CHAPTER 15

ACCIDENT ANALYSES

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15.0 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 scope of the situations analyzed includes anticipated (expected) operational occurrences (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 Analytical Objective

The spectrum of postulated initiating events is 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 are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate within regulatory guidelines, without undue risk to the public health and safety.

15.0.2 Analytical Categories

Transient and accident events contained in this report are discussed in individual categories as required by Reference 1. The results of the events are summarized in Table 15.0.1. Each event evaluated is assigned to one of the following applicable categories.

15.0.2 Analytical Categories (Continued)

(1) Decrease in Core Coolant Temperature: Reactor vessel water (moderator) temperature reduction results in an increase in core reactivity. This could lead to fuel-cladding damage.

(2) Increase in Reactor Pressure: Nuclear system pressure increases threaten to rupture the reactor coolant pressure boundary (RCPB). Increasing pressure also collapses the voids in the core-moderator, thereby increasing core reactivity and power level which threaten fuel cladding due to overheating.

(3) Decrease in Reactor Core Coolant Flow Rate: A reduction in the core coolant flow rate threatens to overheat the cladding as the coolant becomes unable to adequately remove the heat generated by the fuel.

(4) Reactivity and Power Distribution Anomalies: Transient events included in this category are those which cause rapid increases in power which are due to increased core flow disturbance events. Increased core flow reduces the void content of the moderator, thereby increasing core reactivity and power level.

(5) Increase in Reactor Coolant Inventory: Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

(6) Decrease in Reactor Coolant Inventory: Reductions in coolant inventory could threaten the fuel as the coolant becomes less able to remove the heat generated in the core.

15.0.2 Analytical Categories (Continued)

(7) Radioactive Release from a Subsystem or Component: Loss of integrity of a radioactive containment component is postulated.

(8) Anticipated Transients Without Scram: In order to determine the capability of plant design to a ccommodate an extremely low probability event, a multi-system maloperation situation is postulated.

15.0.3 Event Evaluation

15.0.3.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized based upon 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 the following frequency groups.

(1) Incidents of moderate frequency - these are incidents that may occur during a calendar year to once per 20 years for a particular plant. This event is referred to as an "anticipated (expected) operational transient".

(2) Infrequent incidents - these are incidents that may occur during the life of the particular plant (spanning once in 20 years to once in 100 years). This event is referred to as an "abnormal (unexpected) operational transient".

15.0.3.1 Identification of Causes and Frequency Classification (Continued)

(3) Limiting faults - these are incidents that are not expected to occur but are postulated because their consequences may result in the release of significant amounts of radioactive material. This event is referred to as a "design basis (postulated) accident".

(4) Normal operation - operations of high frequency are not discussed here but are examined along with (1), (2), and (3) in the nuclear systems operational analyses in Appendix A to Chapter 15.

15.0.3.1.1 Unacceptable Results for Incidents of Moderate Frequency (Anticipated (Expected) Operational Transients)

The following are considered to be unacceptable safety results for incidents of moderate frequency (anticipated operational transients):

(1) a release of radioactive material to the environs that exceeds the limits of 10CFR20;

(2) reactor operation induced fuel cladding failure;

(3) Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes; and

(4) containment stresses in excess of that allowed for the transient classification by applicable industry codes.

15.0.3.1.2 Unacceptable Results for Infrequent Incidents (Abnormal (Unexpected) Operational Transients)

The following are considered to be unacceptable safety results for infrequent incidents (abnormal operational transients):

(1) release of radioactivity which results in dose consequences that exceed a small fraction of 10CFR100;

(2) fuel damage that would preclude resumption of normal operation after a normal restart;

(3) generation of a condition that results in consequential loss of function of the reactor coolant system; and

(4) generation of a condition that results in a consequential loss of function of a necessary containment barrier.

15.0.3.1.3 Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents)

The following are considered to be unacceptable safety results for limiting faults (design basis accidents):

(1) radioactive material release which results in dose consequences that exceed the guideline values of 10CFR100;

(2) failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited;

(3) nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes;

15.0.3.1.3 Unacceptable Results for Limiting Faults (Design Basis (Postulated) Accidents) (Continued)

(4) containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required; and

(5) radiation exposure to plant operations personnel in the main control room in excess of 5 Rem whole body, 30 Rem inhalation and 75 Rem skin.

15.0.3.2 Sequence of Events and Systems Operations

Each transient or accident 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 and controls are assumed to function;

(3) the extent to which 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; and

(6) the effect of a single failure or an operator error on the event.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

This paragraph discusses a very important concept pertaining to the application of single failures and operator errors analyses of the postulated events. Single active component failure (SACF) criteria have been required and successfully applied on past NRC approved docket applications to design basis accident (DBA) categories only. Reference 1 infers that a "single failures and operator errors" requirement should be applied to transient events (both high, moderate and low probability occurrences), as well as accident (very low probability) situations.

Transient evaluations have been judged against a criterion of one single equipment failure "or" one single operator error as the initiating event with no additional single failure assumptions to the protective sequences although a great majority of these protective sequences utilized safety systems which can accommodate single active component failure aspects. Even under these postulated events, the plant damage allowances or limits were very much the same as those for normal operation.

Reference 1 suggests that the transient and accident scenarios should now include "and" (multi-failure) event sequences. The format request follows:

For initiating occurrence	(1)	an equipment failure or an operator error, and
For single equipment failure or operator error analysis	(2)	another equipment failure or failures and/or another operator error or errors.

15.0.3.2.1.1 General (Continued)

This certainly is considered a new requirement and the impact will need to be completely evaluated. While this is under consideration, GE has evaluated and presented the transients and accidents in this chapter in the above new requirement manner.

Event categorization relative to transient and accident analysis is discussed here. If the evaluation is done per the new multi-failure methods, the event frequency categories should be modified.

The original categorization of events was based on frequency of the initiating event alone, and, thus, the allowance or limit was accordingly established based on that high frequency level. With the introduction of additonal assumptions and conditions (initial event and single component failure and/or single operator error), the total event would now fall into a lower frequency/probability category. Thus, less restrictive limits or allowances should be applied in the analysis of transients and accidents. This certainly needs to be considered and evaluated.

GE has evaluated and presented the transients and accidents in this chapter by the more restrictive old allowances and limits of the event categorization presently in effect.

Most events postulated for consideration are already the results of single equipment failures or single operator errors that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operator errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs.

15.0.3.2.1.2 Initiating Event Analysis

(1) the undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow); or

(2) the undesired starting or stopping of any single component; or

(3) the malfunction or maloperation of any single control device; or

(4) any single electrical component failure; or

(5) any single operator error.

Operator error is defined as an active deviation from written operating procedures or nuclear plant standard operating practices. A single operator error is the set of actions that is a direct consequence of a single erroneous decision. The set of actions is limited as follows:

(1) those actions that could be performed by one person;

(2) those actions that would have constituted a correct procedure had the initial decision been correct; and

(3) those actions that are subsequent to the initial operator error and have an effect on the designed operation of the plant, but are not necessarily directly related to the operator error.

15.0.3.2.1.2 Initiating Event Analysis (Continued)

Examples of single operator errors are as follows:

(1) an increase in power above the established flow control power limits by control rod withdrawal in the specified sequences;

(2) the selection and complete withdrawal of a single control rod out of sequence;

(3) an incorrect calibration of an average power range monitor; and

(4) manual isolation of the main steam lines as a result of operator misinterpretation of an alarm or indication.

15.0.3.2.1.3 Single Active Component Failure or Single Operator Failure Analysis

(1) the undesired action or maloperation of a single active component, or

(2) any single operator error where operator errors are defined as in Subsection 15.0.3.2.1.2.

15.0.3.3 Core and System Performance

15.0.3.3.1 Introduction

Section 4.4 (Thermal and Hydraulic Design) describes the various fuel failure mechanisms. Avoidance of unacceptable results (1) and (2) (Subsection 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to data calculation, manufacturing and operating uncertainties. An acceptable criterion was determined to be that 99.9% of the fuel rods in the

15.0.3.3.1 Introduction (Continued)

core would not be expected to experience boiling transition (Reference 2). This criterion is met by demonstrating that incidents of moderate frequency do not result in a minimum critical power ratio (MCPR) less than 1.06 for the initial core and 1.07 for subsequent reload cores. The reactor steady-state CPR operating limit is derived by determining the decrease in MCPR for the most limiting event. All other events result in smaller MCPR decreases and are not reviewed in depth in this chapter. The MCPR during significant abnormal events is calculated using a transient core heat transfer analysis computer program. The computer program is based on a multinode, single channel thermal-hydraulic model which requires simultaneous solution of the partial differential equations for the conservation of mass, energy and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time.

A detailed description of the analytical model may be found in Appendix C of Reference 2. The initial condition assumed for all full power transient MCPR calculations is that the bundle is operating at or above the MCPR limit of 1.20 for the initial core and subsequent reload cores. Maintaining MCPR greater than 1.06 for the initial core and 1.07 for subsequent reload cores is a sufficient, but not necessary, condition to assure that no fuel damage occurs. This is discussed in Section 4.4.

For situations in which fuel damage is sustained, the extent of damage is determined by correlating fuel energy content, cladding temperature, fuel rod internal pressure and cladding mechanical characteristics.

15.0.3.3.1 Introduction (Continued)

These correlations are substantiated by fuel rod failure tests and are discussed in Sections 4.4 and 6.3 (Emergency Core Cooling System).

15.0.3.3.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed within this section have values for input parameters and initial conditions as specified in Table 15.0-2. Analyses which assume data inputs different than these values are designated accordingly in the appropriate event discussion.

15.0.3.3.3 Initial Power/Flow Operating Constraints

The analyses basis for most of the transient safety analyses is the thermal power at rated core flow (100%) corresponding to 105% nuclear boiler rated (NBR) steam flow. This operating point is the apex of a bounded operating power/flow map which, in response to any classified abnormal operational transients, will yield the minimum pressure and thermal margins of any operating point within the bounded map. 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 (104.2% rod line A-D), the lower bound is the zero power line H-J, the right bound is the rated 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 acceptable operational constraints for abnormal operational transient evaluations.

Any other constraint which may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump

15.0.3.3.3 Initial Power/Flow Operating Constraints (Continued)

cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100% NBR, the power/flow map is truncated by the line B- and reactor operation must be confined within the boundary B- C- D- J- L- K- B. If the maximum operating power level has to be limited (such as point F) to satisfy pressure margin criteria, the upper constraint on power/flow is correspondingly reduced to the rod line (such as line F-G), which intersects the power/flow coordinate of the new operating basis. In this case, the operating bounds would be F- G- J- L- K- F. Operation would not be allowed at any point along line F- E, removed from point F, at the derated power but at reduced flow. If, however, operating limitations are imposed by GETAB derived from transient data with an operating basis at point A, the power/flow boundary for 100% NBR licensed power would be B- C- D- J- L- K- B. This power/flow boundary would be truncated by the MCPR operating limit for which there is no direct correlation to a line on the power/flow map. Operation is allowed within the defined power/flow boundary and within the constraints imposed by GETAB. If operation is restricted to point F by the MCPR operating limit, operation at point E would be allowed provided the MCPR limit is not violated.

Consequently, 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.

15.0.3.3.3 Initial Power/Flow Operating Constraints (Continued)

Reactor operation up to the APRM rod block line, which is above the power levels corresponding to the design flow control line except at low drive flows, is assumed for ECCS analysis.

General Compliance or Alternate Approach Assessment for Regulatory Guide 1.49

For commitment and revision number, see Section 1.8.

Regulatory Guide 1.49 requires that the proposed licensed power level be restricted to a reactor core power level of 3800 MWt thermal or less, and that analyses and evaluations in support of the application should be made at 1.02 times the proposed licensed power level.

The rated thermal power for the standard 238 size reactor is 3579 MWt. The safety analyses and evaluations were made for a 104.2% power level of 3729 MWt. Both of these are in compliance with the subject Guide Requirements.

15.0.3.3.4 Results

The results of analytical evaluations are provided for each event. In addition, critical parameters are shown in Table 15.0-1. From the data in Table 15.0-1, an evaluation of the limiting event for that particular category and parameter can be made. In Table 15.0-3, a summary of applicable accidents is provided. This table compares the GE calculated amount of failed fuel to that used in worst-case radiological calculations.

15.0.3.4 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 overpressure below which no damage can occur is defined as the pressure increase over design pressure allowed by Section III of the ASME Boiler and Pressure Vessel Code for the reactor vessel and the high pressure nuclear system piping. Because this ASME Code permits pressure transient up to 10% over design pressure, the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1375 psig (110% x 1250 psig). Comparing the events considered in this section with those used in the mechanical design of equipment reveals that either the accidents are the same or that the accident in this section results in less severe stresses than those assumed for mechanical design.

The Low-Low Set (LLS) Relief Function, armed upon relief actuation of any S/R valve in the second lowest relief setpoint group, will cause a greater magnitude blowdown (in the relief mode) for certain specified S/R valves and a subsequent cycling of a single low set valve. The effect of the LLS design on reactor coolant pressure is demonstrated (Chapter 5) on the MSIV closure event. This is considered bounding for all other pressurization events and, therefore, is not simulated in the analysis presented in this chapter.

15.0.3.5 Radiological Consequences

In this chapter, the consequences of radioactivity release during the three types of events: (a) incidents of moderate frequency (anticipated operational transients); (b) infrequent incidents (abnormal operational transients); and (c) limiting faults (design basis accidents) are considered. For all events whose consequences are limiting, a detailed quantitative evaluation is

15.0.3.5 Radiological Consequences (Continued)

presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event..

For limiting faults (design basis accidents), two quantitative analyses are considered:

(1) The first is based on conservative assumptions considered to be acceptable to the MRC for the purposes of worst-case bounding the event and determining the adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

(2) The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis".

Results for both are shown to be within NRC guidelines.

15.0.4 Nuclear Safety Operational Analysis (NSOA) Relationship

Appendix 15A is a comprehensive, total plant, system-level, qualitative FMEA, relative to all the Chapter 15 events considered, the protective sequences utilized to accommodate the events and their effects, and the systems involved in the protective actions.

Interdependency of analysis and cross-referral of protective actions is an integral part of this chapter and the appendix.

Contained in Appendix 15A is a summary table which classifies events by frequency only (i.e., not just within a given category such as Decrease in Core Coolant Temperature).

15.0.5 References

1. United States Nuclear Regulatory Commission Regulatory Guide 1.70 Revision 3 (Preliminary), September 1975, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Light Water Reactor Edition."

2. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973 (NEDO-10958 and NEDE-10958).

Table 15.0-1

RESULTS SUMMARY OF APPLICABLE TRANSIENT EVENTS

Subsection I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum CPR -	Frequency Category*	Duration of Blowdown: No. of Valves First Blow-down	Duration of Blowdown: Duration of Blow-down (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of Feedwater Heater, Auto Flow Control	111.5	1045	1087	1034	105.8	**	a	0	0
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	124.2	1060	1102	1047	113.7	1.08	a	0	0
15.1.2	15.1-3	Feedwater Cntl Failure, Max Demand	124.3	1161	1193	1159	105	1.15	a	19	5
15.1.3	15.1-4	Pressure Controller Fail - Open	104.2	1138	1161	1136	100	**	a	10	5
15.1.4		Inadvertent Opening of Safety or Relief Valve	See Text								
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temp	See Text								
15.2		INCREASE IN REACTOR PRESSURE	See Text								
15.2.1	15.2-1	Pressure Controller Downscale Failure	136.8	1187	1221	1181	102.6	1.16	a	19	7
15.2.2	15.2-2	Generator Load Rejection, Bypass-On,	128.2	1160	1189	1157	100	1.19	a	19	5
15.2.2	15.2-3	Generator Load Rejection, Bypass-Off,	198.7	1203	1231	12^2	102.7	1.17	b	19	7
15.2.3	15.2-4	Turbine Trip, Bypass-On	114.5	1158	1188	1155	100	1.19	a	19	5
15.2.3	15.2-5	Turbine Trip, Bypass-Off	179.4	1202	1231	1201	101.3	1.19	b	19	7
15.2	15.2-6	Inadvertent MSIV Closure	105.3	1177	1207	1174	100	**	a	19	5
15.2.5	15.2-7	Loss of Condenser Vacuum	111.7	1157	1186	1153	100	**	a	19	5
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104.2	1100	1112	1098	100	**	a	1	5
15.2.6	15.2-9	Loss of All Grid Connections	105	1159	1184	1156	100	**	a	19	7
15.2.7	15.2-10	Loss of All Feedwater Flow	104.2	1045	1086	1034	100	**	a	0	0
15.2.8		Feedwater Piping Break	See Table 15.0-2, event 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								

*Frequency definition is discussed in Subsection 15.0.1.1.
**See Subsection 15.0.1.1.1.
a Moderate frequency
b Infrequent

Table 15.0-1 (Continued)

RESULTS SUMMARY OF APPLICABLE TRANSIENT EVENTS (Continued)

Sub-section I.D.	Figure I.D.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Bottom Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	Minimum CPR -	Frequency Category*	Duration of Blowdown: No. of Valves First Blow-down	Duration of Blowdown: Dura-tion of Blow-down (sec)
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104.1	1046	1087	1035	100	**	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104.2	1141	1155	1129	103	**	a	10	5
15.3.2	15.3-3	Fast Closure of One Main Recirc. Valve	104.2	1135	1149	1133	100	**	a	10	5
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves	104.2	1142	1151	1129	100	**	a	10	5
15.3.3	15.3-5	Seizure of One Recirculation Pump	104.2	1139	1153	1137	100	**	c	10	5
15.3.4		Recirc. Pump Shaft Break	See Subsection 15.3.3								
15.4		REACTIVITY AND POWER DISTRIBUTION ANOMALIES									
15.4.1.1		RWE - Refueling			See Text				b		
15.4.1.2		RWE - Startup			See Text				b		
15.4.2		RWE - At Power			See Text				a		
15.4.3		Control Rod Misoperation	See Subsections 15.4.1 and 15.4.2								
15.4.4	15.4-1	Abnormal Startup of Idle Recirculation Loop	109.3	985	1002	983	118.7	***	a	0	0
15.4.5	15.4-2	Fast Opening of One Main Recirc. Valve	235.7	978	99	974	[illegible]	***	a	0	0
15.4.5	15.4-3	Fast Opening of Both Main Recirc. Valves	162.2	974	990	971	122.4	***	a	0	0
15.4.7		Misplaced Bundle Accident			See Text			1.10	b		
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104.2	1045	1083	1034	100	***	a	0	0
15.5.2		BWR Transients	See appropriate Events in Sections 15.1 and 15.2								

*Frequency classification is discussed in Subsection 15.0.3.1.
**See Subsection 15.0.3.1.1.
***Transients initiated from low power.
a Moderate frequency
b Infrequent
c Unexpected

Table 15.0-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

1. Thermal Power Level (MWt)	
Warranted Value	3579
Analysis Value	3729
2. Steam Flow (lb/hr)	
Warranted Value	$15.40 * 10^6$
Analysis Value	$16.17 * 10^6$
3. Core Flow (lb/hr)	$104.0 * 10^6$
4. Feedwater Flow Rate (lb/sec)	
Warranted Value	4269
Analysis Value	4483
5. Feedwater Temperature (°F)	425
6. Vessel Dome Pressure (psig)	1045
7. Vessel Core Pressure (psig)	1056
8. Turbine Bypass Capacity (% NBR)	35
9. Core Coolant Inlet Enthalpy (But/lb)	528.9
10. Turbine Inlet Pressure (psig)	960
11. Fuel Lattice	P8 x 8R
12. Core Average Gap Conductance (Btu/sec-ft2-°F)	0.1892
13. Core Leakage Flow (%)	12.9
14. Required MCPR Operating Limit	
First Core	1.20
Reload Core	1.20
15. MCPR Safety Limit	
First Core	1.06
Reload Core	1.07
16. Doppler Coefficient ( )¢/°F	
Analysis Data (REDY only)*	0.132

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

Table 15.0-2 (Continued)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

17.	Void Coefficient (-)¢/% Rated Voids Analysis Data for Power Increase Events (REDY only)* Analysis Data for Power Decrease Events (REDY only)*	 14.0 4.0
18	Core Average Rated Void Fraction (%) (REDY only)*	 42.54
19.	Scram Reactivity, $\$\Delta k$ Analysis Data (REDY only)*	 Figure 15.0-2
20.	Control Rod Drive Position versus time	 Figure 15.0-3
21.	Nuclear characteristics used in ODYN simulations	EOEC**
22.	Jet Pump Ratio (M)	2.257
23.	Safety/Relief Valve Capacity (% NBR) at 1210 psig Manufacturer Quantity Installed	108.5 *** 19
24.	Relief Function Delay (sec)	0.4
25.	Relief Function Response Time Constant (sec)	 0.1
26.	Safety Function Delay (sec)	0.0
27.	Safety Function Response Time Constant (sec)	 0.2
28.	Set Points for Safety/Relief Valves Safety Function (psig) Relief Function (psig)	 1175,1185,1195,1205,1215 1125,1135,1145,1155
29.	Number of Valve Groupings Simulated Safety Function (No.) Relief Function (No.)	 5 4

*For transients simulated on the ODYN model, this input is calculated by ODYN.
**EOEC = End of Equilibrium Cycle.
***Applicant to Supply

Table 15.0-2 (Continued)

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

30.	S/R Valve Reclosure Setpoint - Both Modes (% of setpoint)	
	- Maximum Safety Limit (used in analysis)	98
	- Minimum Operational Limit	89
31.	High Flux Trip (% NBR)	
	Analysis setpoint (122 x 1.042)	127.2
32.	High Pressure Scram Setpoint (psig)	1095
33.	Vessel Level Trips (ft above bottom of separator skirt bottom)	
	Level 8 - (L8)(ft)	5.89
	Level 4 - (L4)(ft)	4.04
	Level 3 - (L3)(ft)	2.165
	Level 2 - (L2)(ft)	(-)1.739
34.	APRM Simulated Thermal Power Trip Scram % NBR	
	Analysis Setpoint (114 x 1.042)	118.8
	Time Constant (sec)	7
35.	Recirculation Pump Trip Delay (sec)	0.14
36.	Recirculation Pump Trip Inertia Time Constant for Analysis (sec)***	5
37.	Total Steamline Volume (ft^3)	3850
38.	Set pressure of Recirculation pump trip (psig) (Nominal)	1135

*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 (lb-ft);
n = rated pump speed (rps);
g = gravitational constant (ft/sec^2); and
T_o = pump shaft torque (lb-ft).

Table 15.0-3

SUMMARY OF ACCIDENTS

Subsection I.D.	Title	Failed Fuel Rods GE Calculated Value	NRC Worst-Case Assumption
15.3.3	Seizure of One Recirculation Pump	None	
15.3.4	Recirculation Pump Shaft Break	None	
15.4.9	Rod Drop Accident	<770	770
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	None
15.8	ATWS	SPECIAL EVENT STILL UNDER NEGOTIATION	

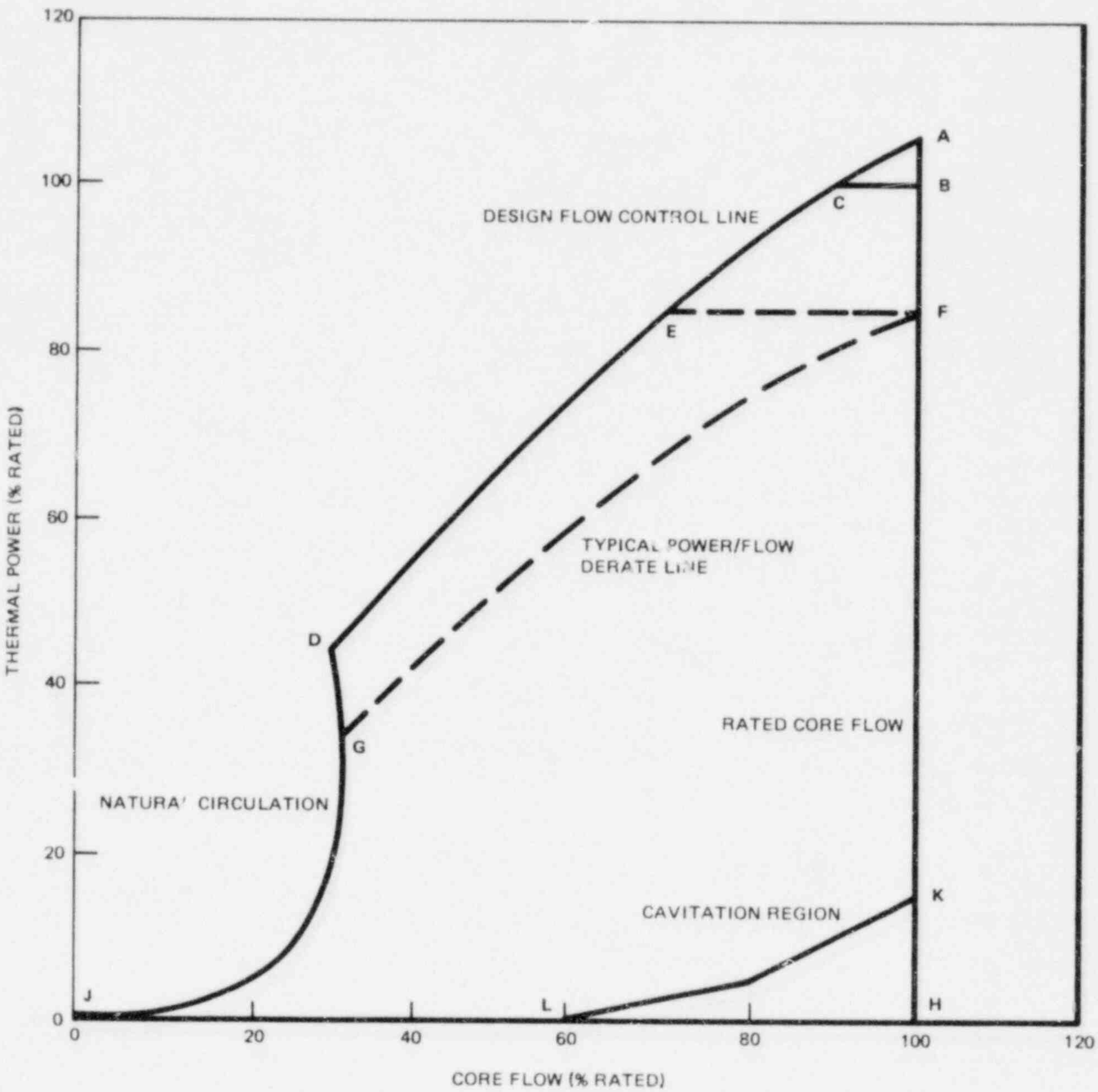

Figure 15.0-1. Typical Power/Flow Map

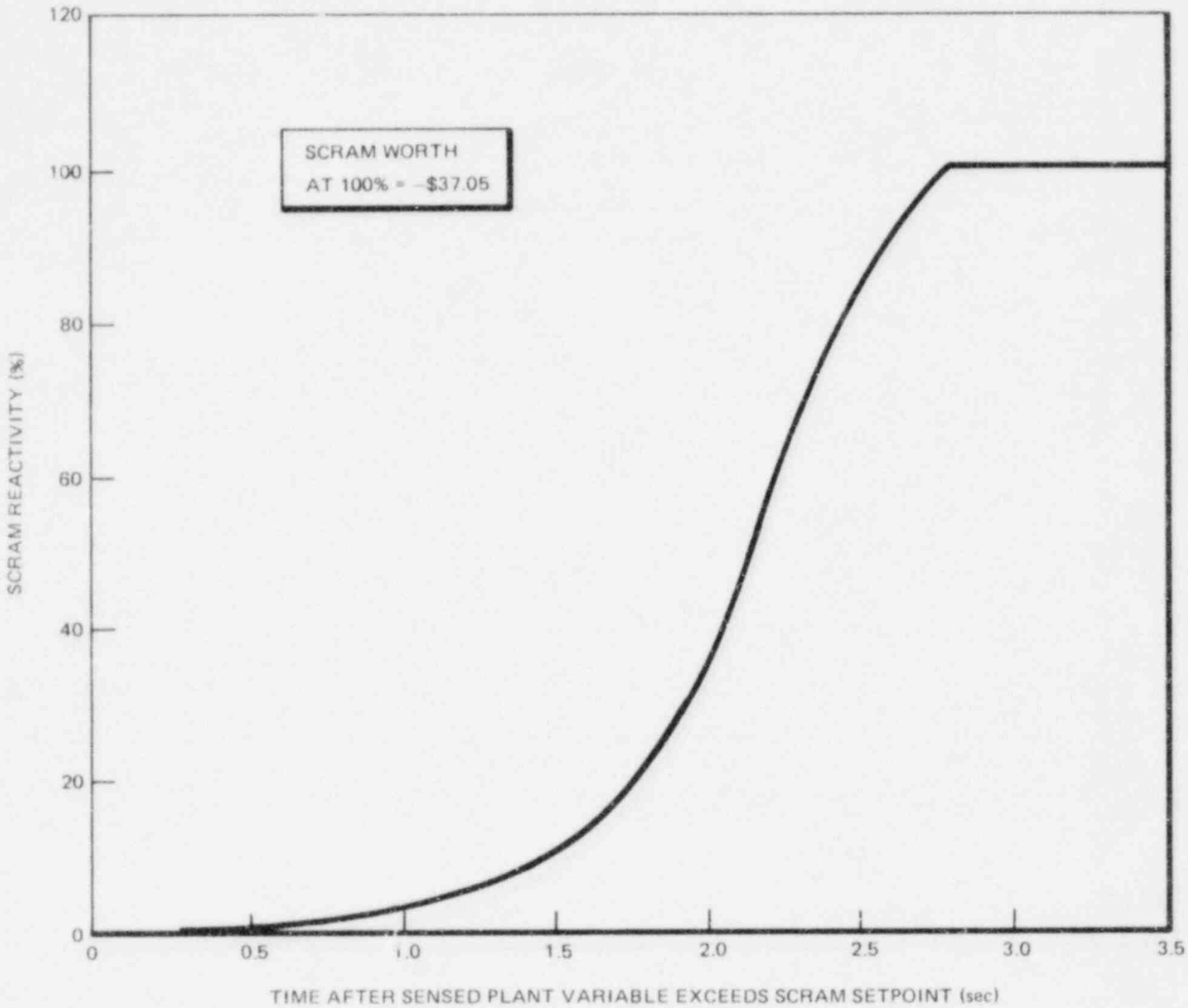

Figure 15.0-2. Scram Reactivity Characteristics

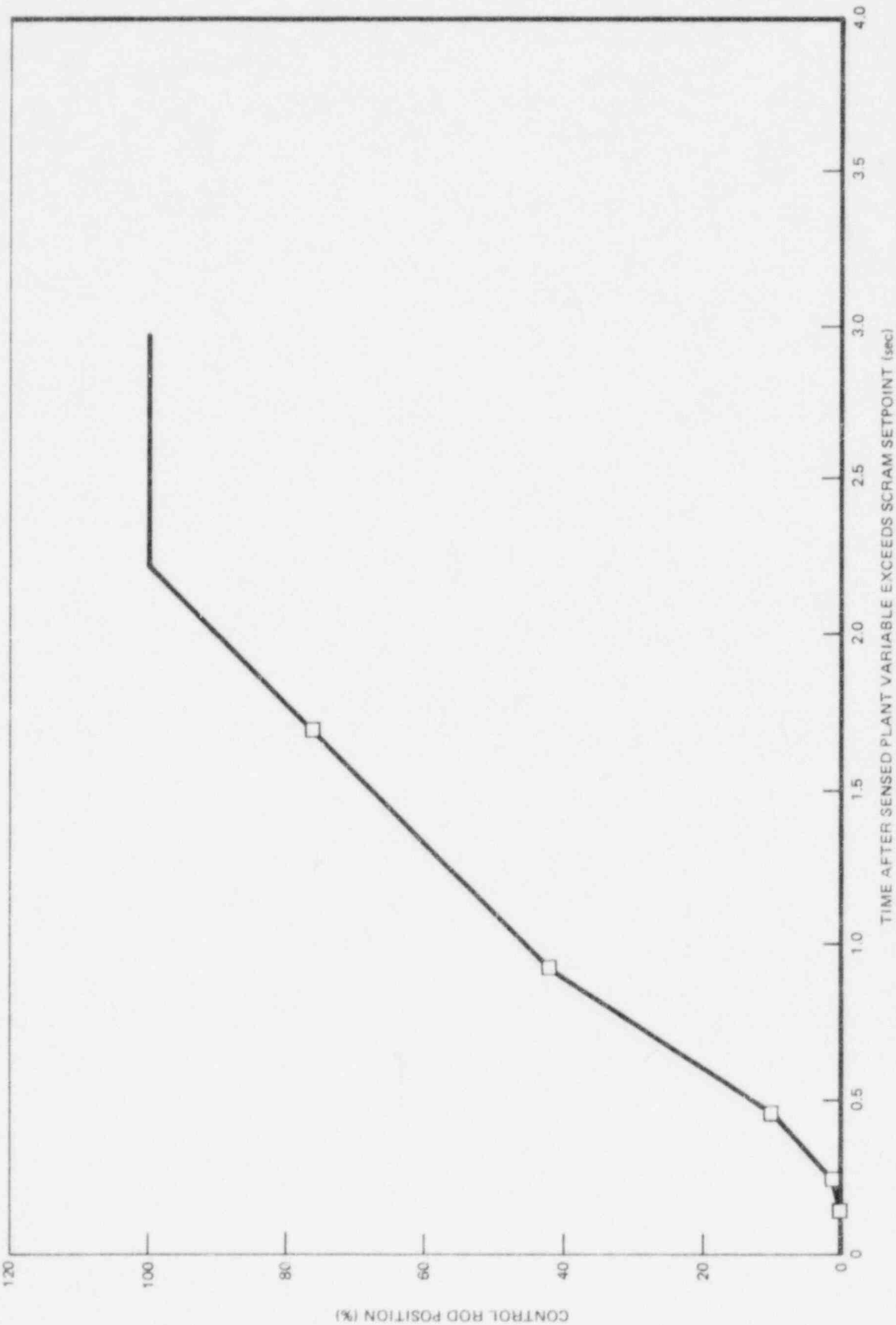

Figure 15.0-3. Scram Time Characteristics

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SECTION 15.1

TABLES

ILLUSTRATIONS

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, and

(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. This event has been conservatively estimated to incur a loss of up to 100°F 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. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control.

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

Tables 15.1-1 and 15.1-2 list the sequence of events for this transient, and its effect on various parameters is shown in Figures 15.1-1 and 15.1-2.

15.1.1.2.1.1 Identification of Operator Actions

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm will alert the operator that he should insert control rods to get back down to the rated flow control line, or that he should reduce flow if in the manual mode. The operator should determine from existing tables the maximum allowable T-G output with feedwater heaters out of service. If reactor scram occurs, as it does in manual flow control mode, the operator should monitor the reactor water level and pressure controls and the T-G auxiliaries during coastdown.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assured 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.

Required operation of Engineered Safeguard Features (ESF) is not expected for either of these transients.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The STPT mentioned in Subsection 15.1.1.2.2 is the mitigating system and is designed to be single failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed. (See Appendix 15A for a detailed discussion of this subject.)

15.1.1.3 Core and System Performance

15.1.1.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 1. This computer model has been verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear, computer-simulated, analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

(1) A point kinetic model is assumed with reactivity feedbacks from control rods (absorption), voids (moderation) and Doppler (capture) effects.

(2) The fuel is represented by three four-node cylindrical elements, each enclosed in a cladding node. One of the cylindrical elements is used to represent core average power and fuel temperature conditions, providing the source of Doppler feedback. The other two are used to represent "hot Spots" in the core, to simulate peak fuel center temperature and cladding temperature.

(3) Four primary system pressure nodes are simulated. The nodes represent the core exit pressure, vessel dome

15.1.1.3.1 Mathematical Model (Continued)

pressure, steamline pressure (at a point representative of the S/R valve location) and turbine inlet pressure.

(4) The active core void fraction is calculated from a relationship between core exit quality, inlet subcooling and pressure. This relationship is generated from multinode core steady-state calculations. A second-order void dynamic model with the void boiling sweep time calculated as a function of core flow and void conditions is also utilized.

(5) Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure and load demand are represented together with their dominant nonlinear characteristics.

(6) The ability to simulate necessary reactor protection system functions is provided.

15.1.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The plant is assumed to be operating at 105% of NBR steam flow and at thermally limited conditions. Both automatic and manual modes of flow control are considered.

The same void reactivity coefficient conservatism used for pressurization transients is applied, since a more negative value conservatively increases the severity of the power increase. The values for both the feedwater heater time constant and the feedwater volume between the heaters and the spargers are adjusted to reduce the time delays, since they are not critical to the calculation of

15.1.1.3.2 Input Parameters and Initial Conditions (Continued)

this transient. The transient is simulated by programming a change in feedwater enthalpy corresponding to a 100°F loss in feedwater heating.

15.1.1.3.3 Results

In the automatic flux/flow control mode, the recirculation flow control system responds to the power increase by reducing core flow so that steam flow from the reactor vessel to the turbine remains essentially constant. In order to maintain the initial steam flow with the reduced inlet temperature, reactor thermal power increases above the initial value and settles at about 110% NBR (106% of initial power), below the flow-referenced APRM simulated thermal power scram setting and core flow is reduced to approximately 80% of rated flow. The MCPR reached in the automatic control mode is greater than for the more limiting manual flow control mode.

The increased core inlet subcooling aids thermal margins, and smaller power increase makes this event less severe than the manual flow control case given below. Nuclear system pressure does not change and, consequently, the reactor coolant pressure boundary is not threatened. If scram occurs, the results become very similar to the manual flow control case. This transient is illustrated in Figure 15.1-1.

In the manual mode, no compensation is provided by core flow, and thus the power increase simulated is greater than in the automatic mode. A scram on high APRM simulated thermal power may occur. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 114% of its initial value and average fuel temperature increases 128°F. The increased core inlet subcooling aids core thermal margins and minimum MCPR remains above the safety limit. Therefore, the design basis is

15.1.1.3.3 Results (Continued)

satisfied. The transient responses of the key plant variables for this mode of operation are shown in Figure 15.1-2.

After the reactor scrams, water level drops to the low level trip point (L3) for recirculation pump trip (not shown in Table 15.1-2).

This transient is less severe from lower initial power levels for two main reasons: (1) lower initial power levels will have CPR values greater than the limiting initial CPR value assumed, and (2) the magnitude of the power rise decreases with lower initial power conditions. Therefore, transients from lower power levels will be less severe.

15.1.1.3.4 Considerations of Uncertainties

Important factors (such as reactivity coefficient, scram characteristics, magnitude of the feedwater temperature change) are assumed to be at the worst configuration so that any deviations seen in the actual plant operation reduce the severity of the event.

15.1.1.4 Barrier Performance

As noted above and shown in Figures 15.1-1 and 15.1-2, 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

Since this event does not result in any additional 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 most severe applicable event is a feedwater controller failure during maximum flow demand. The feedwater controller is forced to its upper limit at the beginning of the event.

15.1.2.1.2 Frequency Classification

This event is considered to be an incident of moderate frequency.

15.1.2.2 Sequence of Events and Systems Operation

15.1.2.2.1 Sequence of Events

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 15.1-3 lists the sequence of events for Figure 15.1-3. The figure shows the changes in important variables during this transient.

15.1.2.2.1.1 Identification of Operator Actions

The operator should:

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

15.1.2.2.1.1 Identification of Operator Actions (Continued)

(2) switch the feedwater controller from auto to manual control in order 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

In order 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 scram and tripping of the main turbine and feedwater pumps, recirculation pump trip (RPT), and low water level initiation of the reactor core isolation cooling (RCIC) system and the high pressure core spray (HPCS) system to maintain long-term water level control following tripping of feedwater pumps.

15.1.2.2.3 The Effect of Single Failures and Operator Errors

In Table 15.1-3, the first sensed event to initiate corrective action to the transient is the vessel high water level (L8) scram. Scram trip signals from Level 8 are designed such that a single failure will neither initiate nor impede a reactor scram trip initiation. Therefore, single failures are not expected to result in a more severe event than analyzed. (See Appendix 15A for a detailed discussion of this subject.)

15.1.2.3 Core and System Performance

15.1.2.3.1 Mathematical Model

The predicted dynamic behavior has been determined using a computer simulated, analytical model of a generic direct-cycle BWR. This model is described in detail in Reference 2. This computer model has been improved and verified through extensive comparison of its predicted results with actual BWR test data.

The nonlinear computer simulated analytical model is designed to predict associated transient behavior of this reactor. Some of the significant features of the model are:

(1) An integrated one-dimensional core model is assumed which includes a detailed description of hydraulic feedback effects, axial power shape changes and reactivity feedbacks.

(2) The fuel is represented by an average cylindrical fuel and cladding model for each axial location in the core.

(3) The steamlines are modeled by eight pressure nodes incorporating mass and momentum balances which will predict any wave phenomenon present in the steamline during pressurization transient.

(4) The core average axial water density and pressure distribution is calculated using a single channel to represent the heated active flow and a single channel to represent the bypass flow. A model, representing liquid and vapor mass and energy conservation and mixture momentum conservation, is used to describe the thermal-hydraulic behavior. Changes in the flow split between the bypass and active channel flow are accounted for during transient events.

15.1.2.3.1 Mathematical Model (Continued)

(5) Principal controller functions such as feedwater flow, recirculation flow, reactor water level, pressure and load demand, are represented together with their dominant non-linear characteristics.

(6) The ability to simulate necessary reactor protection system functions is provided.

(7) The control systems and reactor protection system models are, for the most part, identical to those employed in the point reactor model, which is described in detail in Reference 1 and used in analysis for other transients.

15.1.2.3.2 Input Parameters and Initial Conditions

These analysis have been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-2.

End-of-equilibrium-cycle (all rods out) characteristics are assumed. The S/R valve action is conservatively assumed to occur with higher than nominal setpoints. The transient is simulated by programming an upper limit failure in the feedwater system such that 130% NBR feedwater flow occurs at a system design pressure of 1065 psig.

15.1.2.3.3 Results

The simulated feedwater controller transient is shown in Figure 15.1-3. The high water level turbine trip and feedwater pump trip are initiated at approximately 12 sec. Scram occurs simultaneously and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above the safety limit. The turbine bypass system opens to limit peak pressure in the steamline

15.1.2.3.3 Results (Continued)

near the S/R valves to 1159 psig and the pressure at the bottom of the vessel to about 1193 psig.

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

15.1.2.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief set points, scram stroke time and reactivity characteristics). Expected plant behavior is therefore expected to lead to a less severe transient.

15.1.2.4 Barrier Performance

As noted above, 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.

15.1.2.5 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 Classification

15.1.3.1.1 Identification of Causes

The total steam flow rate to the main turbine resulting from a pressure regulator malfunction is limited by a maximum flow limiter imposed at the turbine controls. This limiter is set to limit maximum steam flow demand to 130% NB rated in the analysis.

If either the controlling pressure regulator or the backup regulator fails to the open position, the turbine admission valves can be fully opened and the turbine bypass valves can be partially opened until the maximum steam flow demand is satisfied.

15.1.3.1.2 Frequency Classification

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

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

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

15.1.3.2.1.1 Identification of Operator Actions

When regulator trouble is preceded by spurious or erratic behavior of the controlling device, it may be possible for the operator to transfer operation to the backup controller in time to prevent the full transient. If the reactor scrams as a result of the isolation caused by the low pressure at the turbine inlet (825 psig) in the run mode, the following sequence of operator actions is expected during the course of the event. Once isolation occurs, the pressure

15.1.3.2.1.1 Identification of Operator Actions (Continued)

will increase to a point where the relief valves 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 and HPCS initiate on low-water level;

(6) secure both HPCS and RCIC when reactor pressure and level are under control;

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

(8) complete the scram report and initiate a maintenance survey of pressure regulator before reactor restart.

15.1.3.2.2 Systems Operation

In order 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 HPCS and RCIC system functions will occur when the vessel water level reaches the L2 setpoint. Normal startup and actuation can take up to 30 seconds before effects are realized.

15.1.3.2.2 Systems Operation (Continued)

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.

15.1.3.2.3 The Effect of Single Failures and Operator Errors

This transient leads to a loss of pressure control such that the increased steam flow demand causes a depressurization. Instrumentation for pressure sensing of the turbine inlet pressure is designed to be single-failure proof for initiation of MSIV closure.

Reactor scram sensing, originating from either the high water level sensors or from the limit switches on the main steamline isolation valves, is designed to be single-failure proof. It is therefore concluded that the basic phenomenon of pressure decay is adequately terminated (see Appendix 15A).

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.1.3.3.2 Input Parameters and Initial Conditions

This transient is simulated by setting the controlling regulator otuput to a high value, which causes the turbine admission valves to open fully and the turbine bypass valves to open partially. Since the controlling and backup regulator outputs are gated by a high value gate, the effect of such a failure in the backup regulator would be exactly the same. A regulator failure with 130% steam flow was simulated as a worst case, since 115% is the normal

15.1.3.3.2 Input Parameters and Initial Conditions (Continued)

maximum flow limit. A reactor scram and trip of the main and feedwater turbines occur on high water level.

A 5-sec isolation valve closure instead of a 3-sec 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.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-2.

15.1.3.3.3 Results

Figure 15.1-4 shows 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. In this simulation, the depressurization rate is large enough such that water level swells to the sensed level trip setpoint (L8), initiating reactor scram and main turbine and feedwater turbine trips. Position switches on the turbine stop valves initiate recirculation pump trip (RPT). After the turbine trip, the failed pressure regulator now signals the bypass to open to full bypass flow of 35% NBR steam flow. After the 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.

15.1.3.3.3.1 Considerations of Uncertainties

If the maximum flow limiter were set higher or lower than normal, there would result a faster or slower loss in nuclear steam pressure. The rate of depressurization may be limited by the bypass capacity, but is is unlikely.

For example, the turbine valves will open to the valves-wide-open state, admitting slightly more than the rated steam flow and, with the limiter in this analyses set to fail at 130%, we would expect something less than 23% to be bypassed. This is therefore not a limiting factor on this plant. If the rate of depressurization does change, it will be terminated by the low turbine inlet pressure trip setpoint.

Depressurization rate has a proportional effect upon the voiding action of the core and the flashing in the vessel bulkwater regions. If the rate is low enough, the water level may not swell to the high water level trip setpoint and the isolation will occur earlier when pressure at the turbine decreases below 825 psig. The reactor will scram as a result of the main steam isolation valve closure.

15.1.3.4 Barrier Performance

Barrier performance analyses were not required since 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. Peak pressure in the bottom of the vessel reaches 1161 psig, which is below the ASME code limit of 1375 psig for the reactor coolant pressure boundary. Vessel dome pressure reaches 1138 psig, just slightly above the setpoint of the second pressure relief group. Minimum vessel dome pressure of 790 psig occurs at about 30 sec.

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 actuaction. 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. Opening and closing circuitry at the individual valve level (as opposed to groups of valves) is subject to a single failure. It is therefore simply postulated that a failure occurs and the event is analysed 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 but, due to a lack of a comprehensive data basis, it is being analyzed as an incident of moderate frequency.

15.1.4.2 Sequence of Events and Systems Operation

15.1.4.2.1 Sequence of Events

Table 15.1-5 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.

15.1.4.2.3 The Effect of Single Failures and Operator Errors

Failure of additional components (e.g., pressure regulator, feedwater flow controller) is discussed elsewhere in this chapter. In addition, a detailed discussion of such effects is given in Appendix 15A.

15.1.4.3 Core and System Performance

15.1.4.3.1 Mathematical Model

The reactor model briefly described in Subsection 15.1.1.3.1 was previously used to simulate this event in earlier FSARs. This model is discussed in detail in Reference 2. It was determined that this event is not limiting from a core performance standpoint. Therefore, a qualitative presentation of results is described below.

15.1.4.3.2 Input Parameters and Initial Conditions

It is assumed that the reactor is operating at an initial power level corresponding to 105% of rated steamflow conditions when a S/R valve is advertently opened. Manual recirculation flow control

15.1.4.3.2 Input Parameters and Initial Conditions (Continued)

is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 520 metric tons per hour.

15.1.4.3.3 Qualitative Results

The opening of a S/R valve 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 pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power 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.

15.1.4.4 Barrier Performance

As discussed above, 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 consequences 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. Since this activity is contained in the primary containment, there will be no exposures to operating personnel. Since 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

15.1.4.5 Radiological Consequences (Continued)

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 of 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 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 moderator 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

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

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 will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-6.

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 since the nuclear system pressure is too high to permit operation of the shutdown cooling (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 would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

15.1.6.2.3 Effect of Single Failures and Operator Action

No single failures can cause this event to be more severe. If the operator takes action, the slow power rise will be controlled in the normal manner. If no operator action is taken, scram will terminate the power increase before thermal limits are reached. (See Appendix 15A for details.)

15.1.6.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise would be terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here.

15.1.6.4 Barrier Performance

As noted above, 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

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

15.1.7 References

1. R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor." April 1973 (NEDO-10802).

2. "Qualification of the One-Dimensional Core Transient Model for BWR," October 1978 (NEDO-24154).

Table 15.1-1

SEQUENCE OF EVENTS FOR FIGURE 15.1-1

Time (sec)	Event
0	Initiate a 100°F temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level but AFC system automatically reduces core flow to maintain initial steam flow.
100	Reactor variables settle into new steady state.

Table 15.1-2

SEQUENCE OF EVENTS FOR FIGURE 15.1-2

Time (sec)	Event
0	Initiate a 100°F temperature reduction into the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level and steam flow.
7	Turbine control valves start to open to regulate pressure.
36	APRM initiates reactor scram on high thermal power.
44.0	Narrow Range (NR) sensed water level reaches Level 3 (L3) setpoint. Recirculation pumps tripped to low frequency speed.
>50 (est)	Recirculation Pump Trip initiated due to Level 2 Trip. (not included in simulation).
>50 (est)	Wide Range (WR) sensed water level reaches Level 2 (L2) setpoint.
>80 (est)	HPCS/RCIC flow enters vessel (not simulated).
>90 (est)	Reactor variables settle into limit cycle.

