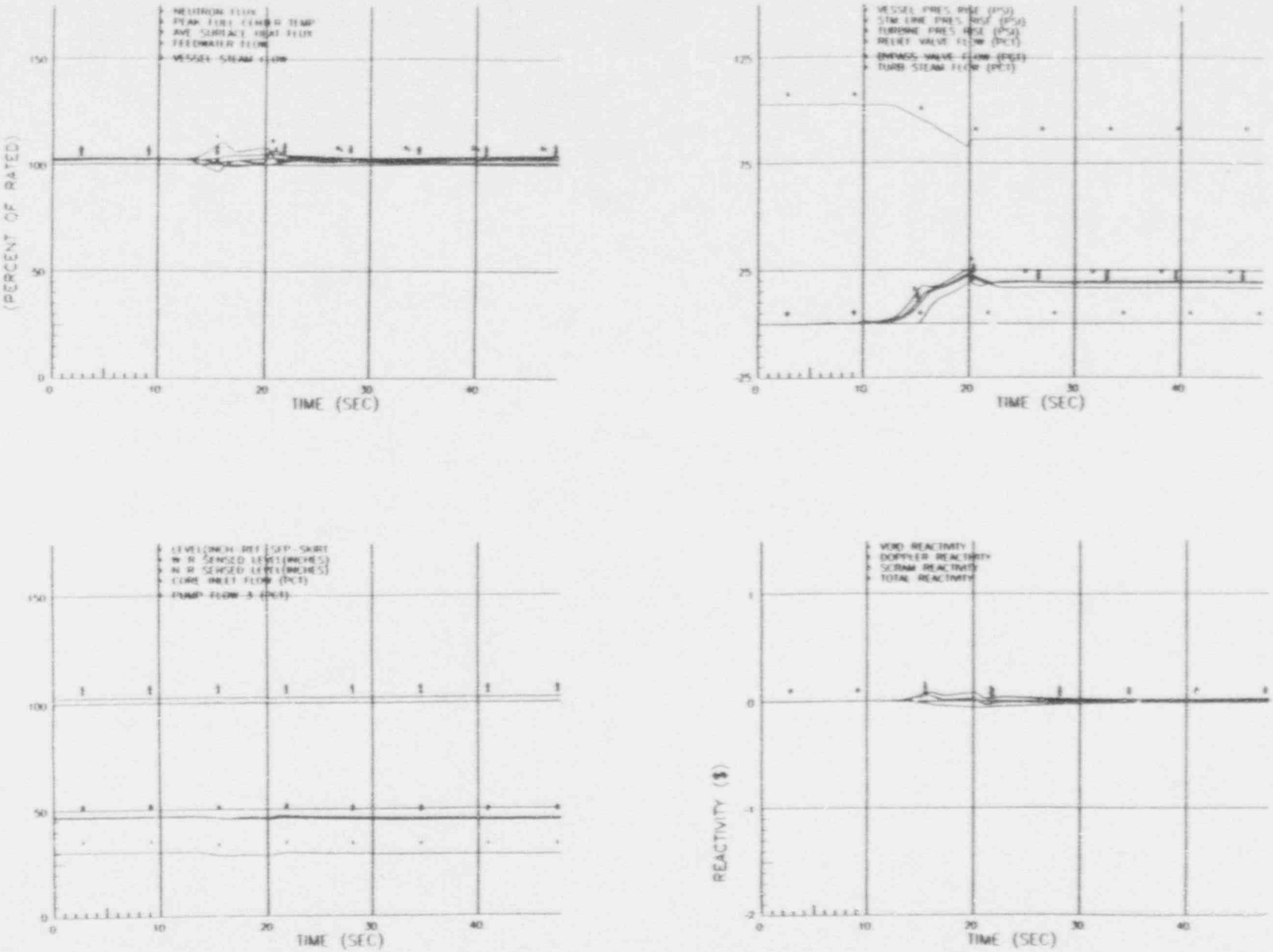
Replaced with the new figure.