Table 15.1-3

SEQUENCE OF EVENTS FOR FIGURE 15.1-3

Time (sec)	Event
0	Initiate simulated failure of 130% upper limit at system design pressure of 1065 psig on feedwater flow.
11.8	L8 vessel level setpoint initiates reactor scram and trips main turbine and feedwater pumps.
11.9	Recirculation pump trip (RPT) actuated by stop valve position switches.
11.9	Main turbine bypass valves opened due to turbine trip.
13.2	Safety/relief valves open due to high pressure.
18.2	Safety/relief valves close.
>20 (est)	Water level dropped to low water level setpoing (Level 2).
>50 (est)	RCIC and HPCS flow into vessel (not simulated).

Table 15.1-4

SEQUENCE OF EVENTS FOR FIGURE 15.1-4

Time (sec)	Event
0	Simulate steam flow demand to 130%.
2.1	Turbine control valves wide open.
2.28	Vessel water level (L8) trip initiates reactor scram and main turbine and feedwater turbine trips.
2.28	Turbine trip initiates bypass operation to full flow.
2.29	Main turbine stop valves reach 90% open position and initiates recirculation pump trip (RPT).
2.38	Turbine stop valves closed. Turbine bypass valves opening to full flow.
2.4	Recirculation pump motor circuit breakers open causing decrease in case flow to natural circulation.
5.2	Group 1 S/R valves open again to relieve decay heat.
10.2	Group 1 S/R valves close again.
25	Vessel water level reaches L2 setpoint.
28	Low turbine inlet pressure trip initiates main steamline isolation.
33	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
55 (est)	HPCS and RCIC flow enters vessel (not simulated).

Table 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

Time (sec)	Event
0	Initiate opening of 1 S/R valve.
0.5 (est.)	Relief flow reaches full flow.
15 (est.)	System establishes new steady-state operation.

Table 15.1-6

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.

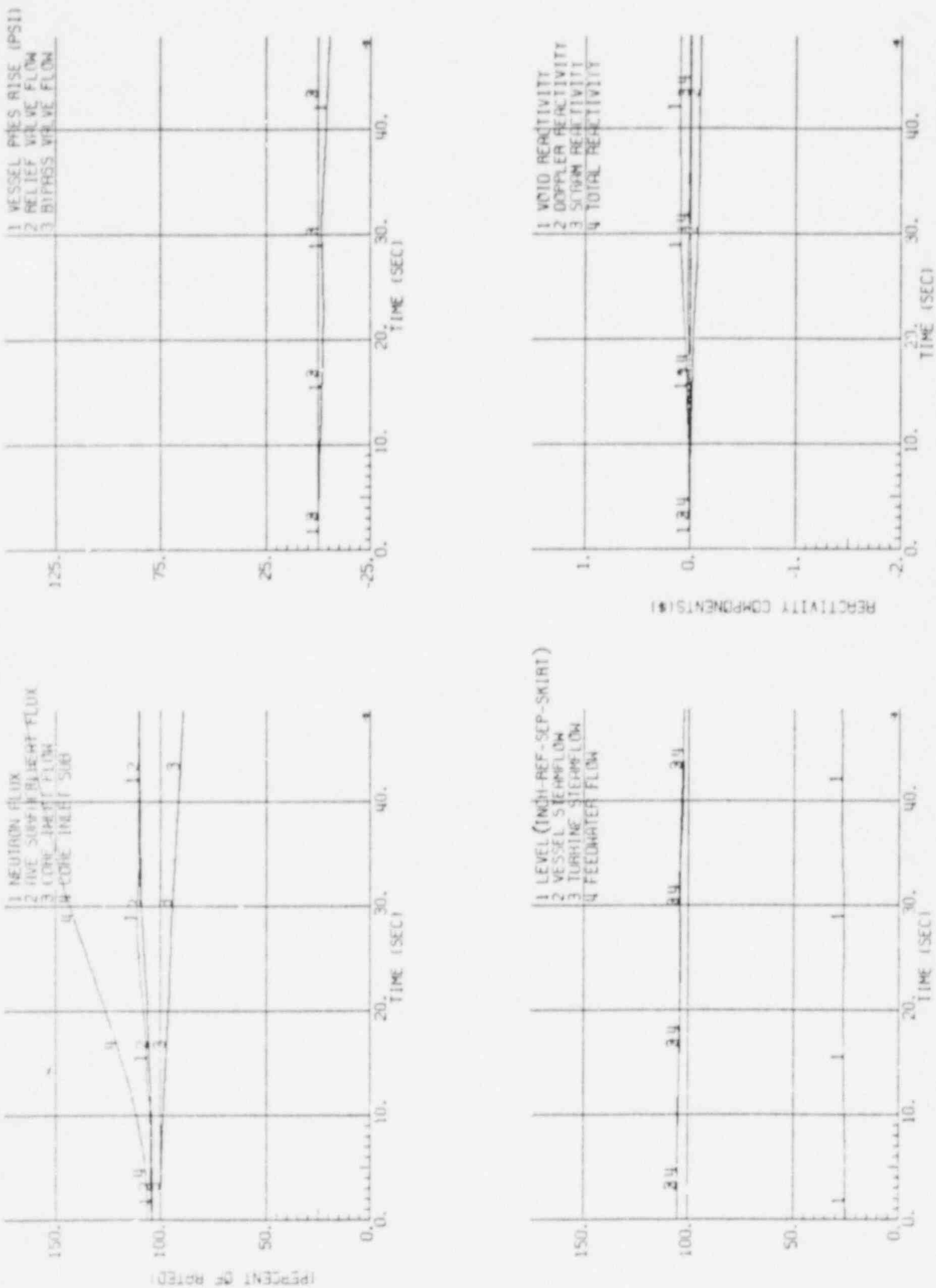

Figure 15.1-1. Loss of 100°F Feedwater Heating
(Automatic Flow Control Mode)

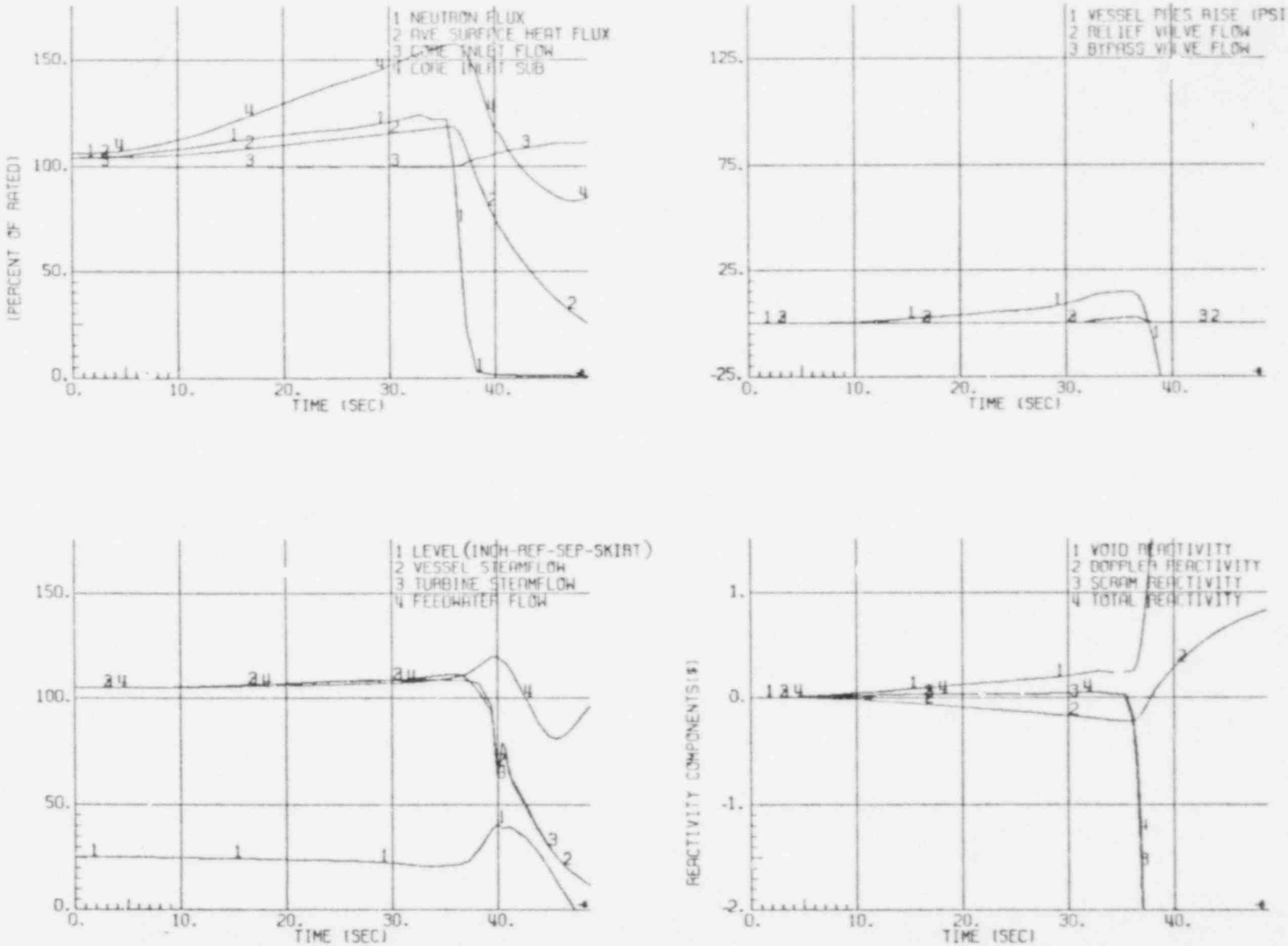

Figure 15.1-2. Loss of 100°F Feedwater Heating
(Manual Flow Control Mode)

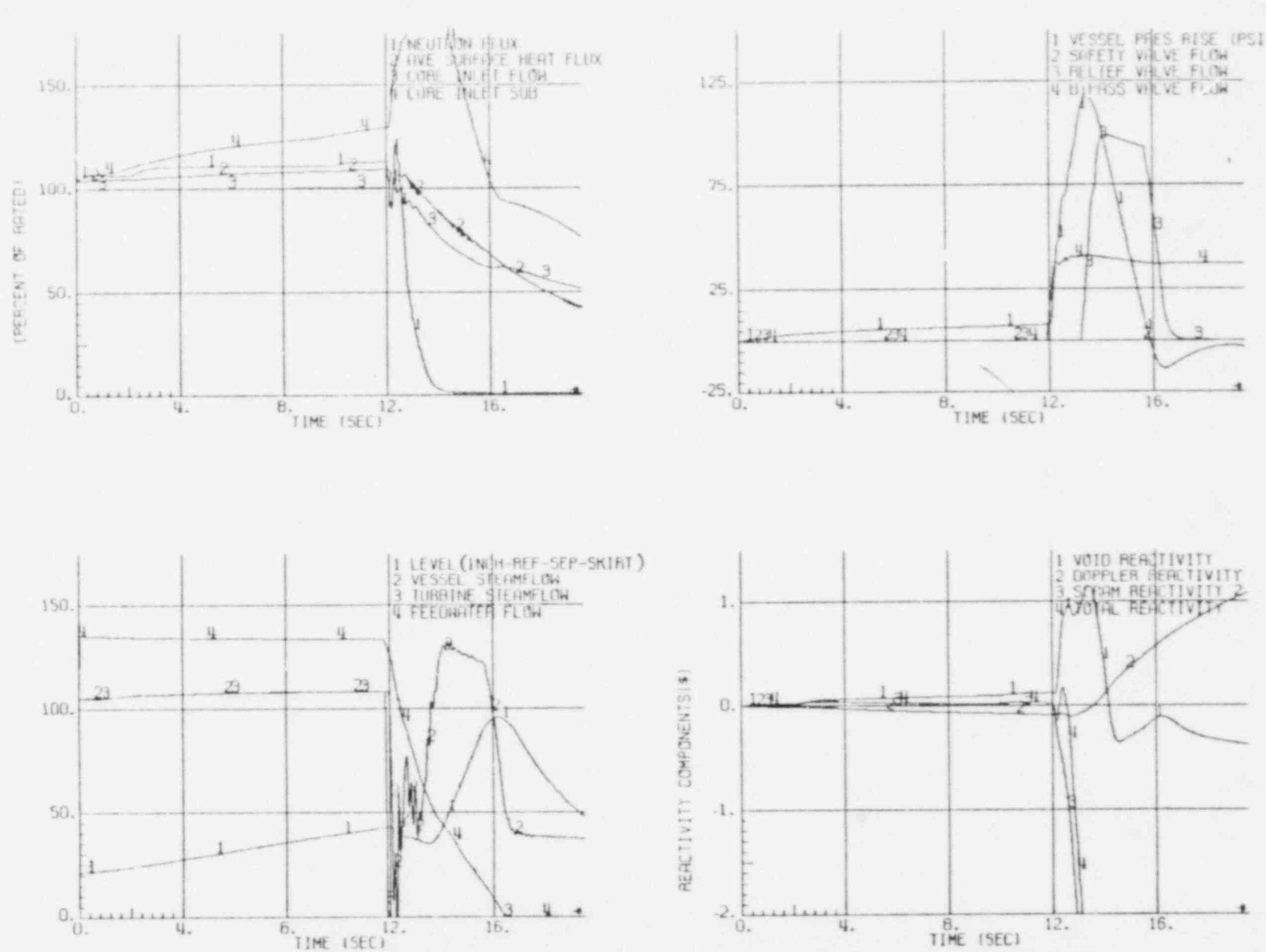

Figure 15.1-3. Feedwater Controller Failure, Maximum Demand, With High Water Level Trips

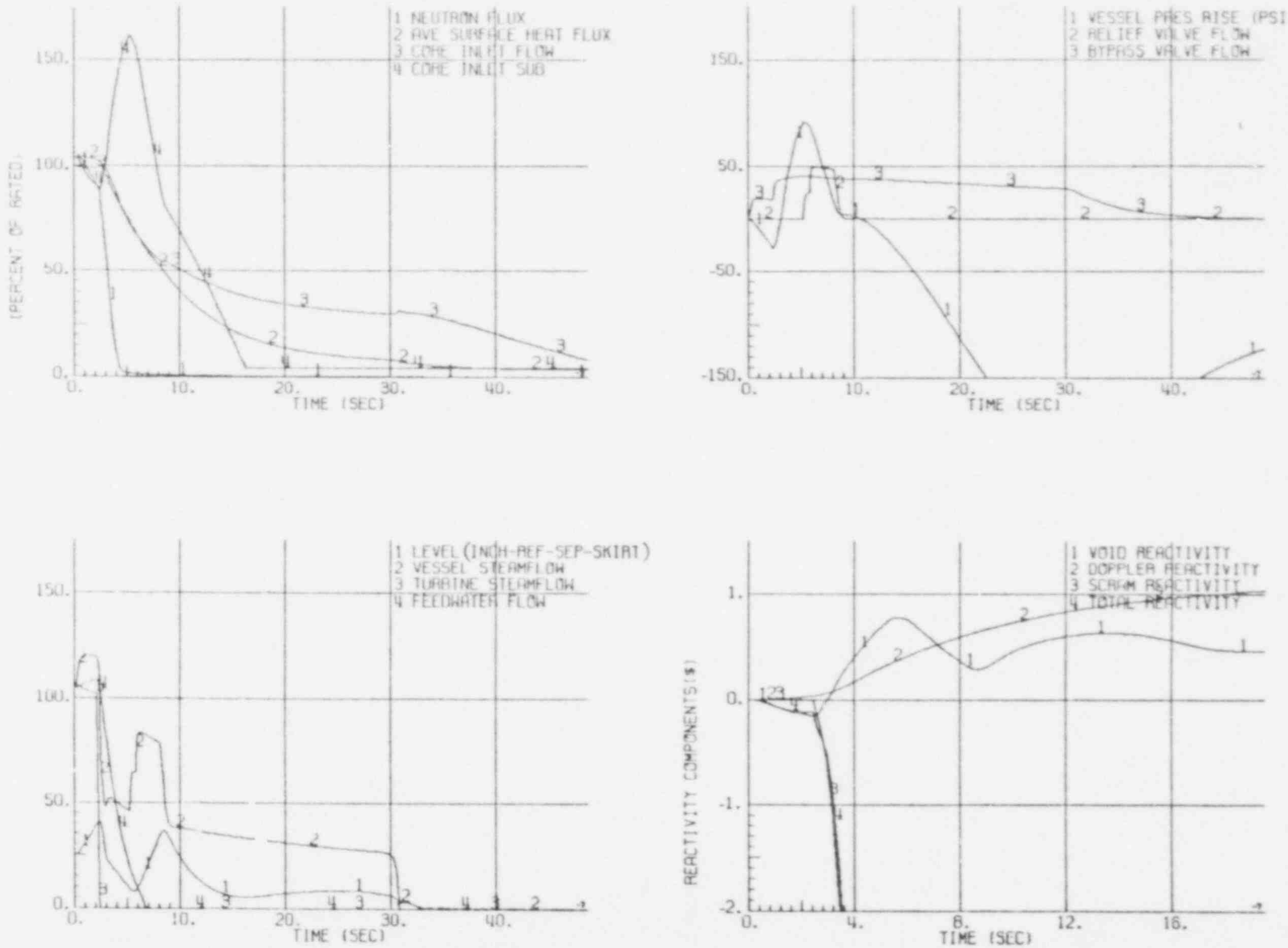

Figure 15.1-4. Pressure Regulator Failure Open to 130%

SECTION 15.2

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SECTION 15.2

TABLES

ILLUSTRATIONS

ILLUSTRATIONS (Continued)

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

Two identical pressure regulators are provided to maintain primary system pressure control. They independently sense pressure just upstream of the main turbine stop valves and compare it to two separate setpoints to create proportional error signals that produce each regulator output. The output of both regulators feeds into a high value gate. The regulator with the highest output controls the main turbine control valves. The lowest pressure setpoint gives the largest pressure error and, thereby, largest regulator output. The backup regulator is set 5 psi higher giving a slightly smaller error and a slightly smaller effective output of the controller.

It is assumed for purposes of this transient analysis that a single failure occurs which erroneously causes the controlling regulator to close the main turbine control valves and thereby increases reactor pressure. If this occurs, the backup regulator is ready to take control.

It is also assumed for purpose of this transient analysis that a single failure occurs which causes a downscale failure of the pressure regulation demand to zero (e.g., high value gate downscale failure). Should this occur, it could cause full closure of turbine control 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 high neutron flux scram setpoint is reached.

15.2.1.1.2 Frequency Classification

15.2.1.1.2.1 One Pressure Regulator Failure - Closed

This event is treated as a moderate frequency event.

15.2.1.1.2.2 Pressure Regulation Downscale Failure

This event is treated as a moderate frequency event.

15.2.1.2 Sequence of Events and System Operation

15.2.1.2.1 Sequence of Events

15.2.1.2.1.1 One Pressure Regulator Failure - Closed

Postulating a failure of the primary or controlling pressure regulator in the closed mode as discussed in Subsection 15.2.1.1.1 will cause the turbine control valves to close moementarily. The pressure will increase, because the reactor is still generating the initial steam flow. The backup regulator will reopen the valves and reestablish steady-state operation above the initial pressure equal to the setpoint difference of 5 psi.

15.2.1.2.1.2 Pressure Regulation Downscale Failure

Table 15.2-1 lists the sequence of events for Figure 15.2-1.

15.2.1.2.1.3 Identification of Operator Actions

15.2.1.2.1.3.1 One Pressure Regulator Failure - Closed

The operator should verify that the backup regulator assumes proper control. However, these actions are not required to terminate the event as discussed in Subsection 15.2.1.2.3.2.

15.2.1.2.1.3.2 Pressure Regulation Downscale Failure

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; 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 One Pressure Regulator Failure - Closed

Normal plant instrumentation and control are assumed to function. This event requires no protection system or safeguard systems operation.

15.2.1.2.2.2 Pressure Regulation Downscale Failure

Analysis of this event assumes normal functioning of plant instrumentation and controls, and plant protection and reactor protection systems. Specifically, this transient takes credit for high

15.2.1.2.2.2 Pressure Regulation Downscale Failure (Continued)

neutron flux scram to shut down the reactor. High system pressure is limited by the pressure relief valve system operation.

15.2.1.2.3 The Effect of Single Failures and Operator Errors

15.2.1.2.3.1 One Pressure Regulation Failure - Closed

The nature of the first assumed failure produces a slight pressure increase in the reactor until the backup regulator gains control, since no other action is significant in restoring normal operation. If we fail the backup regulator at this time (the second assumed failure), the control valves would start to close, raising reactor pressure to the point where a flux scram trip would be initiated to shut down the reactor. This event is similar to that described in Subsection 15.2.1.2.3.2 (details in Appendix 15A).

15.2.1.2.3.2 Pressure Regulation Downscale Failure

This transient leads to a loss of pressure control such that the zero steam flow demand causes a pressurization. The high neutron flux scram is the mitigating system and is designed to be single-failure proof. Therefore, single failures are not expected to result in a more severe event than analyzed (details in Appendix 15A).

15.2.1.3 Core and System Performance

15.2.1.3.1 Mathematical Model

The nonlinear, dynamic model described briefly in Subsection 15.1.2.3.1 is used to simulate this event.

15.2.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions listed in Table 15.0-1.

15.2.1.3.3 Results

15.2.1.3.3.1 One Pressure Regulator Failure - Closed

Qualitative evaluation provided only.

Response of the reactor during this regulator failure is such that pressure at the turbine inlet increases quickly, in less than approximately 2 sec, due to the sharp closing action of the turbine control valves which reopen when the backup regulator gains control. This pressure disturbance in the vessel is not expected to exceed flux or pressure scram trip setpoints.

15.2.1.3.3.2 Pressure Regulation Downscale Failure

A pressure regulation downscale failure is simulated at 105% NBR steam flow condition in Figure 15.2-1.

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 157% NBR by the reactor scram. Peak fuel surface heat flux does not exceed 102.6% of its initial value. MCPR for this transient is still above the safety MCPR limit. Therefore, the design basis is satisfied.

15.2.1.3.4 Consideration of Uncertainties

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief

15.2.1.3.4 Consideration of Uncertainties (Continued)

setpoints, scram stroke time, and worth characteristics). Expected plant behavior is, therefore, expected to reduce the actual severity of the transient.

15.2.1.4 Barrier Performance

15.2.1.4.1 One Pressure Regulator Failure - Closed

As noted above, 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 (Table 15.0-1); therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2 Pressure Regulation Downscale Failure

Peak pressure at the S/R valves reaches 1181 psig. The peak nuclear system pressure reaches 1221 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

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

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.

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 Bypass Failure

This event is categorized as an infrequent incident with the following characteristics:

Frequency: 0.0036/plant year
MTBE: 278 years

Frequency Basis: Thorough searches 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.

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 15.2-2.

15.2.2.2.1.2 Generator Load Rejection with Failure of Bypass

A loss of generator electrical load at high power with bypass failure produces the sequence of events listed in Table 15.2-3.

15.2.2.2.1.3 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) reactor peak power and pressure; and

(5) verify relief valve operation.

15.2.2.2.2 System Operation

15.2.2.2.2.1 Generator Load Rejection with Bypass

In order 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, recirculation pump trip (RPT) 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 Bypass

Same as Subsection 15.2.2.2.2.1, except that failure of the main turbine bypass valves is assumed for the entire transient.

15.2.2.2.3 The Effect of Single Failures and Operator Errors

Mitigation of pressure increase (the basic nature of this transient) is accomplished by the reactor protection system functions. TCV trip scram and RPT are designed to satisfy the single-failure criterion. An evaluation of the most limiting single failure (i.e., failure of the bypass system) was considered in this event. Details of single failure analysis can be found in Appendix 15A.

15.2.2.3 Core and System Performance

15.2.2.3.1 Mathematical Model

The computer model described in Subsection 15.1.2.3.1 was used to simulate this event.

15.2.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with the plant conditions tabulated in Table 15.0-2.

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

The closure chacteristics of the TCVs are assumed such that the valves operate in the full arc (FA) mode and have a full stroke closure time, from fully open to fully closed, of 0.15 sec.

Auxiliary power is independent of any T-G overspeed effects and is continuously supplied at rated frequency, assuming automatic fast transfer to auxiliary power supplies.

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 initiation of HPCS and RCIC core cooling system functions are not included in the simulation. Should these events occur, they will follow sometime after the primary concerns of fuel thermal margin and overpressure

15.2.2.3.2 Input Parameters and Initial Conditions (Continued)

effects have occurred, and are expected to be less severe than those already experienced by the system.

15.2.2.3.3 Results

15.2.2.3.3.1 Generator Load Rejection with Bypass

Figure 15.2-2 shows the results of the generator trip from 105% rated steam flow conditions. Peak neutron flux rises 24% above NB rated conditions.

The average surface heat flux shows no increase from its initial value, and MCPR does not significantly decrease below its initial value.

15.2.2.3.3.2 Generator Load Rejection with Failure of Bypass

Figure 15.2-3 shows that, for the case of bypass failure, peak neutron flux reaches about 199% of rated, and average surface heat flux reaches 102.7% of its initial value. Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria, and the MCPR limit is permitted to fall below the safety limit for the incidents of moderate frequency. However, the MCPR for this event, with a value of 1.17, is well above the safety limit.

15.2.2.3.4 Consideration of Uncertainties

The full stroke closure time of the turbine control valve of 0.15 sec is conservative. Typically, the actual closure time is more like 0.2 sec. Clearly, the less time it takes to close, the more severe the pressurization effect.

15.2.2.3.4 Consideration of Uncertainties (Continued)

All systems utilized for protection in this event were assumed to have the most conservative allowable response (e.g., relief setpoints, scram stroke time and worth characteristics). Anticipated plant behavior is therefore expected to reduce the actual severity of the transient.

15.2.2.4 Barrier Performance

15.2.2.4.1 Generator Load Rejection

Peak pressure remains within normal operating range and no threat to the barrier exists.

15.2.2.4.2 Generator Load Rejection with Failure of Bypass

Peak pressure at the S/R valves reaches 1202 psig. The peak nuclear system pressure reaches 1233 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1375 psig.

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.

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. In order 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 the Bypass

This transient disturbance is categorized as an infrequent incident. Frequency is expected to be as follows:

Frequency: 0.0064/plant year
MTBE: 156 years

15.2.3.1.2.2 Turbine Trip with Failure of the Bypass (Continued)

Frequency Basis: As discussed in the section "Generator Load Rejection with Bypass Failure", the failure rate of the bypass is 0.0048. Combining this with the turbine trip frequency of 1.22 events/plant year yields the frequency of 0.0064/plant year.

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 15.2-4.

15.2.3.2.1.2 Turbine Trip with Failure of the Bypass

Turbine trip at high power with bypass failure produces the sequence of events listed in Table 15.2-5.

15.2.3.2.1.3 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;

15.2.3.2.1.3 Identification of Operator Actions (Continued)

(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 <850 psig;

(6) secure the RCIC operation if auto initiation occurred due to low water level;

(7) monitor control rod drive positions and insert both the IRMs and SRMs;

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

(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.

15.2.3.2.2.1 Turbine Trip (Continued)

Turbine stop valve closure initiates recirculation pump trip (RPT), thereby terminating the jet pump drive 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 the Bypass

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

15.2.3.2.2.3 Turbine Trip at Low Power with Failure of the Bypass

Same as Subsection 15.2.3.2.2.1, except that failure of the main turbine bypass system is assumed.

It should be noted that below 40% NBR power level, a main stop valve scram trip inhibit signal derived from the first-stage pressure of the turbine is activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of shutting down the reactor. All other protection system functions remain operational as before and credit is taken for those protection system trips.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater than 40% NBR

Mitigation of pressure increase, the basic nature of this transient, is accomplished by the reactor protection system functions. Main stop valve closure scram trip and RPT are designed to satisfy the single-failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less than 40% NBR

Same as Subsection 15.2.3.2.3.1, except RPT and stop valves closure scram trip is normally inoperative. Since protection is still provided by high flux, high pressure, etc., these will also continue to function and scram the reactor should a single failure occur.

15.2.3.3 Core and System Performance

15.2.3.3.1 Mathematical Model

The computer model described in Subsection 15.1.2.3.1 was used to simulate these events.

15.2.3.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

Turbine stop valves full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 40% of NBR power level.

15.2.3.3.2 Input Parameters and Initial Conditions (Continued)

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

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip

A turbine trip with the bypass system operating normally is simulated at 105% NBR steam flow conditions in Figure 15.2-4.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 114.5% of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105% NB rated steam flow conditions in Figure 15.2-5.

Peak neutron flux reaches 180% of its rated value, and average surface heat flux reaches 101% of its initial value. Therefore, this transient is less severe than the generator load rejection with failure of bypass transient described in Subsection 15.2.2.3.3.2.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 40% of rated power, the turbine stop valve closure and turbine control valve closure scrams and Recirculation pump trip

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power (Continued)

(RPT) are automatically bypassed. At these lower power levels, turbine first-stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high neutron flux or high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief setpoints are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve setpoints and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Nuclear characteristics for all-rods-out EOEC conditions is assumed.

(3) Minimum specified valve capacities are utilized for overpressure protection.

(4) Setpoints of the S/R valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1188 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1158 psig. 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 the Bypass

The S/R valves 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 1231 psig at the vessel bottom; therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1202 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in Subsection 15.2.3.3.3.3.

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 much 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 MSLIV 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 (MSLIV) 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: manual action (purposely or inadvertent); spurious signals such as low pressure, low reactor water level, low condenser vacuum; and, finally, equipment malfunctions such as faulty valves or operating mechanisms. A closure of one MSLIV may cause an immediate closure of all the other MSLIVs depending on reactor conditions. If this occurs, it is also included in this category. During the main steamline isolation valve closure, position switches on the valves provide a reactor scram if the valves in two or more main steamlines are less than 90% 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 MSLIV may be closed at a time for testing purposes; this is

15.2.4.1.2.2 Closure of One Main Steamline Isolation Valve (Continued)

done manually. Operator error or equipment malfunction may cause a single MSLIV to be closed inadvertently. If reactor power is greater than about 80% when this occurs, a high flux scram may result (if all MSLIVs close as a result of the single closure, the event is considered as a closure of all MSLIVs).

15.2.4.2 Sequence of Events and Systems Operation

15.2.4.2.1 Sequence of Events

Table 15.2-6 lists the sequence of events for Figure 15.2-6.

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/HPCS auto starts on the impending low reactor water level condition;

(4) switch the feedwater controller to the manual position;

(5) initiate operation of the RHR system in the steam condensing mode only.

(6) When the reactor vessel level has recovered to a satisfactory level, secure RCIC/HPCS,

15.2.4.2.1.1 Identification of Operator Actions (Continued)

(7) when the reactor pressure has decayed sufficiently for RHR operation, put it into service per procedure,

(8) before resetting the MSLIV isolation, determine the cause of valve closure,

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

(10) not reset and open MSLIVs unless conditions warrant and be sure the pressure regulator setpoint is above vessel pressure, and

(11) 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

MSLIV closures initiate 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 MSLIV at any given time will not initiate a reactor scram. 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.

15.2.4.2.3 The Effect of Single Failure and Operator Errors

Mitigation of pressure increase is accomplished by initiation of the reactor scram via MSIV position switches and the protection system. Relief valves also operate to limit system pressure. All of these aspects are designed to single-failure criteria, and additional single failures would not alter the results of this analysis.

Failure of a single relief valve to open is not expected to have any significant effect. Such a failure is expected to result in less than a 5 psi increase in the maximum vessel pressure rise. The peak pressure will still remain considerably below 1375 psig. The design basis and performance of the pressure relief system is discussed in Chapter 5.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in Subsection 15.1.2.3.1 was used to simulate these transient events.

15.2.4.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

The main steam isolation valves close in 3 to 5 sec. The worst case (the 3-sec closure time) is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90% open. Closure of these valves inhibits steam flow to the feedwater turbines terminating feedwater flow. Because of the loss of feedwater flow, water level within the vessel decreases sufficiently to initiate trip of the recirculation pump and initiate the HPCS and RCIC systems.

15.2.4.3.3 Results

15.2.4.3.3.1 Closure of All Main Steamline Isolation Valves

Figure 15.2-6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steamlines while the reactor is operating at 105% of NBR steam flow. Neutron flux increases slightly, and fuel surface heat flux shows no increase.

Water level decreases sufficiently to cause a recirculation system trip on the Level 3 (L3) trip at 1.9 sec and initiation of the HPCS and RCIC system on the Level 2 (L2) trip at some time greater than 10 sec. However, there is a delay up to 30 sec before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins.

15.2.4.3.3.2 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

15.2.4.3.3.2 Closure of One Main Steamline Isolation Valve (Continued)

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-sec closure of one main steam isolation valve during 105% rated power conditions, the steam flow disturbance raises 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.

Inadvertent closure of one or all of the isolation valves while the reactor is shut down (such as operating state C, as defined in Appendix 15A) will produce no significant transient. Closures during plant heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) discussed in Subsection 15.2.4.3.3.1.

15.2.4.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Nuclear characteristics for all-rod-out EOEC conditions is assumed.

15.2.4.3.4 Considerations of Uncertainties (Continued)

(3) Minimum specified valve capacities are utilized for overpressure protection.

(4) Setpoints of the S/R valves are assumed to be 1 to 2% higher than the valve's nominal setpoint.

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 2.7 sec 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 1207 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steamline is 1174 psig.

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 unacceptably high operating power level, a flux of pressure scram will result. The main turbine bypass system will continue to regulate system pressure via the other three "live" steamlines.

15.2.4.5 Radiological Consequences

15.2.4.5.1 General Observations

The radiological impact of many transients and accidents involves the consequences: (1) which do not lead to fuel rod damage as a direct result of the event itself; (2) additionally, many events

15.2.4.5.1 General Observations (Continued)

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, (3) 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; and (4) the time duration of the transient varies from several minutes to more than four hours.

The above observations lead to the realization that radiological aspects can involved 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 time tables (4 hours), the radiological impact is greater.

In order 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 No. 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

15.2.4.5.2.1 Fission Product Release from Fuel (Continued)

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 1).

Since 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 1. The results of this analyses are presented in Table 15.2-7, which was used in evaluating the radiological dose consequences in this section.

15.2.4.5.2.2 Fission Product Release to Environment

Since 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%. Reference 2 contains a description of the containment purge release model used. The integrated release to the environment is presented in Table 15.2-8.

15.2.4.5.3 Radiological Exposures

15.2.4.5.3.1 Offsite

The radiological doses for this event are presented in Table 15.2-9. It should be noted that the radiological doses in the above table

15.2.4.5.3.1 Offsite (Continued)

are exposures per event. For the isolation transient, this event is expected to occur 2.5 times per year; therefore, the yearly commitments for these transients will be ~2.5 times the individual values.

15.2.4.5.3.2 Onsite: Egress Dose

Egress doses from four containment work areas are calculated: (1) the control and instrumentation (C&I) panels area, (2) the traversing incore probe (TIP) area; (3) the reactor water sample station area; and (4) the control rod drive (CRD) hydraulic control unit area. These areas are considered to give a representative sampling of the potential egress situations from containment following an isolation transient. The CRD and sample station areas are among the most heavily occupied, and egress from the C&I panels and TIP areas should result in the highest doses. The locations of the areas described above, as well as times required for egress, are given in Table 15.2-10. The egress doses are summarized in Table 15.2-11. The doses are given with and without the air shower concept.

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 15.2-12.

15.2.5.1.2 Frequency Classification

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 15.2-13 lists the sequence of events for Figure 15.2-7.

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 <850 psig,

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

15.2.5.2.1.1 Identification of Operator Actions (Continued)

(7) monitor control rod drive positions and insert both the IRMs and SRMs,

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

(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 pressure are designated in Table 15.2-14.

15.2.5.2.3 The Effect of Single Failures and Operator Errors

This event does not lead to a general increase in reactor power level. Mitigation of power increase is accomplished by the protection system initiation of scram.

Failure of the integrity of the condenser gas treatment system is considered to be an accident situation and is described in Subsection 15.7.1.

Single failures will not affect the vacuum monitoring and turbine trip devices which are redundant. The protective sequences of the anticipated operational transient are shown to be single-failure proof (see Appendix 15A for details).

15.2.5.3 Core and System Performance

15.2.5.3.1 Mathematical Model

The computer model described in Subsection 15.1.2.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions listed in Table 15.0-2 unless otherwise noted.

Turbine stop valves full stroke closure time is 0.1 sec.

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% 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 2 in. Hg/sec vacuum decay rate. Thus, the bypass system is available for several seconds, since the bypass is signaled to close at a vacuum level of about 10 in. Hg less than the stop valve closure.

15.2.5.3.3 Results

Under this hypothetical 2 in. Hg/sec vacuum decay condition, the turbine bypass valve and main steamline isolation valve closure would follow main turbine and feedwater turbine trips about 5 sec after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steamline isolation valve closure tends to be minimal, since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steamline flow. Figure 15.2-7

15.2.5.3.3 Results (Continued)

shows the transient expected for this event. It is assumed that the plant is initially operating at 105% of NBR steam flow conditions. Peak neutron flux reaches 114% of NBR power, while average fuel surface heat flux shows no increase. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Considerations of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the main steamline isolation valves and bypass valves. While these are the major events occurring, other resultant actions will include scram and recirculation pump trip (RPT) (from stop valve closure) and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is directly dependent upon the rate at which the vacuum pressure is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problem produces a very slow rate of loss of vacuum (minutes, not seconds). (See Table 15.2-12.) If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSLIVs, will occur.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

15.2.5.3.4 Considerations of Uncertainties (Continued)

Other uncertainties in these analyses involved protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

(1) Slowest allowable control rod scram motion is assumed.

(2) Nuclear characteristics for all-rod-out EOEC conditions is assumed.

(3) Minimum specified valve capacities are utilized for overpressure protection.

(4) Setpoints of the S/R valves are assumed to be at the upper limit of Technical Specifications for all valves.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1186 psig at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1157 psig. A comparison of these values to those for Turbine Trip at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and 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

15.2.5.5 Radiological Consequences (Continued)

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 these consequences of this event.

15.2.6 Loss of Offsite AC Power

15.2.6.1 Identification of Causes and Frequency Classification

15.2.6.1.1 Identification of Causes

15.2.6.1.1.1 Loss of Auxiliary Power Transformer

Causes for interruption or loss of the auxiliary power transformer can arise from normal operation or malfunctioning of transformer protection circuitry. These can include high transformer oil temperature, reverse or high current operation, as well as opertor error which trips the transformer breakers.

15.2.6.1.1.2 Loss of All Grid Connections

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

15.2.6.1.2 Frequency Classification

15.2.6.1.2.1 Loss of Auxiliary Power Transformer

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

15.2.6.1.2.2 Loss of All Grid Connections

This transient disturbance is categorized 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 Auxiliary Power Transformer

Table 15.2-15 lists the sequence of events for Figure 15.2-8.

15.2.6.2.1.2 Loss of All Grid Connections

Table 15.2-16 lists the sequence of events for Figure 15.2-9.

15.2.6.2.1.3 Identification of Operator Actions

The operator should maintain the reactor water level by use of the RCIC or HPCS system, control reactor pressure by use of the relief valves and steam condensing mode of the RHR. Verify that the turbine d-c oil pump is operating satisfactory to prevent turbine bearing damage. Also, he should verify proper switching and loading of the emergency diesel generators.

15.2.6.2.1.3 Identification of Operator Actions (Continued)

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) following the scram, verify all rods in.,

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

(3) check that both RCIC and HPCS start when reactor vessel level drops to the initiation point after the relief opens,

(4) break vacuum before the loss of sealing steam occurs,

(5) check T-G auxiliaries during coastdown,

(6) when both the reactor pressure and level are under control, secure both HPCS and RCIC as necessary,

(7) continue cooldown per the normal procedure; and

(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 Auxiliary Power Transformer

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

15.2.6.2.2.1 Loss of Auxiliary Power Transformer (Continued)

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 auxiliary transformer) provide the following simulation sequence (assuming a solid state reactor trip system):

(1) all electrical pumps are tripped at a reference time, t=0, with normal coastdown times for the recirculation pumps.

(2) Within 8 sec, the loss of main condenser circulating water pumps causes condenser vacuum to drop to the main turbine and feedwater turbine trip setting, causing stop valve closure and scram when the stop valves are less than 90% open, assuming 0.5 in. Hg/sec vacuum decay rate. However, scram, main turbine and feedwater turbine tripping may occur earlier than this time, if water level reaches the high water level (Level 8) setpoint before 8 sec.

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

Operation of the HPCS and RCIC system functions are not simulated in this analysis. Their 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 All Grid Connections

Same as Subsection 15.2.6.2.2.1 with the following additional concern:

The loss of all grid connections is another feasible, although improbable, way to lose all auxiliary power. This event would add a generator load rejection to the above sequence at time, t=0. The load rejection immediately forces the turbine control valves closed, causes a scram and initiates recirculation pump trip (RPT) (already tripped at reference time t=0).

15.2.6.2.3 The Effect of Single Failures and Operator Errors

Loss of the auxiliary power transformer in general leads to a reduction in power level due to rapid pump coastdown with pressurization effects due to turbine trip occurring after the reactor scram has occurred. Additional failures of the other systems assumed to protect the reactor would not result in an effect different from those reported. Failures of the protection systems have been considered and satisfy single-failure criteria and, as such, no change in analyzed consequences is expected. See Appendix 15A for details on single failure analysis.

15.2.6.3 Core and System Performance

15.2.6.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate the "Loss of Auxiliary Power Transformer" event and the computer model described in Subsection 15.1.2.3.1 was used to simulate the "Loss of All Grid Connections" event.

Operation of the RCIC or HPCS systems is not included in the simulation of this transient, since startup of these pumps does not permit flow in the time period of this simulation.

15.2.6.3.2 Input Parameters and Initial Conditions

15.2.6.3.2.1 Loss of Auxiliary Power Transformer

These analyses have been performed, unless otherwise noted, with plant conditions listed in Table 15.0-2 and under the assumed systems constraints described in Subsection 15.2.6.2.2.

15.2.6.3.2.2 Loss of All Grid Connections

Same as Subsection 15.2.6.3.2.1.

15.2.6.3.3 Results

15.2.6.3.3.1 Loss of Auxiliary Power Transformer

Figure 15.2-8 shows graphically the simulated transient. The initial portion of the transient is similar to the recirculation pump trip transient. At 4 sec turbine trip, scram, and feedwater turbines trip on high water level. Main steamline isolation valves and turbine bypass valves close at 28 sec on their condenser vacuum setpoint.

Sensed level drops to the RCIC and HPCS initiation setpoint at approximately 27 sec after loss of auxiliary power. The RHRS, in the steam condensing mode, is initiated to dissipate the heat.

There is no significant increase in fuel temperature or decrease in the operating MCPR value, fuel thermal margins are not threatened and the design basis is satisfied.

15.2.6.3.3.2 Loss of All Grid Connections

Loss of all grid connections is a more general form of loss of auxiliary power. It essentially takes on the characteristic response of the standard full load rejection discussed in Subsection 15.2.2. Figure 15.2-9 shows graphically the simulated event.

15.2.6.3.4 Consideration of Uncertainties

The most conservative characteristics of protection features are assumed. Any actual deviations in plant performance are expected to make the results of this event less severe.

Operation of the RCIC or HPCS systems is not included in the simulation of the first 50 sec of this transient. Startup of these pumps occurs in the latter part of this time period, but these systems have no significant effect on the results of this transient.

The trip of the feedwater turbines may occur earlier than simulated if the inertia of the condensate and booster pumps is not sufficient to maintain feedwater pump suction pressure above the low suction pressure trip setpoint. The simulation assumes sufficient inertia and, thus, the feedwater pumps are not tripped until the time that level reaches the high water level trip setpoint (L8).

Following main steamline isolation, the reactor pressure is expected to increase until the S/R valve setpoints are reached. During this time, the valves operate in a cyclic manner to discharge the decay heat to the suppression pool.

15.2.6.4 Barrier Performance

15.2.6.4.1 Loss of Auxiliary Power 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.

15.2.6.4.2 Loss of All Grid Connections

Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints. The pressure in the dome is limited to a maximum value of 1184 psig, well below the vessel pressure limit of 1375 psig.

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 these 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, feedwater controller failures, operator errors, or reactor system variables such as 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 15.2-10.

15.2.7.2.1.1 Identification of Operator Actions

The operator should ensure RCIC and HPCS actuation so that water inventory is maintained in the reactor vessel. Initiate the steam condensing mode of the RHR system to complement the RCIC system. 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 HPCS and RCIC initiation,

15.2.7.2.1.1 Identification of Operator Actions (Continued)

(3) verify that the recirculation pumps trip on reactor low level,

(4) secure HPCS when reactor level and pressure are under control,

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

(6) monitor turbine coastdown, break vacuum as necessary, and

(7) complete scram report and survey maintenance requirements.

15.2.7.2.2 Systems Operation

Loss of feedwater flow results in a proportional reduction of vessel inventory, causing the vessel water level to drop. The first corrective action is the low level (L3) scram trip actuation. Reactor protection system responds within 1 sec after this trip to scram the reactor. The low level (L3) scram trip function meets the single-failure criterion.

Containment isolation, if water level reaches (L1), would also initiate a main steamline isolation valve position scram trip signal as part of the normal isolation event. The reactor, however, is already scrammed and shut down by this time.

15.2.7.2.3 The Effect of Single Failures and Operator Errors

The nature of this event, as explained above, results in a lowering of vessel water level. Key corrective effects to shut down

15.2.7.2.3 The Effect of Single Failures and Operator Errors (Continued)

the reactor are automatic and designed to satisfy the single-failure criterion; therefore, any additional failure in these shutdown methods would not aggravate or change the simulated transient (see Appendix 15A for details).

15.2.7.3 Core and System Performance

15.2.7.3.1 Mathematical Model

The computer model described in Subsection 15.1.1.3.1 was used to simulate this event.

15.2.7.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

15.2.7.3.3 Results

The results of this transient simulation are shown in Figure 15.2-10. Feedwater flow terminates at approximately 5 sec. 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 5 sec. 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 recirculation pumps are tripped to low frequency speed. Vessel water level continues to drop to the L2 trip. At this time, the recirculation pumps are tripped, and the HPCS and RCIC operation is initiated. MCPR remains considerably above the safety limit, since increases in heat flux are not experienced.

15.2.7.3.4 Considerations of Uncertainties

End of Equilibrium-cycle scram characteristics are assumed.

This transient is more severe from high power conditions, because the rate of level decrease is greatest and the amount of stored and decay heat to be dissipated are highest.

Operation of the RCIC or HPCS systems is not included in this simulation of the first 50 sec of this transient, since startup of these pumps occurs in the latter part of this time period and, therefore, these systems have no significant effects on the results of this transient except perhaps as discussed in Subsection 15.2.7.2.3.

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

Normally, in evaluating component failure considerations associated with the RHRS-Shutdown Cooling mode operation, active pumps

15.2.9 Failure of RHR Shutdown Cooling (Continued)

or instrumentation (all of which are redundant for safety system portions of the RHRS aspects) would be assumed to be the likely failed equipment. For purposes of worst-case analysis, the single recirculation loop suction valve to the redundant RHRS loops is assumed to fail. This failure would, of course, still leave two complete RHRS loops for LPCI, pool, and containment cooling minus the normal RHRS-Shutdown Cooling loop connection. Although the valve could be manually manipulated open, it is assumed failed indefinitely. If it is now assumed that the single active failure criterion is applied, the plant operator has one complete RHRS loop available with the further selective worst-case assumption that the other RHRS loop is lost.

Recent analytical evaluations of this event have required additional worst-case assumptions. These included:

(1) loss of all offsite AC power;

(2) utilization of safety shutdown equipment only; and

(3) operator involvement only after 10 min after coincident assumptions.

These accident-type assumptions certainly would change the initial incident (malfunction of RHRS suction valve) from a moderate frequency incident to a classification in the design basis accident status. However, the event is evaluated as a moderate frequency event with its subsequent limits.

15.2.9.1 Identification of Causes and Frequency Classification

15.2.9.1.1 Identification of Causes

The plant is operating at 105% NBR steam flow when a long-term loss of offsite power occurs, causing multiple SRV actuation (Subsection 15.2.6) and subsequent heatup of the suppression pool. Reactor vessel depressurization is initiated to bring the reactor pressure to approximately 100 psig. Concurrent with the loss of offsite power, an additional (divisional) single failure occurs which prevents the operator from establishing the normal shutdown cooling path through the RHR shutdown cooling lines. The operator then establishes a shutdown cooling path for the vessel through the ADS valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event. However, for the following reasons, it could be considered an infrequent incident:

(1) no RHR valves have failed in the shutdown cooling mode in BWR total operating experience, and

(2) the set of conditions evaluated is for multiple failure as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-18.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are identical to those described in Subsection 15.2.6 (Loss of Offsite Power Event with Isolation/Scram). The operator should do the following:

(1) at approximately 10 min into the transient, initiate suppression pool cooling (again for purposes of this analysis, it is assumed that only one RHR heat exchanger is available);

(2) initiate RPV shutdown depressurization by manual actuation of 3 ADS valves;

(3) after the RPV is depressurized to approximately 100 psig, the operator should attempt to open one of the two RHR shutdown cooling suction valves (these attempts are assumed unsuccessful); and

(4) at 100 psig RPV pressure, the operator establishes a closed cooling path as described in the notes for Figure 15.2-11.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation, credit is taken for the plant and reactor protection systems and/or the ESF utilization.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst-case single failure (Loss of Division Power) has already been analyzed in this event. Therefore, no single failure or

15.2.9.2.3 The Effect of Single Failures and Operator Errors (Continued)

operator error can make the consequences of this event any worse (see Appendix 15A for details).

15.2.9.3 Core and System Performance

15.2.9.3.1 Methods, Assumptions, and Conditions

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. During this time, MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-min time period assumed for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action.

15.2.9.3.2 Mathematical Model

In evaluating this event, the important parameters to consider are reactor depressurization rate and suppression pool temperature. Models used for this evaluation are described in References 3 and 4.

15.2.9.3.3 Input Parameters and Initial Conditions

Table 15.2-19 shows the input parameters and initial conditions used in evaluation of this event.

15.2.9.3.4 Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using other, normal shutdown cooling equipment. In cases where both of the RHRS shutdown cooling suction valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-12). An evaluation has been performed assuming the worst single failure that could disable the RHRS shutdown cooling valves.

The analysis demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. The evaluation assures that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the safety function can be accomplished, assuming a worst-case single failure.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (Reference 5 and Figure 15.2-11).

The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety grade systems. Even if it is additionally postulated that all of the ADS or relief valves discharge piping also fails, the shutdown cooling function would eventually be accomplished as the cooling water would run directly out of the ADS or safety/relief valves, flooding into the drywell.

15.2.9.3.4 Results (Continued)

The systems have suitable redundancy in components such that, for onsite electrical power operation (assuming offsite power is not available) and for offsite electrical power operation (assuming onsite power is also not available), the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 psia and 125°F) conditions.

15.2.9.3.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCS systems, together with the nuclear boiler pressure relief system.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event (loss of offsite power), which results in reactor isolation and subsequent relief valve actuation and suppression pool heatup. For this postulated condition, the reactor is shut down and the reactor vessel pressure and temperature are reduced to and maintained at saturated conditions at approximately 100 psig. The reactor vessel is depressurized by manually opening selected SRVs. Reactor vessel makeup water is automatically provided via the RCIC/HPCS systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

15.2.9.3.4.1 Full Power to Approximately 100 psig (Continued)

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC/HPCS and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

(1) the vessel is at 100 psig and saturated conditions;

(2) a worst-case single failure is assumed to occur (i.e., loss of a division of emergency power); and

(3) there is no offsite power available.

In the event that the RHR's shutdown suction line is not available because of single failure, the first action to be taken will be to maintain the 100 psig level while personnel gain access and effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling suction line cannot be repaired, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

The RHR shutdown cooling line valves are in two divisions (Division 1 = the outboard valve, and Division 2 = the inboard valve) to satisfy containment isolation criteria. For evaluation

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown (Continued)

purposes, the worst-case failure is assumed to be the loss of a division of emergency power, since this also prevents actuation of one shutdown cooling line valve. Engineered safety feature equipment available for accomplishing the shutdown cooling function includes (for the selected path):

ADS (DC Division 1 and DC Division 2)

RHR Loop (A) (Division 1)

HPCS (Division 3)

RCIC (DC Division 1)

LPCS (Division 1)

Since availability or failure of Division 3 equipment does not affect the normal shutdown mode, normal shutdown cooling is easily available through equipment powered from only Divisions 1 and 2. It should be noted that, conversely, the HPCS system is always available for coolant injections if either of the other two divisions fails. For failure of Divisions 1 or 2, the following systems are assumed functional:

(1) Division 1 Fails, Divisions 2 and 3 Functional:

Failed Systems	Functional Systems
RHR Loop (A)	HPCS
LPCS	ADS
	RHR Loops B and C
	RCIC

15.2.9.3.4.2 Approximately 100 psig to Cold Shutdown (Continued)

Assuming the single failure is a failure of Division 1 emergency power, the safety function is accomplished by establishing one of the cooling loops described in Activity C1 of Figure 15.2-11.

(2) Division 2 Fails, Divisions 1 and 3 Functional:

Failed Systems	Functional Systems
RHR Loops B and C	HPCS
	ADS
	RHR Loop A
	RCIC
	LPCS

Assuming the single failure is the failure of Division 2, the safety function is accomplished by establishing one of the cooling loops described in Activity C2 of Figure 15.2-11. Figures 15.2-13, 15.2-14, 15.2-15 and 15.2-16 show RHR loops A, B and C (simplified).

Using the above assumptions and following the depressurization rate shown in Figure 15.2-17, the suppression pool temperature is shown in Figure 15.2-18.

15.2.9.4 Barrier Performance

As noted above, 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. Release of coolant to the containment occurs via SRV actuation. Release of radiation to the environment is described below.

15.2.9.5 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 these consequences of this event.

15.2.10 References

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

2. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

3. T. Y. Fukushima, "HEX01 User Manual", July 1976 (NEDE-23014).

4. W. I. Bilanin, R. J. Bodily, and G. A. Cruz, "The General Electric Mark III Pressure Suppression Containment System Analytical Model (Supplement 1)", September 1975 (NEDO-20533, Supplement 1).

5. Letter, R. S. Boyd to I. F. Stuart; dated November 12, 1975, Subject: Requirements Delineated for RHRs - Shutdown Cooling System--Single Failure Analysis.

Table 15.2-1

SEQUENCE OF EVENTS FOR FIGURE 15.2-1

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.
2.3	Recirculation pump drive motors are tripped due to high dome pressure.
2.4	Safety/relief valves open due to high pressure.
6.1	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips
6.2	Main turbine stop valves closed.
9.3	Safety/relief valves close.
9.65	Group 1 safety/relief valves open again to relieve decay heat.
>15 (est)	Group 1 safety/relief valves close.

Table 15.2-2

SEQUENCE OF EVENTS FOR FIGURE 15.2-2

Time (sec)	Event
(-) 0.015 (approx.)	Turbine-generator detection of loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure and main turbine bypass system operation.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.
0.1	Turbine bypass valves start to open.
1.5	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

Table 15.2-3

SEQUENCE OF EVENTS FOR FIGURE 15.2-3

Time (sec)	Event
(-) 0.015	Turbine-generator detection of loss of electrical load.
0	Turbine-generator load rejection sensing devices trip to initiate turbine control valve fast closure.
0	Turbine bypass valves fail to operate.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.
1.2	Safety/relief valves open due to high pressure.
5.1	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.4	Safety/relief valves close.
9.3	Group 1 safety/relief valves open again to relieve decay heat.
>10.0 (est)	Group 1 safety/relief valves close again.

Table 15.2-4

SEQUENCE OF EVENTS FOR FIGURE 15.2-4

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine trip initiates bypass operation.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip and a recirculation pump trip (RPT).
0.1	Turbine stop valves close.
0.1	Turbine bypass valves start to open to regulate pressure.
1.6	Safety/relief valves open due to high pressure.
4.0	Vessel water level (L8) trip initiates trip of the feedwater turbines.
6.9	Safety/relief valves close.

Table 15.2-5

SEQUENCE OF EVENTS FOR FIGURE 15.2-5

Time (sec)	Event
0	Turbine trip initiates closure of main stop valves.
0	Turbine bypass valves fail to operate.
0.01	Main turbine stop valves reach 90% open position and initiate reactor scram trip and a recirculation pump trip (RPT).
0.1	Turbine stop valves close.
1.2	Safety/relief valves open due to high pressure.
5.1	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.4	Safety/relief valves close.
9.2	Group 1 safety/relief valves open again to relieve decay heat.
>10.0 (est)	Group 1 safety/relief valves close again.

Table 15.2-6

SEQUENCE OF EVENTS FOR FIGURE 15.2-6

Time (sec)	Event
0	Initiate closure of all main steamline isolation valves (MSLVI).
0.3	MSLIVs reach 90% open.
0.3	MSLIV position trip scram initiated.
1.9	Recirculation pump drive motors are tripped due to low water Level 3 (L3) trip.
2.7	Safety/relief valves open due to high pressure.
8.1	Safety/relief valves close.
9.1	Group 1 safety/relief valves open again to relieve decay heat.
>10 (est)	Vessel water level reaches L2 setpoint.
>10 (est)	Group 1 safety/relief valves close again.
>40 (est)	HPCS and RCIC flow into vessel (not included in simulation).

Table 15.2-7

POST-TRANSIENT RELEASE RATE TO THE CONTAINMENT WITH SUPPRESSION POOL CLEANUP

(μ Ci/sec)

Isotope	8-Hr	12-Hr	1-Day	2-Day	10-Day	30-Day
I131	3.70-4	2.20-4	2.71-5	6.53-7	0	0
I133	7.12-4	3.79-4	3.63-5	6.16-7	0	0
Kr85m	175+3	5.47+2	5.13+1	2.34+0	0	0
Kr85	3.03+2	1.66+2	7.36+1	5.39+1	1.69+1	3.44+0
Kr87	1.57+2	1.26+1	7.59-2	0	0	0
Kr88	2.29+3	5.15+2	2.24+1	3.22-1	0	0
Xe131m	1.72+2	9.40+1	4.28+1	3.00+1	6.32+0	7.36-1
Xe133m	7.45+2	3.91+2	1.56+2	8.67+1	3.84+0	1.03-1
Xe133	2.79+4	1.51+4	6.62+3	4.39+3	5.68+2	3.84+1
Xe135m	2.94-2	4.90-2	3.21-2	0	0	0
Xe135	1.61+4	6.72+3	1.37+3	2.20+2	3.08-2	0

Table 15.2-8

ACTIVITY RELEASED TO THE ENVIRONMENT

(μ Ci)

Isotope	8-12 Hr	12-24 Hr	1-2 Day	2-10 Day	10-30 Day
I131	3.89	3.67	0.285	6.91-3	0
I133	7.20	5.18	0.432	6.91-3	0
Kr85m	1.37+7[a]	6.91+6	9.50+5	6.91+4	0
Kr85	2.88+6	4.32+6	5.36+6	2.42+7	1.38+7
Kr87	6.34+5	6.48+4	4.32+4	0	0
Kr88	1.44+7	5.62+6	2.16+5	3.46+4	0
Xe131m	1.73+6	2.81+6	3.37+6	1.04+7	4.32+6
Xe133m	7.92+6	1.08+7	1.04+7	1.73+7	5.18+5
Xe133	2.59+8	4.10+8	4.58+8	1.11+9	2.42+8
Xe135m	1.30+2	1.73+2	8.62+0	0	0
Xe135	1.44+8	1.17+8	4.32+7	6.91+6	1.73+5

(a) $1.37 + 7 = 1.37 \times 10^7$

Table 15.2-9

ESTIMATED DOSES AND ATMOSPHERIC DISPERSION FACTORS

	Dose (MRem/Event)	Dispersion Factor
Gamma	0.28	4.9 E-6 sec/m^3
Beta	0.56	4.9 E-6 sec/m^3
Total Body	0.12	4.9 E-6 sec/m^3
Skin	0.39	4.9 E-6 sec/m^3
Milk Ingestion	Negligible	-

Table 15.2-10

EGRESS FROM CONTAINMENT WORK AREAS

Area	Location		Egress Time
	Elevation	Azimuth	(sec)
C&I Panels	(+) 11'-0"	315°	175
TIP	(-) 5'-3"	35°	190
Sample Station	(+) 48'-7"	55°	223
CRD	(+) 11'-0"	270°	169

Table 15.2-11

DESIGN TRANSIENT INTEGRATED EGRESS DOSES

Egress	Dose (mRem/Event) Lens of Eye	Beta Skin	Thyroid
Sample Station (egress to upper airlock)	0.80	0.18	0.075
TIP Area (w/o air shower)	11.0	31.0	0.16
TIP Area (w/air shower)	6.8	15.0	0.13
CRD Area (w/o air shower)	3.8	11.0	0.097
CRD Area (w/air shower)	0.97	0.29	0.093
C&I Panels Area (w/o air shower)	4.3	12.0	0.10
C&I Panels Area (w/air shower)	1.5	0.36	0.069

Table 15.2-12

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

Cause	Estimated Vacuum Decay Rate
(1) Failure or Isolation of Steam Jet Air Ejectors	<1 in. Hg/min
(2) Loss of Sealing Steam to Shaft Gland Seals	Approximately 1 to 2 in. Hg/min
(3) Opening of Vacuum Breaker Valves	Approximately 2 to 12 in. Hg/min
(4) Loss of One or More Circulating Water Pumps	Approximately 4 to 24 in. Hg/min

Table 15.2-13

SEQUENCE OF EVENTS FOR FIGURE 15.2-7

Time (sec)	Event
-3.0 (est)	Initiate simulated loss of condenser vacuum at 2 in. Hg/sec.
0.0 (est)	Low condenser vacuum main turbine trip actuated.
0.0 (est)	Low condenser vacuum feedwater trip actuated.
0.01	Main turbine trip initiates recirculation pump trip (RPT) and scram.
1.6	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 bypass valve closure.
6.8	Safety/relief valves close
8.2	Group 1 safety/relief valves open again to relieve decay heat.
13.4	Water level reaches Level 2 setpoint and initiates HPCS and RCIC.
13.9	Group 1 safety/relief valves close again.
17.6	Group 1 heat safety/relief valves open again to relieve decay heat.
23.0	Group 1 safety/relief valves close again.
43.4 (est)	HPCS and RCIC flow enters vessel (not in simulation).

Table 15.2-14

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (in. of Hg)	Protective Action Initiated
27 to 28	Normal Vacuum Range
20 to 23	Main Turbine Trip and Feedwater Turbine Trip (Stop Valve Closures)
7 to 10	Main Steam Line Isolation Valve (MSLIV) Closure and Bypass Valve Closure

Table 15.2-15

SEQUENCE OF EVENTS FOR FIGURE 15.2-8

Time (sec)	Event
0	Loss of auxiliary power transformer occurs.
0	Recirculation system pump motors are tripped.
0	Condensate and booster pumps are tripped.
0	Condenser circulating water pumps are tripped.
4.0	Scram main turbine and feedwater turbines are tripped on L8 high water level.
4.1	Turbine bypass operation initiated by turbine trip.
27	RCIC and HPCS systems initiation on low water level (L2).
28	Low condenser vacuum initiates closure of turbine bypass valves and main steam line isolation valves.
31	Main steam line isolation valves closed.
57 (est)	HPCS and RCIC flow enters vessel (not simulated).

Table 15.2-16

SEQUENCE OF EVENTS FOR FIGURE 15.2-9

Time (sec)	Event
(-) 0.015 (approx.)	Loss of Grid causes turbine-generator to detect a loss of electrical load.
0	Turbine control valve fast closure is initiated.
0	Turbine-generator PLU trip initiates main turbine bypass system operation.
0	Recirculation system pump motors are tripped.
0	Fast control valve closure (FCV) initiates a reactor scram trip.
0.07	Turbine control valves closed.
0.1	Turbine bypass valves open.
1.6	Safety/relief valves open due to high pressure.
3.5	Feedwater pumps trip due to high water level (L8).
8.6	Safety/relief valves close.
>10 (est)	Vessel water level reaches Level 2 setpoint.
28	Closure of MSIV and turbine bypass valves is initiated via low condenser vacuum.
>40 (est)	HPCS and RCIC flow enters vessel (not simulated).

Table 15.2-17

SEQUENCE OF EVENTS FOR FIGURE 15.2-10

Time (sec)	Event
0	Trip of all feedwater pumps initiated.
3.4	Vessel water level reaches Level 4 and initiates recirculation flow runback.
5	Feedwater flow decays to zero.
7	Vessel water level (L3) trip initiates scram trip and recirculation pump trip.
15 (est)	Vessel water level reaches Level 2.
45 (est)	HPCS and RCIC flow enters vessel (not simulated).

Table 15.2-18

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Approximate Elapsed Time (min)	Event
0	Reactor is operated at 102 rated power when loss of offsite power occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
12	Controlled depressurization initiated at suppression pool temperature of 120°F.
13	Suppression pool cooling initiated to prevent overheating from SRV actuation.*
36	Blowdown to approximately 100 psig completed.
66	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
71	RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.

*See Table 15.2-15 for detailed sequence of events for loss of AC power transient.

Table 15.2-19

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF
RHR SHUTDOWN COOLING

Initial Power Corresponding to 105% Rated Power		
Suppression Pool Mass (lbm)		8.696×10^6
RHR (KHX value) (Btu/sec/°F)		610
Initial vessel conditions		
Pressure (psia)		1040
Temperature (°F)		549
Initial primary fluid inventory (lbm)		544,540
Initial pool temperature (°F)		100
Service water temperature (°F)		100
Vessel heat capacity (Btu/lbm/°F)		0.125
HPCS on - off water level (ft)	ON	40.22
	OFF	48
HPCS flow rate, (lbm/sec)		834
LPCI flow rate per loop (lbm/sec)		987
LPCS flow rate (lbm/sec)		834