Figure 15.2-10 FAST CLOSURE OF ONE TURBINE CONTROL VALVE



~~CLOSURE OF ONE TURBINE CONTROL VALVE~~

Figure 15.2-1a



~~SSER UPDATE~~
~~CLOSURE OF ONE TURBINE CONTROL VALVE~~

Figure 15.2-16 SLOW CLOSURE OF ONE TURBINE CONTROL VALVE

SECTION 15.3

CONTENTS (Continued)

Section	Title	Page
15.3.1.3.2.1	Trip of Three Reactor Internal Pumps	15.3-2
15.3.1.3.2.2	Trip of All Reactor Internal Pumps	15.3-2
15.3.1.4	Barrier Performance	15.3-3
15.3.1.4.1	Trip of Three Reactor Internal Pumps	15.3-3
15.3.1.4.2	Trip of All Reactor Internal Pumps	15.3-3
15.3.1.5	Radiological Consequences	15.3-3
15.3.1.5.1	Trip of Three Reactor Internal Pumps	
15.3.2	**Recirculation Flow Control Failure-- Decreasing Flow**	15.3-3
15.3.2.1	Identification of Causes and Frequency Classification	15.3-3
15.3.2.1.1	Identification of Causes	15.3-3
15.3.2.1.2	Frequency Classification	15.3-3
15.3.2.1.2.1	Fast Runback of One Reactor Internal Pumps	15.3-3
15.3.2.1.2.2	Fast Runback of All Reactor Internal Pumps	15.3-3
15.3.2.2	Sequence of Events and Systems Operations	15.3-3
15.3.2.2.1	Sequence of Events	15.3-3
15.3.2.2.1.1	Fast Runback of One Reactor Internal Pumps	15.3-3
15.3.2.2.1.2	Fast Runback of All Reactor Internal Pumps	15.3-4
15.3.2.2.1.3	Identification of Operator Actions	15.3-4
15.3.2.2.1.3.1	Fast Closure of One Main Recirculation Valve	15.3-4
15.3.2.2.1.3.2	Fast Closure of Two Main Recirculation Valves	15.3-4
15.3.2.2.2	Systems Operation	15.3-4
15.3.1.5.2	Trip of All Reactor Internal Pumps	

SECTION 15.3

TABLES

Table	Title	Page
15.3-1	Sequence of Events for Figure 15.3-1	15.3-8
15.3-2	Sequence of Events for Figure 15.3-2	15.3-8
15.3-3	Sequence of Events for Figure 15.3-3	15.3-9
15.3-4	Sequence of Events for Figure 15.3-4	15.3-9
15.3-5	Sequence of Events for Figure 15.3-5	15.3-9

ILLUSTRATIONS

Figure	Title	Page
15.3-1	Three Pump Trip	15.3-10
15.3-2	All Pump Trip	15.3-11
15.3-2a	Cladding Temperature during All Pump Trip	
15.3-3	Fast Runback of One RIP	15.3-12
15.3-4	Fast Runback of All RIPs	15.3-13
15.3-5	One RIP Seizure	15.3-14

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 multiple failures in 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 rapid core flow coastdown. High system pressure is limited by the pressure relief valve system operation.

high simulated thermal power scram, turbine trip due to high water level, or

Insert (D)

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 life (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

Insertion (D)

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 ASD's, which results in a trip of all RIPs.

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 peak clad temperature during this event is calculated to be less than 600°C, which is below the applicable limit of 1200°C.

Insert (E)

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 96.7 Kg/cm^2g 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

Insert → (F)

~~Trip of all 10 internal pumps due to a loss of power supply is considered extremely unlikely to result in perforation of fuel under conditions of boiling transition. The release of fission products would be, however, much less than that assumed in the Loss of Coolant Accident for an event of equal probability. Therefore, the radiological exposures noted in Subsection 15.6.5 cover the consequences of this event.~~

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 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 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%/sec.). 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%/sec. However, the probability of this event occurring is low (less than 7 x 10^{-5} failures per reactor year); and hence, the event should be considered as a limiting fault. However, criteria for moderate 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 about 0.0088 failures per reactor year. 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.