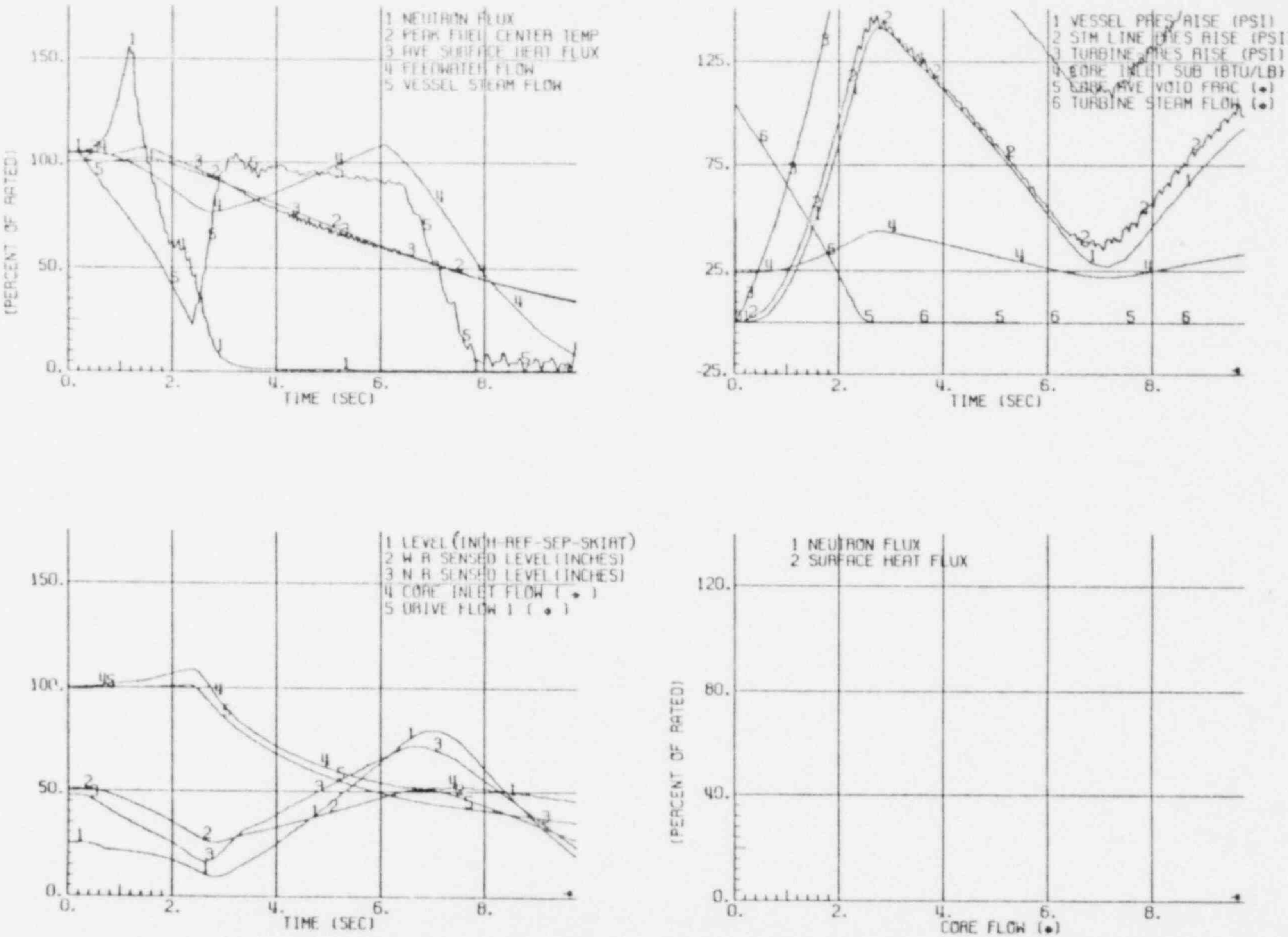

Figure 15.2-1. Pressure Regulator Downscale Failure

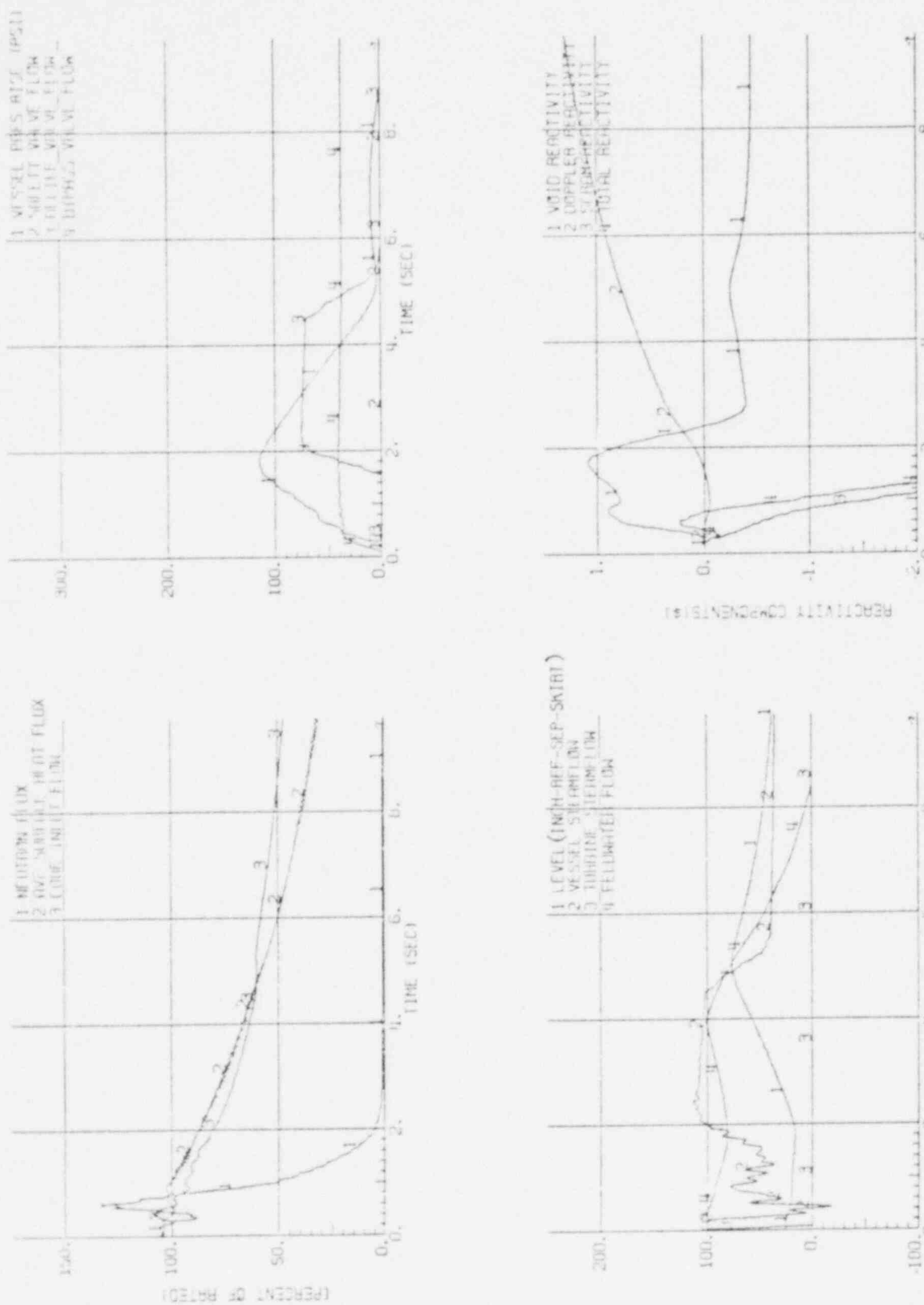

Figure 15.2-2. Generator Load Rejection, with Bypass On

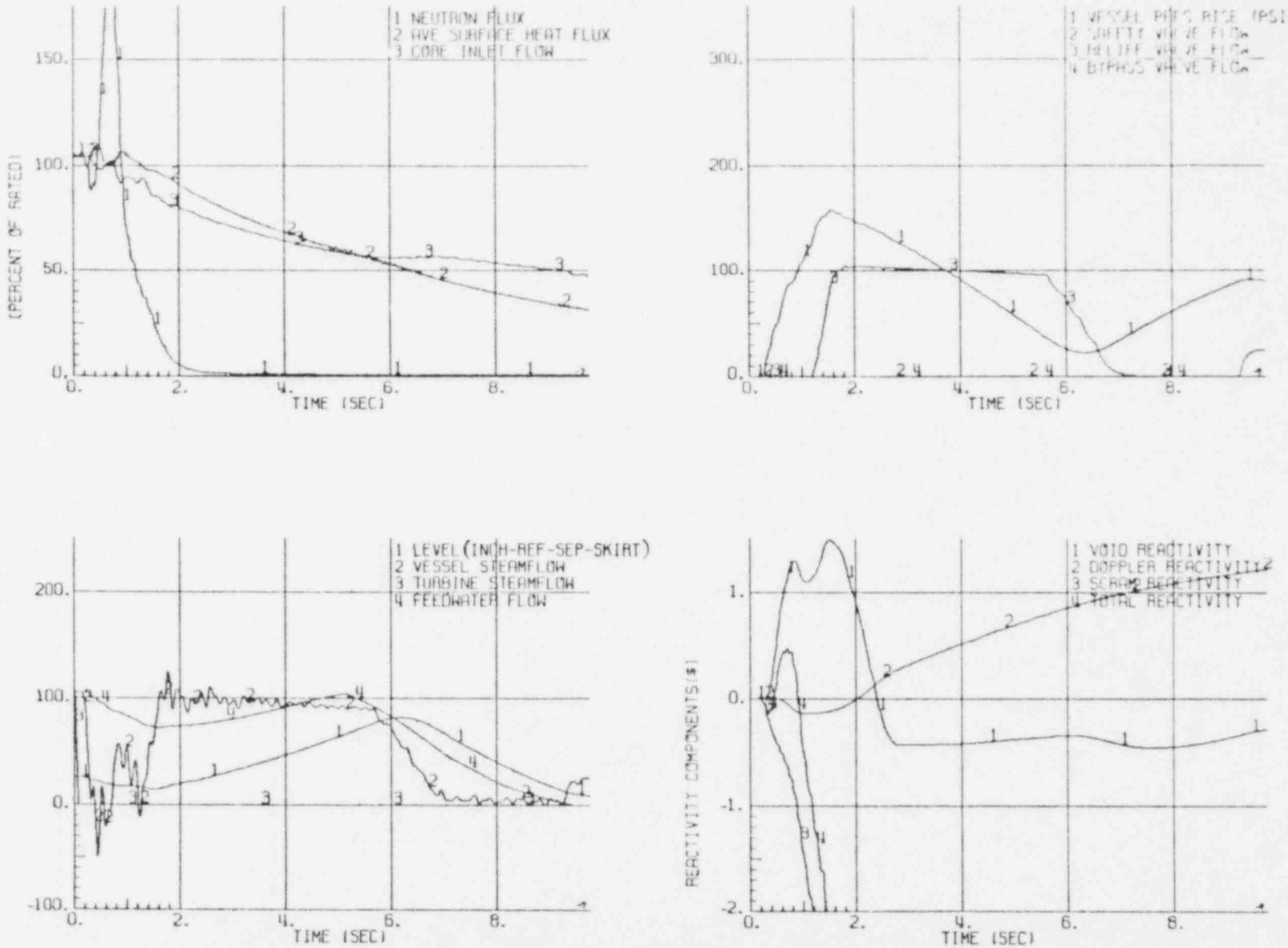

Figure 15.2-3. Generator Load Rejection, without Bypass

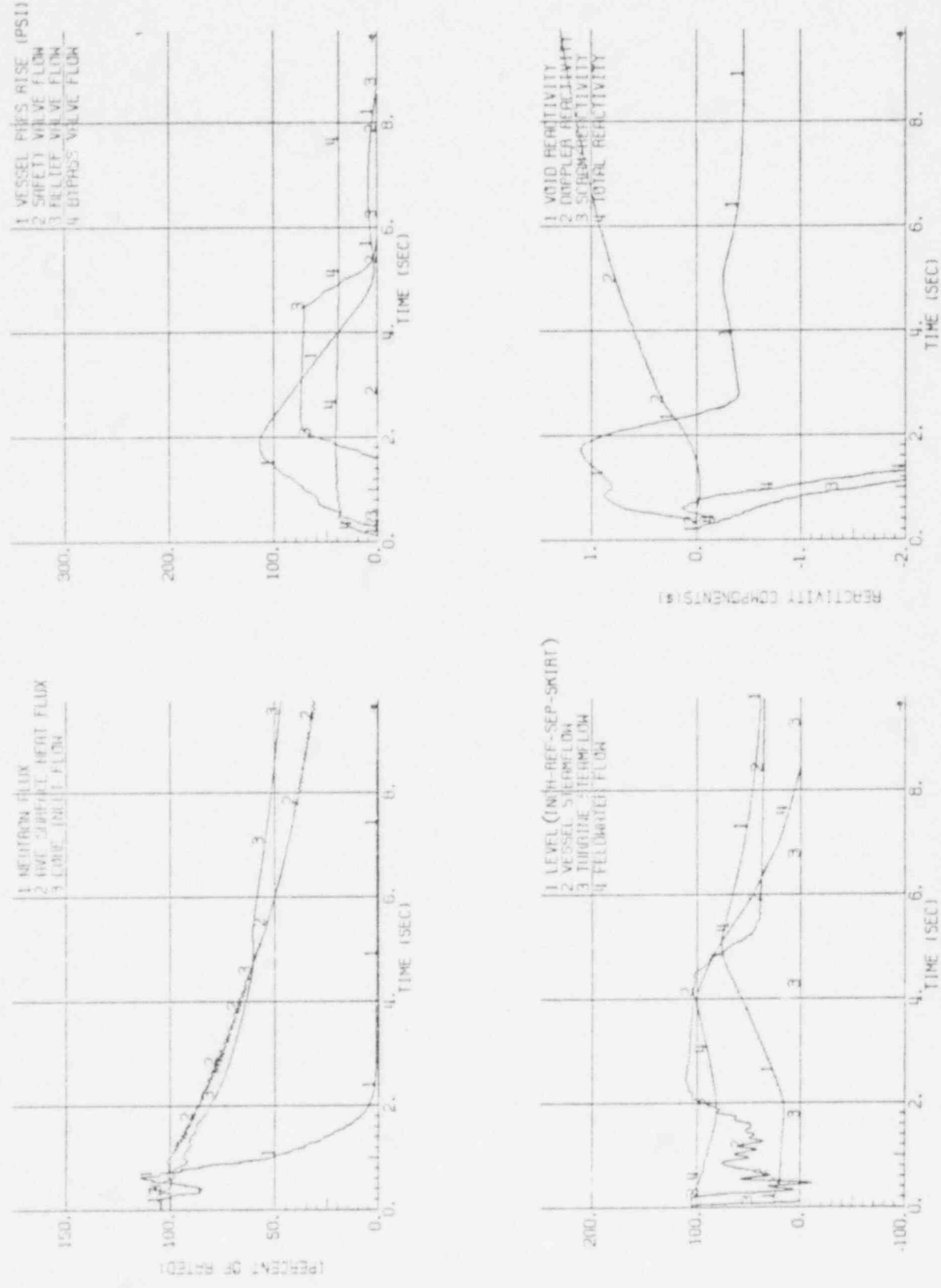

Figure 15.2-4. Turbine Trip with Bypass On, Trip Scram

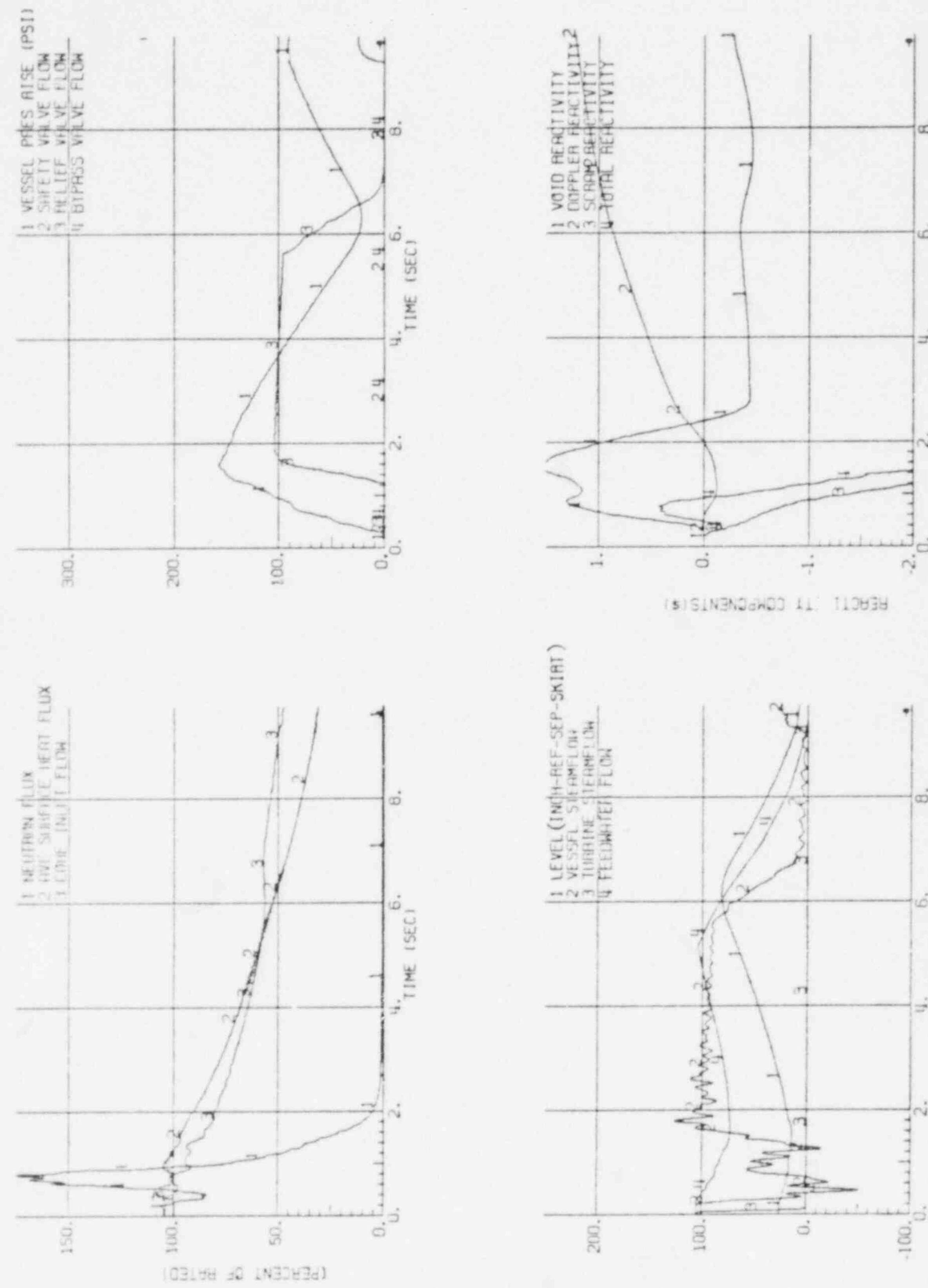

Figure 15.2-5. Turbine Trip without Bypass, Trip Scram

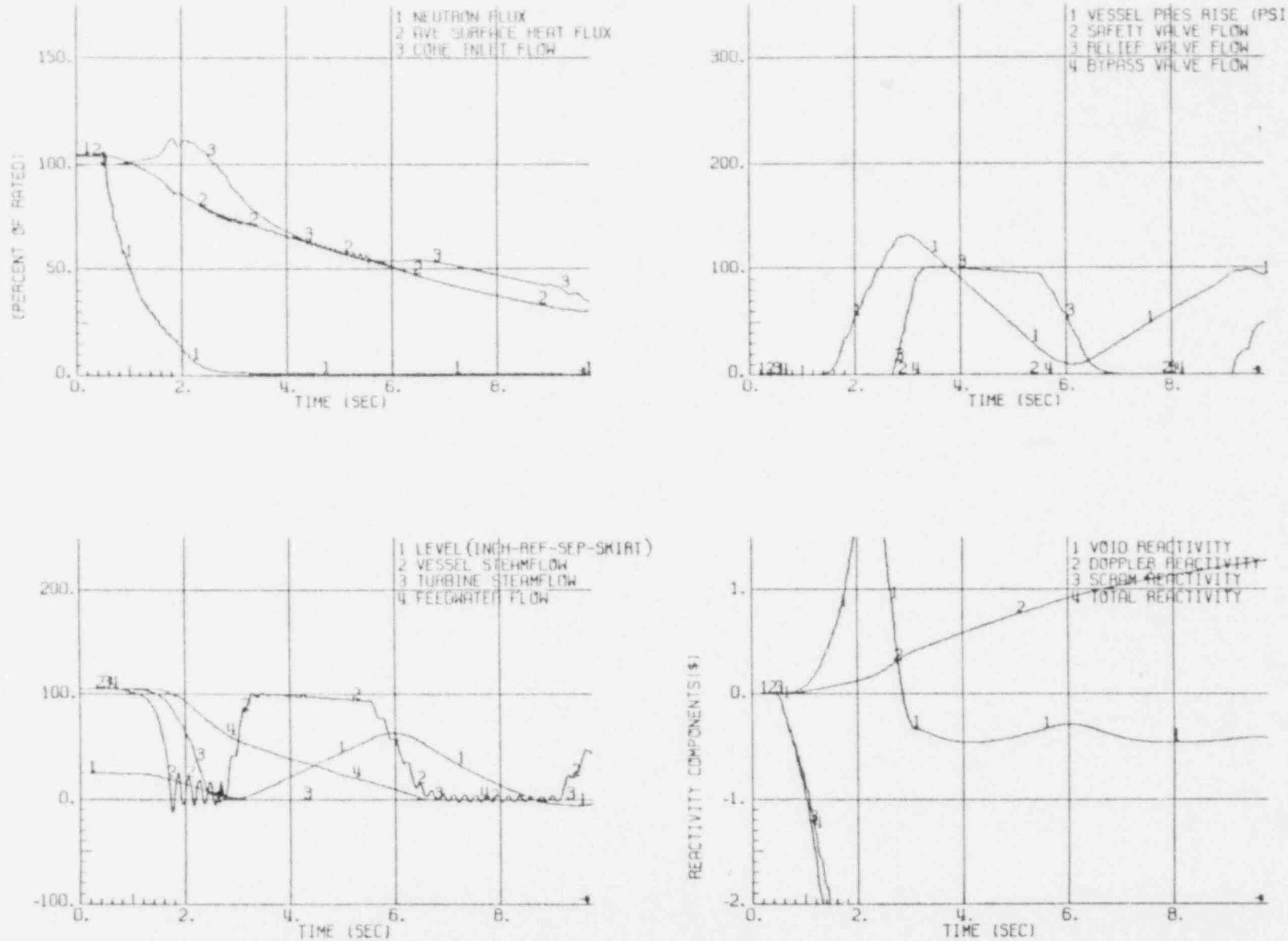

Figure 15.2-6. MSIV Closure, Position Scram

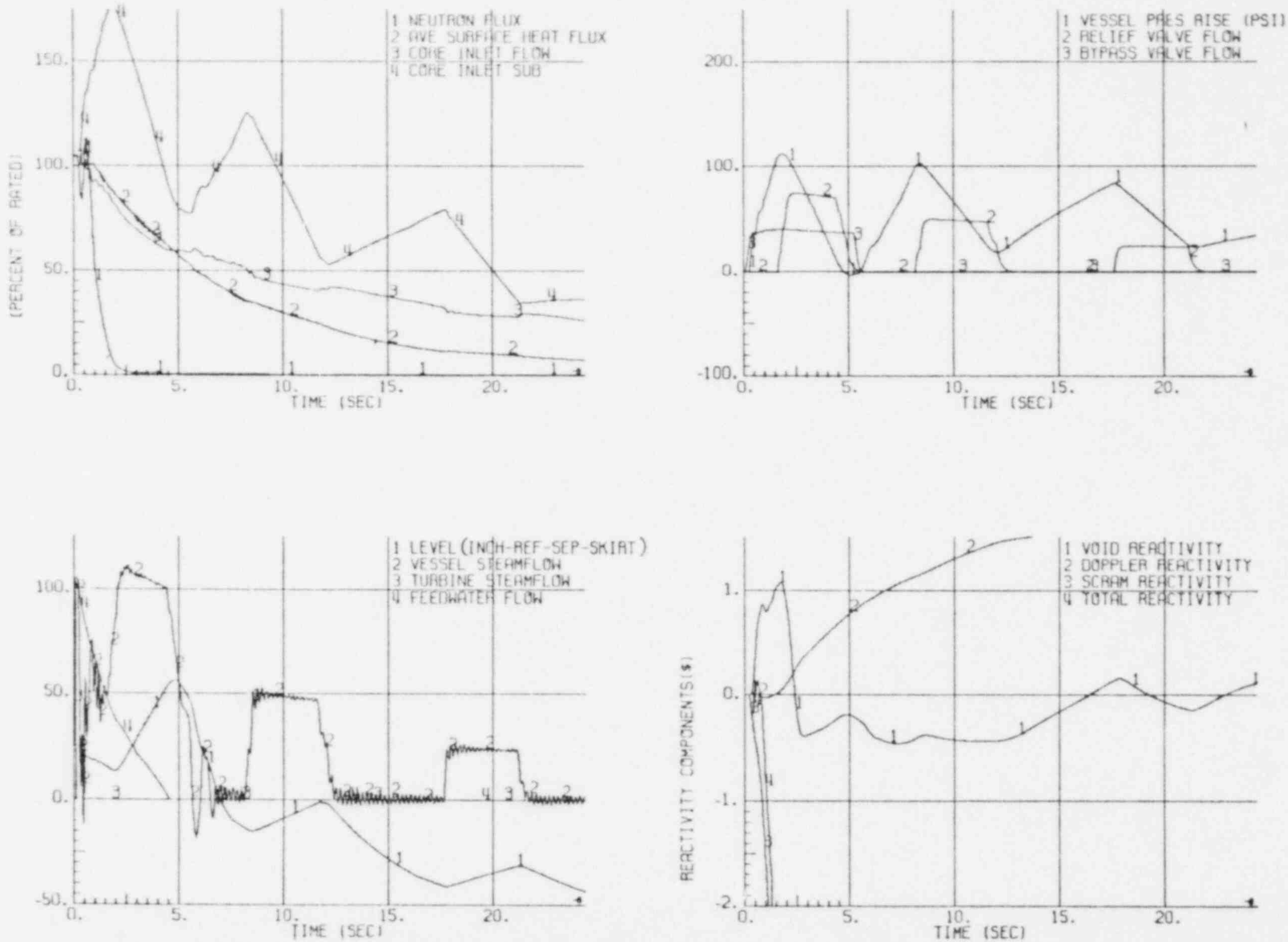

Figure 15.2-7. Loss of Condenser Vacuum at 2 in./sec

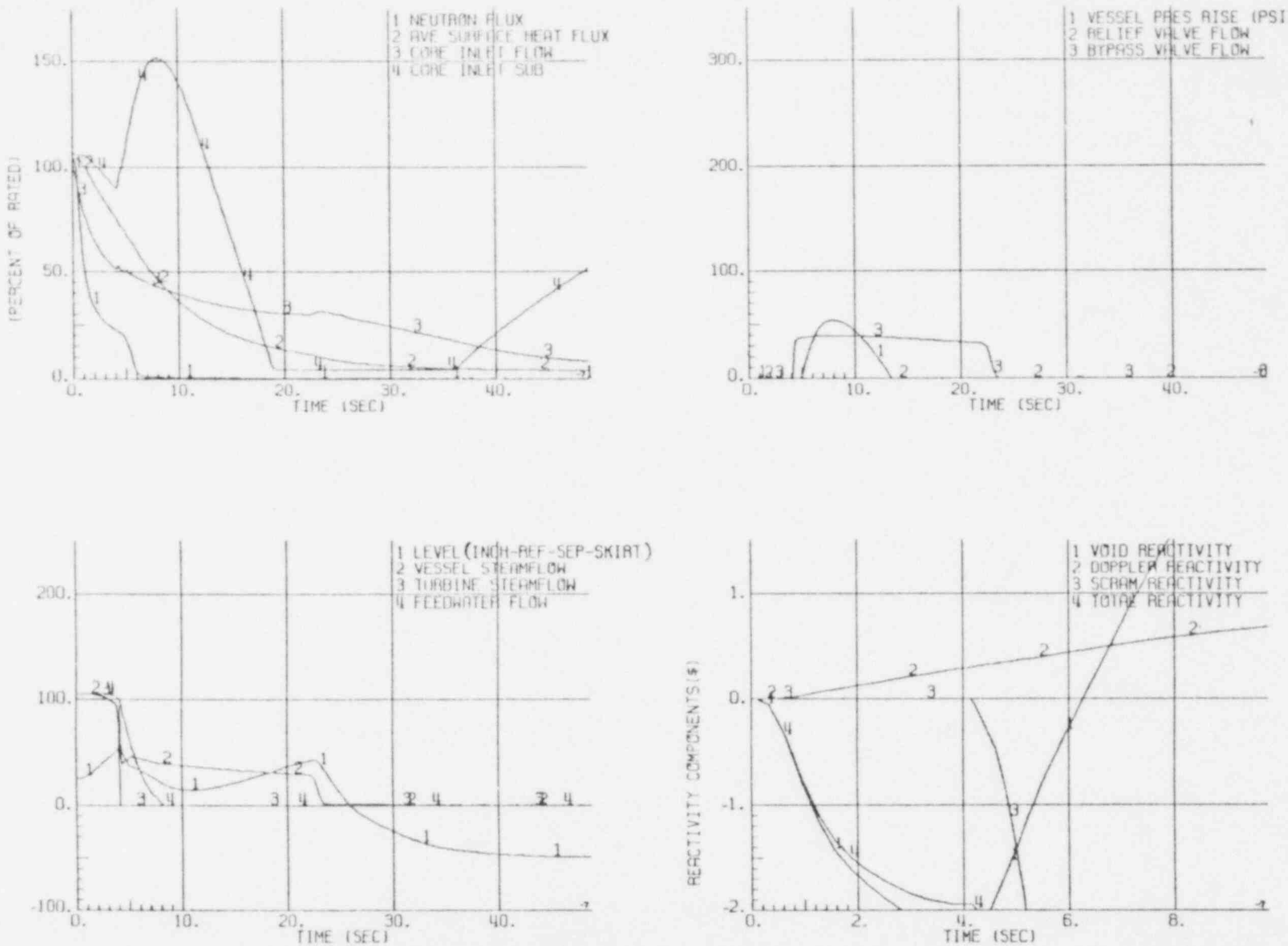

Figure 15.2-8. Loss of Auxiliary Power Transformer

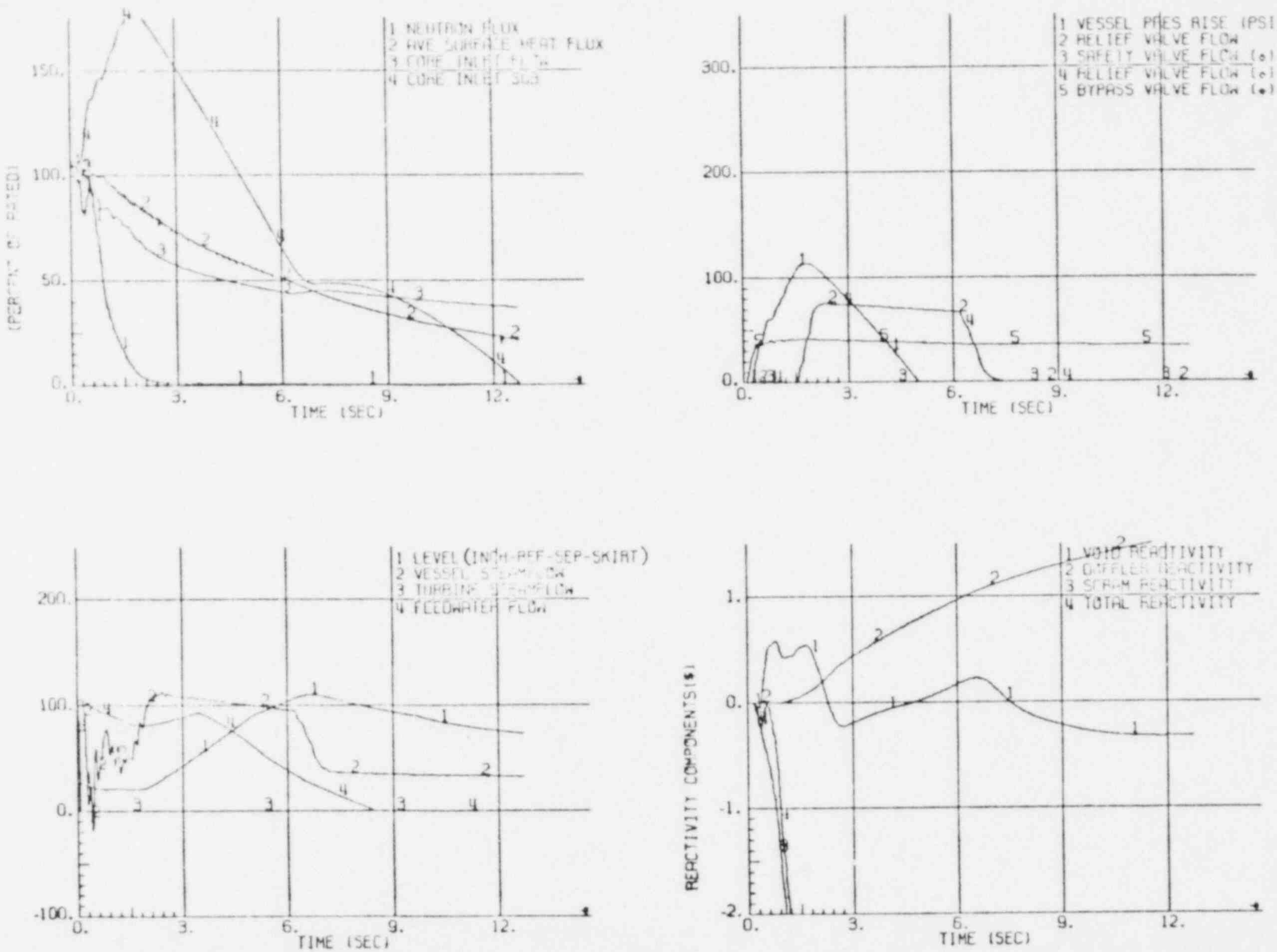

Figure 15.2-9. Loss of All Grid Connections

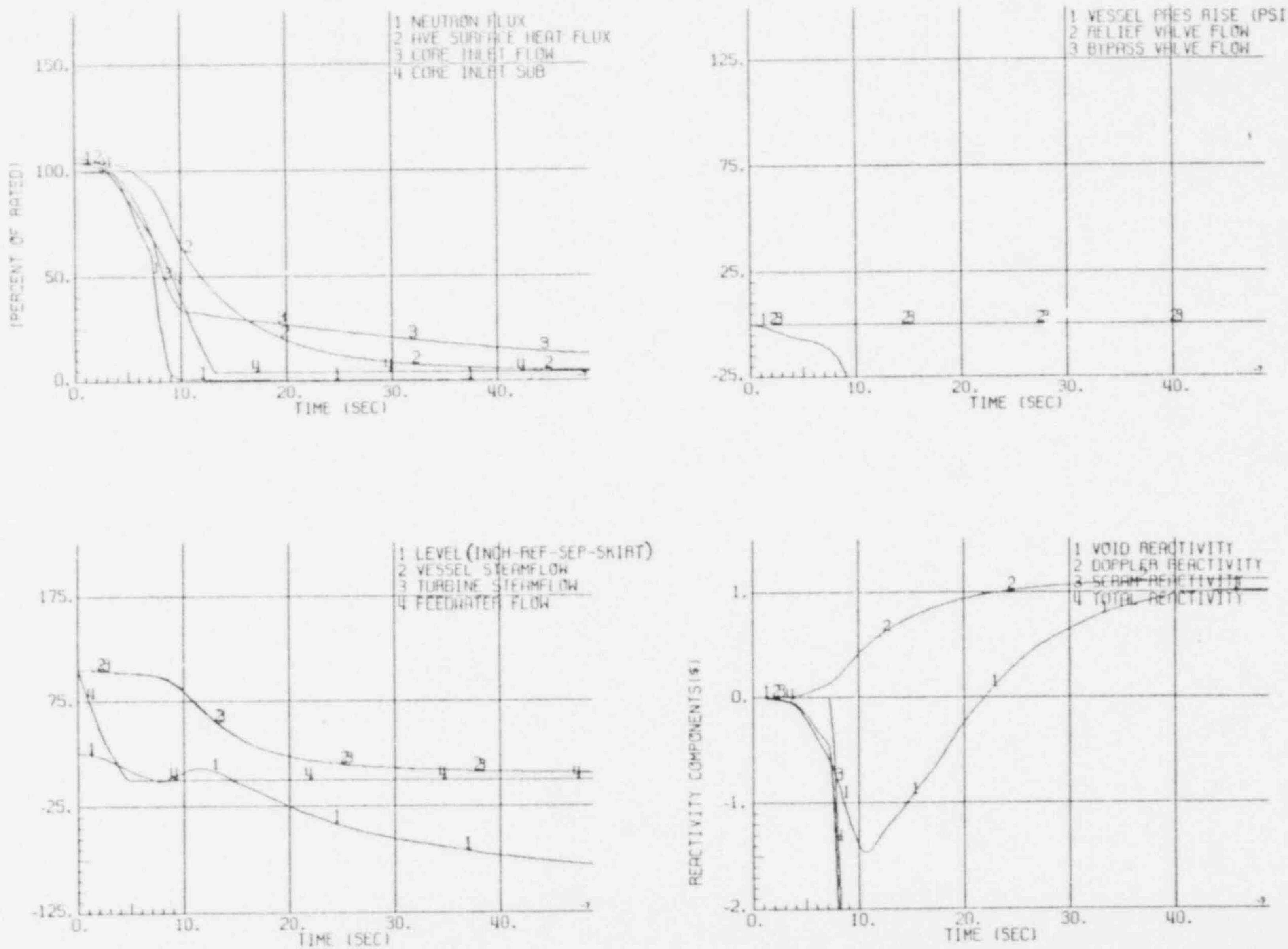

Figure 15.2-10. Loss of All Feedwater Flow

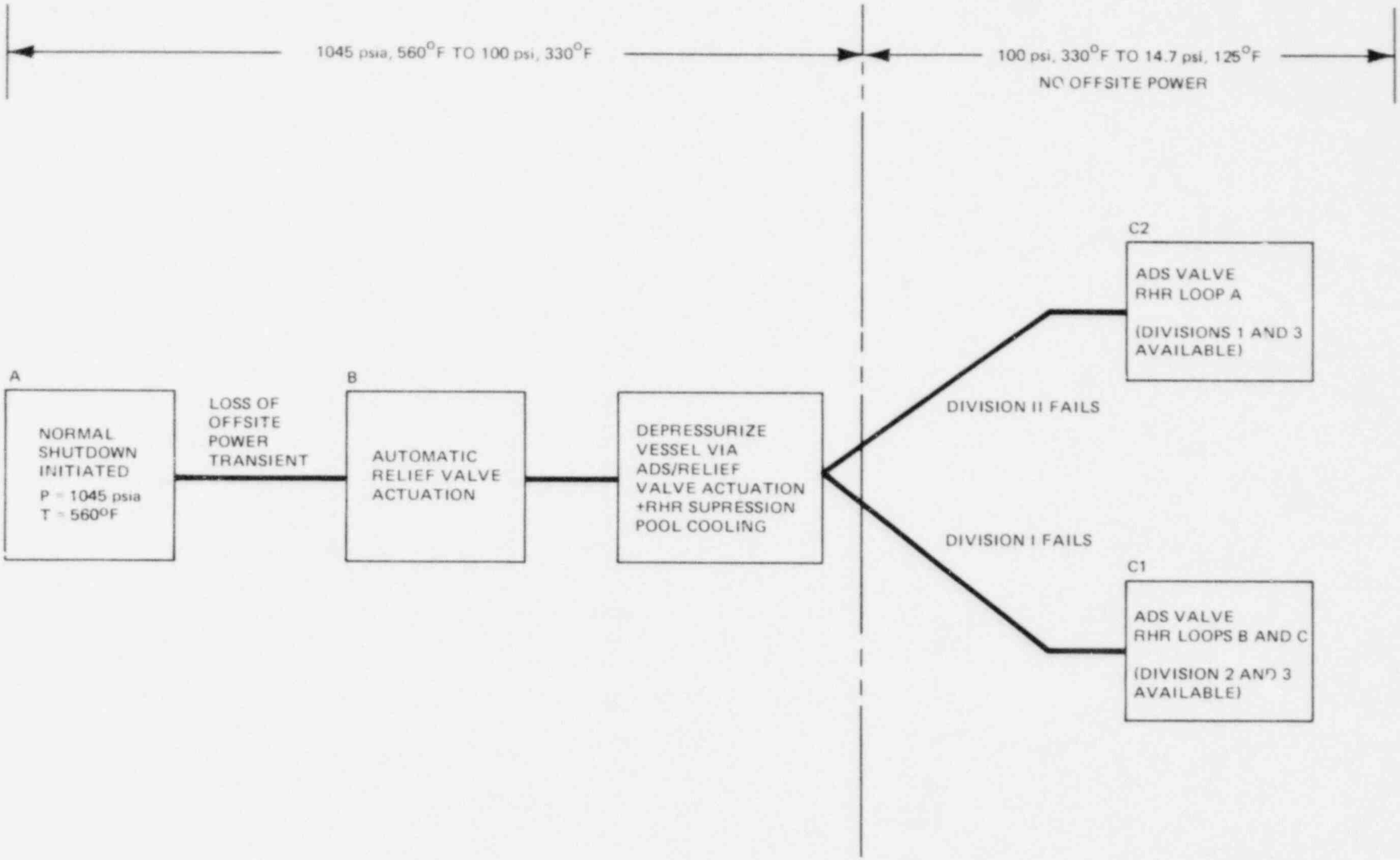

Figure 15.2-11. ADS/RHR Cooling Loops

Notes for Figure 15.2-11

ACTIVITY A

Initial pressure = 1040 psia

Initial temperature = 550°F

For purposes of this analysis, the following worst-case conditions are assumed to exist:

(1) The reactor is assumed to be operating at 102% rated power;

(2) A loss of power transient occurs (Subsection 15.2.6); and

(3) A simultaneous loss of onsite power (Division 1 or Division 2), which eventually results in the operator not being able to open one of the RHR shutdown cooling line suction valves.

ACTIVITY B

Initial system pressure = 1040 psia

Initial system temperature = 550°F

Operator Actions

During approximately the first 12 min, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCS and RCIC system automatic operation.

At approximately 12 min into the transient, the operator initiates depressurization of the reactor vessel. Controlled depressurization procedures consist of controlling vessel pressures and water level by using selected safety/relief valves, RCIC and HPCS systems. After approximately 13 minutes, it is assumed one RHR heat exchanger will be placed in the suppression pool cooling mode to remove decay heat. At this time, the suppression pool will be 125°F.

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown

Notes for Figure 15.2-11 (Continued)

cooling mode. At this time (36 min), the suppression pool temperature will be 159°F.

ACTIVITY C1 (Division 1 fails, Division 2 available)

System pressure = ∿100 psig

System temperature = ∿340°F

Operator Actions

The operator establishes a closed cooling path as follows:

Either of the following cooling paths are established:

(a) Utilizing RHR Loop B, water from the suppression pool is pumped through the RHR heat exchanger (where a portion of the decay heat is removed) into the reactor vessel. The cooled suppression pool water flow through the vessel (picking up a portion of the decay heat) out the ADS valves and back to the suppression pool. This alternate cooling path is shown in Figure 15.2-14. Cold shutdown is achieved approximately 2 hours after transient occurred.

(b) Utilizing RHR loops B and C together, water is taken from the suppression pool and pumped directly into the reactor vessel. The water passes through the vessel picking up decay heat) and out the ADS valves returning to the suppression pool as shown in Figure 15.2-16. Suppression pool water is then cooled by operation of RHR Loop B in the cooling mode (Figure 15.2-15). In this alternate cooling path, RHR Loop C is used for injection and RHR Loop B for cooling. Cold shutdown is achieved approximately 12-1/2 hours after transient occurred.

ACTIVITY C2 (Division 2 fails, Division 1 available)

System pressure = ∿100 psig

System temperature = ∿340°F

Operator Actions

Utilizing RHR Loop A (Figure 15.2-13) instead of Loop B, an alternate cooling path is established as in Activity C1 item 2(a) above. Again, cold shutdown is reached in approximately 2 hours.

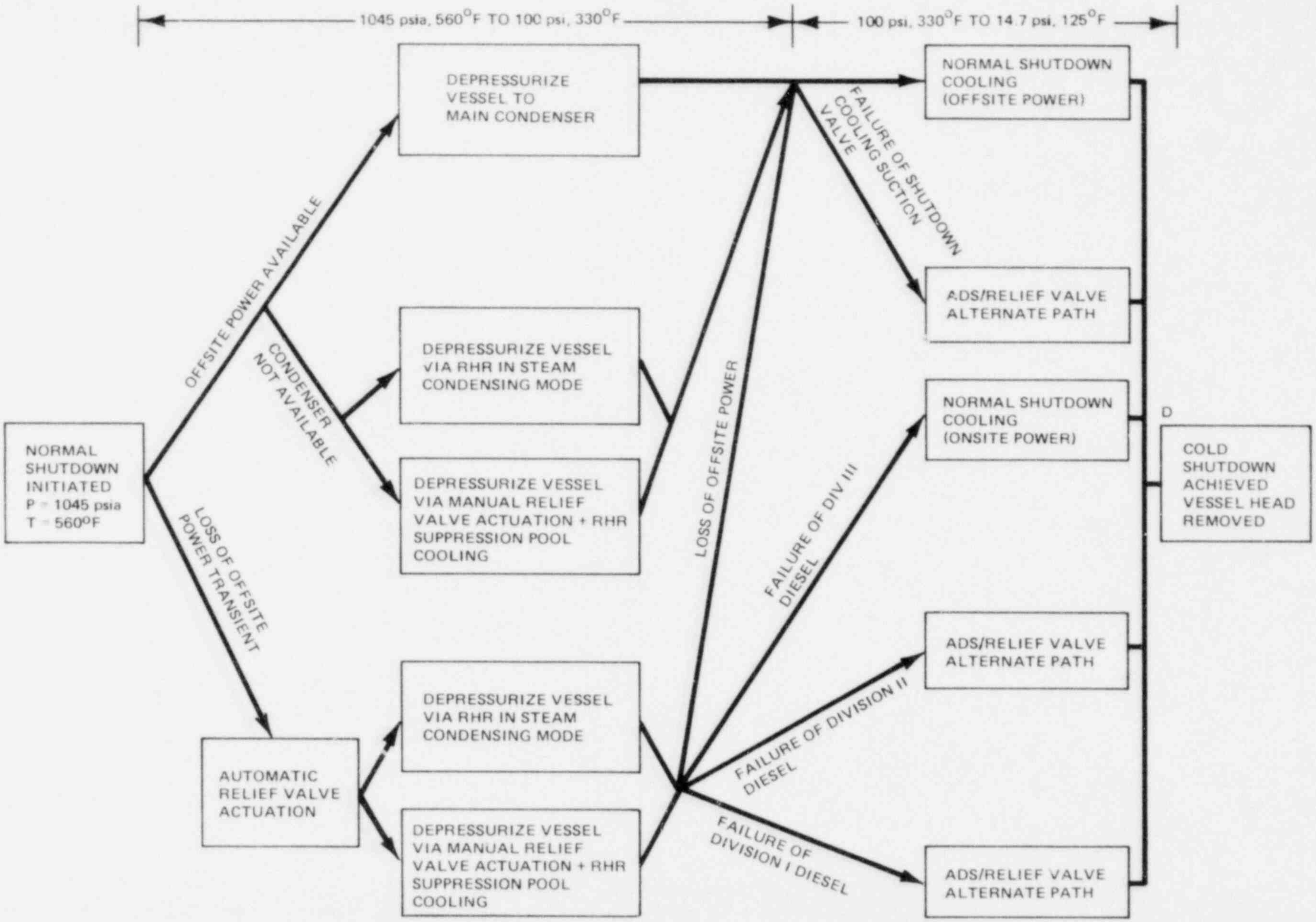

Figure 15.2-12. Summary of Paths Available to Achieve Cold Shutdown

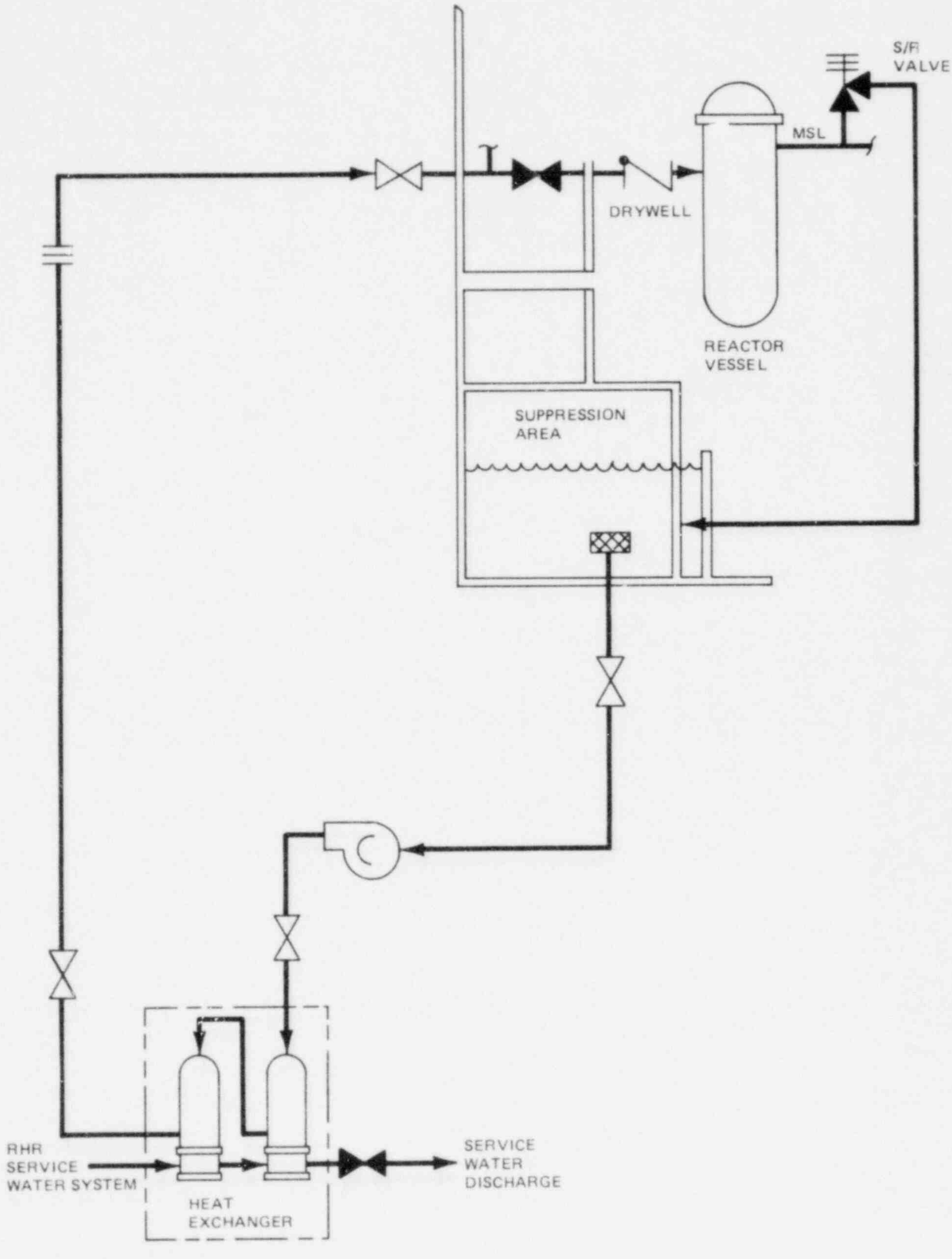

Figure 15.2-13. Activity C2 Alternate Shutdown Cooling Path Utilizing RHR Loop A

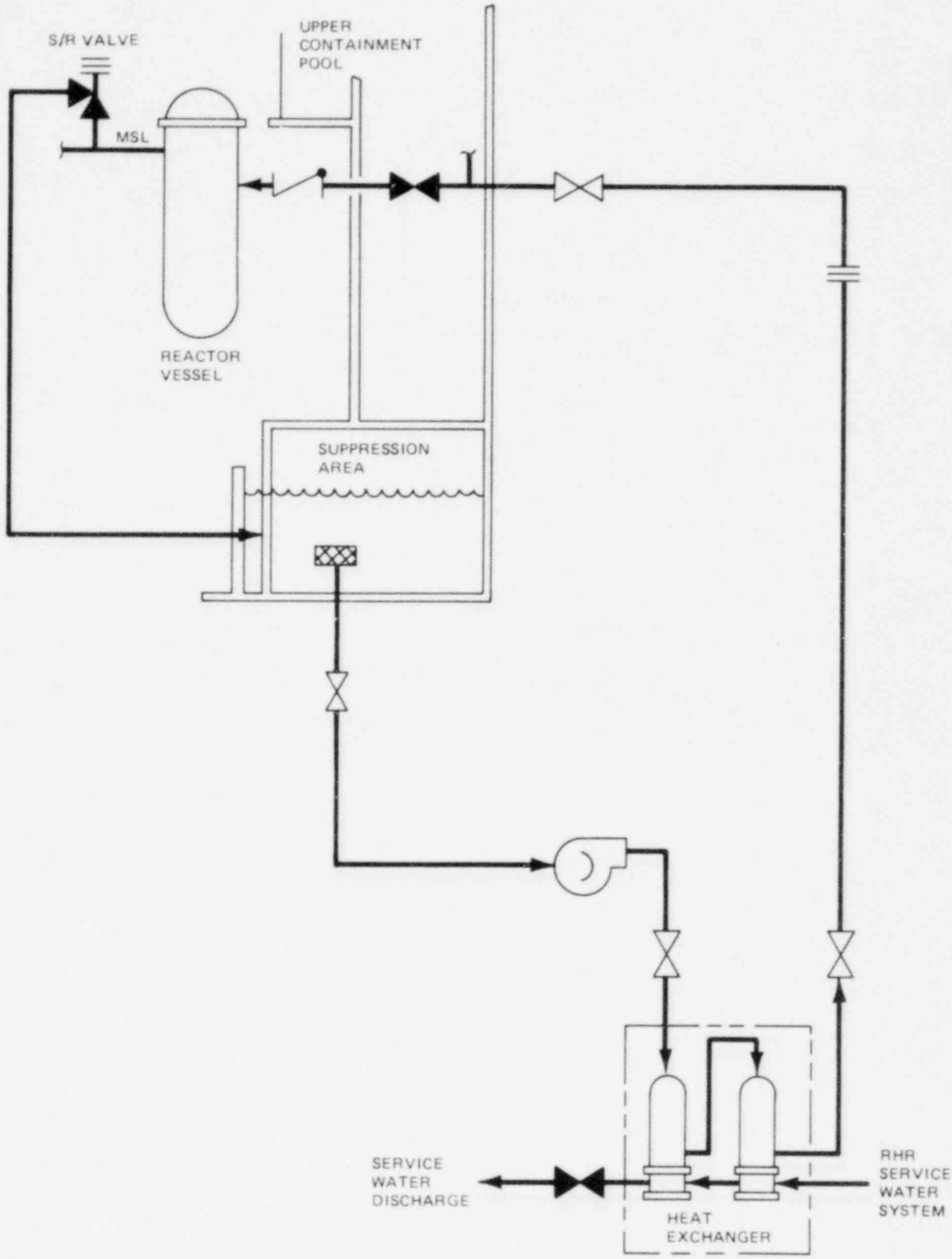

Figure 15.2-14. Activity C1 Alternate Shutdown Cooling Path Utilizing RHR Loop B

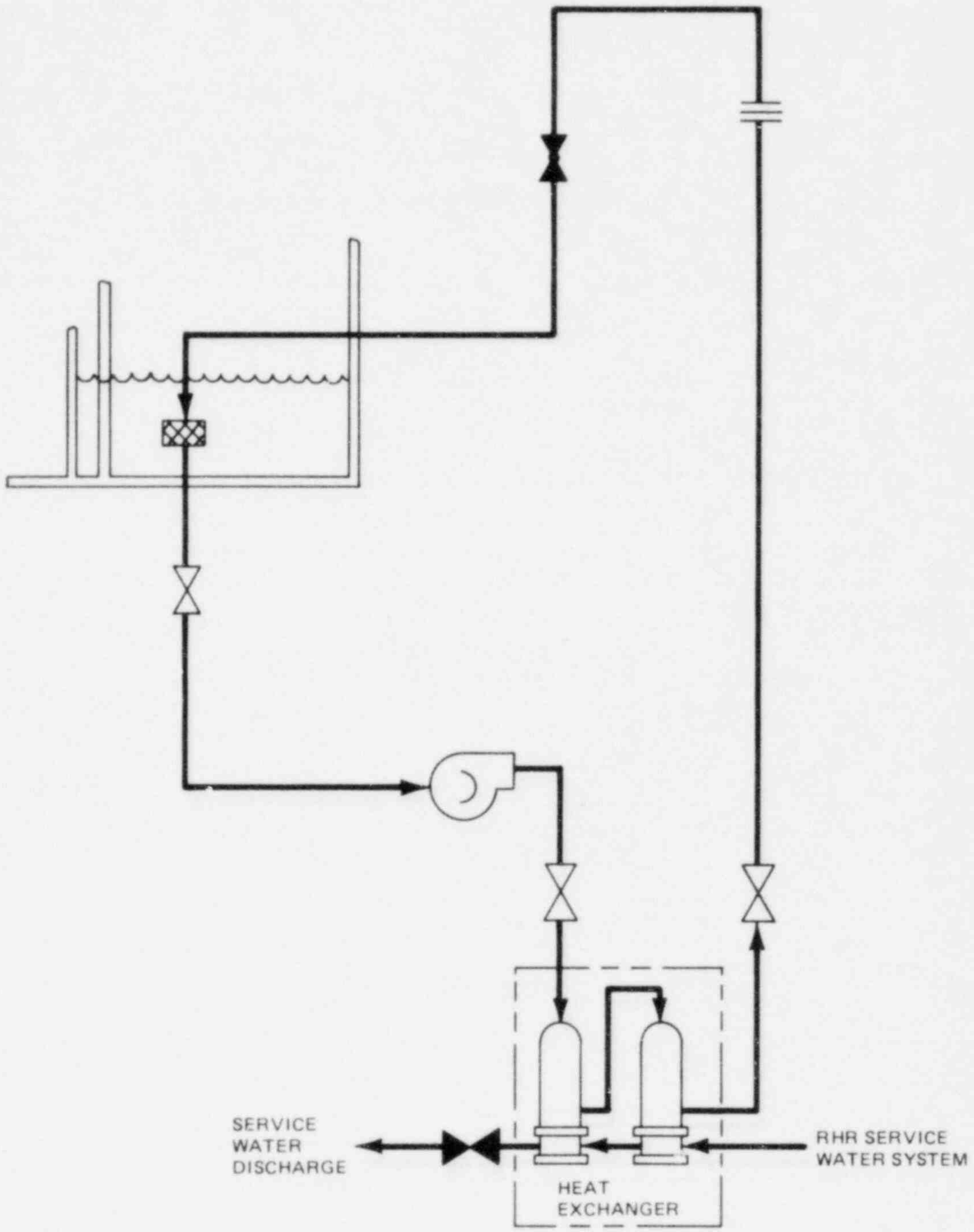

Figure 15.2-15. RHR Loop B (Suppression Pool Cooling Model)

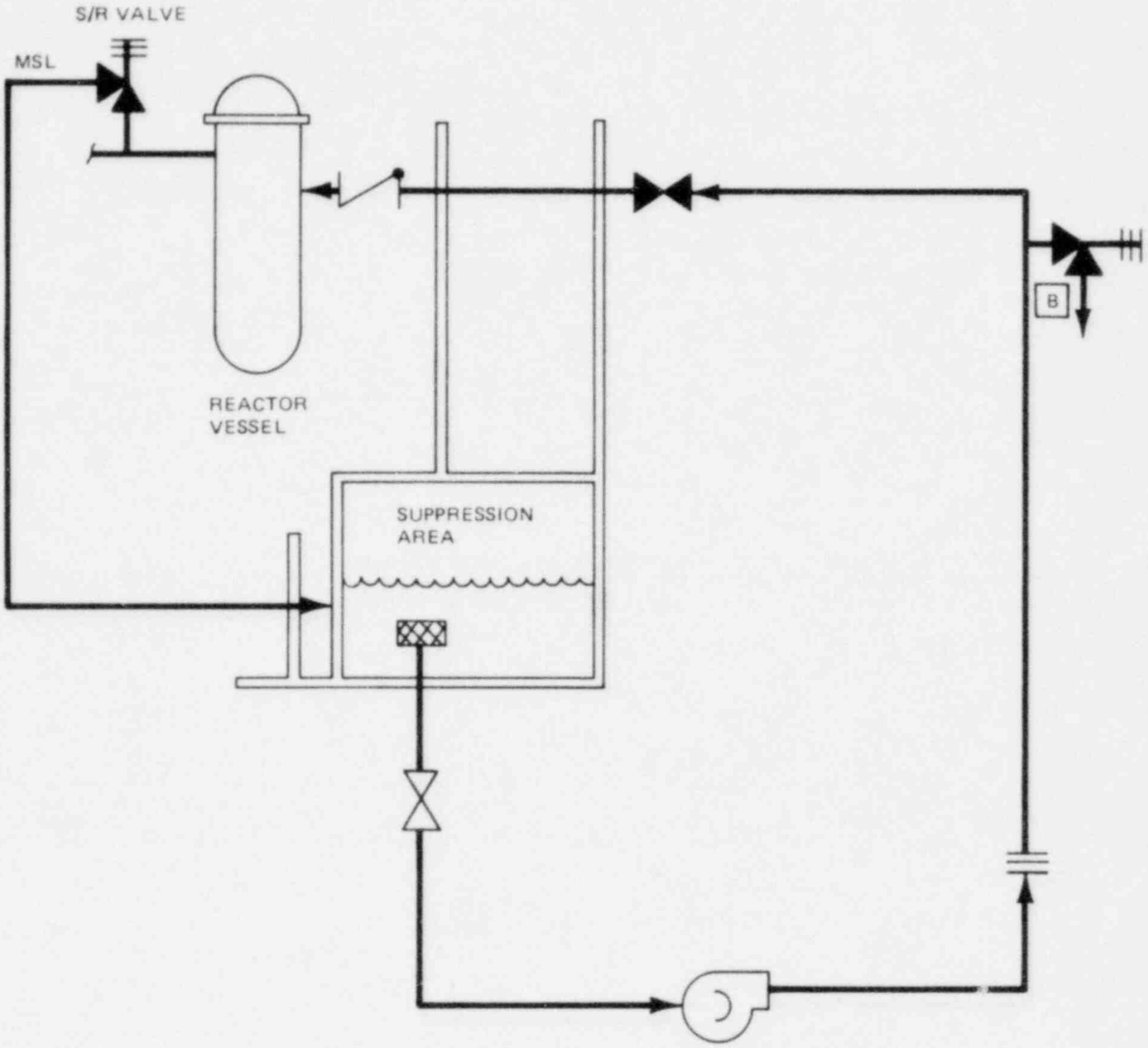

Figure 15.2-16. RHR Loop C

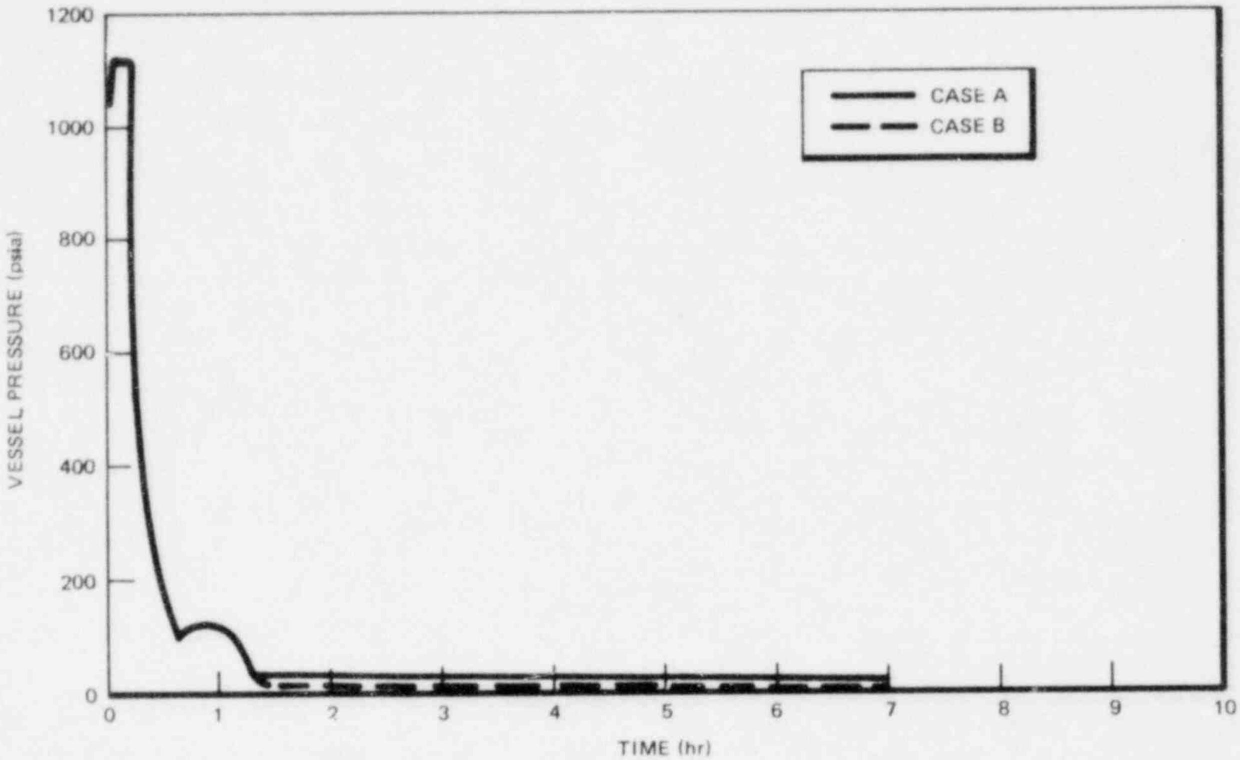

Figure 15.2-17. Vessel Pressure vs. Time

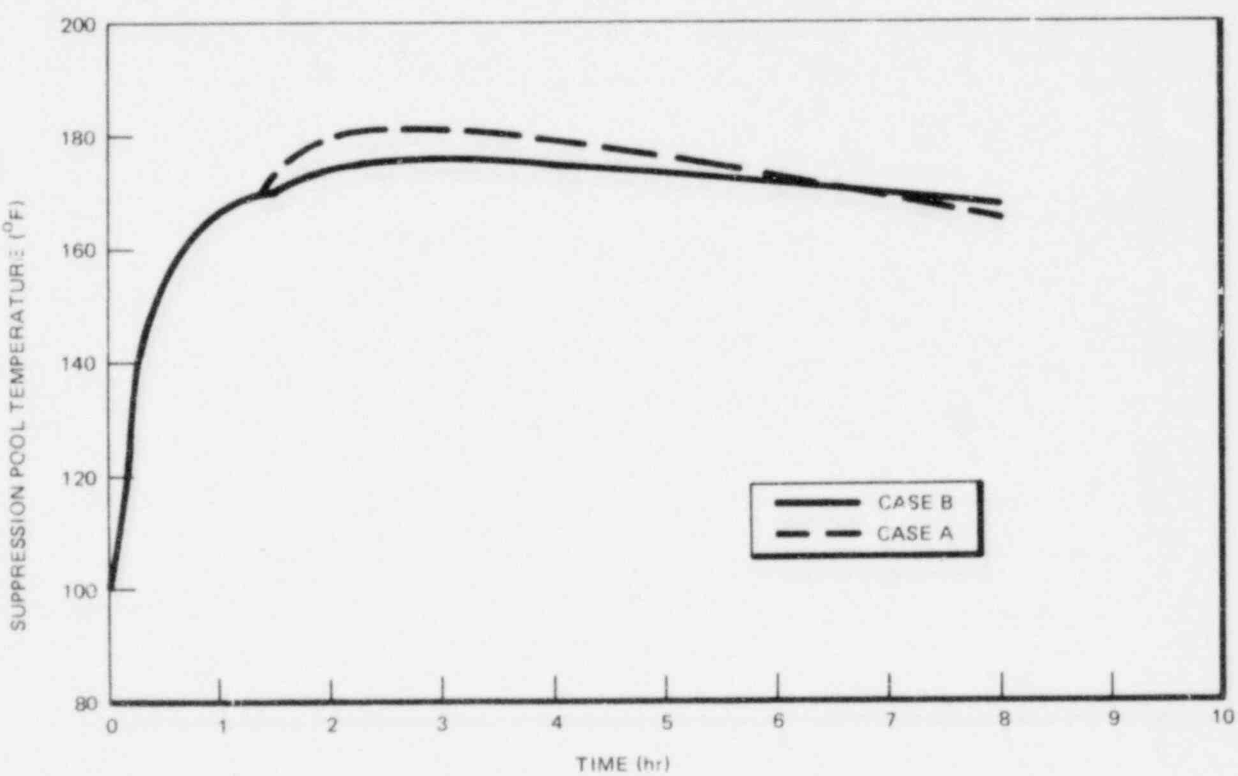

Figure 15.2-18. Suppression Pool Temperature vs. Time

SECTION 15.3

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CONTENTS (Continued)

CONTENTS (Continued)

SECTION 15.3

TABLES

ILLUSTRATIONS

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

15.3.1 Recirculation Pump Trip

15.3.1.1 Identification of Causes and Frequency Classification

15.3.1.1.1 Identification of Causes

Recirculation pump 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 L2 setpoint trip;

(2) TCV fast closure or stop valve closure;

(3) failure to scram high pressure setpoint trip;

(4) motor branch circuit overcurrent protection;

(5) motor overload protection; and

(6) suction block valve not fully open.

Random tripping will occur in response to:

(1) operator error;

(2) loss of electrical power source to the pumps; and

(3) equipment or sensor failures and malfunctions which initiate the above intended trip response.

15.3.1.1.2 Frequency Classification

15.3.1.1.2.1 Trip of One Recirculation Pump

This transient event is categorized as one of moderate frequency.

15.3.1.1.2.2 Trip of Two Recirculation Pumps

This transient event is categorized as one of moderate frequency.

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 One Recirculation Pump

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

15.3.1.2.1.2 Trip of Two Recirculation 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 One Recirculation Pump

Since no scram occurs for trip of one recirculation pump, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded, and reduce flow of the operating pump to confrom to the single pump flow criteria. Also, the operator should determine the cause of failure prior to returning the system to normal and follow the restart procedure.

15.3.1.2.1.3.2 Trip of Two Recirculation Pumps

The operator should ascertain that the reactor scrams with the turbine trip resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation, monitoring reactor water level and pressure control after shutdown. When both reactor pressure and level are under control, the operator should secure both HPCS and RCIC as necessary. The operator should also determine the cause of the trip prior to returning the system to normal.

15.3.1.2.2 Systems Operation

15.3.1.2.2.1 Trip of One Recirculation Pump

Tripping a single recirculation pump 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 Two Recirculation Pumps

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

Specifically, this transient takes credit for vessel level (L8) instrumentation to trip the turbine. Reactor shutdown relies on scram trips from the turbine stop valves. High system pressure is limited by the pressure relief valve system operation.

15.3.1.2.3 The Effect of Single Failures and Operator Errors

15.3.1.2.3.1 Trip of One Recirculation Pump

Since no corrective action is required (Subsection 15.3.1.2.2.1), no additional effects of single failures need be discussed. If additional SACF or SOE is assumed (for envelope purposes, the other pump is assumed tripped), then the following two-pump trip analysis is provided (see Appendix 15A for specific details).

15.3.1.2.3.2 Trip of Two Recirculation Pumps

Table 15.3-2 lists the vessel level (L8) scram as the first response to initiate corrective action in this transient. This scram trip signal is designed such that a single failure will neither initiate nor impede a reactor scram trip initiation (see Appendix 15A for details).

15.3.1.3 Core and System Performance

15.3.1.3.1 Mathematical Model

The nonlinear, dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.3.1.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

Pump motors and pump rotors are simulated with minimum specified rotating inertias.

15.3.1.3.3 Results

15.3.1.3.3.1 Trip of One Recirculation Pump

Figure 15.3-1 shows the results of losing one recirculation pump. The tripped loop diffuser flow reverses in approximately 5.7 sec. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 131% of normal diffuser flow and 54% of rated core flow. 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.

15.3.1.3.3.2 Trip of Two Recirculation Pumps

Figure 15.3-2 graphically shows this transient with minimum specified rotating inertia. MCPR remains unchanged. No scram is initiated directly by pump trip. The vessel water level swell due to rapid flow coastdown is expected to reach the high level trip, thereby shutting down the main turbine and feed pump turbines, and scramming. Subsequent events, such as main steamline isolation and initiation of RCIC and HPCS systems occurring late in this event, have no significant effect on the results.

15.3.1.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than expected under actual plant conditions.

Actual pump and pump-motor driveline rotating inertias are expected to be somewhat greater than the minimum design values assumed in this simulation. Actual plant deviations regarding inertia are expected to lessen the severity as analyzed. Minimum

15.3.1.3.4 Consideration of Uncertainties (Continued)

design inertias were used, as well as the least negative void coefficient, since the primary interest is in the flow reduction.

15.3.1.4 Barrier Performance

15.3.1.4.1 Trip of One Recirculation Pump

Figure 15.3-1 results indicate a basic reduction in system pressures from the initial conditions. Therefore, the RCPB barrier is not threatened.

15.3.1.4.2 Trip of Two Recirculation Pumps

The results shown in Figure 15.3-2 indicate that peak pressures stay well below the 1375 psig limit allowed by the applicable code. Therefore, the barrier pressure boundary is not threatened.

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

15.3.2 <u>Recirculation Flow Control Failure - Decreasing Flow</u>

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

Master controller malfunctions can cause a decrease in core coolant flow. A downscale failure of either the master power controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator has a velocity limiter which limits the maximum valve stroking rate to 11%/sec. A postulated failure of the input demand signal, which is utilized in both loops, can decrease core flow at the maximum valve stroking rate established by the loop limiter.

Failure within either loop's controller can result in a maximum valve stroking rate as limited by the capability of the valve hydraulics.

15.3.2.1.2 Frequency Classification

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

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 Closure of One Main Recirculation Valve

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

15.3.2.2.1.2 Fast Closure of Two Main Recirculation Valves

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 Closure of One Main Recirculation Valve

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

15.3.2.2.1.3.2 Fast Closure of Two Main Recirculation Valves

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

15.3.2.2.2 Systems Operation

15.3.2.2.2.1 Fast Closure of One Main Recirculation Valve

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

15.3.2.2.2.2 Fast Closure of Two Main Recirculation Valves

Normal plant instrumentation and control is assumed to function. Credit is taken for scram in response to vessel high water level (L8) trip.

15.3.2.2.3 The Effect of Single Failures and Operator Errors

The single failure and operator error considerations for this event are the same as discussed in Subsection 15.3.1.2.3.2. The

15.3.2.2.3 The Effect of Single Failures and Operator Errors (Continued)

fast closure of two recirculation valves, instead of one, would be the envelope case for the additional SCF or SOE.

15.3.2.3 Core and System Performance

15.3.2.3.1 Mathematical Model

The nonlinear dynamic model described in Subsection 15.1.1.3.1 is used to simulate these transient events.

15.3.2.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions listed in Table 15.0-2.

The less negative void coefficient in Table 15.0-2 was used for these analyses.

15.3.2.3.2.1 Fast Closure of One Main Recirculation Valve

Failure within either loop controller can result in a maximum stroking rate of 60%/sec as limited by the valve hydraulics.

15.3.2.3.2.2 Fast Closure of Two Main Recirculation Valves

A downscale failure of either the master power controller or the flux controller will generate a zero flow demand signal to both recirculation flow controllers. Each individual valve actuator circuitry has a velocity limiter which limits maximum valve stroking rate to 11%/per sec. Recirculation loop flow is allowed to decrease to approximately 25% of rated before high water level

15.3.2.3.2.2 Fast Clousure of Two Main Recirculation Valves (Continued)

(L8) causes trip of the recirculation pumps due to stop valve closure. This is the flow expected when the flow control valves are maintained at a minimum open position.

15.3.2.3.3 Results

15.3.2.3.3.1 Fast Closure of One Recirculation Valve

Figure 15.3-3 illustrates the maximum valve stroking rate which is limited by hydraulic means. Even though a turbine trip on high water level occurs, the fuel thermal limits are not threatened.

15.3.2.3.3.2 Fast Closure of Two Recirculation Valves

Figure 15.3-4 illustrates the expected transient which is similar to a two-pump trip. This analysis is very similar to the two-pump trip described in Subsection 15.3.1. Design of limiter operation is intended to render this transient to be less severe than the two-pump trip. MCPR remains greater than the safety limit; therefore, no fuel damage occurs.

15.3.2.3.4 Consideration of Uncertainties

Initial conditions chosen for these analyses are conservative and tend to force analytical results to be more severe than otherwise expected.

These analyses, unlike the pump trip series, will be unaffected by deviations in pump/pump motor and driveline inertias, since it is the main valve that causes rapid recirculation decreases.

15.3.2.4 Barrier Performance

15.3.2.4.1 Fast Closure of One Recirculation Valve

The narrow-range level rises to the high level trip setpoint, causing scram and trip of the feedwater pumps and main turbine. Safety/relief valves open in the pressure relief mode and briefly discharge steam to the suppression pool. Peak pressures are less than those for the "Fast Closure of Two Recirculation Valves," given in Subsection 15.3.2.4.2. At approximately 26 sec, the wide-range level falls to the low water level trip setpoint, causing initiation of HPCS and RCIC system. However, there is a delay of up to 30 sec before the water supply from HPCS and RCIC system enters the vessel.

15.3.2.4.2 Fast Closure of Two Recirculation Valves

The narrow-range level rises to the high level trip setpoint causing scram and trip of the feedwater pumps and main turbine. Safety/relief valves open in the pressure relief mode and briefly discharge steam to the suppression pool. Pressure in the vessel bottom is limited to 1153 psig, well below the ASME code limit. At approximately 28 sec, the wide-range level falls to the low water level trip setpoint, causing initiation of the HPCS and RCIC system. However, there is a delay of up to 30 sec before the water supply from HPCS and RCIC system enters the vessel.

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 identified in Subsection 15.2.4.5 for a Type 2 event. Therefore, the radiological

15.3.2.5 Radiological Consequences (Continued)

exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.3.3 Recirculation Pump Seizure

15.3.3.1 Identification of Causes and Frequency Classification

The seizure of a recirculation pump is considered as a design basis accident (DBA) event. It has been evaluated as having a very mild accident in relation to other DBAs such as the LOCA. The analysis has been conducted with consideration to a single- or two-loop operation. (Refer to Section 5.1 for special mechanical considerations and Chapter 7 for electrical aspects.)

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

15.3.3.1.1 Identification of Causes

The case of recirculation pump seizure represents the extremely unlikely event of instantaneous stoppage of the pump motor shaft of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the large hydraulic resistance introduced by the stopped rotor.

15.3.3.1.2 Frequency Classification

This event is considered to be a limiting fault but results in effects which can easily 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

The operator should ascertain that the reactor scrams from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump, and he should monitor reactor water level and pressure control after shutdown.

15.3.3.2.2 Systems Operation

In order 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.

Operation of safe shutdown features, though not included in this simulation, is expected to be utilized in order to maintain adequate water level.

15.3.3.2.3 The Effect of Single Failures and Operator Errors

Single failures in the scram logic originating via the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2 (see Appendix 15A for further details).

15.3.3.3 Core and System Performance

15.3.3.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.3.3.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

For the purpose of evaluating consequences to the fuel thermal limits, this transient event is assumed to occur as a consequence of an unspecified, instantaneous stoppage of one recirculation pump shaft while the reactor is operating at 105% NBR steamflow. 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 15.0-2).

15.3.3.3.3 Results

Figure 15.3-5 represents the results of the accident. MCPR does not decrease significantly before fuel surface heat flux begins dropping enough to restore greater thermal margins. The level swell produces a trip of the main turbine and feedwater pumps and scram at 3.1 sec into the transient. The scram conditions impose no threat to thermal limits. Additionally, the momentary opening of the bypass valves and some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.3.3.1 Considerations of Uncertainties

Considerations of uncertainties are included in the GETAB analysis.

15.3.3.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.3.5 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 activation. 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 cover the consequences of this event.

15.3.4 Recirculation Pump Shaft Break

15.3.4.1 Identification of Causes and Frequency Classification

The breaking of the shaft of a recirculation pump is considered as a DBA event. It has been evaluated as a very mild accident in relation to other DBAs such as the LOCA. The analysis has been conducted with consideration to a single- or two-loop operation. (Refer to Chapter 5 for specific mechanical considerations and Chapter 7 for electrical aspects.)

15.3.4.1 Identification of Causes and Frequency Classification (Continued)

This postulated event is bounded by the more limiting case of recirculation pump 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 recirculation pump shaft breakage represents the extremely unlikely event of instantaneous stoppage of the pump motor operation of one recirculation pump. This event produces a very rapid decrease of core flow as a result of the break of the pump shaft.

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 pump motor shaft of one recirculation pump (Subsection 15.3.4.1.1.) will cause the core flow to decrease rapidly resulting in water level swell in the reactor vessel. When the vessel water level reaches the high water level setpoint (Level 8), scram, main turbine trip and feedwater pump trip will be initiated. Subsequently, the remaining recirculation pump trip will be initiated due to the turbine trip. Eventually, the vessel water level will be controlled by HPCS and RCIC flow.

15.3.4.2.1.1 Identification of Operator Actions

The operator should ascertain that the reactor scrams resulting from reactor water level swell. The operator should regain control of reactor water level through RCIC operation or by restart of a feedwater pump; and he should monitor reactor water level and pressure control after shutdown.

15.3.4.2.2 Systems Operation

Normal operation of plant instrumentation and control is assumed. This event takes credit for vessel water level (L8) instrumentation to scram the reactor and trip the main turbine and feedwater pumps. High system pressure is limited by the pressure relief system operation.

Operation HPCS and RCIC systems is expected in order to maintain adequate water level control.

15.3.4.2.3 The Effect of Single Failures and Operator Errors

Effects of single failures in the high vessel level (L8) trip are similar to the considerations in Subsection 15.3.1.2.3.2.

Assumption of SEF or SOE in other equipment has been examined, and this has led to the conclusion that no other credible failure exists for this event. Therefore, the bounding case has been considered (see Appendix 15A for details).

15.3.4.3 Core and System Performance

The severity of this pump shaft break event is bounded by the pump seizure event described in Subsection 15.3.3. This can be easily demonstrated by consideration of the two events discussed in Subsection 15.3.4.3.1. Since this event is less limiting than

15.3.4.3 Core and System Performance (Continued)

the event described in Subsection 15.3.3, only qualitative evaluation is provided. Therefore, no discussion of mathematical model, input parameters and considerations of uncertainties, etc, is necessary.

15.3.4.3.1 Qualitative Results

If this extremely unlikely event occurs, core coolant flow will drop rapidly. The level swell produces a reactor scram and trip of the main and feedwater turbines. Since heat flux decreases much more rapidly than the rate at which heat is removed by the coolant, there is no threat to thermal limits. Additionally, the bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

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 loop decreases rapidly. In the case of the pump seizure event, the loop 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.

15.3.4.4 Barrier Performance

The bypass valves and momentary opening of some of the safety/relief valves limit the pressure well within the range allowed by the ASME vessel code. Therefore, the reactor coolant pressure boundary is not threatened by overpressure.

15.3.4.5 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 activation. 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 cover the consequences of this event.

Table 15.3-1

SEQUENCE OF EVENTS FOR FIGURE 15.3-1

Time (sec)	Event
0	Trip of one recirculation pump initiated
5.7	Jet pump diffuser flow reverses in the tripped loop
40	Core flow and power level stabilize at new equilibrium conditions

Table 15.3-2

SEQUENCE OF EVENTS FOR FIGURE 15.3-2

Time (sec)	Event
0	Trip of both recirculation pumps initiated
4.0	Vessel water level (L8) trip initiates scram, turbine trip and feedwater pump trip
4.1	Turbine trip initiates bypass operation
6.3	Safety/relief valves open due to high pressure
11.3	Safety/relief valves close
24.8	Vessel water level (L2) setpoint reached
58.6 (est)	HPCS and RCIC flow enters vessel (not simulated)

Table 15.3-3

SEQUENCE OF EVENTS FOR FIGURE 15.3-3

Time (sec)	Event
0	Initiate fast closure of one main recirculation valve
1.5	Jet pump diffuser flow reverses in the affected loop
4.1	Vessel level (L8) trip initiates scram, turbine trip and trip of the feedwater turbines
4.2	Turbine trip initiates bypass operation and recirculation pump trip (RPT)
6.9	Safety/relief valves open due to high pressure
12.4	Safety/relief valves close
26	Vessel water level reaches Level 2 (L2) setpoint
56 (est)	HPCS and RCIC flow enters vessel (not simulated)

Table 15.3-4

SEQUENCE OF EVENTS FOR FIGURE 15.3-4

Time (sec)	Event
0	Initiate fast closure of both main recirculation valves
6.1	Vessel level (L8) trip initiates scram and turbine trip
6.1	Feedwater pumps tripped off
6.2	Turbine trip initiates bypass operation and recirculation pump trip (RPT)
8.5	Safety/relief valves open due to high pressure
13.6	Safety/relief valves close
28.	Vessel water level reaches Level 2 setpoint
58. (est)	HPCS and RCIC flow enters vessel (not simulated)

Table 15.3-5

SEQUENCE OF EVENTS FOR FIGURE 15.3-5

Time (sec)	Event
0	Single pump seizure was initiated
0.6	Jet pump diffuser flow reverses in seized loop
3.1	Vessel level (L8) trip initiates scram
3.1	Vessel level (L8) trip initiates turbine trip
3.1	Feedwater pumps are tripped off
3.2	Turbine trip initiates bypass operation
3.2	Turbine trip initiates recirculation pumps trip
5.5	Safety/relief valves open due to high pressure
10.5	Safety/relief valves close
21.7	Main bypass valves close to regain pressure regulator control
24.2	Vessel water level reaches Level 2 (L2) setpoint
54.2 (est)	HPCS/RCIC flow enters the vessel (not simulated)

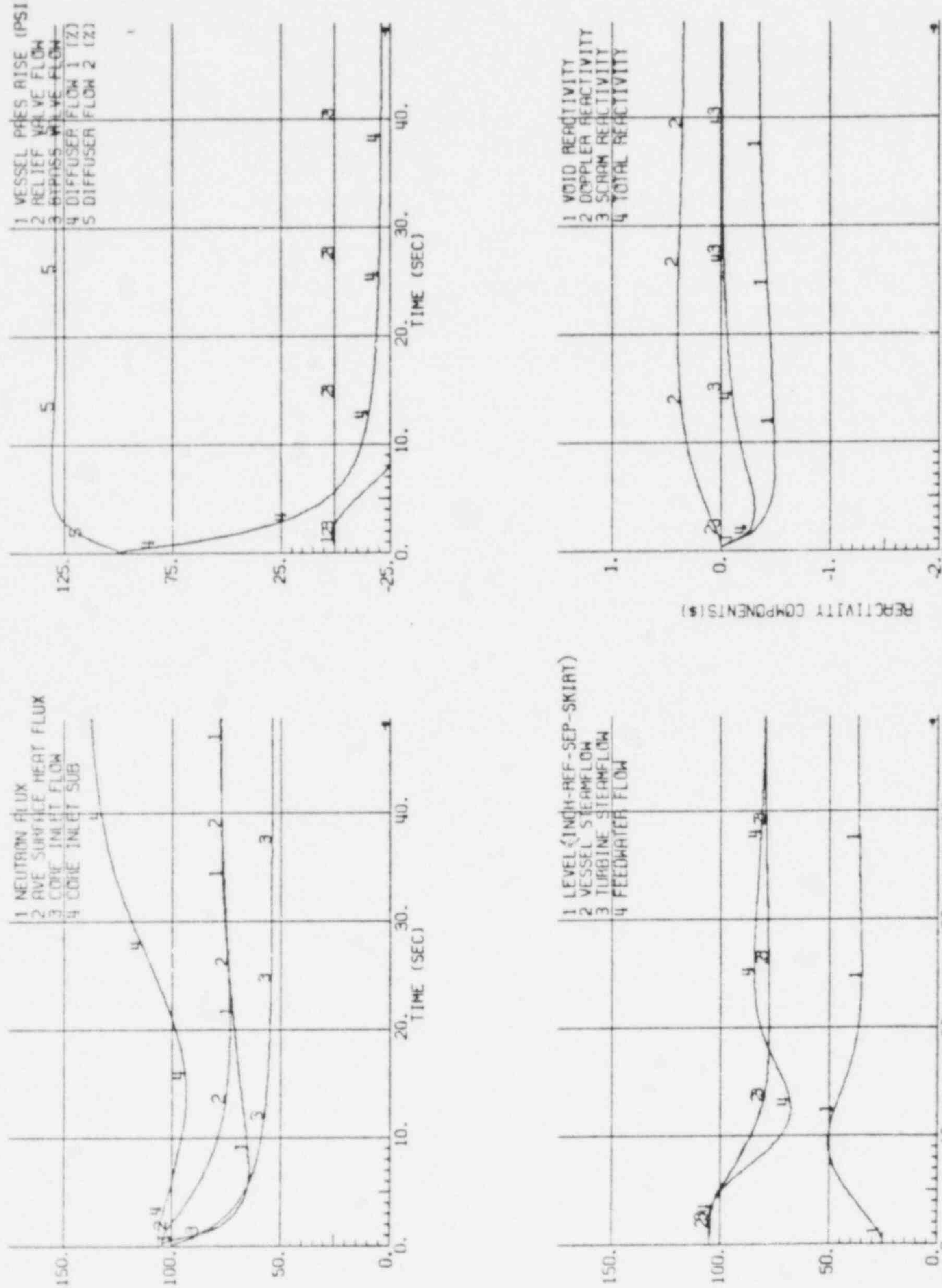

Figure 15.3-1. Trip of One Recirculation Pump

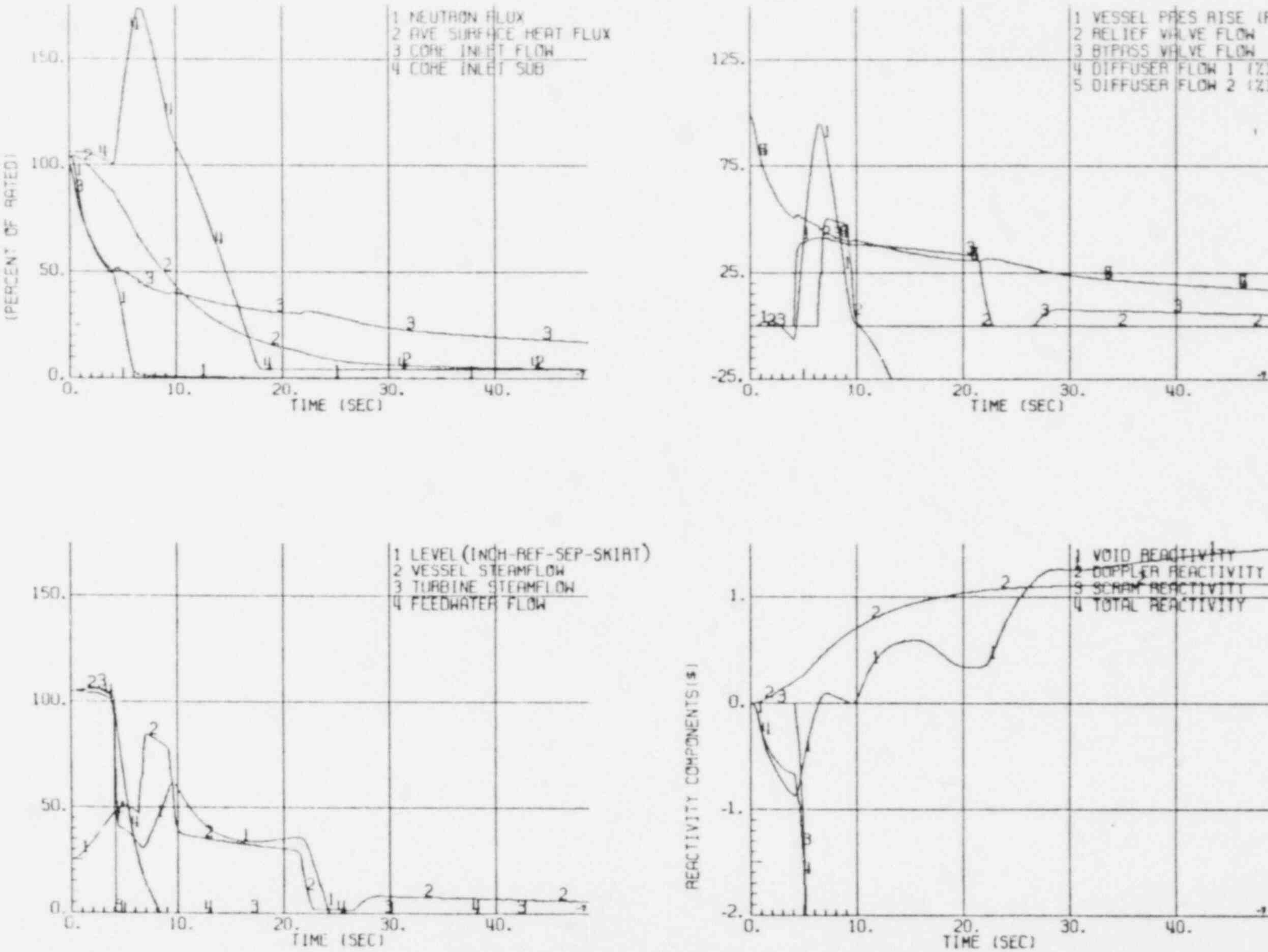

Figure 15.3-2. Two Recirculation Pump Trip

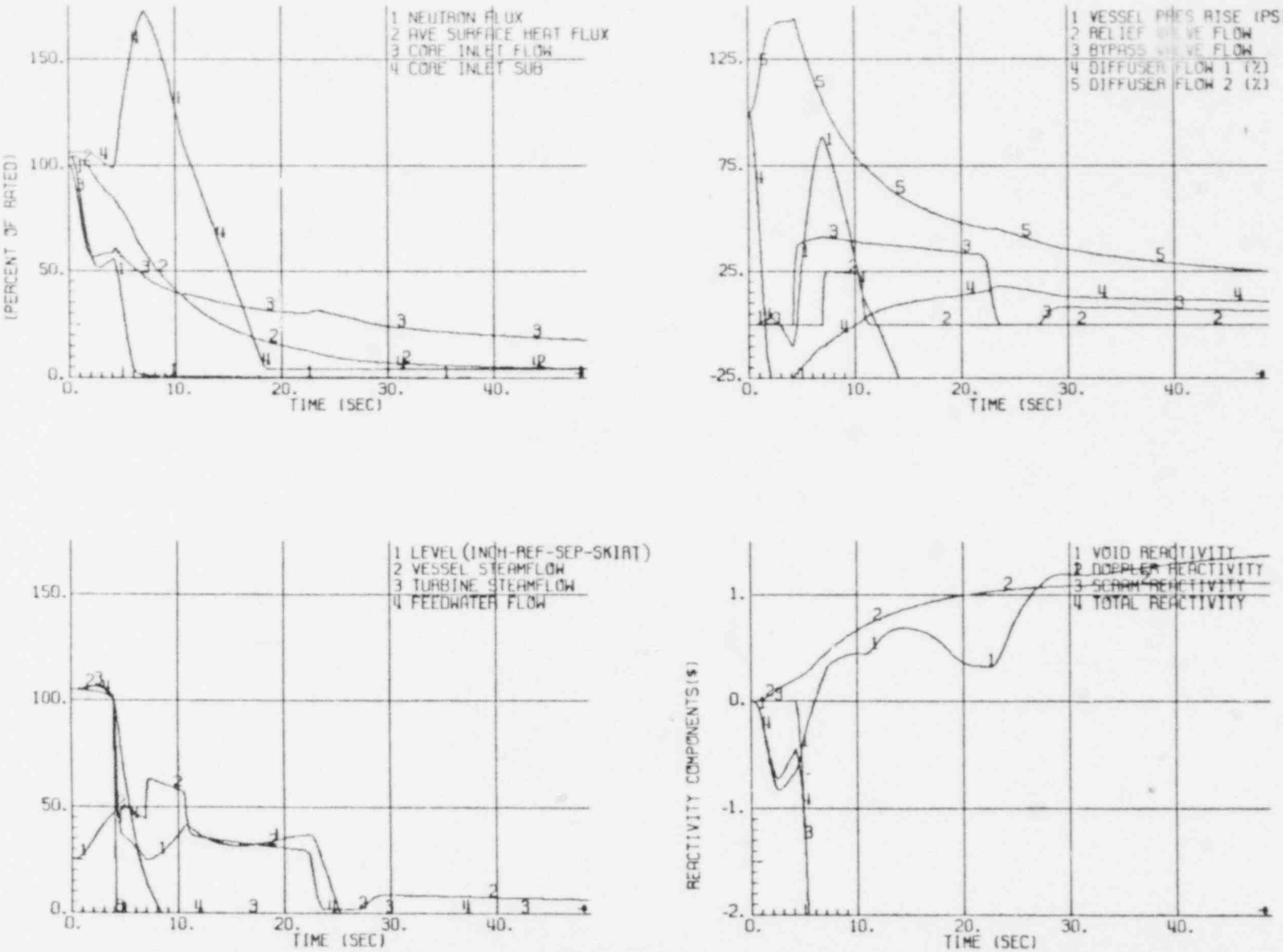

Figure 15.3-3. Fast Closure of One Recirculation Valve at 60%/sec

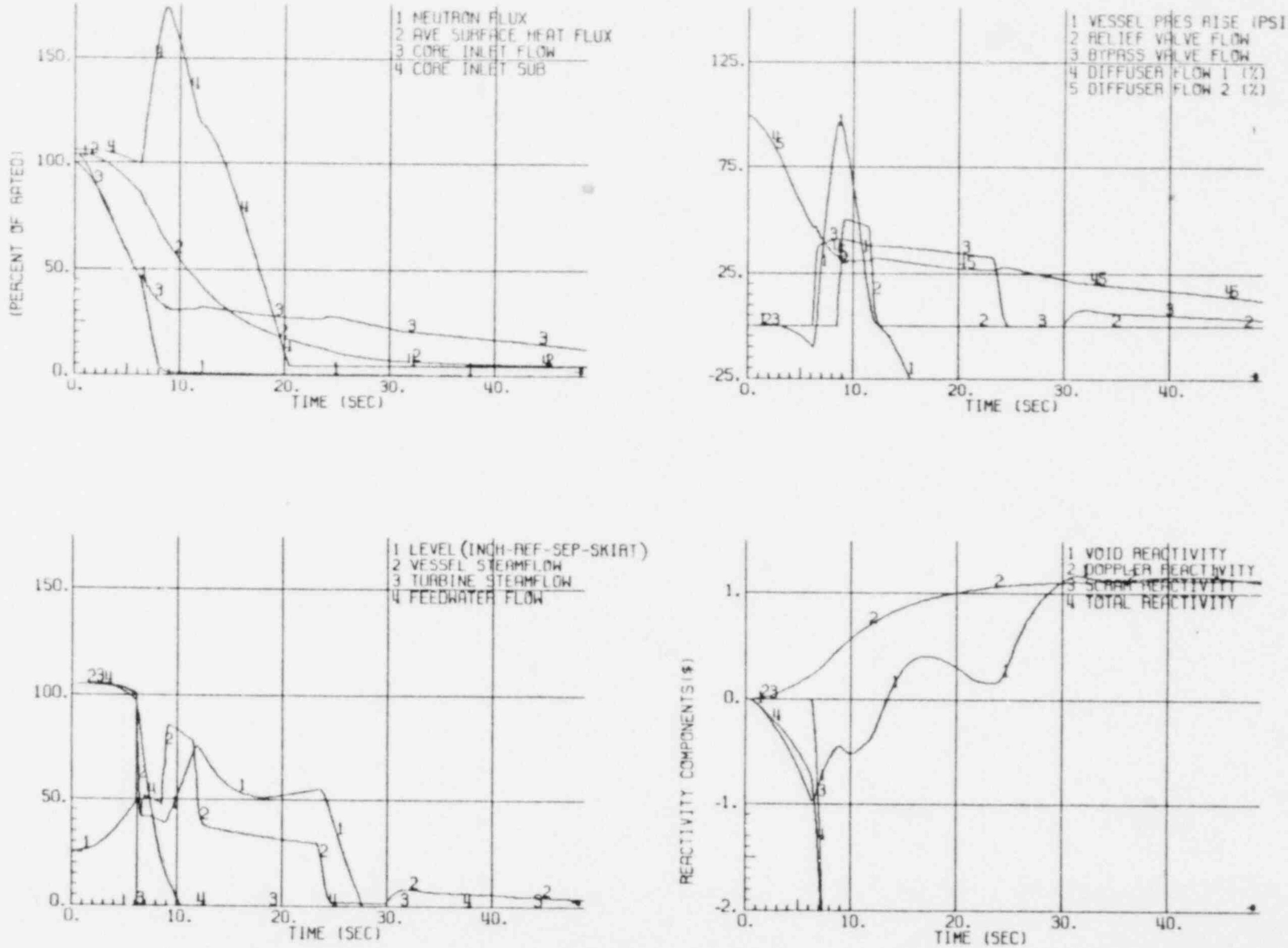

Figure 15.3-4. Fast Closure of Both Recirculation Valves at 11%/sec

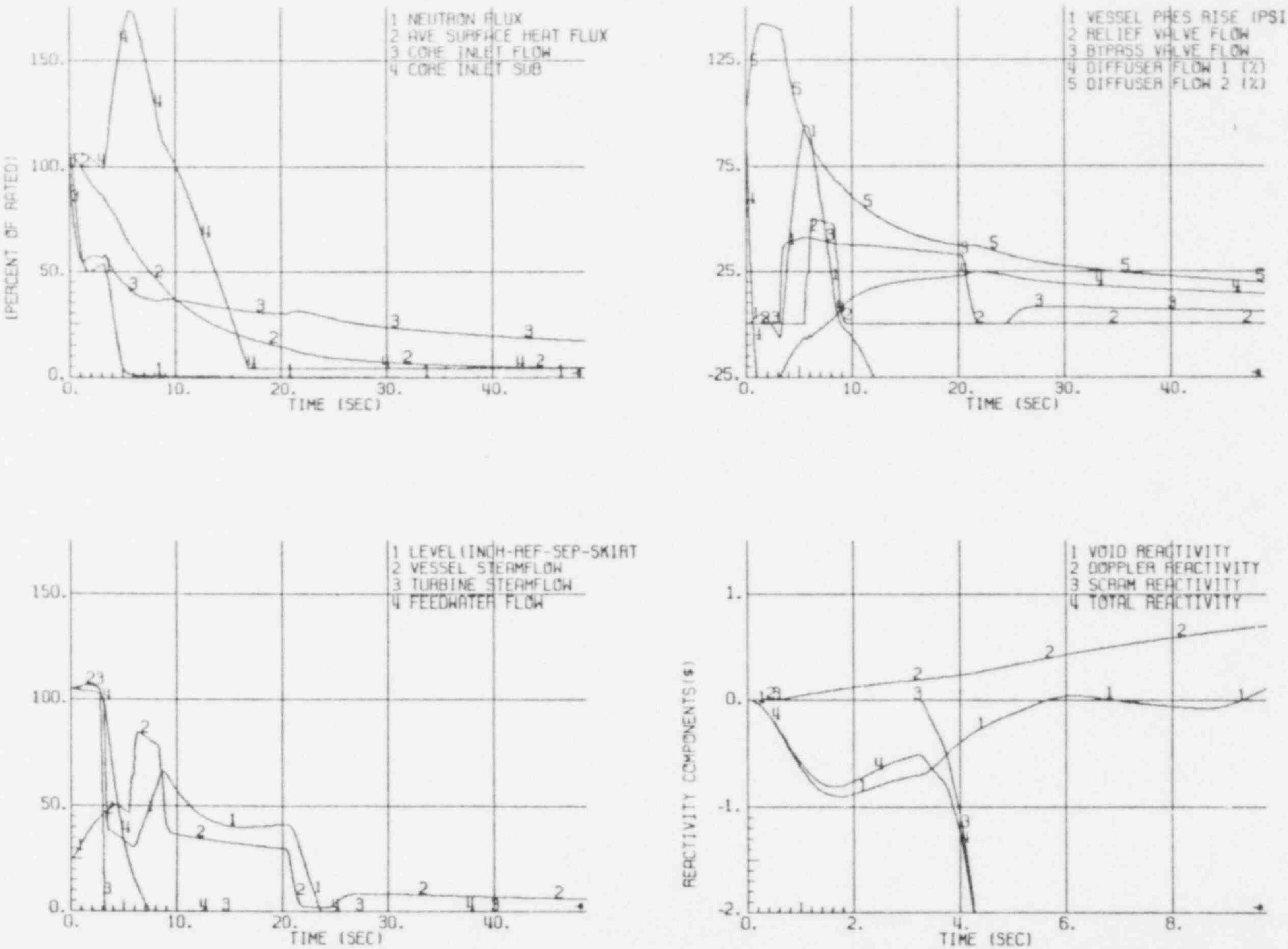

Figure 15.3-5. Seizure of One Recirculation Pump

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SECTION 15.4

TABLES

SECTION 15.4

ILLUSTRATIONS

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, since 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 operations, 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 containing no 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

15.4.1.1.2.2 Fuel Insertion With Control Rod Withdrawn (Continued)

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. Any attempt to withdraw a second rod results in a rod block by the refueling interlocks. Since the core is designed to meet shutdown requirements with the highest worth rod withdrawn, the core remains subcritical even with one rod withdrawn.

15.4.1.1.2.4 Control Rod Removal Without Fuel Removal

Finally, the design of the control rod, incorporating the velocity limiter, does not physically permit the upward removal of the control rod without 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, since the protection system design as discussed above will prevent its occurrence.

15.4.1.1.2.6 Effect of Single Failure and Operator Errors

If any one of the operations involved in initial failure or error is followed by any other single component failure or single operator error, the necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation (see Appendix 15 for details).

15.4.1.1.3 Core and System Performance

Since the possibility of inadvertent criticality during refueling is precluded, the core and system performances were not analyzed. The withdrawal of the highest worth control rod during refueling will not result in criticality. This is verified experimentally by performing shutdown margin checks (see Subsection 4.3.2 for a description of the methods and results of the shutdown margin analysis). Additional reactivity insertion is precluded by refueling interlocks. Since no fuel damage can occur, no radioactive material will be released from the fuel.

No mathematical models are involved in this event. The need for input parameters or initial conditions is eliminated, as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.1.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since there is not a postulated set of circumstances for which this event could occur.

15.4.1.1.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event, since 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

The probability of the initial causes of error of this event, alone, is considered low enough to warrant its being categorized

15.4.1.2.1 Identification of Causes and Frequency Classification (Continued)

as an infrequent incident. The probability of further single failures postulated for this event is even lower because it is contingent upon the simultaneous failure of two redundant inputs to the rod control and information system (RCIS), concurrent with a high worth rod, out-of-sequence rod selection, plus operator nonacknowledgment of continuous alarm annunciations prior to safety system actuations.

15.4.1.2.2 Sequence of Events and Systems Operation

15.4.1.2.2.1 Sequence of Events

Continuous control rod withdrawal errors during reactor startup are precluded by the RCIS. The RCIS prevents the withdrawal of an out-of-sequence control rod in the 100%-75% control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75% rod density to the low power setpoint. Since only in-sequence control rods can be withdrawn in the 100%-75% control rod density and control rods are withdrawn in the banked position mode from the 75% control rod density point to the low power setpoint, there is no basis for the continuous control rod withdrawal error in the startup and low power range. (See Subsection 15.4.2 for description of continuous control rod withdrawal above the low power setpoint. The bank position mode of the RCIS is described in Reference 1.

15.4.1.2.2.2 Identification of Operator Actions

No operator actions are required to preclude this event, since the plant design as discussed above prevents its occurrence.

15.4.1.2.2.3 Effects of Single Failure and Operator Errors

If any one of the operations involved the initial failure or error and is followed by another SCF or SOE, the necessary safety actions are automatically taken (e.g., rod blocks) prior to any limit violation (see Appendix 15A for details).

15.4.1.2.3 Core and System Performance

The performance of the RCIS prevents erroneous selection and withdrawal of an out-of-sequence control rod. Thus, the core and system performance is not affected by such an operator error.

No mathematical models are involved in this event. The need for input parameters or initial conditions is not required as there are no results to report. Consideration of uncertainties is not appropriate.

15.4.1.2.4 Barrier Performance

As evaluation of the barrier performance was not made for this event, since there is no postulated set of circumstances for which this error could occur.

15.4.1.2.5 Radiological Consequences

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

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

The Rod Withdrawal Error (RWE) transient results from a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously until the Rod Withdrawal Limiter (RWL) function of the Rod Control and Information System (RCIS) blocks further withdrawal.

15.4.2.1.2 Frequency Classification

The frequency of occurrence for the RWE is assumed to be moderate, since definite data do not exist. The frequency of occurrence diminishes as the reactor approaches full power by virtue of the reduced number of control rod movements. A statistical approach, using appropriate conservative acceptance criteria, shows that consequences of the majority of RWEs would be very mild and hardly noticeable.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for this transient is presented in Table 15.4-1.

15.4.2.2.2 System Operations

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod or gang of control rods continuously until the RWL inhibits further withdrawal. The RWL utilizes rod position indications of the selected rod as input.

15.4.2.2.2 System Operations (Continued)

During the course of this event, normal operation of plant instrumentation and controls is assumed, although no credit is taken for this except as described above. No operation of any engineered safety feature (ESF) is required during this event.

15.4.2.2.3 Single Failure or Single Operator Error

The effect of operator errors has been discussed above. It was shown that operator errors (which initiated this transient) cannot impact the consequences of this event due to the RCIS system. The RCIS system is designed to be single-failure proof; therefore, termination of this transient is assured (see Appendix 15A for details).

15.4.2.3 Core and System Performance

15.4.2.3.1 Mathematical Model

The consequences of a RWE are calculated utilizing a three-dimensional, coupled nuclear-thermal-hydraulics computer program (Reference 2). This model calculates the changes in power level, power distribution, core flow and critical power ratio under steady-state conditions, as a function of control blade position. For this transient, the time for reactivity insertion is greater than the fuel thermal time constant and core-hydraulic transport times, so that the steady-state assumption is adequate.

15.4.2.3.2 Input Parameters and Initial Conditions

The reactor core is assumed to be on MCPR and MLHGR technical specification limits prior to RWE initiation. A statistical analysis of the rod withdrawal error results (Appendix 15B) initiated from a wide range of operating conditions (exposure, power, flow, rod

15.4.2.3.2 Input Parameters and Initial Conditions (Continued)

patterns, xenon conditions, etc) has been performed, establishing allowable rod withdrawal increments applicable to all BWR/6 plants. These rod withdrawal increments were determined such that the design basis ΔMCPR (minimum critical power ratio) for rod withdrawal errors initiated from the technical specification operating limit and mitigated by the RWL system withdrawal restrictions, provides a 95% probability at the 95% confidence level that any randomly occurring RWE will not result in a larger ΔMCPR. MCPR was verified to be the limiting thermal performance parameter and therefore was used to establish the allowable withdrawal increments. The 1% plastic strain limit on the clad was always a less limiting parameter.

15.4.2.3.3 Results

The calculated results demonstrate that, should a rod or gang be withdrawn a distance equal to the allowable rod withdrawal increment, there exists a 95% probability at the 95% confidence level that the resultant ΔMCPR will not be greater than the design basis ΔMCPR. Furthermore, the peak LHGR will be substantially less than that calculated to yield 1% plastic strain in the fuel clad.

These results of the generic analyses in Appendix 15B show that a control rod or gang can be withdrawn in increments of 12 in. at power levels ranging from 70-100% of rated, and 24 in. at power levels ranging from 20-70% (Table 15.4-2). See Subsection 15.4.1.2 for RWE's below 20% reactor power. The 20% and 70% reactor core power levels correspond to the Low Power Set Point (LPSP) and High Power Set Point (HPSP) of the RWL.

15.4.2.3.4 Consideration of Uncertainties

The most significant uncertainty for this transient is the initial control rod pattern and the location of the rods or gang improperly

15.4.2.3.4 Consideration of Uncertainties (Continued)

selected and withdrawn. Because of the near-infinite combinations of control patterns and reactor states, all possible states cannot be analyzed. However, because only high worth gangs were included in the statistical analysis, enough points have been evaluated so as to clearly establish the 95%/95% confidence level. This effectively bounds the results from any actual operator error of this type with the indicated probabilities.

Quasi-steady-state conditions were assumed for thermal-hydraulic conditions. Although the uncertainty introduced by this assumption is not conservative, the magnitude of the effects neglected is insignificant relative to the result of the transient.

15.4.2.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since this is a localized event with very little change in the gross core characteristics. Typically, an increase in total core power for RWEs initiated from rated conditions is less than 4% and the changes in pressure are negligible.

15.4.2.5 Radiological Consequences

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

15.4.3 Control Rod Maloperation (System Malfunction or Operator Error)

This event is covered with evaluation cited in Subsections 15.4.1 and 15.4.2.

15.4.4 Abnormal Startup of Idle Recirculation Pump

15.4.4.1 Identification of Causes and Frequency Classification

15.4.4.1.1 Identification of Causes

This action results directly from the operator's manual action to initiate pump operation. It assumes that the remaining loop is already operating.

15.4.4.1.1.1 Normal Restart of Recirculation Pump at Power

This transient is categorized as an incident of moderate frequency.

15.4.4.1.1.2 Abnormal Startup of Idle Recirculation Pump

This transient is categorized as an incident of moderate frequency.

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 Figure 15.4-1.

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 loop start;

(2) determine that the idle recirculation pump suction and discharge block values are open and that the flow control valve in the idle loop is at minimum position and, if not, place them in this configuration;

15.4.4.2.1.1 Operator Actions (Continued)

(3) readjust flow of the running loop downward to less than half of the rated flow;

(4) determine that the temperature difference between the two loops is no more than 50°F;

(5) start the idle loop pump and adjust flow to match the adjacent loop flow (monitor reactor power); and

(6) readjust power, as necessary, to satisfy plant requirements per standard procedure.

NOTE: The time to do the above work is approximately 1/2 hour.

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 transient.

15.4.4.2.3 The Effect of Single Failures and Operator Errors

Attempts by the operator to start the pump at higher power levels will result in a reactor scram on flux (see Appendix 15A for details).

15.4.4.3 Core and System Performance

15.4.4.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.4.4.3.2 Input Parameters and Initial Conditions

This analysis has been performed, unless otherwise noted, with plant conditions tabulated in Table 15.0-2.

One recirculation loop is idle and filled with cold water (100°F). (Normal procedure when starting an idle loop with one pump already running requires that the indicated idle loop temperature be no more than 50°F lower than the indicated active loop temperature.)

The active recirculation loop is operating with the flow control valve position that produces about 70% of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 33% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

The idle recirculation pump suction and discharge block valves are open and the recirculation flow control valve is closed to its minimum open position. (Normal procedure requires leaving an idle loop in this condition to maintain the loop temperature within the required limits for restart.)

15.4.4.3.3 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15.4-1. Shortly after the pump begins to move, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply. The motor approaches synchronous speed in approximately 3 sec because of the assumed minimum pump and motor inertia.

A short-duration neutron flux peak is produced as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at 80% of rated

15.4.4.3.3 Results (Continued)

before decreasing after the cold water washed out of the loop at about 18 sec. No damage occurs to the fuel barrier and MCPR remains above the safety limit as the reactor settles out at its new steady-state condition.

15.4.4.3.4 Consideration of Uncertainties

This particular transient is analyzed for an initial power level that is much higher than that expected for the actual event. The much slower thermal response of the fuel mitigates the effects of the rather sharp neutron flux spike and, even in this high range of power, no threat to thermal limits is possible.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient (Figure 15.4-1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since 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

Failure of the master controller of neutron flux controller can cause an increase in the core coolant flow rate. Failure within

15.4.5.1.1 Identification of Causes (Continued)

a loop's flow controller can also cause an increase in core coolant flow rate.

15.4.5.1.2 Frequency Classification

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

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 Opening of One Recirculation Valve

Table 15.4-4 lists the sequence of events for Figure 15.4-2.

15.4.5.2.1.2 Fast Opening of Two Recirculation Valves

Table 15.4-5 lists the sequence of events for Figure 15.4-3.

15.4.5.2.1.3 Identification of Operator Actions

Initial action by the operator should include:

(1) transfer flow control to manual and reduce flow to minimum, and

(2) identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has

15.4.5.2.1.3 Identification of Operator Actions (Continued)

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 MSLIVs from isolating;

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

(4) continue to 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; and

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

NOTE: Time required from first trouble alarm to restart would be approximately 1 hr.

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.

15.4.5.2.3 The Effect of Single Failures and Operator Errors

Both of these transients lead to a quick rise in reactor power level. Corrective action first occurs in the high flux trip which, being part of the reactor protection system, is designed to single-failure criteria (see Appendix 15A for details). Therefore, shutdown is assured. Operator errors are not of concern here since automatic shutdown events follow so quickly after the postulated failure.

15.4.5.3 Core and System Performance

15.4.5.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this event.

15.4.5.3.2 Input Parameters and Initial Conditions

These analyses have been performed, unless otherwise noted, with plant conditions listed in Table 15.0-2.

In each of these transient events, the most severe transient results when initial conditions are established for operation at the low end of the rated flow control rod line. Specifically, this is 54% NBR power and 33% core flow. The maximum stroking rate of the recirculation loop valves for a master controller failure driving two loops is limited by individual loop controls to 11%/sec

Maximum stroking rate of a single recirculation loop value for a loop controller failure is limited by hydraulics to 30%/sec.

15.4.5.3.3 Results

15.4.5.3.3.1 Fast Opening of One Recirculation Valve

Figure 15.4-2 shows the analysis of a failure where one recirculation loop main valve is opened at its maximum stroking rate of 30%/sec. Table 15.4-4 provides the sequence of events of this failure.

The rapid increase in core flow causes a sharp rise in neutron flux, initiating a reactor scram at approximately 1.3 sec. The peak neutron flux reached was 235% of NBR value, while the accompanying average fuel surface heat flux reaches 73% of NBR at approximately 2.2 sec. MCPR remains considerably above the safety limit and average fuel temperature increases only 108°F. Reactor pressure is discussed in Subsection 15.4.5.4.

15.4.5.3.3.2 Fast Opening of Two Recirculation Valves

Figure 15.4-2 illustrates the failure where both recirculation loop main valves are opened at a maximum stroking rate of 11%/sec. Table 15.4-5 shows the sequence of events for this failure. It is very similar to the above transient. Flux scram occurs at approximately 1.6 sec, peaking at 162% of NB rated, while the average surface heat flux reaches 67% of NB rated at approximately 2.3 sec. MCPR remains considerably above the safety limit and average fuel temperature increases 80°F.

As indicated above, this is the most severe set of conditions under which this transient may occur. The results expected from an actual occurrence of this transient will be less severe than those calculated.

15.4.5.3.4 Considerations of Uncertainties

Some uncertainties in void reactivity characteristics, scram time and worth are expected to be more optimistic and will therefore lead to reducing the actual severity over that which is simulated herein.

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Opening of One Recirculation Valve

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

15.4.5.4.2 Fast Opening of Two Recirculation Valves

This transient results in a very 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, since 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 Misplaced 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 improper loading of a fuel bundle 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 misloaded into a wrong location in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to also be put in an incorrect location or discharged. Third, the misplaced bundles would have to be overlooked during the core verification process performed following core loading.

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 the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequency incident based on the following data:

Expected Frequency: 0.002 events/operating cycle

The above number is 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 misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, 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.

15.4.7.2.3 Effect of Single Failure and Operator Errors

This analysis already represents the worst case [i.e., operation of a misplaced bundle with three single operator errors (SOE)].

15.4.7.3 Core and System Performance

15.4.7.3.1 Mathematical Model

A three-dimensional BWR simulator model is used to calculate the core performance resulting from this event. This model is described in detail in Reference 2.

The procedure for analysis of this event is based on the fact that the loading error has its worst consequences at end of cycle, where the k-infinity difference between the mislocated bundle and its mirror image is greatest. Each possible fuel loading error is evaluated based on the expected MCPR, rather than at the operating limit. The expected MCPR is the calculated value corrected with observed calculation versus field operating data biases. The procedure is described in References 3 and 4.

15.4.7.3.2 Input Parameters and Initial Conditions

The equilibrium core consists of one bundle type loaded in four separate cycles (see the reference loading pattern, Figure 4.3-1).

15.4.7.3.2 Input Parameters and Initial Conditions (Continued)

The bundles have a range of power distributions depending on their accumulated exposure and position in the core. The fuel bundle loading error involves interchanging a fresh fuel bundle with a highly exposed bundle near a high radial peaking core position. Such an area is indicated via the MCPR position, MLHGR position, or by the two-dimensional radial power map.

After the loading errors are made and have gone undetected, the operator assumes that the mislocated bundle is operating at the same power as the instrumented bundle in the mirror image location and operates the plant until EOC. Applying the operating versus calculation bias to the EOC MCPRs results in the calculation of the expected MCPR in the event of a mislocated bundle.

A summary of input parameters for this analysis is given in Table 15.4-7.

15.4.7.3.3 Results

An analysis was performed to quantify the worst fuel bundle loading error for this equilibrium cycle. A summary of the results of that analysis is presented in Table 15.4-8. As can be seen, MCPR remains well above the MCPR safety limit, and MLHGR does not exceed the 1% plastic strain limit for the clad. Therefore, no violation of fuel limits occurs as a result of this event.

15.4.7.3.4 Considerations of Uncertainties

The model uncertainties are accounted for by the correction to observed operating plant characteristics.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event, since it is very mild and highly localized event. No perceptable change in the core pressure would be observed.

15.4.7.5 Radiological Consequences

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

15.4.8 Spectrum of Rod Ejection Assemblies

Not applicable to BWRs. This is a PWR event.

The BWR has precluded this event by incorporating into its design mechanical equipment which restricts any movement of the CRD system assemblies. The CRD housing support assemblies are described in Chapter 4.

15.4.9 Control Rod Drop Accident (CRDA)

15.4.9.1 Identification of Causes and Frequency Classification

15.4.9.1.1 Identification of Causes

The control rod drop accident (CRDA) is the result of a postulated event in which a high worth control rod, within the constraints of the rod pattern control (RPC), drops from the fully inserted or intermediate position in the core. The high worth rod becomes decoupled from its drive mechanism. The mechanism is withdrawn but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops to the CRD position. This can result in the removal of large negative reactivity from the core and results in a localized power excursion.

15.4.9.1.1 Identification of Causes (Continued)

A more detailed discussion is given in Reference 5.

15.4.9.1.2 Frequency Classification

The CRDA is categorized as a limiting fault because it is not expected to occur during the lifetime of the plant; but if postulated to occur, it has consequences that include the potential for the release of radioactive material from the fuel.

15.4.9.2 Sequence of Events and System Operation

15.4.9.2.1 Sequence of Events

Before the CRDA is possible, the sequence of events presented in Table 15.4-9 must occur. No operator actions are required to terminate this transient.

15.4.9.2.2 Systems Operation

The unlikely set of circumstances, referenced above, makes possible the rapid removal of a control rod. The dropping of the rod results in high reactivity in a small region of the core. For large, loosely coupled cores, this could result in a highly peaked power distribution and subsequent operation of shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion.

The RPC function of the RCIS limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This function prevents the movement of an out-of-sequence rod in the 100 to 75% rod density range, and from the 75% rod density point to the preset power level. The RCIS will only allow bank position mode rod withdrawals or insertions. The banked position mode of this system is described in Reference 1 for a typical BWR.

15.4.9.2.2 Systems Operation (Continued)

The RCIS uses redundant input to provide absolute assurance on CRD position. If either of the diverse inputs were to fail, the other would provide the necessary information.

The termination of this excursion is accomplished by automatic safety features of inherent shutdown mechanisms. Therefore, no operator action during the excursion is required. Although other normal plant instrumentation and controls are assumed to function, no credit for their operation is taken in the analysis of this event.

15.4.9.2.3 Effect of Single Failures and Operator Errors

Systems mitigating the consequences of this event are the RPC function of the RCIS and APRM scram. The RCIS is designed as a redundant system network and therefore provides single failure protection. The APRM scram system is designed to single-failure criteria. Therefore, termination of this transient within the limiting results discussed below is assured.

No operator error (in addition to the one that initiates this event) can result in a more limiting case, since the RCIS will prevent the inadvertent withdrawal of an out-of-sequence control rod and the reactor protection system will automatically terminate the transient.

Appendix 15A provides a detailed discussion on this subject.

15.4.9.3 Core and System Performance

15.4.9.3.1 Mathematical Model

The analytical methods, assumptions and conditions for evaluating the excursion aspects of the CRDA are described in detail in

15.4.9.3.1 Mathematical Model (Continued)

References 5, 6 and 7. They are considered to provide a conservative assessment of the associated consequences. The data presented in Reference 1 show that the RPC function of the RCIS reduces the control rod worths to the degree that the detailed analyses presented in References 5, 6 and 7, or the bounding analyses presented in Reference 8, are not necessary. Compliance checks are instead made to verify that the maximum rod worth does not exceed 1% Δk.

If this criterion is not met, then the bounding analysis is performed. The rod worths are determined using the BWR simulator model (Reference 2). Detailed evaluations, if necessary, are made using the methods described in References 5, 6 and 7.

15.4.9.3.2 Input Parameters and Initial Conditions

The core at the time of rod drop accident is assumed to be at the point in cycle which results in the highest incremental rod worth, to contain no xenon, to be in a hot-startup condition, and to have the control rods in sequence A at 50% rod density (groups 1-4 withdrawn). Removing xenon, which competes well for neutron absorptions, increases the fractional absorptions, or worth, of the control rods. The 50% control rod density ("black and white" rod pattern), which nominally occurs at the hot-startup condition, ensures that withdrawal of a rod results in the maximum increment of reactivity.

Since the maximum incremental rod worth is maintained at very low values, the postulated CRDA cannot result in peak enthalpies in excess of 280 calories per gram (cal/gm) for any plant condtion. The data presented in Subsection 15.4.9.3.3 show the maximum

15.4.9.3.2 Input Parameters and Initial Conditions (Continued)

control rod worth. Other input parameters and initial conditions are shown in Table 15.4-10.

15.4.9.3.3 Results

The radiological evaluations are based on the assumed failure of 770 fuel rods. The number of rods which exceed the damage threshold is less than 770 for all plant operating conditions or core exposure, provided the peak enthalpy is less than the 280 cal/gm design limit.

The results of the compliance-check calculation (Table 15.4-11) indicate that the maximum incremental rod worth is well below the worth required to cause a CRDA which would result in 280 cal/gm peak fuel enthalpy (References 5, 6). The conclusion is that the 280 cal/gm design limit is not exceeded and the assumed failure of 770 rods for the radiological evaluation is conservative.

15.4.9.4 Barrier Performance

An evaluation of the barrier performance was not made for this accident, since this is a highly localized event with no significant change in the gross core temperature or pressure.

15.4.9.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CRF100 guidelines. This analysis is referred to as the "Design Basis Analysis".

15.4.9.5 Radiological Consequences (Continued)

(2) The second analysis is based on assumptions considered to provide a realistic yet conservative estimate of ratiological consequences. This analysis is referred to as the "Realistic Analysis".

A schematic of the leakage path is shown in Figure 15.4-4.

15.4.9.5.1 Design Basis Analysis

The specific models, assumptions and the program used for computer evaluation are described in Reference 9. Specific parametric values used in the evaluation are presented in Table 15.4-12.

15.4.9.5.1.1 Fission Product Release from Fuel

The failure of 770 fuel rods is used for this analysis. The mass fraction of the fuel in the damaged rods which reaches or exceeds the initiation temperature of fuel melting (taken as 2842°C) is estimated to be 0.0077.

Fuel reaching melt conditions is assumed to release 100% of the noble gas inventory and 50% of the iodine inventory. The remaining fuel in the damaged rods is assumed to release 10% of both the noble gas and iodine inventories.

A maximum equilibrium inventory of fission products in the core is based on 1000 days of continuous operation at 3651 MWt. No delay time is assumed, but it is assumed that the failed rods have been operated at power level 1.5 times that of the average power level of the core.

15.4.9.5.1.2 Fission Product Transport to the Environment

The transport pathway is shown in Figure 15.4-4 and consists of carryover with steam to the turbine condenser prior to MSLIV closure, and leakage from the condenser to the environment. No credit is taken for the turbine building.

Of the activity released from the fuel, 100% of the noble gases and 10% of the iodines are assumed to be carried to the condenser before MSLIV closure is complete.

Of the activity reaching the condenser, 100% of the noble gases and 10% of the iodines (due to partitioning and plateout) remain airborne. The activity airborne in the condenser is assumed to leak directly to the environment a rate of 1.0% per day. Radioactive decay is accounted for during residence in the condenser; however, it is neglected after release to the environment.

The activity airborne in the condenser is presented in Table 15.4-13. The cumulative release of activity to the environment is presented in Table 15.4-14.

15.4.9.5.1.3 Results

The calculated exposures from the design basis analysis are presented in Table 15.4-15 and are well within the guidelines of 10CFR100.

15.4.9.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in

15.4.9.5.2 Realistic Analysis (Continued)

Reference 10. Specific values of parameters used in the evaluation are presented in Table 15.4-12.

15.4.9.5.2.1 Fission Product Release from Fuel

The following assumptions are used in calculating the fission product activity released from the fuel:

(1) The reactor has been operating at design power for 1000 days until 30 min prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30-min time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity in a failed fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 11).

(3) The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the fission products already present in the fuel.

15.4.9.5.2.2 Fission Product Transport to the Environment

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

(1) The recirculation flow rate is 25% of rated, and the steam flow to the condenser is 5% of rated. The 25% recirculation flow and 5% steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5% steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steamlines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steamline isolation valves to achieve full closure.

(2) The main steamline isolation valves are assumed to receive an automatic closure signal 0.5 sec after detection of high radiation in the main steamlines and to be fully closed at 5 sec from the receipt of the closure signal. The signal originates from the main steamline radiation monitors. The total amount of fission product activity transported to the condenser before the steamlines are isolated is, therefore, governed by the 5.5-sec isolation time and the conditions in (1) above.

(3) All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

15.4.9.5.2.2 Fission Product Transport to the Environment (Continued)

(4) The mass ratio of the halogen concentration in steam, to that of the water, is assumed to be 2%.

(5) Fission product plate-out is neglected in the reactor vessel, main steamlines, turbine and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100% of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor, volume of air, and volume of water. The partition factor assumed applicable is 100, while the ratio of air volume to water volume is taken as 3. Based on the above conditions, the activity airborne in the condenser is presented in Table 15.4-16.

The following assumptions and conditions are used to evaluate the activity released to the environment:

(1) The leak rate out of the condenser is 0.5% of the combined condenser and turbine free volume (2.47E5 ft^3) per day.

(2) The activity released from the condenser becomes airborne in the turbine building. The turbine building ventilation rate is 1327% per day.

Based on the above assumptions, the integrated fission product release to the environment is presented in Table 15.4-17.

15.4.9.5.2.3 Results

The calculated off-site exposures for the realistic analysis are presented in Table 15.4-18 and demonstrate the wide margin of conservatism in the design basis analysis.

15.4.9.6 References

1. C. J. Paone, "Bank Position Withdrawal Sequence", September 1976 (NEDO-21231).

2. J. A. Woolley, "Three-Dimensional Boiling Water Reactor Simulator", May 1976 (NEDO-20953).

3. Letter, Ronald Engel to Darrell Eisenhut, "Fuel Assembly Loading Error", June 1, 1977.

4. Letter, Ronald Engel to Darrell Eisenhut, "Fuel Assembly Loading Error", November 30, 1977.

5. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", March 1976 (NEDO-10527).

6. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", July 1972, Supplement 1 (NEDO-10527).

7. R. C. Stirn, et al., "Rod Drop Accident Analysis for Large BWRs", January 1973, Supplement 2 (NEDO-10527).

8. "GE BWR Generic Reload Application for 8x8 Fuel", Supplement 3 to Revision 1 (NEDO-20360).

9. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143).

15.4.9.6 References (Continued)

10. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

11. N. R. Horton, W. A. Williams, K. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors", March 1976 (APED-5756).

Table 15.4-1

SEQUENCE OF EVENTS

Elapsed Time (sec)	Event
0	Core is operated in a typical control rod pattern on limits
0	Operator withdraws a single rod or gang of rods continuously
∿1	The local power in the vicinity of the withdrawn rod (or gang) increases. Gross core power increases.
∿4*	RWL blocks further withdrawal
∿25	Core stabilizes at slightly higher core power level

*For a 1.0 ft RWL incremental withdrawal bloc. Time would be longer for a larger block, since rods are withdrawn at approximately 3 in./sec.