Insertion (E)

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.

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

Insertion (F) -1

15.2.1.5.1 Trip of Three Reactor Internal Pumps

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.

15.3.1.5.1 Trip of All Reactor Internal Pumps

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

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

(b) For fuel rods with greater than 20 GWD/T exposure, rods getting into transition boiling shall be assumed to fail for radiological dose calculation.

(to be cont.)

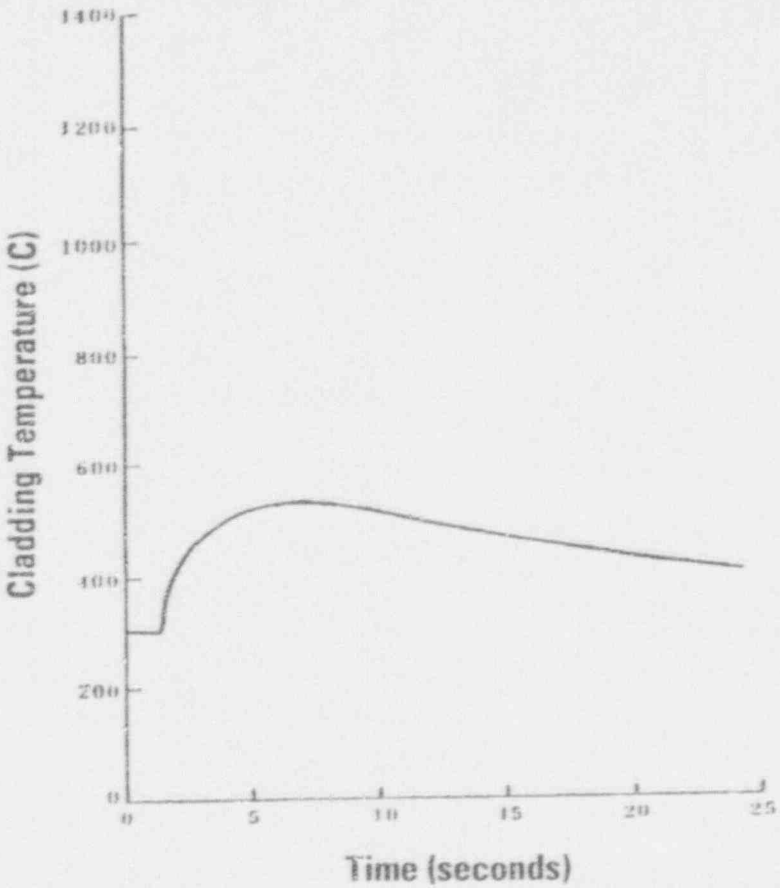
Insertion (E) - 2

(c) The radiological doses shall be less than 10% of 10CFR 100 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 stay in the core for at least 2 fuel cycle. In the equilibrium cycle, these fuel bundles only account for about 45% of total bundles. The power generated by these bundles are usually 20% less than the hottest bundles. Therefore, less than 0.2% of these rods is expected to get into transition boiling. Therefore, the requirements of 10% of 10CFR100 are met.

~~Cladding Temperature during APT (Licensing Calculation)~~



~~Assumption: No rewetting, Conservative inputs~~

New

Figure 15.3-2a Cladding Temperature during All pump Trip

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

15.8.1 Requirements

SRP 15.8 requires a 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 to 86 gpm of 13 weight percent sodium pentaborate solution.

15.8.2 Plant Capabilities

For ATWS prevention/mitigation for ABWR, the following are provided:

a. An ARI system that utilizes sensors and logic which are diverse and independent of the reactor protection system,

b. Electrical insertion of FMCRDs that also utilize sensors and logic which are diverse and independent of the reactor protection system,

c. Automatic recirculation pump trip under conditions indicative of an ATWS, and

d. Automatic initiation of ~~Backup manual~~/SLCS with 100 gpm capacity 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, the 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 fine motion control rod drive (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 simutaneous 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 Criterion 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 modem 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.

~~More information is provided in the response to Question 440.103~~. Supporting analysis is documented in Appendix 15E.