Table 15.4-2

ROD BLOCK ALARM DISTANCES (BWR/6)

Power Range (% of rated)	Allowable Withdrawal Distance (ft)
60 - 100	1.0
20 - 70	2.0
0 - 20	no restrictions*

*The BPWS function of the RCIS provides control of rod withdrawals below the 20% power setpoint and allows a maximum withdrawal distance of 9 ft.

Table 15.4-3

SEQUENCE OF EVENTS FOR FIGURE 15.4-1

Time (sec)	Event
0	Start pump motor
0.30	Jet pump diffuser flows on started pump side become positive
3.0	Pump motor at full speed and drive flow at about 21% of rated
18.0 (est)	Last of cold water leaves recirculation drive loop
18.1	Peak value of core inlet subcooling
50	Reactor variables settle into new steady state

Table 15.4-4

SEQUENCE OF EVENTS FOR FIGURE 15.4-2

Time (sec)	Event
0	Simulate failure of single loop control
1.3	Reactor APRM high-flux scram trip initiated
3.0 (est)	Turbine control valves start to close upon falling turbine pressure
6.5	Recirculation pump drive motors trip due to L3
25	Turbine control valves closed. Turbine pressure below pressure regulator setpoints
>100 (est)	Reactor variables settle into new steady-state

Table 15.4-5

SEQUENCE OF EVENTS FOR FIGURE 15.4-3

Time (sec)	Event
0	Initiate failure of master controller
1.6	Reactor APRM high-flux scram trip initiated
3.5 (est)	Turbine control valves start to close upon falling turbine pressure
5.6	Recirculation pump drive motors trip due to L3
32.0 (est)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints
>100 (est)	Reactor variables settle into new steady-state

Table 15.4-6

SEQUENCE OF EVENTS FOR THE MISPLACED 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 bundle.

(5) The plant continues to operate throughout the cycle.

Table 15.4-7

INPUT PARAMETERS AND INITIAL CONDITIONS FOR THE FUEL BUNDLE LOADING ERROR

(1)	Power (% rated)	100
(2)	Flow (% rated)	100
(3)	MCPR operating limit*	1.20
(4)	MLHGR operating limit (kW/ft)*	13.4
(5)	Core Exposure	End of Cycle

*These are the current operating limits. Since these limits do not go into the calculation of the MCPR associated with a mislocated bundle, future changes in the safety operating limits will not effect these results.

Table 15.4-8

RESULTS OF MISPLACED BUNDLE ANALYSIS
EQUILIBRIUM CYCLE

(1)	MCPR Safety Limit	1.07
(2)	MCPR with misplaced bundle	1.14
(3)	LHGR 1% plastic strain limit	>20 kW/ft
(4)	LHGR with misplaced bundle*	14.9

*Does not include any densification penalty.

Table 15.4-9

SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time (sec)	Event
	Reactor is operating at 50% rod density pattern.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod either individually or along with other control rods assigned to the RCIS group.
	Decoupled control rod sticks in the fully inserted or an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1	Reactor goes on a positive period and the initial power increase is terminated by the doppler coefficient.
<1	APRM 120% power signal scrams reactor.
<5	Scram terminates accident.

Table 15.4-10

INPUT PARAMETERS AND INITIAL CONDITIONS FOR ROD WORTH COMPLIANCE CALCULATION

1.	Reactor Power (% rated)	1
2.	Reactor Flow (% rated)	100
3.	Core Average Exposure (MWd/t)	Most reactive Point in cycle
4.	Control Rod Fraction	~0.50
5.	Average Fuel Temperature (°C)	286
6.	Average Moderator Temperature (°C)	286
7.	Xenon State	None
8.	Core Average Void Fraction (%)	0

Table 15.4-11

INCREMENT WORTH OF THE MOST REACTIVE ROD USING BPWS

Core Condition	Control Rod Group	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase (k_{eff})
3000	7	04	(26,35)	00-08	0.00248
3000	7	08	(26,35)	00-12	0.00278
3000	7	12	(26,35)	00-48	0.00269
3000	7	48	(26,35)	00-48	0.00198

NOTE: The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:

(a) BOC

(b) Hot Startup

(c) No Xenon

Table 15.4-12

CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

		Design Basis Assumptions	Realistic Basis Assumptions
I.	Data and assumptions used to estimate radioactive source from postulated accidents:		
	A. Power level	3651 MWt	3651 MWt
	B. Burnup	NA	NA
	C. Fuel damaged	770 rods	770 rods
	D. Release of activity by nuclide	Table 15.4-14	Table 15.4-17
	E. Iodine fractions:		
	(1) Organic	0	0
	(2) Elemental	1	1
	(3) Particulate	0	0
	F. Reactor coolant activity before the accident	NA	NA
	G. Peaking factor	1.5	1.0
II.	Data and assumptions used to estimate activity released:		
	A. Condenser leak rate (%/day)	1.0	0.5
	B. Turbine building leak rate (%/day)	NA	1327
	C. Valve closure time (sec)	NA	5
	D. Absorption and filtration efficiencies:		
	(1) Organic iodine	NA	NA
	(2) Elemental iodine	NA	NA
	(3) Particulate iodine	NA	NA
	(4) Particulate fission products	NA	NA
	E. Recirculation system parameters:		
	(1) Flow rate	NA	NA
	(2) Mixing efficiency	NA	NA
	(3) Filter efficiency	NA	NA
	F. Containment spray parameters (flow rate, drop size, etc)	NA	NA
	G. Containment volumes	NA	NA
	H. All other pertinent data and assumptions	None	None

Table 15.4-12 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data:		
A. Site Boundary and LPZ distances (m)	*	*
B. X/Q's for time intervals of:		
(1) 0-1 hr - SB/LPZ	2.0E-3/1.0E-2	2.0E-3/1.0E-3
(2) 1-8 hr - SB/LPZ	3.8E-4	3.8E-4
(3) 8-16 hr - SB/LPZ	1.0E-4	1.0E-4
(4) 16 hr-3 days - LPZ	3.4E-5	3.4E-5
(5) 3-26 day - LPZ	7.5E-6	7.5E-6
IV. Dose Data:		
A. Method of dose calculation	Reference 15.4.9.6-9	Reference 15.4.9.6-10
B. Dose conversion assumptions	Reference 15.4.9.6-9	Reference 15.4.9.6-10
C. Peak activity concentrations in condenser	Table 15.4-13	Table 15.4-16
D. Doses	Table 15.4-15	Table 15.4-18

*Applicant to Supply

Table 15.4-13

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.1E 03	2.1E 03	2.0E 03	1.5E 03	1.2E 02
I132	3.6E 03	3.1E 03	2.7E 03	2.0E 03	1.1E 03	3.2E 02	9.4E 01	2.5E 00	7.5E-10	0.
I133	3.3E 03	3.3E 03	3.2E 03	3.1E 03	2.9E 03	2.6E 03	2.2E 03	1.5E 03	1.3E 02	9.5E-08
I134	5.6E 03	3.8E 03	2.6E 03	1.2E 03	2.4E 02	1.0E 01	4.2E-01	3.1E-05	0.	0.
I135	4.7E 03	4.5E 03	4.2E 03	3.8E 03	3.1E 03	2.0E 03	1.3E 03	3.7E 02	1.8E-01	0.
Total I	1.9E 04	1.7E 04	1.5E 04	1.2E 04	9.5E 03	7.0E 03	5.7E 03	3.9E 03	1.6E 03	1.2E 02
Kr83m	2.5E 04	2.1E 04	1.8E 04	1.2E 04	5.7E 03	1.3E 03	2.8E 02	3.2E 00	5.8E-12	0.
Kr85m	6.1E 04	5.6E 04	5.2E 04	4.5E 04	3.3E 04	1.8E 04	9.5E 03	1.5E 03	2.0E-02	0.
Kr85	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.5E 03	1.5E 03	1.2E 03
Kr87	1.2E 05	9.5E 04	7.3E 04	4.2E 04	1.4E 04	1.6E 03	1.8E 02	2.5E 01	0.	01
Kr88	1.8E 05	1.6E 05	1.4E 05	1.1E 05	6.6E 04	2.4E 04	9.1E 03	4.6E 02	7,8E-06	0.
Kr89	1.8E 05	3.1E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe131m	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe133m	6.1E 04	6.1E 04	6.0E 04	6.0E 04	5.8E 04	5.5E 04	5.2E 04	4.4E 04	1.7E 04	4.1E 00
Xe133	3.6E 05	3.5E 05	3.5E 05	3.5E 05	3.5E 05	3.4E 05	3.3E 05	3.1E 05	2.0E 05	5.1E 03
Xe135m	9.7E 04	2.6E 04	6.7E 03	4.4E 02	1.9E 00	3.6E-05	6.9E-10	9.	0.	0.
Xe135	6.5E 04	6.2E 04	6.0E 04	5.6E 04	4.8E 04	3.5E 04	2.6E 04	1.0E 04	4.3E 01	0.
Xe137	3.9E 05	2.1E 03	9.2E 00	1.8E-04	7.0E-14	0.	0.	0.	0.	0.
Xe138	4.3E 05	1.0E 05	2.4E 04	1.3E 03	3.6E 00	2.9E-05	2.4E-10	0.	0.	01
Total NG	2.0E 06	9.4E 05	7.9E 05	6.8E 05	5.7E 05	4.8E 05	4.3E 05	3.7E 05	2.2E 05	6.5E 03

Table 15.4-14

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO ENVIRONMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	1.5E-02	4.6E-01	9.1E-01	1.8E 00	3.6E 00	7.2E 00	1.1E 01	2.1E 01	7.3E 01	2.1E 02
I132	2.5E-02	7.0E-01	1.3E 00	2.3E 00	3.5E 00	4.5E 00	4.8E 00	4.9E 00	4.9E 00	4.9E 00
I133	2.3E-02	6.9E-01	1.4E 00	2.7E 00	5.2E 00	9.8E 00	1.4E 01	2.3E 01	4.0E 01	4.1E 01
I134	3.9E-02	0.8E-01	1.6E 00	2.4E 00	2.9E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00
I135	3.3E-02	9.5E-01	1.9E 00	3.5E 00	6.4E 00	1.1E 01	1.3E 01	1.7E 01	1.9E 01	1.9E 01
Total I	1.4E-01	3.8E 00	7.1E 00	1.3E 01	2.2E 01	3.5E 01	4.6E 01	6.9E 01	1.4E 02	2.8E 02
Kr83m	1.8E-01	4.9E 00	8.9E 00	1.5E 01	2.2E 01	2.7E 01	2.8E 01	2.8E 01	2.8E 01	2.8E 01
Kr85m	4.2E-01	1.2E 01	2.4E 01	4.4E 01	2.7E 00	1.2E 00	1.4E 02	1.6E 02	1.6E 02	1.6E 02
Kr85	1.1E-02	3.3E-01	6.5E-01	1.3E 00	2.6E 00	5.2E 00	7,8E 00	1.6E 01	6.1E 01	4.0E 02
Kr87	8.7E-01	2.3E 01	4.0E 01	6.4E 01	8.5E 01	9.4E 01	9.5E 01	9.5E 01	9.5E 01	9.5E 01
Kr88	1.2E 00	3.5E 01	6.6E 01	1.2E 02	1.9E 02	2.6E 02	2.8E 02	3.0E 02	3.0E 02	3.0E 02
Kr89	1.4E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00
Xe131m	1.1E-02	3.2E-01	6.3E-01	1.3E 00	2.5E 00	5.0E 00	7.5E 00	1.5E 01	5.4E 01	2.0E 02
Xe133m	4.3E-01	1.3E 01	2.5E 01	5.0E 01	9.9E 01	1.9E 02	2.8E 02	5.2E 02	1.4E 03	1.9E 03
Xe133	2.5E 00	7.4E 01	1.5E 02	2.9E 02	5.9E 02	1.2E 03	1.7E 03	3.3E 03	1.1E 04	2.5E 04
Xe135m	6.9E -01	1.2E 01	1.4E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01
Xe135	4.5E-01	1.3E 01	2.6E 01	5.0E 01	9.3E 01	1.6E 02	2.1E 02	3.0E 02	3.5E 02	3.5E 02
Xe137	3.0E 00	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01
Xe138	3.0E 00	4.9E 01	6.0E 01	6.3E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01
Total NG	1.4E 01	2.6E 02	4.4E 02	7.4E 02	1.3E 03	2.1E 03	2.9E 03	4.9E 03	1.3E 04	2.8E 04

TABLE 15.4-15

CONTROL ROD DROP ACCIDENT
(DESIGN BASIS ANALYSIS)
Radiological Effects

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area		
*	0.22	2.55
Low Population Zone	0.16	4.08
*		

*Applicant to Supply

Table 51.4-16

CONTROL ROD DROP ACCIDENT (REALISTIC ANALYSIS) ACTIVITY AIRBORNE IN THE CONDENSER (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	2.92E-01	2.91E-01	2.90E-01	2.84E-01	2.68E-01	2.07E-01	2.20E-02
I132	4.47E-02	3.31E-02	2.45E-02	3.96E-03	3.08E-05	9.65E-15	0.
I133	1.43E-01	1.38E-01	1.34E-01	1.09E-01	6.42E-02	5.80E-03	5.46E-12
I134	3.36E-02	1.54E-02	6.99E-03	6.04E-05	1.89E-10	0.	0.
I135	1.09E-01	9.84E-02	8.86E-02	4.72E-02	8.77E-03	4.48E-06	0.
Total	6.23E-01	5.77E-01	5.44E-01	4.45E-01	3.41E-01	2.13E-01	2.20E-02
Kr83m	3.35E 01	2.32E 01	1.59E 01	1.68E 00	4.20E-03	7.78E-15	0.
Kr85m	2.28E 02	1.95E 02	1.67E 02	6.60E 01	5.53E 00	7.73E-05	0.
Kr85	2.26E 02	2.26E 02	2.26E 02	2.26E 02	2.25E 02	2.21E 02	1.94E 02
Kr87	1.91E 02	1.12E 02	6.46E 01	2.42E 00	3.80E-04	0.	0.
Kr88	4.30E 02	3.37E 02	2.63E 02	5.94E 01	1.13E 00	1.94E-08	0.
Kr89	1.13E-01	2.67E-07	5.06E-13	0.	0.	0.	0.
Xe131m	2.87E 01	2.86E 01	2.85E 01	2.81E 01	2.69E 01	2.23E 01	4.36E 00
Xe133m	4.41E 02	4.35E 02	4.29E 02	3.97E 02	3.21E 02	1.24E 02	3.39E-02
Xe133	4.27E 03	4.25E 03	4.22E 03	4.08E 03	3.73E 03	2.47E 03	7.17E 01
Xe135m	3.23E 00	2.23E-01	1.47E-02	1.22E-09	9.	0.	0.
Xe135	8.99E 02	8.34E 02	7.73E 02	4.91E 02	1.46E 02	6.17E-01	0.
Xe137	4.01E-01	9.45E-06	1.86E-10	0.	0.	0.	0.
Xe138	6.17E 01	3.46E 00	1.84E-01	4.26E-09	0.	0.	0.
Total	6.81E 03	6.44E 03	6.19E 03	5.35E 03	4.45E 03	2.84E 03	2.70E 02

Table 15.4-17

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	4.66E-09	1.41E-05	4.79E-05	3.72E-04	1.29E-03	4.84E-03	1.55E-02
I132	7.14E-10	1.78E-06	5.07E-06	1.72E-05	1.99E-05	1.99E-05	1.99E-05
I133	2.28E-09	6.75E-06	2.25E-05	1.59E-04	4.41E-04	8.05E-04	8.42E-04
I134	5.39E-10	1.00E-06	2.21E-06	3.67E-06	3.69E-06	3.69E-06	3.69E-06
I135	1.74E-09	4.93E-06	1.58E-05	8.88E-05	1.65E-04	1.82E-04	1.82E-04
Total	9.94E-09	2.86E-05	9.34E-05	6.41E-04	1.92E-03	5.85E-03	1.66E-02
Kr83m	5.35E-07	1.28E-03	3.50E-03	1.02E-02	1.12E-02	1.12E-02	1.12E-02
Kr85m	3.63E-06	9.97E-03	3.09E-02	1.51E-01	2.32E-01	2.40E-01	2.40E-01
Kr85	3.61E-06	1.09E-02	3.72E-02	2.92E-01	1.04E 00	4.40E 00	3.13E 01
Kr87	3.06E-06	6.59E-03	1.64E-02	3.60E-02	3.69E-02	3.69E-02	3.69E-02
Kr88	6.86E-06	1.78E-02	5.21E-02	2.01E-01	2.50E-01	2.50E-01	2.50E-01
Kr89	1.94E-09	8.96E-08	8.96E-08	8.96E-08	8.96E-08	8.96E-08	8.96E-08
Xe131m	4.57E-07	1.38E-03	4.70E-03	3.67E-02	1.28E-01	4.97E-01	1.93E 00
Xe133m	7.03E-06	2.11E-02	7.14E-02	5.37E-01	1.73E 00	4.85E 00	6.82E 00
Xe133	6.81E-05	2.06E-01	6.98E-01	5.39E 00	1.84E 01	6.43E 01	1.52E 02
Xe135m	5.23E-08	3.49E-05	4.30E-05	4.38E-05	4.38E-05	4.38E-05	4.38E-05
Xe135	1.43E-05	4.14E-02	1.35E-01	8.29E-01	1.77E 00	2.17E 00	2.18E 00
Xe137	6.80E-09	4.48E-07	4.48E-07	4.48E-07	4.48E-07	4.48E-07	4.48E-07
Xe138	1.00E-06	6.04E-04	7.21E-04	7.31E-04	7.31E-04	7.31E-04	7.31E-04
Total	1.09E-04	3.17E-01	1.05E 00	7.48E 00	2.36E 01	7.68E 01	1.95E 02

TABLE 15.4-18

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area *	9.4E-05	5.4E-05
Low Population Zone *	1.7E-04	2.0E-04

*Applicant to Supply

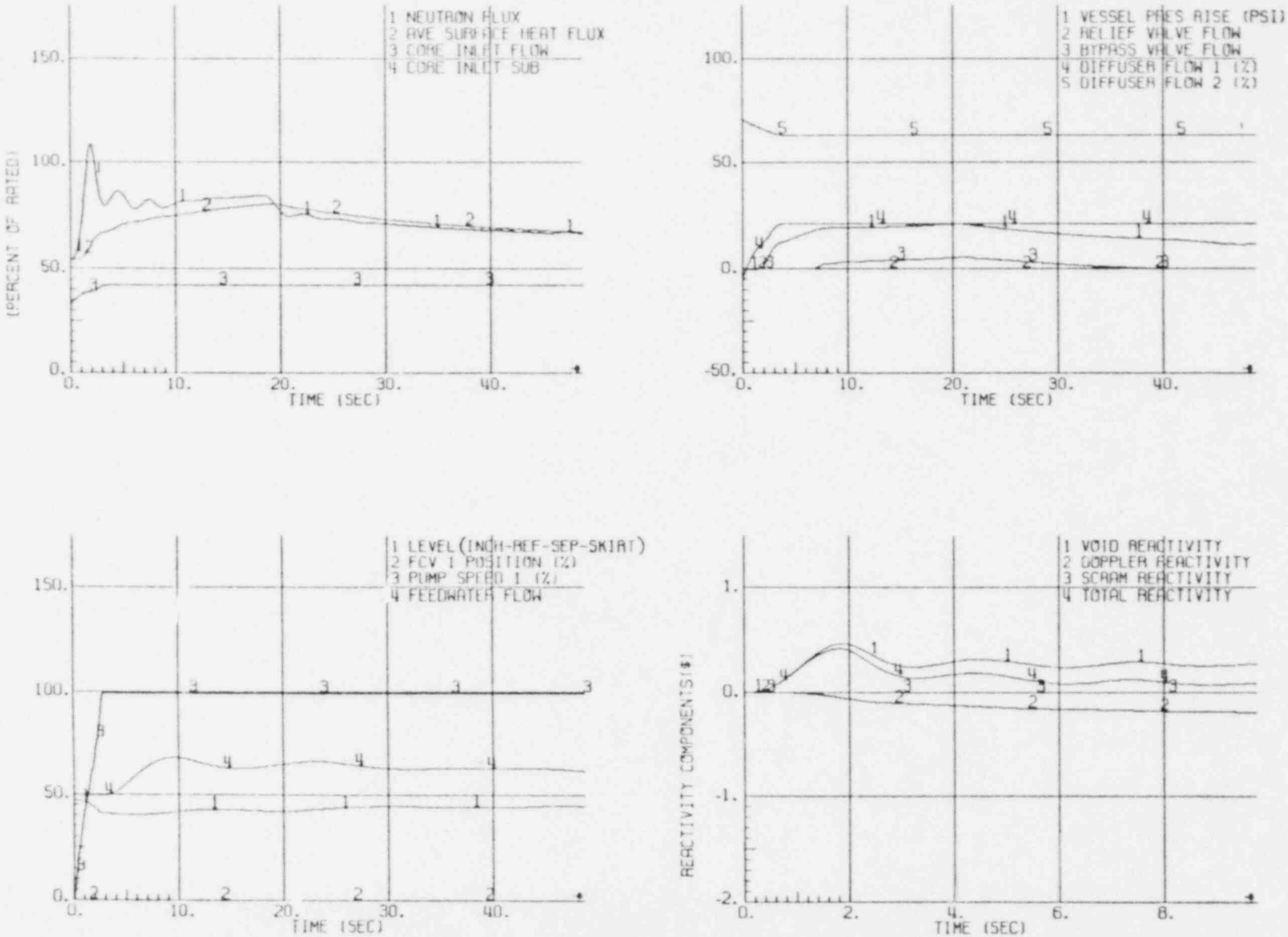

Figure 15.4-1. Startup of Idle Recirculation Loop Pump

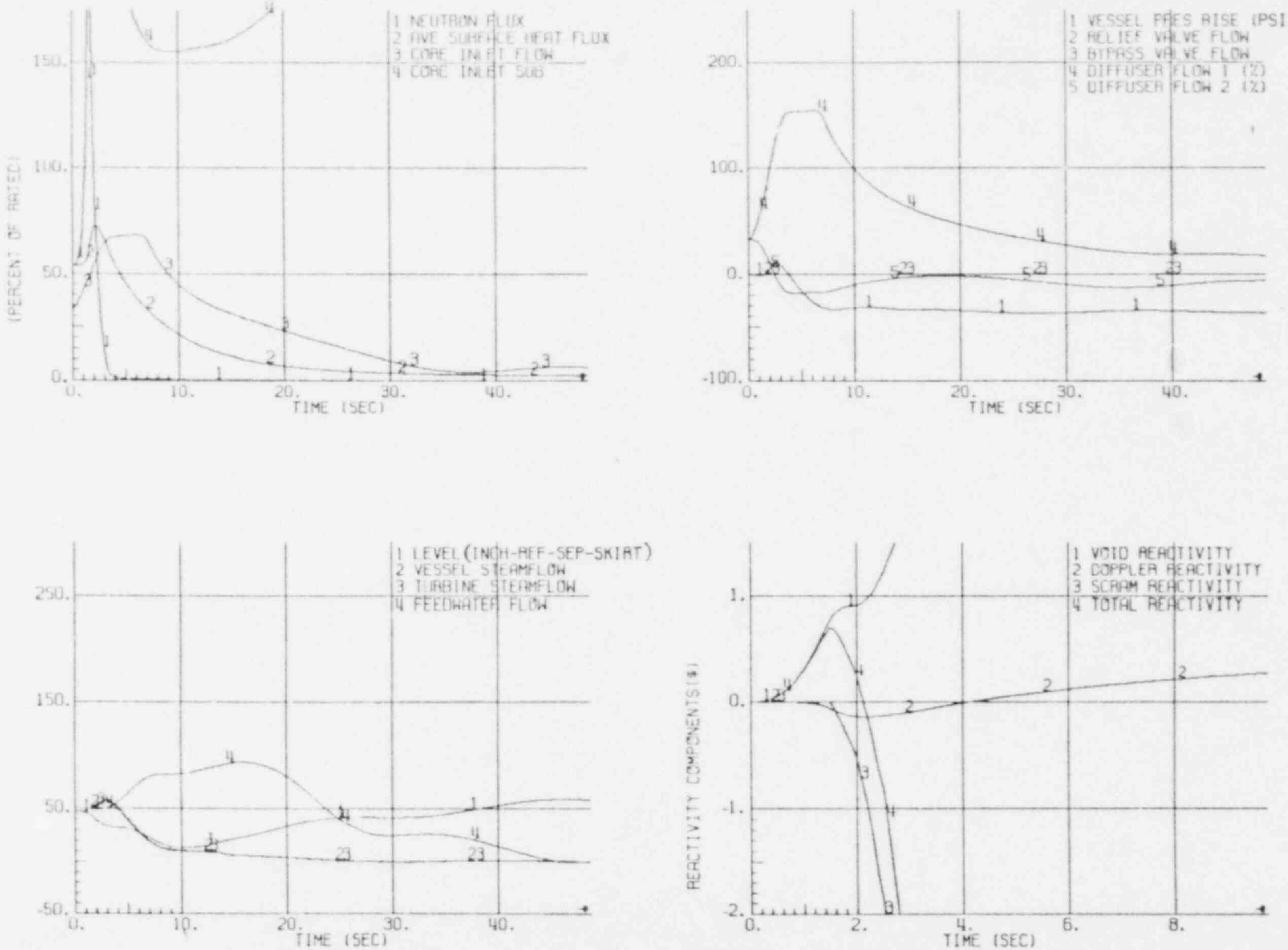

Figure 15.4-2. Fast Opening of One Recirc Valve at 30%/sec

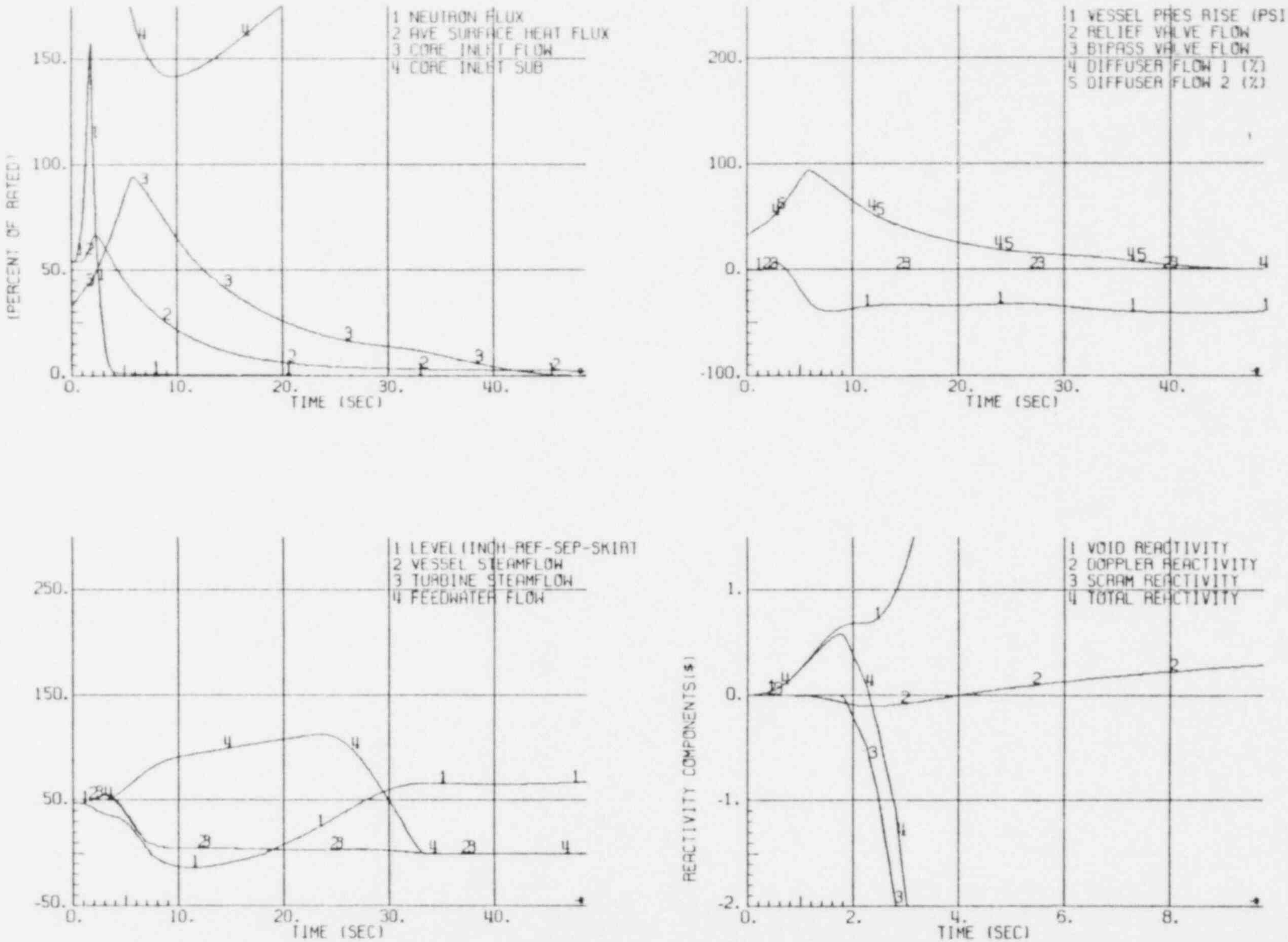

Figure 15.4-3. Fast Opening of Both Recirc Valves at 11%/sec

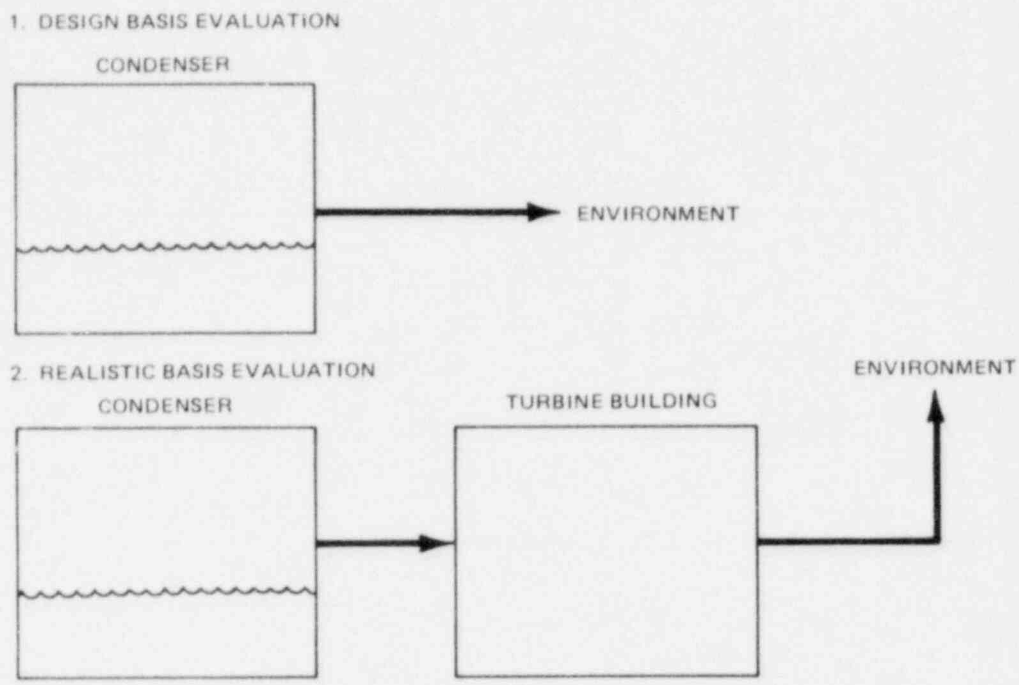

Figure 15.4-4. Leakage Path Model for Rod Drop Accident

SECTION 15.5

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ILLUSTRATIONS

15.5 INCREASE IN REACTOR COOLANT INVENTORY

15.5.1 Inadvertent HPCS Startup

15.5.1.1 Identification of Causes and Frequency Classification

15.5.1.1.1 Identification of Causes

Manual startup of the HPCS 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 Systems 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

With the recirculation system in either the automatic or manual mode, relatively small changes would be experienced in plant conditions. The operator should, after hearing the alarm that the HPCS has commenced operation, check reactor water level and drywell pressure. If conditions are normal, the operator should shut down the system.

15.5.1.2.2 System Operation

In order to properly simulate the expected sequence of events, the analysis of this event assumes normal functioning of plant

15.5.1.2.2 System Operation (Continued)

instrumentation and controls--specifically, the pressure regulator 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.

15.5.1.2.3 The Effect of Single Failures and Operator Errors

Inadvertent operation of the HPCS results in a mild depressurization. Corrective action by the pressure regulator and/or level control is expected to establish a new stable operating state. The effect of a single failure in the pressure regulator will aggravate the transient, depending upon the nature of the failure. Pressure regulator failures are discussed in Subsections 15.1.3 and 15.2.1.

The effect of a single failure in the level control system has rather straightforward consequences, including level rise or fall by improper control of the feedwater system. Increasing level will trip the turbine and automatically trip the HPCS system off. This trip signature is already described in the failure of feedwater controller with increasing flow. Decreasing level will automatically initiate scram at the L3 level trip and will have a signature similar to loss of feedwater control - decreasing flow.

15.5.1.3 Core and System Performance

15.5.1.3.1 Mathematical Model

The detailed nonlinear dynamic model described briefly in Subsection 15.1.1.3.1 is used to simulate this transient.

15.5.1.3.2 Input Parameter and Initial Conditions

This analysis has been performed unless otherwise noted with plant conditions tabulated in Table 15.0-2.

The water temperature of the HPCS system was assumed to be 40°F with an enthalpy of 11 Btu/lb.

Inadvertent startup of the HPCS system was chosen to be analyzed, since it provides the greatest auxiliary source of cold water into the vessel.

15.5.1.3.3 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 3 sec, the full HPCS flow is established at approximately 5.1% of the rated feedwater flow rate. This flow is nearly 102% of the HPCS flow at rated pressure. No delays were 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. In the automatic flow control mode, following a momentary decrease, neutron power settles out at a level slightly above operating level. In manual mode the flux level settles out slightly below operating level. In either case, 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.

15.5.1.3.3.1 Consideration of Uncertainties

Important analytical factors, including reactivity coefficient and feedwater temperature change, have been assumed to be at the

15.5.1.3.3.1 Consideration of Uncertainties (Continued)

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 indicates 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

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

15.5.2 <u>Chemical Volume Control System Malfunction (or Operator Error)</u>

This section is not applicable to BWR. This is of PWR interest.

15.5.3 <u>BWR Transients Which Increase Reactor Coolant Inventory</u>

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

Table 15.5-1

SEQUENCE OF EVENTS FOR FIGURE 15.5-1

Time (sec)	Event
0	Simulate HPCS cold water injection.
3	Full flow established for HPCS.
7	Depressurization effect stabilized.

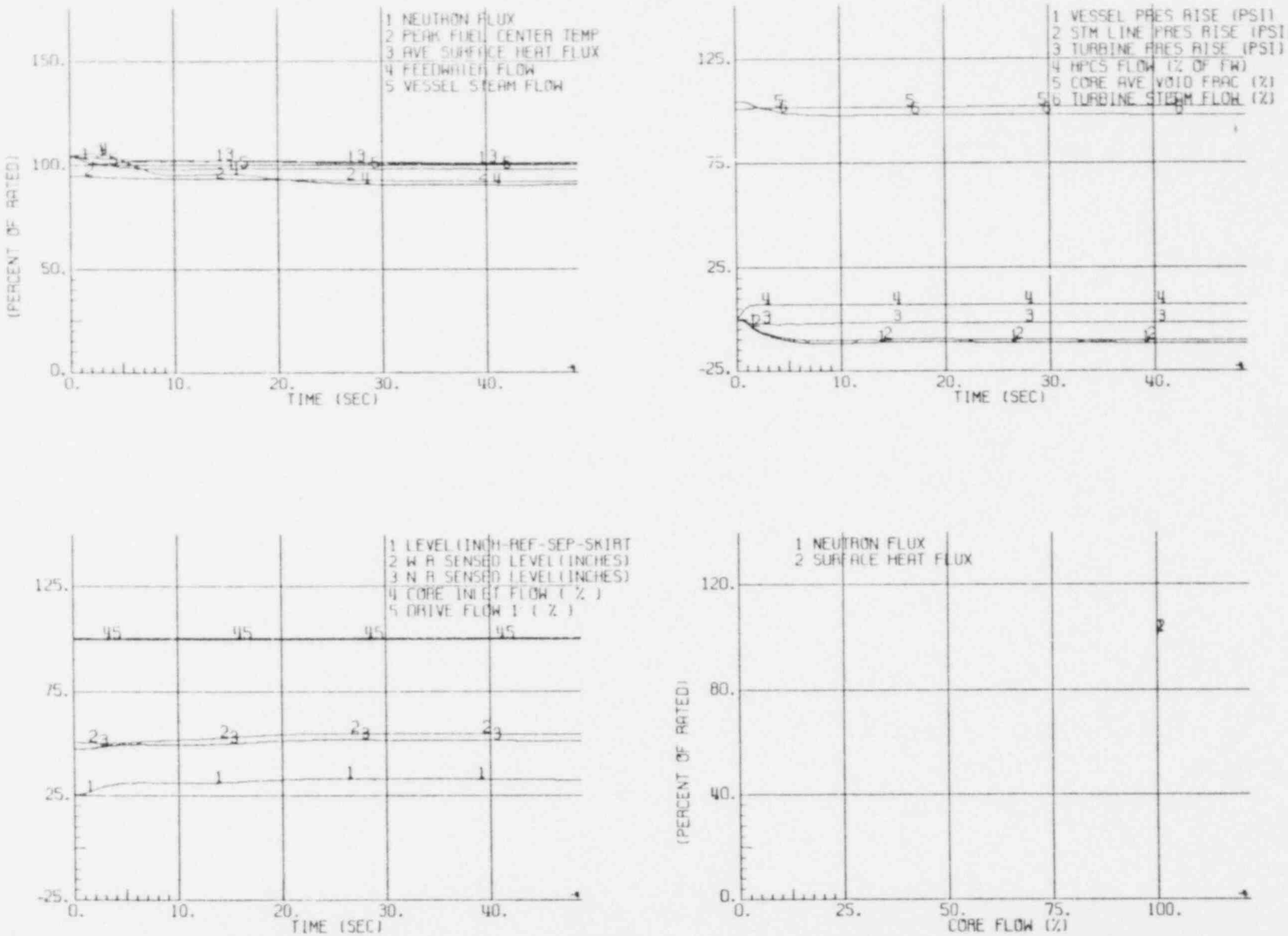

Figure 15.5-1. Inadvertent Startup of HPCS Pump

SECTION 15.6

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CONTENTS (Continued)

CONTENTS (Continued)

SECTION 15.6

TABLES

TABLES (Continued)

SECTION 15.6

ILLUSTRATIONS

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 Inadvertent Safety Relief Valve Opening

This event is discussed and analyzed in Subsection 15.1.4.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The Standard 238 Nuclear Island design has no instrument or sample lines connected to the reactor coolant pressure boundary which penetrate the primary containment. Therefore, radiological consequences as a result of this analysis were not analyzed.

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 the postulation of 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

15.6.4.1.1 Identification of Causes (Continued)

codes and standards, and to restrictive 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 fault event for breaks outside the containment is a complete severence of one of the four main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.6-1.

15.6.4.2.1.1 Identification of Operator Actions

Normally, the reactor operator will maintain reactor vessel water inventory and, therefore, core cooling with the RCIC system. Without operator action, the RCIC would initiate automatically on low water level following isolation of the main steam supply system (i.e., MSLIV closure). The core would be covered throughout the accident and there would be no fuel damage. Without taking credit for the RCIC water makeup capability and assuming HPCS failure, the operator should initiate the ADS or manual relief valve system to ensure termination of the accident without 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 upstream of the inboard isolation valve. Flow from the downstream side is initially limited by the total area of the flow restrictors in the three unbroken lines. Subsequent closure of the MSLIVs 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.

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

15.6.4.2.3 The Effect of Single Failures and Operator Errors

The effect of single failures has been considered in analyzing this event. The ECCS aspects are covered in Section 6.3. The break detection and isolation considerations are defined in Sections 7.3 and 7.6. All of the protective sequences for this event are capable of SCF and SOE accommodation and yet completion of the necessary safety action (see Appendix 15A for further details).

15.6.4.3 Core and System Performance

Quantitative results (including math models, input parameters, and consideration of uncertainties) for this event are given 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

Refer to Section 6.3 for initial conditions.

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.

15.6.4.3.3 Considerations of Uncertainties

Sections 6.3 and 7.3 contain discussions of the uncertainties associated with the ECCS performance and the containment isolation systems, respectively.

15.6.4.4 Barrier Performance

Since 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 (Secondary Containment Functional Design).

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 MSLIVs:

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

(2) nuclear system pressure is 1040 psia and remains constant during closure;

(3) an instantaneous circumferential break of the main steamline occurs;

(4) isolation valves start to close at 0.5 sec on high flow signal and are fully closed at 5.5 sec;

15.6.4.4 Barrier Performance (Continued)

(5) the Moody critical flow model (Reference 1) is applicable;

(6) level rise time conservatively assumed to be 1 sec. Mixture quality is conservatively taken to be a constant seven (steam weight percentage) during mixture flow; and

(7) AC power is available.

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 170% 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 60,699 lb, of which 46,213 lb is liquid and 14,486 lb is steam.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first 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".

(2) The second is based on assumptions concerned to provide a realistic yet conservative estimate of the radiological consequences. This analysis is referred to as the "realistic analysis".

15.6.4.5 Radiological Consequences (Continued)

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 2. Specific values of parameters used in the evaluation are presented in Table 15.6-2.

General Compliance or Alternate Approach Statement (RG 1.5):

For commitment and revision number, see Section 1.8.

This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a steam line 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 X/Q of 2.0E-3 (sec/m^3); and

3. LPZ X/Q of 1.0E-3 (sec/m^3).

Some of the models and conditions that are prescribed are demonstrably inconsistent with actual physical phenomena. The impact of the conservative bias that is introduced is generally limited to plant design choices outside the GE scope.

For this reason, additional analyses are provided in Subsection 15.6.4, which utilizes realistic assumptions to demonstrate the conservative bias in the regulatory guide requirements.

15.6.4.5.1 Design Basis Analysis (Continued)

In either case, regardless of the model used for evaluation, the dose resultant 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 100 μCi/sec - MWt after 30 min delay (365,100 μCi/sec). The iodine concentration in the reactor coolant is then given by (μCi/gm):

I-131	2.03E-2
I-132	2.59 E-1
I-133	1.49 E-1
I-134	4.75 E-1
I-135	2.39 E-1

Because of its short half-life, N-16 is not considered in the analysis.

15.6.4.5.1.2 Fission Product Transport to the Environment

The transport pathway is a direct unfiltered release to the environment. The MSLIV detection and closure time of 5.5 sec results in a discharge of 14,486 lb of steam and 46,213 lb 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 15.6-3.

15.6.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-4 and are a small fraction of the guidelines of 10CFR100.

15.6.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and program used for computer evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.6-2.

15.6.4.5.2.1 Fission Product Release from Fuel

There is no fuel rod damage as a consequence of this event; therefore, the only activity released to the environment is that associated with the steam and liquid discharged from the break.

15.6.4.5.2.2 Fission Product Transport to the Environment

The activity released from the accident is a function of the coolant activity, valve closure time and mass of coolant released. A portion of the released coolant exists as steam prior to the blowdown and, as such, does not contain the same concentration per unit of mass as does the steam generated as a consequence of the blowdown. Therefore, it is necessary to subtract the initial steam mass from the total mass released and assign to it only 2.0% of the iodine activity contained by an equivalent mass of primary coolant.

15.6.4.5.2.2 Fission Product Transport to the Environment (Continued)

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the reactor coolant pressure boundary:

(1) The amount of coolant discharged is that calculated in the analysis of the nuclear system transient.

(2) The concentrations of biologically significant radionuclides contained in the primary coolant are as follows:

I-131	2.03 E-2	μCi/gm
I-132	2.59 E-1	μCi/gm
I-133	1.49 E-1	μCi/gm
I-134	4.75 E-1	μCi/gm
I-135	2.39 E-1	μCi/gm

Measurements made on current generation BWRs show the activity ratio between the main turbine condensate and reactor coolant is on the order of 0.5% to 2%. For the purpose of this evaluation, the conservative assumption is made that the activity per pound of steam is equal to 2.0% of the activity per pound of reactor water.

(3) The noble gas discharge rate, after 30 min holdup, is assumed to be 0.1 Ci/sec, an unusually high normal discharge rate. This assumption permits direct computation of the amount of noble gas activity leaving the reactor vessel at the time of the accident. The result is that 0.45 Ci of noble gas activity leaves the reactor vessel during each second that the isolation valve is open.

15.6.4.5.2.2 Fission Product Transport to the Environment (Continued)

(4) Because of the short half-life of nitrogen-16, the radiological effects from this isotope are of no major concern and are not considered in the analysis.

Based on the above considerations, the amount of activity which is available for atmospheric dispersion is presented in Table 15.6-5.

15.6.4.5.2.3 Results

The calculated exposures for this event are presented in Table 15.6-6. As noted, these values are a small fraction of 10CFR100.

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

This event involves the postulation of 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 an SSE earthquake.

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 new information not presented in the subject sections. All other information is covered by cross-referencing.

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

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 safe shutdown earthquake plus SACF criteria requirements. The subject piping is designed of high quality, to strict industry code and standard criteria, and for severe seismic and environmental conditions. However, since such an accident provides an upper limit estimate to 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 shown Table 6.3-2 for core system performance and Table 6.2-8 for barrier (containment) performance.

Following the pipe break and scram, the MSLIV will begin closing on the low-low level signal. The low-low water level or high

15.6.5.2.1 Sequence of Events (Continued)

drywell pressure signal will initiate HPCS and LPCS systems at time 0 plus approximately 30 sec.

15.6.5.2.1.1 Identification of Operator Actions

Since automatic actuation and operation of the ECCS is a system design basis, no operator actions are required for the accident. However, the operator should perform the following described actions.

The operator should, after assuring that all rods have been inserted at time 0 plus approximately 10 sec, determine plant conditions by observing the annunciators. After observing that the ECCS flows are initiated, the operator should check that the diesel generators have started and are on standby condition. When possible (less than half an hour later), the operator should initiate operation of the RHR system heat exchangers in the suppression pool cooling mode and give instructions to put the service water systems in service. After the RHR system and other auxiliary systems are in proper operation, the operator should monitor the hydrogen concentration in the drywell for proper activation of the recombiner and mixer, if necessary.

15.6.5.2.2 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 examined in Sections 6.2 and 6.3, including the severance of small process system lines, the main steamlines upstream of the

15.6.5.2.2 Systems Operations (Continued)

flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the containment result from a complete circumferential break of one of the two recirculation loop pipelines. The minimum required functions of any Reactor and Plant Protection System are discussed in Sections 6.2, 6.3, 7.3, 7.6 and 8.3, and Appendix 15A.

15.6.5.2.3 The Effect of Single Failures and Operator Errors

Single failures and operator errors have been considered in the analysis of the entire spectrum of primary system breaks. The consequences of a LOCA with considerations for single failures are shown to be fully accommodated without the loss of any required safety function (see Appendix 15A for details).

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

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 very 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 given in Table 6.3-1.

15.6.5.3.3 Results

Results of this event are given 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 input is minimized and within limits. Continued operator control and surveillance is examined and guaranteed.

15.6.5.3.4 Consideration of Uncertainties

This event was conservatively analyzed (see Sections 6.3, 7.5, 7.6, 8.3 and Appendix 15A for details).

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 the largest single 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 separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet

15.6.5.5 Radiological Consequences (Continued)

10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

(2) The second is based on assumptions considered to provide a realistic estimate of radiological consequences. This analysis is referred to as the "realistic analysis".

A schematic of the transport pathway is shown in Figure 15.6-2.

15.6.5.5.1 Design Basis Analysis

The methods, assumptions and conditions used to evaluate this accident are in accordance with those guidelines set forth in Regulatory Guides 1.3 and 1.7. The specific models, assumptions and computer code used to evaluate this event based on the above criteria are presented in Reference 2. Specific values of parameters used in this evaluation are presented in Table 15.6-7.

15.6.5.5.1.1 Fission Product Release from Fuel

It is assumed that 100% of the noble gases and 50% of the iodine are released from an equilibrium core operating at a power level of 3651 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3, the assumed release of 100% of the core noble gas activity and 50% of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases and 50% of the iodine become airborne. The remaining 50% of the iodine is removed by plate-out and condensation; therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in Table 15.6-8.

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the containment to the secondary containment-like structures by several different mechanisms and discharge to the environment through the Standby Gas Treatment System (SGTS):

(1) Containment leakage - The design basis leak rate of the primary containment and its penetrations (excluding the main steamlines) is 1.0%/day for the duration of the accident. All of this leakage is to the secondary containment and from there to the environment via a 99% SGTS. Credit is taken for mixing and holdup within the secondary containment. The Shield Building exhaust rate, leakage rate, and mixing ratio are given on Tables 15.6-9 and 15.6-10.

(2) Leakage from engineered safety feature (ESF) components outside primary containment.

(3) Hydrogen purge - In the event of failure of the Hydrogen Recombiner System, purging of the containment may be necessary to control hydrogen concentration inside the primary containment. The earliest this purge may be utilized is one hour after the accident rate of 100 scfm minimum. The purge would be processed by SGTS prior to release to the environment.

Fission product release to the environment based on the above assumption is given in Table 15.6-11.

15.6.5.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-12 and are well within the guidelines of 10CFR100.

15.6.5.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.6-7.

15.6.5.5.2.1 Fission Product Release from Fuel

Since this accident does not result in any fuel damage, the only activity released to the drywell is that activity contained in the reactor coolant plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

While there are vaious activation and corrosion products contained in the reactor coolant, the products of primary importance are the iodine isotopes I-131 to I-135. The coolant concentration for these isotopes is:

I-131	2.03 E-2	µCi/gm
I-132	2.59 E-1	µCi/gm
I-133	1.49 E-1	µCi/gm
I-134	4.75 E-1	µCi/gm
I-135	2.39 E-1	µCi/gm

Considering that approximately 40% of the released liquid flashes to steam, it is conservatively assumed that 40% of the released iodine activity is airborne initially. However, as a result of plate-out and condensation effects, only 50% of the activity initially airborne remains available for release to the environment.

15.6.5.5.2.1 Fission Product Release from Fuel (Continued)

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. Measurements performed (Reference 4) at operating BWRs during reactor shutdown have been used to develop an analytical model for the prediction of iodine and noble gas spiking as a consequence of reactor scram and vessel depressurization. Based on the 95th percentile (i.e., only 5% of the time will the release be greater) probability, the I-131 release is calculated to be 2.14 Ci/bundle and Xe-133 to be 11.55 Ci/bundle. Other iodine and noble gas isotopes are determined in accordance with their cumulative fission yields and are tabulated in Table 15.6-13.

While no measurements have been obtained during a pressure transient as rapid as the LOCA, it is difficult to predict the actual release rate from the fuel as a consequence of iodine spiking. Therefore, it is arbitrarily assumed that 100% of the spiking source term is released during the time period that 40% of the discharge coolant is flashing to steam.

It is also assumed that plate-out and condensation removes 50% of the airborne iodine activity. The total activity airborne in the containment is presented in Table 15.6-14.

15.6.5.5.2.2 Fission Product Transport to the Environment

The leak rate from the primary containment to the secondary containment is 1.0%/day, where 100% mixing is assumed to occur. Release from the secondary containment to the environment via a 99.9% iodine efficient SGTS is presented in Table 15.6-15. The integrated isotopic activity released to the environment is presented in Table 15.6-15.

15.6.5.5.2.3 Results

The calculated radiological exposures for this event are presented in Table 15.6-16 and as shown are a small fraction of 10CFR100.

15.6.5.5.3 Control Room

A dose analysis has been performed to demonstrate that the ventilation system satisfies the NRC radiation guidelines. The results of the analysis show that the ventilation system design does satisfy their guideline. A schematic of the control room intake vents is shown in Figure 15.6-3.

The doses received during a 30-day period after a loss-of-coolant accident are:

	Dose (Rem)	U.S. NRC Limit (Rem)
Whole Body	2.56	5
Thyroid	29.4	30
Beta	53.8	75

A factor of 1/4 was taken into account for a dual inlet with manual override capabilities. The methods used to calculate these doses are presented in Reference 5. A complete list of assumptions and input data follows:

(1) Source Terms

The source terms used in this analysis are consistent with R.G. 1.3 (i.e., 25% halogens and 100% noble gases airborne in the containment) and were presented in Table 15.6-8.

15.6.5.5.3 Control Room (Continued)

(2) Ventilation Parameter

Inlet air flows	
filtered	0.944 m^3/sec
unfiltered	0.0014 m^3/sec
Filter efficiency	99%
Control Room Volume	1.102 E+4 m^3
Occupancy factors	
0-2 hr	1.0
2-8 hr	1.0
8-24 hr	1.0
1-4 day	0.6
4-30 day	0.4

(3) Meteorology Data

X/Q Values	sec/m^3
0-2 hr	8.0 E-3
2-8 hr	1.6 E-3
8-24 hr	1.4 E-3
24 hr-4 day	1.1 E-3
4-30 day	1.1 E-3

15.6.6 Feedwater Line Break - Outside Containment

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope

15.6.6 Feedwater Line Break - Outside Containment (Continued)

evaluation relative to 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 in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is covered by cross-referencing to appropriate Chapter 6 subsections.

15.6.6.1 Identification of Causes and Frequency Classification

15.6.6.1.1 Identification of Causes

A feedwater 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 severe 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 shown in Table 15.6-17.

15.6.6.2.1.1 Identification of Operator Actions

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

(1) The operator should determine that a line break has occurred and evacuates the area of the turbine building.

(2) The operator is not required to take any action to prevent primary reactor system mass loss, but should ensure that the reactor is shut down and that RCIC and/or HPCS are operating normally.

(3) The operator should implement site radiation incident procedures.

(4) If possible, the operator should shut down the feedwater system and de-energize any electrical equipment which may be damaged by water from the feedwater system in the turbine building.

(5) The operator should continue to monitor reactor water level and the performance of the ECCS systems while the radiation incident procedure is being implemented and begins normal reactor cooldown measures.

(6) When the reactor pressure has decreased below 150 psia, the operator should initiate RHR in the shutdown cooling mode to continue cooling down the reactor.

The above operator procedures occur over an elapsed time of 3-4 hours.

15.6.6.2.2 Systems Operations

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 system. 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 RCIC/HPCS systems are assumed to operate normally.

15.6.6.2.3 The Effect of Single Failures and Operator Errors

The feedwater line 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 discussed in detail in Subsection 6.3.3.3. For the feedwater line break outside the containment, since the break is isolatable, either the RCIC or the HPCS can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either the HPCS or the RCIC would still provide 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 of the feedwater piping lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions as given in Table 6.3-1.

15.6.6.3.2 Qualitative Results

The feedwater 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), the feedwater line break inside the containment (analysis presented in Subsections 6.3.3 and 15.6.5). It certainly is far less limiting than the design basis accident (the recirculation line break analysis presented in Subsections 6.3.3 and 15.6.5).

The reactor vessel is isolated on low-low water level, and the RCIC and the HPCS together restore the reactor water level to the normal elevation. 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 results 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 fault event for breaks outside the containment is a complete severance of one of the main steam lines as described in Subsection 15.6.4. The feedwater 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, no design basis analysis will be presented.

15.6.6.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.6-18. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-4.

15.6.6.5.2.1 Fission Product Release

There is no fuel damage as a consequence of this accident. In addition, an insignificant quantity of activity (compared to that existing in the main condenser hotwell prior to occurrence of the break) is released from the contained piping system prior to isolation closure.

The iodine concentration in the main condenser hotwell is consistent with an offgas release rate of 100,000 μCi/sec at 30 min delay and is 0.02 (2% carryover) times the concentration in the reactor coolant. Noble gas activity in the condensate is negligible, since the air ejectors remove practically all noble gas from the condenser.

15.6.6.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of liquid release from the break, carryover to the turbine building atmosphere due to flashing and partitioning and unfiltered release to the environment through the turbine building ventilation system.

Of the 901,760 lb of condensate released from the break, 18,035 lb flashes to steam. Of the activity remaining in the unflashed liquid, 2.0E-6% is assumed to become airborne. Normally, all feedwater reaching the break location will have passed through condensate demineralizers.

However, as a result of the increased feedwater flow caused by the break, differential pressure across the demineralizer is assumed to initiate flow through the demineralizer bypass line. This bypass line then carries 35% of the total flow, resulting in an effective iodine removal efficiency for all flow of 58%.

Taking no credit for holdup, decay or plate-out during transport through the turbine building, the release of activity to the environment is presented in Table 15.6-19. The release is assumed to take place within 2 hours of the occurrence of the break.

15.6.6.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.6-20 and are a small fraction of 10CFR100 guidelines.

15.6.6.6 References

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

2. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143).

3. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

4. F. J. Prutschy, G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup", August 1972 (NEDO-10585).

5. D. G. Weiss and V. O. Nguyen, "Control Room Accident Exposure Evaluation - CRDS Program", January 1979 (NEDO-23909)

Table 15.6-1

SEQUENCE OF EVENTS FOR STEAMLINE BREAK OUTSIDE CONTAINMENT

Time (sec)	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.
≤5.5	Main steamline isolation valves fully closed.
9.1	Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 1100 psi.
14.5	RCIC and HPCS would initiate on low water level (RCIC considered unavailable, HPCS assumed single failure and therefore not available).
225	Reactor water level above core begins to drop slowly due to loss of steam through the safety valves. Reactor pressure still at approximately 1100 psi.
600	Operator initiates ADS or manually controls relief valves. Vessel depressurizes rapidly.
(Subsection 6.3.3)	Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
(Subsection 6.3.3)	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

Table 15.6-2

STEAMLINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	NA	NA
B. Burnup	NA	NA
C. Fuel damaged	None	None
D. Release of activity by nuclide	Table 15.6-3	Table 15.6-5
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident	Subsection 15.6.4.5.1.1	Subsection 15.6.4.5.2.2
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Isolation valve closure time (sec)	5	5
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None

Table 15.6-2 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	*	*
B. χ/Q's for Total dose - SB/LPZ	2.0E-3/ 1.0E-3	2.0E-3/ 1.0E-3
IV. Dose Data		
A. Method of dose calculation	Reference 2	Reference 3
B. Dose conversion assumptions	Reference 2	Reference 3
C. Peak activity concentrations in containment	NA	NA
D. Doses	Table 15.6-4	Table 15.6-6

*Applicant to Supply

Table 15.6-3

STEAMLINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (Ci)

Isotope	Curies
I131	1.564E-00
I132	1.995E-01
I133	1.148E-01
I134	3.659E-01
I135	1.841E-01
Total Halogens	8.799E-01
Kr83m	8.458E-03
Kr85m	1.482E-01
Kr85	5.783E-04
Kr87	4.618E-01
Kr88	4.735E-01
Kr89	1.970E 00
Xe131m	4.726E-04
Xe131m	7.066E-03
Xe133	1.979E-01
Xe135m	5.792E-01
Xe135	5.340E-01
Xe137	2.602E-00
Xe138	1.970E-00
Total Noble Gases	9.030E-00

Table 15.6-4

STEAM LINE BREAK ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area *	9.03E-2	4.77
Low Population Zone *	4.51E-2	2.38

*Applicant to Supply

Table 15.6-5

STEAMLINE BREAK ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Isotope	Activity
I131	1.7E-01
I132	2.2E-00
I133	1.2E-00
I134	4.0E-00
I135	2.0E-00
Total	9.5E-00
Kr83m	2.3E-02
Kr85m	4.1E-02
Kr85	1.6E-04
Kr87	1.3E-01
Kr88	1.3E-01
Kr89	5.4E-01
Xe131m	1.3E-04
Xe133m	1.9E-03
Xe133	5.4E-02
Xe135m	1.6E-01
Xe135	1.5E-01
Xe137	7.1E-01
Xe138	5.4E-01
Total	2.5E-00

Table 15.6-6

STEAMLINE BREAK ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area *	1.1E-2	5.2E-1
Low Population Zone *	5.4E-3	2.6E-1

*Applicant to Supply

Table 15.6-7

LOSS-OF-COOLANT ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3651 MWt	3651 MWt
B. Burnup	NA	NA
C. Fuel damage	100%	0
D. Release of activity by nuclide	Table 15.6-9	Table 15.6-14
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident	NA	15.6.5.5.2.1
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	1.0	1.0
B. Secondary containment leak rate (%/day)	0.2 hrs 319.3	123
	2-10 hrs 59.9	123
	>10 hrs 43.9	123
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies (%)		
(1) Organic iodine	NA	NA
(2) Elemental iodine	99	99.9
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters		
(1) Flow rate (CFM)	5000	5000
(2) Mixing efficiency	50	100
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment Volumes	NA	NA
H. All other pertinent data and assumptions	None	None

Table 15.6-7 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Rate		
A. Boundary and LPZ distance (m)	*	*
B. χ/Q's for time intervals of		
(1) 0-2 hr - SB/LPS	2.0E-3/1.0E-3 1.0E-3	2.0E-3/1.0E-3 1.0E-3
(2) 2-8 hr - LPZ	3.8E-4	3.8E-4
(3) 8-24 hr - LPZ	1.0E-4	1.0E-4
(4) 1-4 days - LPZ	3.4E-5	3.4E-5
(5) 4-30 days - LPZ	7.5E-6	7.5E-6
IV. Dose Data		
A. Method of dose calculation	Reference 2	Reference 3
B. Dose conversion assumptions	Reference 2	Reference 3
C. Peak activity concentrations in containment	Table 15.6-8	Table 15.6-14
D. Doses	Table 15.6-12	Table 15.6-16

*Applicant to Supply

Table 15.6-8

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.0E 07	1.9E 07	1.5E 07	1.2E 06
I132	3.5E 07	3.0E 07	2.6E 07	1.9E 07	1.0E 07	3.1E 06	9.2E 05	2.4E 04	7.3E-06	0.
I133	3.3E 07	3.2E 07	3.1E 07	3.0E 07	2.8E 07	2.5E 07	2.2E 07	1.4E 07	1.3E 06	9.3E-04
I134	5.5E 07	3.7E 07	2.5E 07	1.1E 07	2.3E 06	9.8E 04	4.1E 03	3.0E-01	0.	0.
I135	4.6E 07	4.3E 07	4.1E 07	3.7E 07	3.0E 07	2.0E 07	1.3E 07	3.6E 06	1.8E 03	0.
Total I	1.9e 08	1.6E 08	1.4E 08	1.2E 08	9.2E 07	6.8E 07	5.6E 07	3.7E 07	1.6E 07	1.2E 06
Kr83m	9.5E 06	8.0E 06	6.6E 06	4.5E 06	2.1E 06	4.8E 05	1.1E 05	1.2E 03	2.2E-09	0.
Kr85m	2.3E 07	2.1E 07	2.0E 07	1.7E 07	1.2E 07	6.6E 06	3.6E 06	5.5E 05	7.6E 00	0.
Kr85	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.8E 05	5.8E 05	5.6E 05	4.3E 05
Kr87	4.7E 07	3.6E 07	2.7E 07	1.6E 07	5.3E 06	5.9E 05	6.6E 04	9.2E 01	6.5E-16	0.
Kr88	6.7E 07	5.9E 07	5.2E 07	4.1E 07	2.5E 07	9.2E 06	3.4E 06	1.7E 05	2.9E-03	0.
Kr89	6.7E 07	1.2E 05	1.6E 02	3.0E-04	1.1E-15	0.	0.	0.	0.	0.
Xe131m	5.7E 05	5.7E 05	5.7E 05	5.7E 05	5.7E 05	5.6E 05	5.5E 05	5.4E 05	4.4E 05	7.5E 04
Xe133m	2.3E 07	2.3E 07	2.3E 07	2.2E 07	2.2E 07	2.1E 07	2.0E 07	1.7E 07	6.4E 06	1.5E 03
Xe133	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.2E 08	1.2E 08	7.6E 07	1.9E 06
Xe135m	3.6E 07	9.8E 06	2.5E 06	1.7E 05	7.2E 02	1.4E-02	2.6E-07	0.	0.	0.
Xe135	2.4E 07	2.3E 07	2.3E 07	2.1E 07	1.8E 07	1.3E 07	9.8E 06	3.9E 06	1.6E 04	0.
Xe137	1.5E 08	7.8E 05	3.4E 03	6.8E-02	2.6E-11	0.	0.	0.	0.	0.
Xe138	1.6E 08	3.9E 07	9.0E 06	4.8E 05	1.4E 03	1.1E-02	8.9E-08	0.	0.	0.
Total NG	7.4E 08	3.5E 08	3.0E 08	2.5E 08	2.2E 08	1.8E 08	1.6E 08	1.4E 08	8.3E 07	2.4E 06

Table 15.6-9

SHIELD BUILDING EXHAUST RATE

Time (hr)	Average Exhaust Flow Rate to SGTS (SCFM)
0 - 2	480
2 - 10	90
>10	66

Table 15.6-10

LEAKAGE RATES AND MIXING RATIO

Parameter	Numerical Value	
	Design Basis	Realistic
A. Primary to Secondary Containment (%/day)		
0 - 2 hr	0.832	1.0
2 - 10 hr	0.903	1.0
>10 hr	0.908	1.0
B. Primary Containment Leakage to SGTS (%/day)		
0 - 2 hr	0.168	NA
2 - 10 hr	0.097	NA
>10 hr	0.092	NA
C. Secondary Containment Leakage to SGTS (%/day)		
0 - 2 hr	319.3	123
2 - 10 hr	59.9	123
>10 hr	43.6	123
D. Mixing Efficiency (%)		
Primary Containment	100	100
Shield Building Annulus	50	100

Table 15.6-11

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.5E-01	8.7E 00	2.0E 01	4.8E 01	7.5E 01	1.5E 02	2.4E 02	5.7E 02	3.9E 03	1.8E 04
I132	4.1E-01	1.3E 01	2.8E 01	5.7E 01	7.5E 01	9.6E 01	1.0E 02	1.1E 02	1.1E 02	1.1E 02
I133	3.8E-01	1.3E 01	2.9E 01	7.0E 01	1.1E 02	2.0E 02	3.0E 02	6.0E 02	1.6E 03	1.7E 03
I134	6.4E-01	1.8E 01	3.4E 01	5.7E 01	6.4E 01	6.6E 01	6.6E 01	6.6E 01	6.6E 01	6.6E 01
I135	5.4E-01	1.8E 01	4.0E 01	9.1E 01	1.3E 02	2.2E 02	2.9E 02	4.1E 02	4.8E 02	4.8E 02
Total I	2.2E 00	7.1E 01	1.5E 02	3.2E 02	4.6E 02	7.4E 02	1.0E 03	1.8E 03	6.2E 03	2.0E 04
Kr83m	1.1E 01	3.5E 02	7.3E 02	1.4E 03	1.9E 03	2.2E 03	2.3E 03	2.4E 03	2.4E 03	2.4E 03
Kr85m	2.7E 01	8.9E 02	1.9E 03	4.3E 03	6.2E 03	9.4E 03	1.2E 04	1.4E 04	1.5E 04	1.5E 04
Kr85	6.9E-01	2.4E 01	5.4E 01	1.3E 02	2.1E 02	4.2E 02	6.7E 02	1.7E 03	1.3E 04	1.4E 05
Kr87	5.5E 01	1.7E 03	3.3E 03	6.0E 03	7.2E 03	7.9E 03	8.0E 03	8.1E 03	8.1E 03	8.1E 03
Kr88	7.8E 01	2.5E 03	5.4E 03	1.1E 04	1.6E 04	2.1E 04	2.4E 04	2.5E 04	2.5E 04	2.5E 04
Kr89	8.8E 01	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02
Xe131m	6.7E-01	2.3E 01	5.3E 01	1.3E 02	2.0E 02	4.0E 02	6.5E 02	1.6E 03	1.1E 04	6.4E 04
Xe133m	2.7E 01	9.3E 02	2.1E 03	5.1E 03	7.9E 03	1.6E 04	2.4E 04	5.5E 04	2.6E 05	4.5E 05
Xe133	1.6E 02	5.4E 03	1.2E 04	3.0E 04	4.7E 04	9.3E 04	1.5E 05	3.5E 05	2.2E 06	7.3E 06
Xe135m	4.4E 01	8.2E 02	1.1E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03
Xe135	2.8E 01	9.7E 02	2.1E 03	5.0E 03	7.5E 03	1.3E 04	1.8E 04	2.8E 04	4.0E 04	4.0E 04
Xe137	1.9E 02	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03
Xe138	1.9E 02	3.5E 03	4.5E 03	4.8E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03
Total NG	9.0E 02	1.9E 04	3.5E 04	7.1E 04	1.0E 05	1.7E 05	2.4E 05	4.9E 05	2.6E 05	8.1E 06

Table 15.6-12

LOSS-OF-COOLANT ACCIDENT (DESIGN BASE ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area *	19.3	66.7
Low Population Zone *	14.1	14.9

Table 15.6-13

ISOTOPIC SPIKING ACTIVITY

Isotope Name	The 95th Cumulative Probability Spiking Activity (Ci/bundle)
I131	2.14
I132	3.21
I133	5.03
I134	5.44
I135	4.79
Kr83m	9.04-1*
Kr85m	2.23+0
Kr85	4.90-1
Kr87	4.33+0
K488	6.12+0
Kr89	7.96+0
Xe131m	6.60-2
Xe133m	3.26-1
Xe133	1.16+1
Xe135m	1.80+0
Xe135	1.10+1
Xe137	1.05+1
Xe138	1.06+1

*9.04-1 = 9.04 x 10-1
**Applicant to Supply

Table 15.6-14

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE CONTAINMENT (Ci)

Isotope	1 min	1 hr	2 hr	8 hr	1 day	3 day	26 day
I131	8.59E 01	8.55E 01	8.52E 01	8.33E 01	7.84E 01	5.96E 01	5.64E 00
I132	1.41E 02	1.05E 02	7.73E 01	1.25E 01	9.69E-02	2.99E-11	0.
I133	2.08E 02	2.01E 02	1.94E 02	1.59E 02	9.28E 01	8.26E 00	6.91E-09
I134	2.39E 02	1.10E 02	4.97E 01	4.29E-01	1.34E-06	0.	0.
I135	2.03E 02	1.83E 02	1.65E 02	6.75E 01	1.62E 01	8.16E-03	0.
Total	8.77E 02	6.84E 02	5.71E 02	3.43E 02	1.88E 02	6.79E 01	5.64E 00
Kr83m	6.72E 02	4.65E 02	3.20E 02	3.37E 01	8.38E-02	1.53E-13	0.
Kr85m	1.67E 03	1.43E 03	1.23E 03	4.83E 02	4.03E 01	5.55E-04	0.
Kr85	3.67E 02	3.66E 02	3.66E 02	3.65E 02	3.63E 02	3.52E 02	2.73E 02
Kr87	3.21E 03	1.87E 03	1.08E 03	4.05E 01	6.35E-03	0.	0.
Kr88	4.56E 03	3.57E 03	2.79E 03	6.29E 02	1.19E 01	2.02E-07	0.
Kr89	4.78E 03	1.13E-02	2.14E-08	0.	0.	0.	0.
Xe131m	4.94E 01	4.92E 01	4.91E 01	4.83E 01	4.61E 01	3.77E 01	6.53E 00
Xe133m	2.44E 02	2.41E 02	2.37E 02	2.19E 02	1.77E 02	6.75E 01	1.63E-02
Xe133	8.64E 03	8.59E 03	8.54E 03	8.24E 03	7.50E 03	4.91E 03	1.26E 02
Xe135m	1.28E 03	8.87E 01	5.85E 00	4.83E-07	0.	0.	0.
Xe135	8.19E 03	7.60E 03	7.04E 03	4.46E 03	1.32E 03	5.51E 00	0.
Xe137	6.54E 03	1.54E-01	3.03E-06	0.	0.	0.	0.
Xe138	7.58E 03	4.25E 02	2.27E 01	5.23E-07	0.	0.	0.
Total	4.78E 04	2.47E 04	2.17E 04	1.45E 04	9.46E 03	5.37E 03	4.06E 02

Table 15.6-15

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	2.54E-10	8.98E-07	3.52E-06	5.04E-05	3.44E-04	2.24E-03	8.20E-03
I132	4.19E-10	1.22E-06	3.95E-06	2.07E-05	2.80E-05	2.81E-05	2.81E-05
I133	6.14E-10	2.13E-06	8.19E-06	1.04E-04	5.39E-04	1.46E-03	1.56E-03
I134	7.12E-10	1.53E-06	3.76E-06	7.66E-06	7.74E-06	7.74E-06	7.74E-06
I135	6.01E-10	1.99E-06	7.30E-06	7.15E-05	2.2E-04	2.63E-04	2.63E-04
Total	2.60E-09	7.76E-06	2.67E-05	2.55E-04	1.13E-03	4.00E-03	1.01E-02
Kr83m	1.99E-06	5.54E-03	1.72E-02	7.48E-02	9.03E-02	9.04E-02	9.04E-02
Kr85m	4.94E-06	1.58E-02	5.62E-02	4.64E-01	1.03E 00	1.11E 00	1.11E 00
Kr85	1.08E-07	3.84E-03	1.51E-02	2.19E-01	1.55E 00	1.15E 01	9.29E 01
Kr87	9.53E-06	2.38E-02	6.68E-02	1.99E-01	2.11E-01	2.11E-01	2.11E-01
Kr88	1.35E-05	4.07E-02	1.37E-01	8.40E-01	1.30E 00	1.32E 00	1.32E 00
Kr89	1.52E-05	7.28E-04	7.28E-04	7.28E-04	7.28E-04	7.28E-04	7.28E-04
Xe131m	1.46E-07	5.16E-04	2.03E-03	2.91E-02	2.01E-01	1.36E 00	6.01E 00
Xe133m	7.22E-07	2.53E-03	9.88E-03	1.36E-01	8.48E-01	3.95E 00	6.07E 00
Xe133	2.56E-05	9.02E-02	3.53E-01	5.01E 00	3.36E 01	2.03E 02	5.44E 02
Xe135m	3.86E-06	2.89E-03	3.70E-03	3.81E-03	3.81E-03	3.81E-03	3.81E-03
Xe135	2.42E-05	8.17E-02	3.06E-01	3.33E 00	1.21E 01	1.81E 01	1.81E 01
Xe137	2.06E-05	1.42E-03	1.42E-03	1.42E-03	1.42E-03	1.42E-03	1.42E-03
Xe138	2.28E-05	1.54E-02	1.90E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02
Total	1.44E-04	2.85E-01	9.88E-01	1.03E 01	5.10E 01	2.40E 02	6.70E 02

Table 15.6-16

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	2.2E-4	5.5E-6
*		
Low Population Zone	5.8E-4	6.6E-5
*		

Table 15.6-17

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK
OUTSIDE CONTAINMENT

Time	Event
0 sec	One feedwater line breaks.
0+ sec	Feedwater line check valves isolate the reactor from the break.
<30 sec	At low-low water reactor level RCIC would initiate, HPCS would initiate, MSLIV closure would initiate, reactor scram would initiate and recirculation pumps would trip.
~2 min	The safety/relief valves would open and close and maintain the reactor vessel pressure at approximately 1100 psig.
1-2 hr	Normal reactor cooldown procedure established.

*Applicant to Supply

Table 15.6-18

FEEDWATER LINE BREAK ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	NA	NA
B. Burnup	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Table 15.6-19
E. Iodine fractions		
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	Subsection 15.6.6.5.2.1
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Isolation valve closure time (sec)	NA	NA
D. Adsorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	NA	None

Table 15.6-18 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	NA/ NA	*
B. χ/Q's for Total dose - SB/LPZ (sec/m^3)	NA/ NA	2.0E-3/ 1.0E-3
IV. Dose Data		
A. Method of dose calculation	NA	Reference 3
B. Dose conversion assumptions	NA	Reference 3
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.6-20

*Site specific

*Applicant to Supply

Table 15.6-19

FEEDWATER LINE BREAK (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I131	2.1E-03
I132	2.6E-02
I133	1.5E-02
I134	4.8E-02
I135	2.4E-02
Total	1.2E-01

Table 15.6-20

FEEDWATER LINE BREAK (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	1.2E-04	6.3E-03
Low Population Zone	5.9E-05	3.2E-03

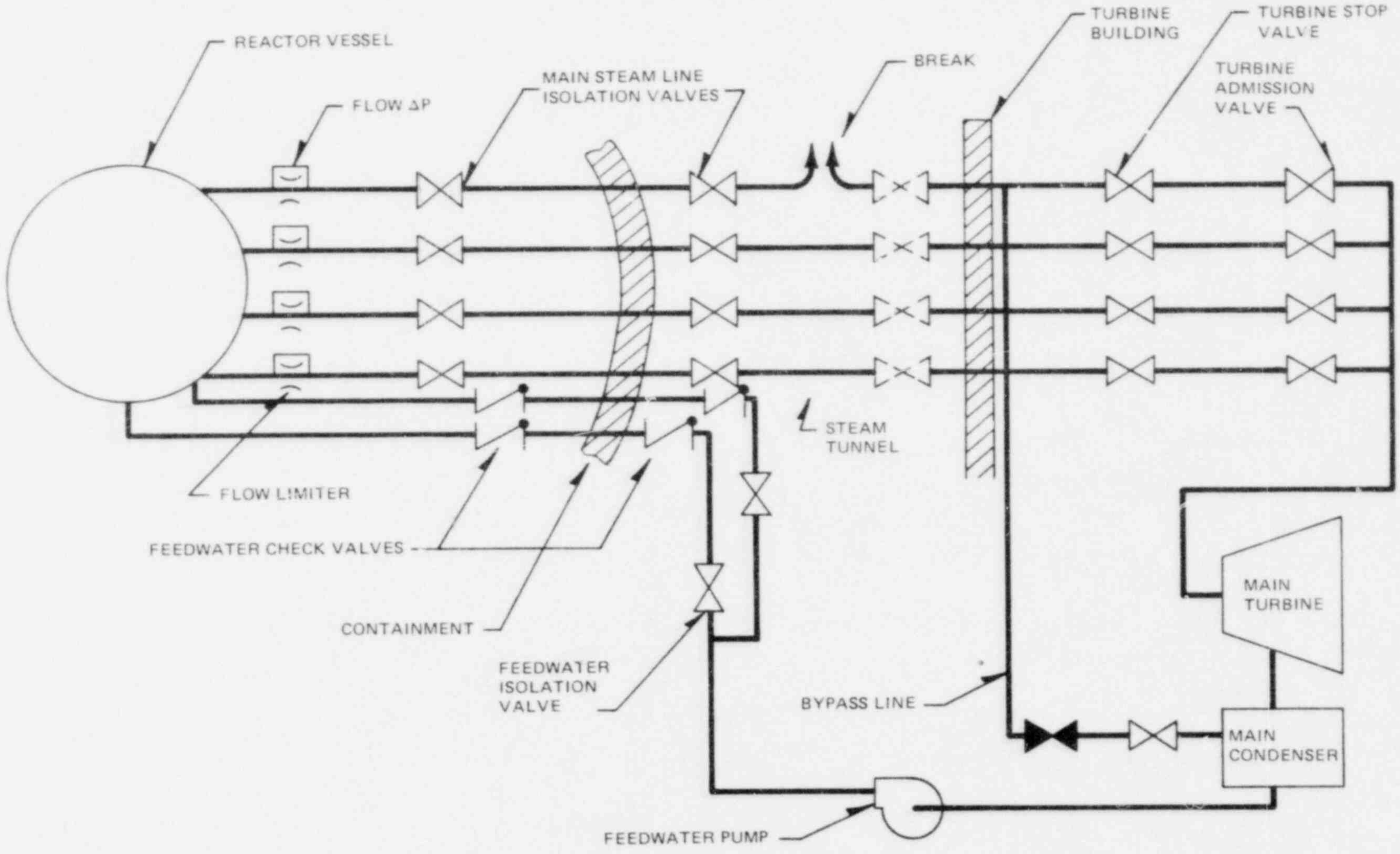

Figure 15.6-1. Steam Flow Schematic for Steam Break Outside Containment

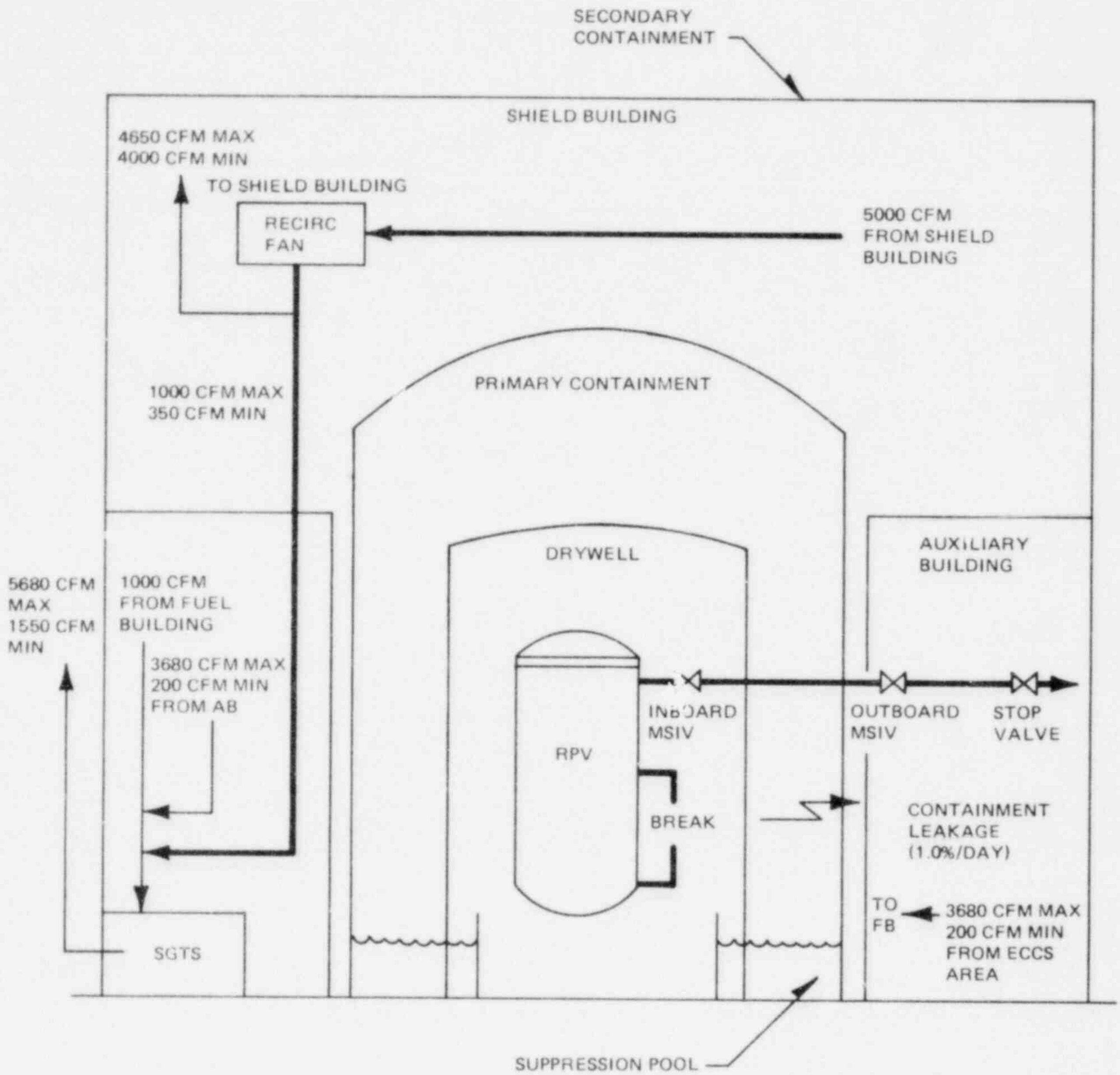

Figure 15.6-2. Post-LOCA Leakage Pathways

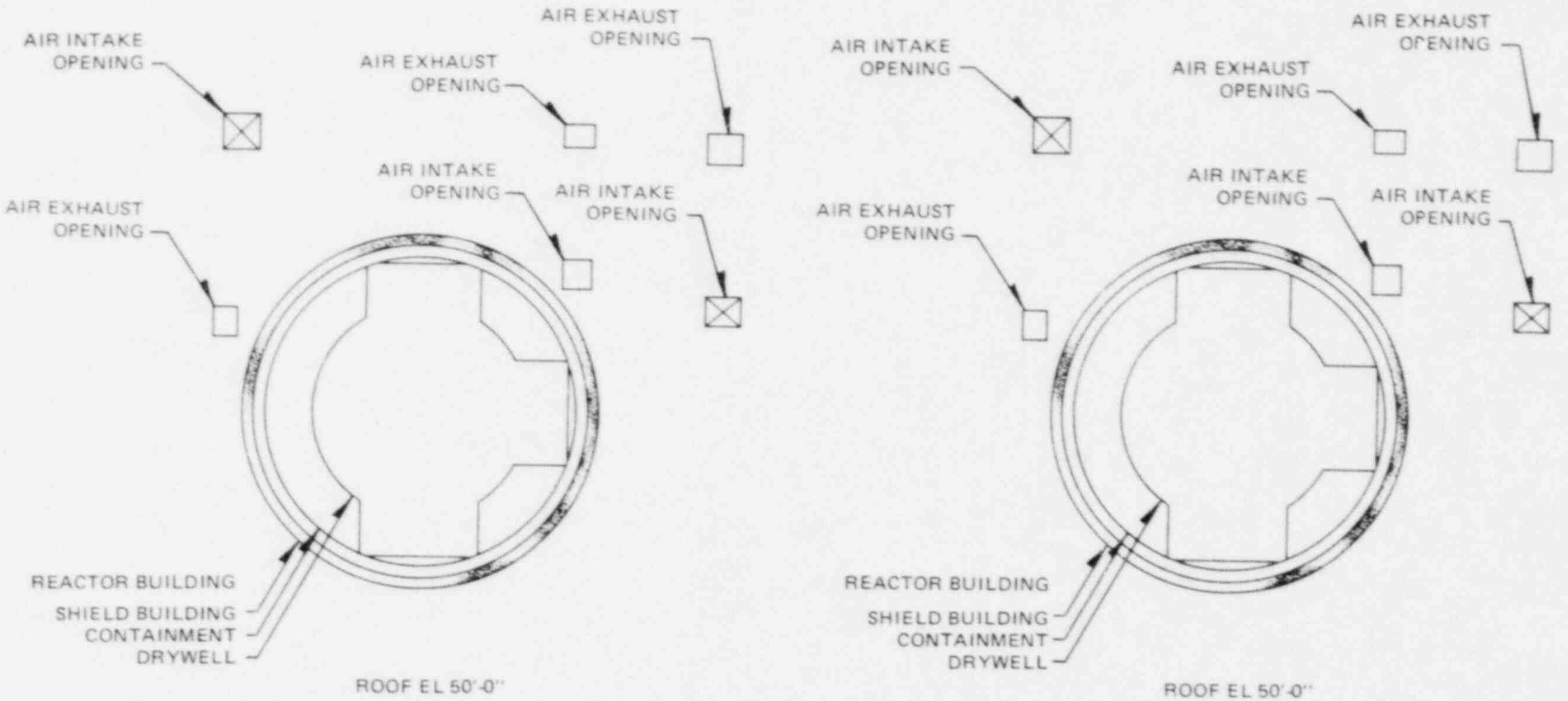

Figure 15.6-3. Plan at Elevation 54'-7" (Plant A)

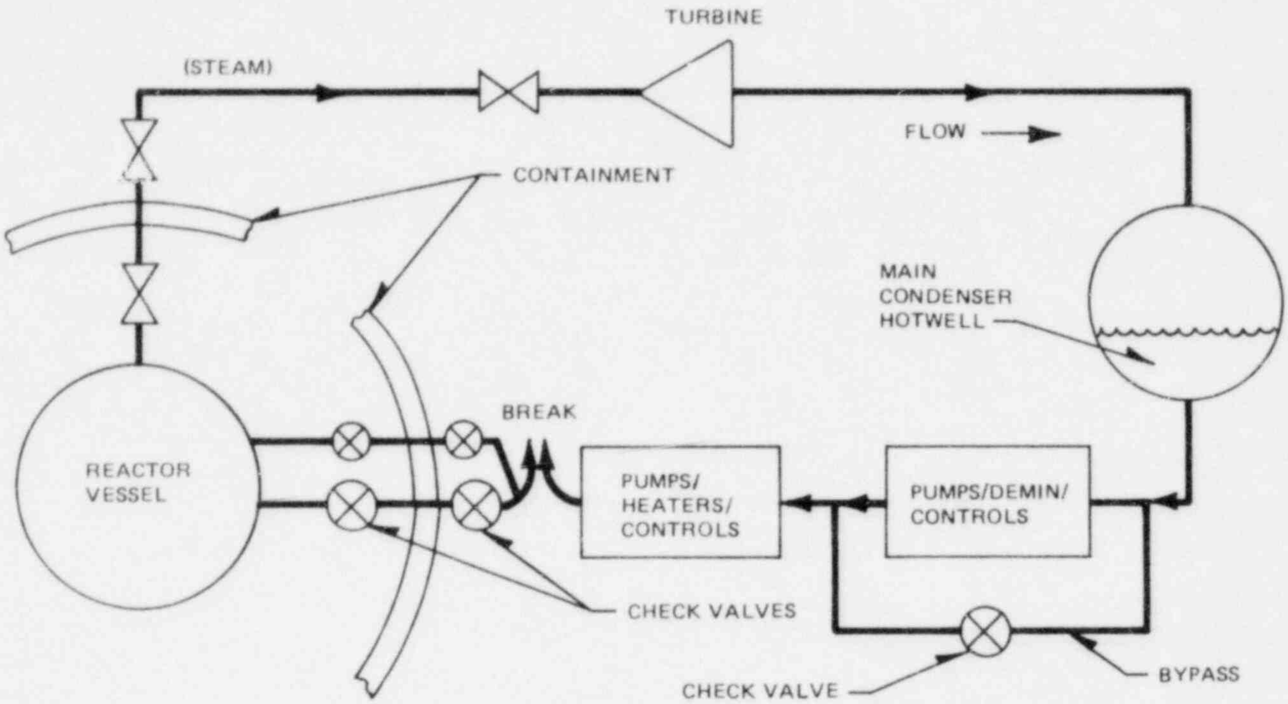

Figure 15.6-4. Leakage Path for Feedwater Line Break Outside Containment

SECTION 15.6

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CONTENTS (Continued)

CONTENTS (Continued)

SECTION 15.7

TABLES

TABLES (Continued)

SECTION 15.7

ILLUSTRATIONS

15.7 RADIOACTIVE RELEASE FROM SUBSYSTEMS AND COMPONENTS

15.7.1 Radioactive Gas Waste System Leak or Failure

The following radioactive gas waste system components are examined under severe failure mode conditions for effects on the plant safety profile:

(1) main condenser gas treatment system failure;
(2) malfunction of main turbine gland sealing system; and
(3) failure of air ejector lines.

15.7.1.1 Main Condenser Offgas Treatment System Failure

15.7.1.1.1 Identification of Causes and Frequency Classification

15.7.1.1.1.1 Identification of Causes

Those events which could cause a gross failure in the offgas treatment system are:

(1) a seismic occurrence - exceeding the seismic capabilities of the equipment;

(2) a hydrogen detonation which ruptures the system pressure boundary;

(3) a fire in the filter assemblies; and

(4) failure of adjacent equipment which could subsequently cause failure of offgas equipment.

The seismic event is considered to be the only conceivable event which could cause significant system damage.

15.7.1.1.1.1 Identification of Causes (Continued)

The equipment and piping are designed to contain any hydrogen-oxygen detonation which has a reasonable probability of occurring. A detonation is not considered as a possible failure mode.

The decay heat on the filters is insignificant and cannot serve as an ignition source for the filters.

The system is isolated from other systems or components which could cause any serious interaction or failure. The only credible event which could result in the release of significant activity to the environment is an earthquake, causing building damage and subsequent rupture of offgas components from falling building debris.

Even though the offgas [illegible] is designed to uniform building code seismic requirements, an event more severe than the design requirements is arbitrarily assumed to occur, resulting in the failure of the offgas system.

The design basis, description and performance evaluation of the subject system is given in Section 11.3.

15.7.1.1.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.1.2 Sequence of Events and System Operation

15.7.1.1.2.1 Sequence of Events

The sequence of events following this failure is shown in Table 15.7-1.

15.7.1.1.2.2 Identification of Operator Actions

Gross failure of this system may require manual isolation of this system from the main condenser. This isolation results in high condenser pressure and a reactor scram. The operator should monitor the turbine-generator auxiliaries and break vacuum as soon as possible. The operator should notify personnel to evacuate the area immediately and notify radiation protection personnel to survey the area and determine requirements for reentry. The time needed for these actions is about 2 min.

15.7.1.1.2.3 Systems Operation

In analyzing the postulated offgas system failure, no credit is taken for the operation of plant and reactor protection systems, or of engineered safety features. Credit is taken for functioning of normally operating plant instruments and controls and other systems only in assuming the following:

(1) capability to detect the failure itself - indicated by an alarmed increase in radioactivity levels seen by Area Radiation Monitoring System, in an alarmed loss of flow in the Offgas System, and in an alarmed increase in activity at the vent release;

(2) capability to isolate the system and shutdown the reactor; and

(3) operational indicator and annunciators in the main control room.

15.7.1.1.2.4 The Effect of Single Failures and Operator Errors

After the initial system gross failure, the inability of the operator to actuate a system isolation could affect the analysis.

15.7.1.1.2.4 The Effect of Single Failures and Operator Errors (Continued)

However, the seismic event which is assumed to occur beyond the present plant design basis for nonsafety equipment will undoubtedly cause the tripping of turbine or will lead to a load rejection. This will initiate a scram and negate a need for the operator to initiate a reactor shutdown via system isolation (see Appendix 15A for details).

15.7.1.1.3 Core and System Performance

The postulated failure results in a system isolation, necessitating reactor shutdown because of loss of vacuum in the main condenser. This transient has been analyzed in Subsection 15.2.5.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the offgas system pressure boundary. No credit is taken for performance of secondary barriers, except to the extent inherent in the assumed equipment release fractions discussed in Subsection 15.7.1.1.5.

15.7.1.1.5 Radiological Consequences

15.7.1.1.5.1 General

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

15.7.1.1.5.1 General (Continued)

(2) The second is based on assumptions considered to provide a realistic yet conservative estimate of radiological consequences. This analysis is referred to as the "realistic analysis".

Both are based on the following equipment characteristics with respect to retention of radioactive solid daughter products:

(1) Offgas condenser - 100% retained and continuously washed out;

(2) Water separator - 100% retained and continuously washed out;

(3) Holdup pipe - 60% retained and continuously washed out;

(4) Prefilter - 100% retained, element changed approximately annually;

(5) Dryer - 100% retained;

(6) Carbon beds - 100% retained; and

(7) Post filter - 100% retained, element changed approximately annually.

15.7.1.1.5.2 Design Basis Analysis

There are no specific quantitative regulatory guidelines or requirements upon which to perform a Design Basis Analysis. Nevertheless, an evaluation, which is believed to produce very limiting conservative results, has been performed. The primary differences between this analysis and the Realistic Analysis are in the basic

15.7.1.1.5.2 Design Basis Analysis (Continued)

source term and the equipment release fractions. The same analytical techniques used for the realistic analysis are used for this evaluation. Specific parametric values used in this evaluation are presented in Table 15.7-2.

15.7.1.1.5.2.1 Fission Product Release

15.7.1.1.5.2.1.1 Initial Conditions

The activity in the offgas system is based on the following conditions:

(1) 2 SCFM air inleakage, and

(2) 100,000 μCi/sec noble gas after 30 min delay for a period of 11 months, followed by 1 month of 350,000 μCi/sec at 30 min.

15.7.1.1.5.2.1.2 Assumptions

Depending on the assumptions as to radionuclide release fraction, various equipment pieces could be controlling with respect to dose consequences. The assumed released fractions for the design basis analysis are found in Table 15.7-3.

The iodine activity leaving the offgas recombiner has been assumed to be entirely retained in the first charcoal tank. Thus, failure of this tank results in the highest potential iodine release. Iodine absorbs strongly to charcoal. A conservative evaluation leads to an assumption of the release of 1% of the iodine in the first charcoal tank.

15.7.1.1.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of direct release of fission products to the environment from the failed component through the building ventilation system. The release of activity to the environment is presented in Table 15.7-4.

15.7.1.1.5.2.3 Results

The calculated exposures for the Design Basis Analysis are presented in Table 15.7-5 and are well within the guidelines of 10CFR100.

15.7.1.1.5.3 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-2.

15.7.1.1.5.3.1 Fission Product Release

15.7.1.1.5.3.1.1 Initial Conditions

The activity in the offgas system is based on the following normal operating conditions:

(1) 30 SCFM air inleakage, and
(2) 100,000 μCi/sec Noble Gas after 30-min delay.

The activity stored in the various equipment pieces before the postulated failure is given in Table 12A-1 (Appendix 12A).

15.7.1.1.5.3.1.2 Assumptions

The only credible failure that could result in loss of carbon from the vessels is the failure of the concrete structure surrounding the vessel. A circumferential failure of the vessel could result from concrete falling on the vessel in either of two ways:

(1) Pending Load - the vessel being supported in the center and loaded on each end. This could result in a tear around 50% of the circumference.

(2) Shearing Load - the vessel being supported and loaded near the same point from above.

In either case, no more than 10-15% of the carbon would be displaced from the vessel. Iodine is strongly bonded to the charcoal and would not be expected to be removed by exposure to the air. However, the conservative assumption is made that 1% of the iodine activity contained in the absorber tanks is released to the vault containing the offgas equipment.

Measurements made at KRB indicate that offgas is about 30% richer in Kr than air. Therefore, if this carbon is exposed to air, it will eventually reach equilibrium with the noble gases in the air. However, the first few inches of carbon will blanket the underlying carbon from the air. A 10% loss of noble gas activity from a failed vessel is conservative because of the small fraction of carbon exposed to the air.

Prefilters: Because of the design features of the prefilter vessel (approximately 24 in. diameter, 4 ft. height, 350 psig design pressure, 1/2 in. wall thickness and collapsible filter media), a failure mechanism cannot be postulated that will result in emission of filter media or daughter products from this vessel. However, to illustrate the consequences of a radioactivity loss from this vessel, 1% release of particulate activity is assumed.

15.7.1.1.5.3.1.2 Assumptions (Continued)

Holdup Pipe: Pipe rupture and depressurization of the pipe is considered. Normally, the pipe will operate at less than 16 psia and depressurize to 14.7 psia. The possible loss of solid daughters and noble gases and iodines is conservatively taken as 20%. The model used assumes retention and washout of 60% of the particulate daughters for the calculation of the holdup pipe inventory.

Piping: It is assumed that the seismic event causing the pipe failure is accompanied by a reactor isolation, stopping steam flow to the steam jet air ejectors. Therefore, the resulting release from failed piping is not significant compared to those failures previously considered.

15.7.1.1.5.3.2 Fission Product Transport to the Environment

The release of activity to the environment is presented in Table 15.7-6.

15.7.1.1.5.3.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-7.

15.7.1.2 Malfunction of Main Turbine Gland Sealing System

(Applicant to supply.)

15.7.1.3 Failure of Main Turbine Steam Air Ejector Lines

(Applicant to supply).

15.7.2 Liquid Radioactive System Failure

15.7.2.1 Identification of Causes and Frequency Classification

15.7.2.1.1 Identification of Cause

The event which could cause a failure in the liquid radwaste system is a liquid radwaste tank rupture by a seismic occurrence.

Although the system consists of Non-Seismic Category I Equipment, the liquid radwaste tanks are constructed in accordance with sound engineering principles. Therefore, simultaneous failure of all of the tanks is unlikely. However, for purposes of this analysis, a simultaneous failure releasing the contained liquid activity of all tanks is assumed.

15.7.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.2.5 Radiological Consequences

15.7.2.5.1 General

Two radiological analyses are provided for this accident:

(1) The "design basis analysis" is based upon conservative assumptions considered to be acceptable to the NRC for the purpose of determining design adequacy to meet 10CFR100 guidelines.

(2) The conservative "realistic analysis" is considered to provide a realistic estimate of radiological consequences.

15.7.2.5.2 Design Basis Analysis

The liquid radwaste tank failure analysis is evaluated in accordance with the following parameters:

(1) simultaneous rupture of the liquid radwaste tanks and release of all liquid contents;

(2) 10% of total iodine inventory released becomes airborne for release to environs;

(3) release takes place over 2-hr period;

(4) Atmospheric dispersion is 5 percentile probable χ/Q; and

(5) χ/Q at the site boundary is 2.0E-3 sec/m^3.

15.7.2.5.2.1 Fission Product Release

The activity contained as I-131, 132, 133, 134, and 135 in the major radwaste tanks liquid is shown in Table 15.7-8. Activity content is based upon the design basis source term of 100,000 μCi/sec. Tank volumes are presented in Section 11.2

15.7.2.5.2.2 Fission Product Transport to the Environment

It is conservatively assumed that the activity listed in Table 15.7-9 is released from the building at ground level.

15.7.2.5.2.3 Results

The resultant thyroid inhalation exposures from the iodine activity released to the environment are listed in Table 15.7-10. Since very little noble gas activity is released, the whole body dose is negligible. It should be noted that the assumption of

15.7.2.5.2.3 Results (Continued)

release to the environment of 10% of the iodine activity contained in the radwaste tanks, using 5% probable χ/Q will undoubtedly result in an overestimate of real exposure by a factor to 10 to 100. However, exposures are well within the guidelines of 10CFR100.

15.7.2.5.3 Realistic Analysis

Parameters used in the design basis analysis would be pertinent to the realistic analysis with the following exception:

(1) 1% of total iodine inventory released becomes airborne for release to environs, and

(2) only the concentrated waste tank (greatest iodine content) is ruptured and releases all liquid contents.

15.7.2.5.3.1 Fission Product Release

The activity contained in the concentrated waste tank is as shown in the appropriate row and column in Table 15.7-8.

15.7.2.5.3.2 Fission Product Transport to the Environment

It is conservatively assumed that the activity listed in Table 15.7-11 is released from the building at ground level.

15.7.2.5.3.3 Results

Radiological effects from a realistic basis reduces the dose over the design basis effect by a factor of about 20 due to less iodine released from a single tank. Results from the design basis (Table 15.7-10) are already substantially under 10CFR100 guidelines.

15.7.3 Postulated Radioactive Released Due to Liquid Radwaste Tank Failure

15.7.3.1 Identification of Cause and Frequency Classification

15.7.3.1.1 Identification of Causes

An unspecified event causes the complete release of the average radioactivity inventory in the tank containing the largest quantities of significant radionuclides in the liquid radwaste system. This is one of the concentrated waste tanks in the radwaste enclosure. The airborne radioactivity released during the accident passes directly to the environment via the plant vent stack.

Postulated events that could cause release of the radioactive inventory of the concentrated waste tank are cracks in the vessels and operator error. The possiblity of small cracks and consequent low-level release rates receives primary consideration in system and component design. The concentrated waste tank is designed to operate at atmospheric pressure and 200°F maximum temperature so the possibility of failure is considered small. A liquid radwaste release caused by operator error is also considered a remote possiblity. 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 liquid radioactive wastes occur, floc. [illegible] in sump pumps in the floor of the radwaste building will receive a high water level alarm, activate automatically, and remove the spilled liquid.

15.7.3.1.2 Frequency Classification

Much of the exposition concerning the remote likelihood of a leakage or malfunction accident of the concentrates waste tank applies equally to a complete release accident. However, the probability

15.7.3.1.2 Frequency Classification (Continued)

of a complete rupture or complete malfunction accident is considered even lower.

Although not analyzed for the requirements of Seismic Category I equipment, the liquid radwaste tanks are constructed in accordance with the sound engineering principles. Therefore, simultaneous failure of all the tanks is not considered credible.

This accident is expected to occur with the frequency of a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

The sequence of events expected to occur is as follows:

Sequence of Events	Elapsed Time (min)
(1) Event begins - failure occurs.	0
(2) Area radiation alarms alert plant personnel.	~1
(3) Operator actions begin.	~5

The rupture of a concentrated waste tank would leave little recourse to the operator. No method of recontaining the gaseous phase discharge is available; however, isolation of the radwaste area would minimize the results. High radiation alarms both in the radwaste ventilation exhaust and in the radwaste area would alert the operator to the failure.

Normal isolation of the radwaste area ventilation is actuated upon initiation of the above alarms. However, no credit for any operator action or for ventilation isolation has been taken in evaluating this event.

15.7.3.4 Design Basis Accident

The design basis accident is based on a conservative assessment of this accident. Two release pathways are analyzed. An airborne release is analyzed in which 10% of the iodine inventory is assumed to be released to the environment, and a surface water release is analyzed in which 90% of the concentrated waste tank is assumed to be released directly to the surface water. The specific models, assumptions and programs used for computer evaluation are described in References 1 and 2. Specific values of parameters used in the evaluation are presented in Table 15.7-12.

15.7.3.4.1 Fission Product Release

The fission product release is identified in Subsection 15.7.3.5.1 and is based on an offgas release rate of 100,000 μCi/sec at 30 minutes.

15.7.3.4.2 Fission Product Release to the Environment

Tables 15.7-13 and 15.7-14 present the information on activity released to the environment.

15.7.3.4.3 Results

Table 15.7-15 provides the airborne radiological effects from this event. It should be noted that the referenced computer program which is used to evaluate the radiological consequences of this event is based on the assumption that the activity in the aquatic life is at equilibrium levels. Since this assumption will result in an over-estimate of the actual consequence, the radiological doses in Table 15.7-16 are considered to be very conservative.

15.7.3.5 Realistic Analysis

The realistic analysis is based on a realistic (but still conservative) assessment of this accident. The specific models, assumptions and the program used for computer evaluation are also described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-12.

15.7.3.5.1 Fission Product Release

The fission produce release is based on an offgas release rate of 100,000 μCi/sec at 30-min decay.

15.7.3.5.2 Fission Product Transport to the Environment

Table 15.7-17 presents the information on activity released to the environment.

15.7.3.5.3 Results

It should be noted that the referenced computer program which is used to evaluate the radiological consequences of this event is based on the assumption that the activity in the aquatic life is at equilibrium levels. Since this assumption will result in an overestimate of the actual consequence, the radiological doses in Table 15.7-18 are considered to be very conservative.

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 the dropping of a raised fuel assembly onto stored fuel bundles. A variety of events which qualify for the class of accidents termed "fuel-handling accidents" has been investigated. These included considerations for containment upper pool refueling operations, as well as refueling building-pool activities. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle onto the reactor core when the reactor vessel head is off. However, because this takes place only in the containment and the containment leak rate is very low, a fuel-handling accident in the refueling building results in higher offsite radiological releases.

15.7.4.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

The most severe fuel-handling accident from a radiological release viewpoint is the drop of a channeled spent fuel bundle onto

15.7.4.2.1 Sequence of Events (Continued)

unchanneled spent fuel in the spent fuel racks in the refueling building. The sequence of events which is assumed to occur is as follows:

Event	Approximate Elapsed Time (sec)
(1) Channeled fuel bundle is being handled by a crane over spent fuel pool. Crane motion changes from horizontal to vertical and the fuel grapple releases, dropping the bundle. The channeled bundle strikes unchanneled bundles in the rack.	
(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 refueling building.	0
(4) The refueling building ventilation system high radiation alarm alerts plant personnel.	0
(5) Operator actions begin.	

15.7.4.2.2 Identification of Operator Actions

The operator actions are as follows:

(1) initiate the evacuation of the fuel storage building and the locking of the fuel storage building doors;

(2) The fuel handling foreman should give instructions to go immediately to the radiation protection personnel decontamination area;

15.7.4.2.2 Identification of Operator Actions (Continued)

(3) the fuel-handling foreman should make the operations shift engineer aware of the accident;

(4) the shift engineer should immediately determine if the normal ventilation system has isolated and the standby gas treatment is in operation;

(5) the shift engineer should initiate 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 should determine if the standby gas treatment system is performing as designed;

(7) the duty shift engineer should post the appropriate radiological control signs at the entrance of the reactor building; and

(8) before entry to the refueling building is made, a careful study of conditions, radiation levels, etc., will be 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.

15.7.4.2.4 The Effects of Single Failures and Operator Errors

The automatic ventilation isolation system (includes the radiation monitoring detectors, isolation valves, and the SGTS) is designed to single-failure criteria and safety requirements (see Sections 7.6 and Appendix 15A for details).

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 realistic, yet conservative assessment of the consequences.

The kinetic energy acquired by a falling fuel assembly may be dissipated in one or more impacts.

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 spent fuel racks 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 are expected to 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 ft-lb prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 250 ft-lb 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

15.7.4.3.1 Mathematical Model (Continued)

fuel assembly consists of 76% fuel, 19% cladding, and 5% other structural material by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

The energy absorption on successive impacts is estimated by considering a plastic impact. Conservatism of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is:

$$1 - \frac{M_1}{M_1 + M_2}$$

where M_1 is the impacting mass and M_2 is the struck mass.

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 less than 10 ft.

(2) The entire amount of potential energy, referenced to the top of the spent fuel racks, 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 in the water above the rack 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.

15.7.4.3.2 Input Parameters and Initial Conditions (Continued)

(3) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

General Compliance or Alternate Approach Assessment Regulatory Guide 1.25: (For commitment and revision number, see Section 1.8.)

This guide 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 by General Electric in the analyses are as follows:

(1) Site Boundary χ/Q of 2.0E-3 sec/m^3.

(2) SGTS Filter Efficiency 99% for all iodine forms.

(3) All activity released to the environment is via the SGTS.

(4) 101 Fuel Rods damaged.

Some of the models and conditions that are prescribed are demonstratably inconsistent with physical phenomena and for this reason additional analyses are provided in Subsection 15.7.4 to demonstrate the conservative bias of the regulatory requirements.

15.7.4.3.3 Results

15.7.4.3.3.1 Energy Available

Dropping a fuel assembly onto the spent fuel racks from the maximum height of 6 ft results in an impact velocity of 19.7 ft/sec.

The kinetic energy acquired by the falling fuel assembly is less than 4280 ft-lb and is dissipated in one or more impacts.

15.7.4.3.3.2 Energy Loss Per Impact

Based on the fuel geometry in the spent fuel rack, two fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is approximately 67%.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 22 more fuel assemblies, so that after the second impact only 60 ft-lb (approximately 2% of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft-lb in compression before cladding failure, it is unlikely that any fuel rod will fail on a third impact.

If the dropped fuel assembly strikes only one fuel assembly on the first impact, the energy absorption by the fuel rack support structure results in approximately the same energy dissipation on the first impact as in the case where two fuel assemblies are struck. The energy relations on the second and third impacts remain approximately the same as in the original case. Thus, the calculated energy dissipation is as follows:

First impact = 67%
Second impact = 32%
Third impact = 1% (no cladding failures)

15.7.4.3.3.3 Fuel Rod Failures

15.7.4.3.3.3.1 First Impact Failures

The first impact dissipates 0.67 x 4,280 or 2.850 ft-lb of energy. It is assumed that 50% of this energy is absorbed by the dropped fuel assembly and that the remaining 50% is absorbed by the struck fuel assemblies in rack. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 62 rods of the dropped fuel assembly are assumed to fail. Because the eight tie rods of each struck fuel assembly are more susceptible to bending failure than the other 54 fuel rods, it is assumed that they fail on the first impact. Thus, 2 x 8 = 16 tie rods (total in 2 assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the spent fuel racks, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause falure oi all the remaining rods of the two struck assemblies, 250 x 54 x 2 or 27,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the stuck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 2{,}850 \times \frac{19}{19+5}}{250} = 5$$

15.7.4.3.3.3.1 First Impact Failures (Continued)

Thus, during the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	16 tie rods (bending)
Struck assemblies	5 rods (compression)
	83 failed rods

15.7.4.3.3.3.2 Second Impact Failures

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 22 struck assemblies are the tie rods subjected to bending failure. Thus, 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\frac{0.32}{2} \times 4{,}280 \times \frac{19}{19+5}}{250} = 2$$

Thus, during the second impact, the fuel rod failures are as follows:

Struck assemblies	16 tie rods (bending)
Struck assemblies	2 rods (compression)
	18 failed rods

15.7.4.3.3.3.3 Total Failures

The total number of failed rods resulting from the accident is as follows:

First impact	83 rods
Second impact	18 rods
Third impact	0 rods
	101 total failed rods

15.7.4.4 Barrier Performance

This failure occurs in the refueling building outside the normal barriers (RCPB and Containment). Therefore, this section is not directly applicable. The transport of fission products to the environment is discussed in Subsection 15.7.4.5.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first 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. This analysis is referred to as the "Design Basis Analysis".

(2) The second analysis is based on assumptions considered to provide a realistic yet conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis".

For both analyses, the fission product inventory in the fuel rods assumed to be damaged is based on 1000 days of continuous operation at 3651 MWt. A 24-hr period for decay from the above

15.7.4.5 Radiological Consequences (Continued)

power condition is assumed because it is not expected that fuel handling can begin within 24 hr following initiation of reactor shutdown. Figure 15.7-1 indicates 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 evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.7-19.

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 - 3651 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 Fuel Building - 10% of the noble gas activity and 0.1% for the iodine activity.

Based on the above conditions, the activity released to the fuel building is presented in Table 15.7-20.

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 fuel building (Table 15.7-20) 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 15.7-21.

15.7.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-22 and are well within the guidelines of 10CFR100.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 1. Specific values of parameters used in the evaluation are presented in Table 15.7-19.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission release estimates for the fuel-handling accident are based on the following assumptions:

(1) The reactor fuel has an average irradiation time of 1000 days at NBR up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not increase the concentration of biologically significant isotopes. The 24-hr decay period allows time to shut down the reactor, depressurize the nuclear system, remove the reactor vessel head and remove the reactor vessel upper internals.

15.7.4.5.2.1 Fission Product Release from Fuel (Continued)

It is not expected that these operations could be accomplished in less than 24 hr and probably will require at least 48 hrs.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity is in the fuel rod plena and available for release. This assumption is based on fission product release data from defective fuel experiments (Reference 4).

(3) Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products is assumed to be released.

(4) It is assumed that 101 fuel rods fail. This is considered to be conservative because it is expected that much less than 101 rods would be damaged.

15.7.4.5.2.2 Fission Product Transport to the Environment

The following assumptions and conditions are assumed in calculating the release of activity to the environments.

(1) The fission product activity released to the refueling building will be in proportion to the removal efficiency of the water in the fuel pool. Because water has a negligible effect on removal of the noble gases, the gases are assumed to be instantaneously released from the pool to the building.

(2) The iodine activity airborne is in proportion to the partition factor and the ratio of the volume of air (V_a) to the volume of water (V_w) for which the respective

15.7.4.5.2.2 Fission Product Transport to the Environment (Applicability to be confirmed by Applicant) (Continued)

values are applicable. It is assumed that a partition factor of 100 and a V_a/V_w of 3 is applicable for this event. It should be noted that the volume assumed for V_a is not equal to the total volume of air in the refueling building, but is nevertheless considered to be a conservative estimate of the volume of air which may form an equilibrium condition with the activity in the fuel storage pool.

(3) The ventilation rate from the refueling building to the environment via the SGTS is 9 air changes per day. Based on these assumptions, the activity airborne in the refueling building is shown in Table 15.7-23.

Due to isolation of the refueling building and initiation of the SGTS, the release rate to the environment is 9 air changes per day. Considering an SGTS efficiency for iodine of 99.9%, the integrated activity discharged to the environment is presented in Table 15.7-24.

15.7.4.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-25 and are well below the guidelines set forth in 10CFR100.

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

15.7.5.1 Identification of Cause (Continued)

is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fail.

It is assumed that a spent fuel shipping cask containing irradiated fuel assemblies is in the process of being moved with the cask suspended from the crane above the rail care. The fuel assemblies have been out of the reactor for at least 90 days.

Through some unspecified failure, the cask is released from the crane and falls less than 30 ft; however, the cask does not fail and there is no radioisotopic release. Therefore, there is no radiological release to the environment for this event.

15.7.6 References

1. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142).

2. P. P. Stancavage and D. C. Abbott, "Liquid Discharge Doses - LIDSR Code", August 1976 (NEDM-20609-1).

3. P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143).

4. N. R. Horton, W. A. Williams, and K. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors", March 1976 (APED-5756).

Table 15.7-1

SEQUENCE OF EVENTS FOR MAIN CONDENSER GAS TREATMENT SYSTEM FAILURE

Approximate Elapsed Time	Events
0 sec	Event begins - system fails
0 sec	Noble gases are released
< 1 min	Area radiation alarms alert plant personnel
< 1 min	Operator actions begin with: (a) initiation of appropriate system isolations (b) manual scram actuation (c) assurance of reactor shutdown cooling.

Table 15.7-2

GASEOUS RADWASTE SYSTEM FAILURE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source fropostulated accidents		
A. Power level	NA	
B. Burnup	NA	
C. Fuel Damage	None	None
D. Inventory of activity by nuclide	Table 12A-4	Table 12A-1
E. Iodine fractions	NA	NA
(1) Organic	0	0
(2) Elemental	1.0	1.0
(3) Particulate	0	0
F. Reactor coolant activity before the accident		
II. Data and assumptions used to estimate activity released		
A. Containment leak rate (%/day)	NA	NA
B. Secondary containment leak rate ((%/day)	NA	NA
C. Valve Movement times	NA	NA
D. Absorption and filteration efficiencies	NA	NA
(1) Organic Iodine	NA	NA
(2) Elemented Iodine	NA	NA
(3) Particulate Iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing Efficiency	NA	NA
(3) Filter Efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None

Table 15.7-2

GASEOUS RADWASTE SYSTEM FAILURE – PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES (Continued)

		Design Basis Assumptions	Realistic Basis Assumptions
III.	Dispersion Data		
	A. Site Boundary and LPZ distances (m)	*	*
	B. χ/Q's for SB/LPZ	2.0E-3/1.0E-3	2.0E-3/ 1.0E-3
IV.	Dose Data		
	A. Method of dose calculation	NA	
	B. Dose conversion assumptions	Reference 1	Reference 1
	C. Peak activity concentrations in containment	NA	NA
	D. Doses	Table 15.7-5	Table 15.7-7

*Applicant to Supply

Table 15.7-3

EQUIPMENT FAILURE RELEASE ASSUMPTIONS RELEASE FRACTIONS
ASSUMED FOR DESIGN BASIS/REALISTIC ANALYSIS

Equipment Piece	Noble Gases	Solid Daughters	Radioiodine
Preheater	1.00/1.00	1.00/1.00	N/A
Catalytic Recombiner	1.00/1.00	1.00/1.00	N/A
Offgas Condenser	1.00/1.00	1.00/1.00	N/A
Water Separator	1.00/1.00	1.00/1.00	N/A
Holdup Pipe	1.00/0.20	1.00/0.20	1.00/0.20
Cooler Condenser	1.00/1.00	1.00/1.00	N/A
Moisture Separator	1.00/1.00	1.00/1.00	N/A
Dessicant Dryer	1.00/0.10	0.01/0.01	N/A
Prefilter	1.00/1.00	0.01/0.01	N/A
Charcoal Adsorbers	0.10/0.10	0.01/0.01	0.01/0.01
Afterfilter	1.00/1.00	0.01/0.01	N/A

Table 15.7-4

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(DESIGN BASIS ANALYSIS)
FISSION PRODUCT RELEASE TO ENVIRONMENT

Isotope	Ci	Isotope	Ci	Isotope	Ci
Cr23	1.39E-2	H3	1.01E-3	Ru103	1.07E-7
24	3.02E-2	C14	9.81E-5	105	1.08E-6
25	2.96E-2	Na24	4.18E-6	1-6	4.65E-9
I131	9.98E-3	P32	4.01E-8	Ag110m	2.25E-7
132	1.28E-1	Cr51	9.70E-7	Te129m	2.76E-8
133	7.43E-2	Mn54	4.69E-8	129	3.98E-6
134	2.87E-1	56	1.06E-4	131m	1.86E-7
135	1.19E-1	Fr59	1.47E-7	131	2.76E-7
		Co58	8.58E-6	132	1.08E-6
Kr83m	9.00E+1	60	3.74E-7	Co187	4.88E-8
85m	1.54E+2	Ni65	6.84E-7	188	1.72E-8
85	8.24E-2	Zn65	2.71E-9	Co189	6.29
87	4.19E-12	Pb88	1.42E+2	140	8.44E-8
88	4.88E-12	89	6.80E+1	141	2.84E-8
89	1.52E+13	Br89	2.96E-2	142	4.31E-8
90	3.35E+13	90	4.34E-5	La140	1.08E-4
Xe131m	8.56E-1	91	8.12E-2	142	1.09E-8
133m	8.81	92	1.30E-3	Ce141	2.00E-7
133	2.14E+2	Y90	3.51E-7	143	9.87E-7
135m	3.87E+2	91m	1.61E-2	144	7.66E-8
135	6.85E+2	91	1.75E-5	Nd147	1.18E-8
137	1.87E+3	92	5.31E-6	W187	1.90E-8
138	1.26E+3	93	3.48E-6	Np289	1.81E-3
139	3.54E+3	Zr95	1.83E-7		
140	3.17E+3	97	1.73E-6		
		Nb95	1.60E-4		
		Mo99	1.69E-6		
		Te99m	8.97E-4		
		101	1.14E-3		

Table 15.7-4 (Continued)

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(DESIGN BASIS ANALYSIS)
FISSION PRODUCT RELEASE TO ENVIRONMENT

3T	9.991E-01	90RB	1.471E 02	106TC	4.864E-02	143XE	1.815E 07
13N2	3.865E 03	90SR	8.272E-05	106RU	2.583E-06	143SC	9.297E-04
13AM	9.577E 00	90YM	1.262E-08	110AGM	1.250E-04	143BA	1.067E 00
13NO	5.922E-02	90Y	1.046E-06	129SB	3.622E-04	143LA	6.382E-02
14C	3.721E-02	91BR	5.106E-03	129TEM	1.535E-05	143CE	5.483E-04
16N2	8.289E 06	91KR	1.559E 05	129TE	2.184E-03	143PR	1.490E-04
16AM	3.623E 04	91RB	4.750E 02	129I	7.410E-09	144XE	1.743E 01
16NO	8.801E 01	91SR	2.394E-01	131SB	1.496E-02	144CS	5.810E-01
17N2	1 374E 03	91YM	5.578E-03	131TEM	7.547E-05	144BA	1.441E 00
17AM	7.677E 00	91Y	5.226E-05	131TE	1.588E-02	144LA	5.361E-01
17NO	3.422E-02	92BR	1.991E-07	131I	5.545E 00	144CE	4.253E-05
18F	4.951E 00	92KR	6.949E 04	131XEM	1.116E 02	147ND	6.288E-05
19O	7.646E 02	92RB	7.091E 02	132TE	5.977E-04	147PM	6.639E-06
24NA	2.292E-03	92SR	8.352E-01	132I	7.094E 01	149ND	1.517E-03
32P	2.225E-05	92Y	2.876E-03	133SB	8.744E-02	149PM	6.996E-05
51CR	5.391E-04	93KR	5.428E 03	133TEM	1.195E-02	187W	1.056E-02
54MN	2.604E-05	93RB	1.490E 02	133TE	3.510E-02	239NP	7.297E-01
56MN	5.873E-02	93SR	2.659E 00	133IM	1.043E 00		
58CO	4.748E-03	93Y	1.933E-03	133I	4.127E 01		
59FE	8.181E-05	93ZR	6.383E-11	133XEM	1.157E 03		
60CO	2.080E-04	93NBM	2.307E-12	133XE	2.798E 04		
65NI	3.524E-04	94KR	2.366E-05	134TE	2.616E-02		
65ZN	1.505E-06	94RB	6.495E-01	134IM	5.703E 00		
69ZNM	3.440E-05	94SR	8.276E-01	134I	1.316E 02		
83AS	3.917E-02	94Y	5.011E-02	134XEM	6.465E-01		
83SEM	2.113E-02	95KR	8.360E-02	135K	6.615E 01		
83SE	1.389E-03	95RB	3.539E-03	135XEM	1.341E 05		
83BR	7.746E 00	95SR	8.794E-01	135XE	8.437E 04		
83KRM	1.458E 04	95Y	8.721E-02	135CSM	7.169E-05		
84AS	2.611E-02	95ZR	1.014E-04	135CS	4.569E-10		
84SE	3.169E-02	95NBM	2.172E-06	136TE	1.565E-01		
84BRM	2.799E-01	95NB	8.863E-05	136IM	2.151E 01		
84BR	1.680E 01	97ZR	9.621E-04	136I	3.258E 01		
85AS	6.725E-03	97NBM	3.654E-01	137I	2.937E 01		
85SEM	4.135E-02	97NB	1.193E-02	137XE	9.242E 05		
85SE	5.041E-02	99ZR	2.766E-01	137CS	9.029E-05		
85BR	1.644E 01	99NBM	8.237E-02	137BAM	1.285E-04		
85KRM	2.172E 04	99NB	4.750E-01	138XE	4.510E 05		
85KR	1.044E 01	99MO	9.380E-04	138CSM	3.111E-02		
87AS	2.61[illegible]E-04	99TCM	4.985E-01	138CS	3.184E 00		
87SE	9.006E-02	99TC	1.072E-08	139XE	1.953E 06		
87BR	2.253E 01	101MO	4.963E-02	139CS	5.039E 01		
87KR	7.499E 04	101TC	6.342E-01	139BA	3.249E-01		
88SE	6.922E-03	102MO	5.230E-02	140XE	1.759E 06		
88BR	2.011E 01	102TCM	5.445E-01	140CS	4.404E 02		
88KR	7.315E 04	102TC	1.665E-04	140BA	2.551E-02		
88RB	9.242E-01	103TC	3.930E-01	140LA	1.838E-04		
89SE	1.499E-06	103RU	5.939E-05	141XE	4.342E 04		
89BR	8.827E 00	103RHM	6.396E-03	141CS	1.800E 02		
89KR	7.645E 05	104MO	8.987E-02	141BA	1.409E 00		
89RB	1.139E 01	104TC	2.178E-01	141LA	4.119E-03		
89SR	6.810E-03	105MO	5.886E-02	141CE	1.112E-04		
89YM	3.892E-08	105TC	1.213E-01	142XE	3.257E 03		
90BR	1.215E 00	105RU	5.987E-04	142CS	9.245E 01		
90KR	1.854E 06	105RHM	1.990E-02	124BA	2.336E 00		
90RBM	1.161E 01	105RH	8.604E-05	142LA	1.106E-02		

T = Tritium
No = Nitrogen Oxide
AM = Ammonia
N2 = Gaseous Nitrogen

Table 15.7-5

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(DESIGN BASIS ANALYSIS)
OFF-SITE RADIOLOGICAL EFFECTS (mRem)

Site Boundary

	Bone	Liver	Total Body	Thryoid	Kidney	Lung	GI-LLI
Halogen	1.31E-1	2.69E-1	1.02E-1	30.70	0.44	--	1.21E-1
Noble Gas			1.67E+3*				
Other	6.1	15.56	8.29	2.3E-4	7.15	7.60	2.93
Total	6.23	15.83	1.68E+3	30.70	7.59	7.60	3.05

Low Population Zone

	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Halogen	6.6E-2	1.35E-1	5.0E-2	15.4	2.20E-1		0.67
Noble Gas			3.61E+2*				
Other	3.07	7.77	4.14	1.2E-4	3.59	3.81	1.46
Total	3.14	7.90	3.65E+2	15.4	3.81	3.81	2.13

*Decay in flight accounted for.

Table 15.7-6

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(REALISTIC ANALYSIS)
FISSION PRODUCT RELEASE TO ENVIRONMENT

Isotope	Ci	Isotope	Ci	Isotope	Ci
Br83	4.65E-3	H3	2.03E-3	Rn103	4.01E-9
84	1.01E-2	C14	1.90E-4	104	3.89E-7
85	9.87E-3	Na24	4.13E-6	106	2.84E-9
I131	3.38E-3	P32	4.12E-9	Ag110m	1.24E-7
132	4.32E-2	Cr51	103E-6	Te129m	1.08E-8
133	2.48E-2	Mn54	8.22E-8	129	1.41E-6
134	7.90E-2	56	1.06E-4	131m	4.54E-9
135	3.97E-2	Fe59	1.68E-7	131	9.53E-6
		Co58	1.05E-5	122	3.62E-7
Kr83m	8.40	60	9.92E-7	Ce137	1.91E-6
85m	1.25E+1	Ni65	6.34E-7	138	5.86E-1
85	3.17E+2	Zn65	4.07E-9	Br139	1.82E-2
87	4.31E+1	Rb88	2.02E-1	140	4.17E-5
88	4.21E+1	89	1.10	141	8.46E-4
89	4.33E+2	Sr89	3.48E-5	142	1.40E-3
90	1.07E+3	90	4.01E-7	La140	1.30E-7
Xe131m	6.54E-2	91	3.98E-4	142	6.64E-6
133m	6.48E-1	92	5.01E-4	Ce141	7.36E-8
133	1.58E+1	Y90	2.52E-9	143	3.39E-7
135m	7.43E+1	91m	9.61E-6	144	4.35E-8
135	4.70E+1	91	3.97E-8	Nd147	3.88E-8
137	5.08E+2	92	1.78E-6	W187	6.34E-6
138	2.50E+2	93	1.16E-6	Np239	4.39E-9
139	1.11E+3	Zr95	7.35E-8		
140	1.03E+3	97	5.77E-7		
		Nb95	7.27E-8		
		Mo99	5.67E-7		
		Te99m	3.03E-4		
		101	3.81E-4		

Table 15.7-6

GASEOUS RADWASTE SYSTEM FAILURE SYSTEM RUPTURE
(REALISTIC ANALYSIS)
FISSION PRODUCT RELEASE TO ENVIRONMENT FROM
RUPTURE OF SJAE OUTLET LINE ONLY (Continued)

							1.141E 07
3H	1.128E 00	90RB	4.749E 01	106TC	1.621E-02	143XE	3.099E-04
13N2	3.691E 03	90SR	1.442E-04	106RU	1.579E-06	143CS	1.462E-01
13AM	9.577E 00	90YM	4.208E-09	110AGM	6.876E-05	143BA	3.556E-01
13NO	5.922E-02	90Y	1.361E-06	129SB	1.207E-04	143LA	2.127E-02
14C	1.037E-01	91BR	1.702E-03	129TEM	5.988E-06	143CE	1.830E-04
16N2	8.270E 06	91KR	5.157E 05	129TE	7.822E-04	143PR	5.178E-05
16AM	3.623E 04	91RB	1.576E 02	129I	1.574E-08	144XE	5.810E 00
16NO	8.801E 01	91SR	7.977E-02	131SB	4.986E-03	144CS	1.937E-01
17N2	1.37[illegible]E 03	91YM	1.855E-03	131TEM	2.522E-05	144BA	4.804E-01
17AM	7.679E 00	91Y	2.067E-05	131TE	5.295E-03	144LA	1.787E-01
17NO	3.422E-02	92BR	6.636E-08	131I	1.878E 00	144CE	2.418E-05
18F	4.952E 00	92KR	2.316E 04	131XEM	3.481E 01	147ND	2.155E-05
19O	7.677E 02	92RB	2.364E 02	132TE	2.010E-04	147PM	3.395E-06
24NA	2.294E-03	92SR	2.[illegible]85E-01	132I	2.399E-01	149ND	5.058E-04
32P	2.288E-05	92Y	9.591E-04	133SB	2.915E-02	149PM	2.339E-05
51CR	5.852E-04	93KR	1.809E 03	133TEM	3.983E-03	187W	3.523E-03
54MN	4.566E-05	93RB	4.965E 01	133TE	1.170E-02	239NP	2.440E-01
56MN	5.873E-02	93SR	8.862E-01	133IM	1.476E-01		
58CO	5.822E-03	93Y	6.444E-04	131I	1.378E 01		
59FE	9.343E-05	92ZR	6.726E-11	133XEM	3.430E 02		
60CO	5.511E-04	93NBM	7.577E-12	133XE	8.360E 03		
65NI	3.524E-04	94KR	7.887E-06	134TE	8.721E-03		
65ZN	2.263E-06	94RB	2.165E-01	134IM	1.901E 00		
69ZNM	3.442E-05	94SR	2.759E-01	134I	4.388E 01		
83AS	1.306E-02	94Y	1.670E-02	134XEM	2.155E-01		
83SEM	7.043E-03	95KR	2.787E-02	135I	2.207E 01		
83SE	4.631E-04	95RB	1.176E-03	135XEM	3.965E 04		
83BR	2.583E 00	95SR	2.931E-01	135XE	2.486E 04		
83KRM	4.449E 03	95Y	2.907E-02	135CSM	2.390E-05		
84AS	8.704E-03	95ZR	4.084E-05	135CS	1.922E-09		
84SE	1.056E-02	95NBM	8.958E-07	136TE	5.217E-02		
84BRM	9.329E-02	95NM	4.039E-05	136IM	7.170E 00		
84BR	5.602E 00	97ZR	3.208E-04	136I	1.086E 01		
85AS	2.242E-03	97NBM	1.221E-01	137I	9.789E 00		
85SEM	1.378E-02	97NB	3.986E-03	137XE	2.755E 05		
85SE	1.680E-02	99ZR	9.221E-02	137CS	1.493E-04		
85BR	5.481E 00	99NBM	2.746E-02	137BAM	1.705E-04		
85KRM	6.625E 03	99NB	1.583E-01	138XE	1.332E 05		
85KR	1.680E 01	99MO	3.149E-04	138CSM	1.037E-02		
87AS	8.716E-05	99TCM	1.685E-01	138CS	9.669E-01		
87SE	3.002E-02	99TC	2.382E-08	139XE	6.076E 05		
87BR	7.509E 00	101MO	1.654E-02	139CS	1.597E 01		
87KR	2.287E 04	101TC	2.114E-01	139BA	1.078E-01		
88SE	2.307E-03	102MO	1.744E-02	140XE	5.723E 05		
88BR	6.702E 00	102TCM	1.815E-01	140CS	1.445E 02		
88KR	2.231E 04	102TC	5.549E-05	140BA	8.676E-03		
88RB	2.873E-01	103TC	1.310E-01	140LA	6.328E-05		
89SE	4.997E-07	103RU	2.25E-05	141XE	1.447E 04		
89BR	2.942E 00	103RHM	2.442E-03	141CS	6.001E 01		
89KR	2.356E 05	104MO	2.996E-02	141BA	4.697E-01		
89RB	3.572E 00	104TC	7.260E-02	141LA	1.272E-03		
89SR	2.533E-03	105MO	1.962E-02	141CE	4.086E-05		
89YM	1.297E-08	105TC	4.045E-02	142XE	1.086E 03		
90BR	4.051E-01	105RU	1.996E-04	142CS	3.082E 01		
90KR	5.912E 05	105RHM	6.637E-03	142BA	7.785E-01		
90RBM	3.748E 00	105RH	2.874E-05	142LA	3.687E-03		

Table 15.7-7

GASEOUS RADWASTE SYSTEM FAILURE (REALISTIC ANALYSIS) FIRST CHARCOAL TANK RUPTURE RADIOLOGICAL EFFECTS (mrem)

Site Boundary

	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Halogen	4.4E-2	9.05E-2	3.35E-2	1.03	1.47E-1		4.06E-2
Noble Gas			2.88E+2*				
Other	2.18	6.33	3.66E-2	2.83E-4	2.46E-2	2.33E-2	1.73E-2
Total	2.22	6.42	2.88E+2	1.03	1.71E-1	2.33E-2	5.79E-2

Low Propulsion Zone

	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Halogen	4.6E-3	1.2E-2	4.4E-3	7.7E-1	1.9E-2		9.00E-3
Noble Gas			37.0*				
Other	1.09E-2	3.16E-2	1.8E-2	1.42E-4	1.23E-2	1.16E-2	8.6E-3
Total	1.55E-2	4.36E-2	37.0	7.7E-1	3.13E-2	1.16E-2	1.8E-2

*Decay in flight accounted for.

Table 15.7-8

LIQUID ACTIVITY RADWASTE TANKS (Ci)

Tank	I131	I132	I133	I134	I135
Concentrate Waste	2.6E01	6.2E-02	1.5E00	1.0E-04	7.3E-02
Low Conductivity Oil Separator	3.2E-01	9.1E-01	1.0E00	8.4E-01	1.2E00
High Conductivity Oil Separator	3.5E-01	8.0E-02	2.3E-01	4.0E-02	1.5E-01
Low Conductivity Collector	6.3E00	1.6E00	3.0E00	1.1E-01	1.3E00
High Conductivity Collector	1.0E01	7.4E-02	3.3E00	5.3E-03	4.9E-01
Filtrate	2.1E-01	7.3E-03	9.5E-02	1.6E-03	3.7E-02
Distillate	1.7E-04	1.2E-06	5.4E-05	5.8E-08	7.8E-06
Precoat	1.8E-03	4.1E-04	8.3E-04	1.4E-05	3.2E-04
Detergent Waste	1.1E-06	4.3E-07	4.2E-06	4.2E-08	2.1E-06

Table 15.7-9

LIQUID ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Tank	I131	I132	I133	I134	I135
Concentrate Waste	2.6E00	6.2E-03	1.5E-01	1.0E-05	7.2E-03
Low Conductivity Oil Separator	3.2E-02	9.1E-02	1.0E-01	8.4E-02	1.2E-01
High Conductivity Oil Separator	3.5E-02	8.0E-03	2.3E-02	4.0E-03	1.5E-02
Low Conductivity Collector	6.3E-01	1.6E-01	3.0E-01	1.1E-02	1.3E-01
High Conductivity Collector	1.0E00	7.4E-03	3.3E-01	5.3E-04	4.9E-02
Filtrate	2.1E-02	7.3E-04	9.5E-03	1.6E-04	3.7E-03
Distillate	1.7E-05	1.2E-07	5.4E-06	5.8E-09	7.8E-07
Precoat	1.8E-04	4.1E-08	8.3E-05	1.4E-06	3.2E-05
Detergent Waste	1.1E-07	4.3E-07	4.2E-07	4.2E-09	2.1E-07

Table 15.7-10

RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Site Boundary Exclusion Area	8E-05	2E-00
Low Population Zone	3E-05	4E-01

Table 15.7-11

REALISTIC ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Tank	I131	I132	I133	I134	I135
Concentrate Waste	2.6E-01	6.2E-04	1.5E-02	1.0E-06	7.3E-04

Table 15.7-12

LIQUID RADWASTE TANKS FAILURE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power Level	NA	NA
B. Furnup	NA	NA
C. Fuel Damaged	NA	NA
D. Release of activity by nuclide	NA NA	Table 15.7-17
E. Iodine Fractions		
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	NA
G. Iodine Release fractions	10%	1%
II. Data and assumptions used to estimate activity released		
A. Containment Leak rate (%/day)	NA	NA
B. Secondary containment leak rate (%/day)	NA	NA
C. Valve movement times	NA	NA
D. Absorption and filtration efficiencies	NA	NA
(1) Organic Iodine	NA	NA
(2) Elemented Iodine	NA	NA
(3) Particulate Iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters	NA	NA
(1) Flow rate	NA	NA
(2) Mixing Efficiency	NA	NA
(3) Filter Efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions		
(1) Dilution factor afforded by public waterway	150	NA
(2) Dilution of Liquid Ingestion	NA	NA

Table 15.7-12

LIQUID RADWASTE TANKS FAILURE - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(3) Aquatic Life Consumed	85.92 gm/day	NA
III. Dose Data		
A. Method of dose calculation	Reference 1	Reference 1
B. Dose conversion assumption	Reference 1	Reference 1
C. Peak activity concentrations in containment	NA	NA
D. Doses	Tables 15.7-15 & 15.7-16	Table 15.7-18
IV. Dispersion Data		
A. Site Boundary and LPZ distance (m)		
B. χ/Q (sec/m^3)		
(1) Site Boundary	2.0E-3	2.0E-3
(2) LPZ	41.0E-3	1.0E-3

Table 15.7-13

LIQUID RADWASTE SYSTEM FAILURE (DESIGN BASIS)
AIRBORNE ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Tank	I131	I132	I133	I134	I135
All Tanks	4.3	2.7E-1	9.1E-1	1.0E-1	3.3E-1

Table 15.7-14

LIQUID RADWASTE SYSTEM FAILURE (DESIGN BASIS)
LIQUID ACTIVITY RELEASED TO THE SURFACE WATER

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
Br83	2.5E-04	Sr89	8.4E-02	Zr95	1.2E-03	Na24	4.0E-04
Br84	1.9E-06	Sr90	8.9E-03	Zr97	7.5E-06	P32	4.4E-04
Br85	4.7E-10	Sr91	4.8E-03	Nb95	3.7E-03	Cr51	1.4E-02
I131	2.6E 01	Sr92	2.6E-04	Ru103	5.2E-04	Mn54	1.5E-03
I132	6.2E-02	Y90	8.9E-03	Ru106	8.7E-05	Mn56	9.5E-05
I133	1.5E 00	Y91m	[illegible].3E-03	Rh103m	5.2E-04	Co58	1.7E-01
I134	1.0E-04	Mo99	7.3E-02	Rh106	8.7E-05	Co60	2.0E-02
I135	7.3E-02	To99m	1.7E-03	La140	1.8E-01	Fe59	2.8E-03
		Tc101	2.7E-07	Ce141	1.0E-03	Ni65	5.6E-07
Total	2.8E 01	Te129m	7.9E-03	Ce143	3.1E-05	Zn65	8.4E-05
		Te132	5.9E-02	Ce144	1.2E-03	Zn69m	5.0E-06
		Cs134	6.1E-03	Pr143	6.9E-04	Ag110m	2.3E-03
		Cs136	1.9E-03	Nd147	2.2E-04	W187	1.6E-03
		Cs137	9.4E-03				
		Cs138	3.9E-06	Total	1.9E-01	Total	2.1E-01
		Ba137m	9.4E-03				
		Ba139	5.4E-05				
		Ba140	1.6E-01				
		Ba141	6.8E-07				
		Ba142	1.3E-07				
		Np239	5.9E-01				
		Total	1.0E 00				

Table 15.7-15

LIQUID RADWASTE SYSTEM FAILURE (DESIGN BASIS)
AIRBORNE RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area *	1.8E-3	4.65
Low Population Zone *	8.9E-4	2.33

*Applicant to Supply

Table 15.7-16

LIQUID RADWASTE SYSTEM FAILURE (DESIGN BASIS)
LIQUID DISCHARGE RADIOLOGICAL EFFECTS

	Doses (mr/yr)				
Pathway	Body	Skin	GI-LLI	Thyroid	Bone
Drink	3.4E-05	0	3.0E-05	1.6E-02	6.7E-05
Eat plants	0	0	0	0	0
Eat inverts	0	0	0	0	0
Eat fish	4.6E-05	0	6.4E-05	6.7E-03	1.4E-04
Swim	8.3E-06	1.2E-05	0	0	0
Boat	8.3E-06	0	0	0	0
Sunbathe	1.1E-04	1.4E-04	0	0	0
Fish	3.7E-03	4.2E-03	0	0	0
Total	3.9E-03	4.3E-03	9.4E-05	2.2E-02	2.0E-04

Release (Ci/yr) = 2.91E-01

Concentrate (μci/cc) = 1.91E-09

Table 15.7-17

LIQUID RADWASTE SYSTEM FAILURE (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Tank	I131	I132	I133	I134	I135
Concentrate Waste	2.6E-01	6.2E-04	1.5E-02	1.0E-06	7.3E-04

Table 15.7-18

LIQUID RADWASTE SYSTEM FAILURE (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area *	5.50E-5	2.70E-1
Low Population Zone *	2.74E-5	1.36E-1

*Applicant to Supply

Table 15.7-19

FUEL-HANDLING ACCIDENT PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSIS

	Design Basis Conservative (NRC) Assumptions	Realistic Conservative Engineering Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3651 MWt	3651 MWt
B. Radial peaking factor	1.5	1.0
C. Fuel Damaged	rods 101	rods 101
D. Release of Activity by Nuclide	100% iodine 30% Kr85	Subsection 15.7.4.5.2.1
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
II. Data and assumptions used to estimate activity released		
A. Refueling Building leak rate	300%/2 hr	867.6%/day
B. Absorption and filtration efficiencies		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	99%	99.9%
(4) Particulate fission products	NA	NA
C. All other pertinent data and Assumptions	None	None
III. Dispersion Data		
A. Boundary and LPZ distances (m)	*	*
B. χ/Q's for time intervals of		
(1) 0-2 hr - SB/LPZ	2.0E-3/1.0E-3	2.0E-3/1.0E-3
(2) 2-8 hr - LPZ	3.8E-3	3.8E-3
(3) 8-24 hr - LPZ	1.0E-4	1.0E-4
(4) 1-4 days - LPZ	3.4E-5	3.4E-5
(5) 4-30 days - LPZ	7.5E-6	7.5E-6
IV. Dose Data		
A. Method of dose calculation	Reference 3	Reference 1
B. Dose conversion assumptions	Reference 3	Reference 1
C. Peak Activity concentrations in containment	Table 15.7-19	Table 15.7-23
D. Doses	Table 15.7-22	Table 15.7-25

*Applicant to Supply

Table 15.7-20

FUEL-HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN THE REFUELING BUILDING (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.5E 02	1.2E 02	5.7E 01	1.3E 01	6.2E-01	1.5E-03	3.7E-06	5.4E-14	0	0
I132	3.1E-01	1.3E-01	5.2E-02	8.6E-03	2.3E-04	1.7E-07	1.3E-10	0.	0	0
I133	1.9E 02	8.9E 01	4.1E 01	8.9E 00	4.1E-01	9.0E-04	2.0E-06	2.0E-14	0	0
I134	4.0E-06	1.3E-06	4.2E-07	4.2E-08	4.3E-10	4.5E-14	4.7E-18	0.	0	0
I135	4.7E 01	2.1E 01	9.6E 00	1.9E 00	7.8E-02	1.3E-04	2.1E-07	8.9E-16	0	0
Total I	4.8E 02	2.3E 02	1.1E 02	2.3E 01	1.1E 00	2.6E-03	5.9E-06	7.5E-14	0	0
Kr83m	4.0E-01	1.6E-01	6.3E-02	9.6E-03	2.3E-04	1.3E-07	7.0E-11	0.	0	0
Kr85m	1.8E 02	8.0E 01	3.5E 01	6.7E 00	2.4E-01	3.2E-04	4.3E-07	1.0E-15	0	0
Kr85	5.6E 02	2.7E 02	1.3E 02	2.9E 01	1.4E 00	3.5E-03	8.8E-06	1.3E-13	0	0
Kr87	2.9E-02	1.1E-02	3.9E-03	5.1E-04	8.5E-06	2.3E-09	6.5E-13	0.	0	0
Kr88	5.6E 01	2.4E 01	1.0E 01	1.7E 00	5.3E-02	4.9E-05	4.5E-08	3.4E-17	0	0
Kr89	0.	0.	0.	0.	0.	0.	0.	0.	0	0
Xe131m	1.7E 02	8.3E 01	3.9E 01	8.8E 00	4.3E-01	1.1E-03	2.6E-06	3.9E-14	0	0
Xe133m	5.4E 03	2.6E 03	1.2E 03	2.7E 02	1.3E 01	3.1E-02	7.2E-05	9.4E-13	0	0
Xe133	3.7E 04	1.8E 04	8.5E 03	1.9E 03	9.3E 01	2.3E-01	5.5E-04	7.8E-12	0	0
Xe135m	0.	0.	0.	0.	0.	0.	0.	0.	0	0
Xe135	1.3E 03	5.9E 02	2.7E 02	5.5E 01	2.4E 00	4.3E-03	7.9E-06	4.9E-14	0	0
Xe137	0.	0.	0.	0.	0.	0.	0.	0.	0	0
Xe138	0.	0.	0.	0.	0.	0.	0.	0.	0	0
Total NG	4.5E 04	2.2E 04	1.0E 04	2.3E 03	1.1E 02	2.6E-01	6.4E-04	8.9E-12	0	0

Table 15.7-21

FUEL-HANDLING ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	6.3E-02	1.3E 00	2.0E 00	2.4E 00	2.5E 00	2.5E 00	2.5E 00	2.5E 00	2.5E 00	2.5E 00
I132	7.8E-05	1.6E-03	2.2E-03	2.6E-03	2.6E-03	2.6E-03	2.6E-03	2.6E-03	2.6E-03	2.6E-03
I133	4.7E-02	1.0E 00	1.5E 00	1.8E 00	1.9E 00	1.9E 00	1.9E 00	1.9E 00	1.9E 00	1.9E 00
I134	1.0E-09	1.9E-08	2.4E-08	2.7E-08	2.7E-08	2.7E-08	2.7E-08	2.7E-08	2.7E-08	2.7E-08
I135	1.2E-02	2.5E-01	3.6E-01	4.3E-01	4.5E-01	4.5E-01	4.5E-01	4.5E-01	4.5E-01	4.5E-01
Total I	1.2E-01	2.6E 00	3.8E 00	4.6E 00	4.9E 00	4.9E 00	4.9E 00	4.9E 00	4.9E 00	4.9E 00
Kr83m	1.0E-02	2.0E-01	2.8E-01	3.2E-01	3.3E-01	3.3E-01	3.3E-01	3.3E-01	3.3E-01	3.3E-01
Kr85m	4.5E 00	9.3E 01	1.3E 02	1.6E 02	1.7E 02	1.7E 02	1.7E 02	1.7E 02	1.7E 02	1.7E 02
Kr85	1.4E 01	3.0E 02	4.5E 02	5.5E 02	5.7E 02	5.8E 02	5.8E 02	5.8E 02	5.8E 02	5.8E 02
Kr87	7.5E-04	1.4E-02	1.9E-02	2.2E-02	2.2E-02	2.2E-02	2.2E-02	2.2E-02	2.2E-02	2.2E-02
Kr88	1.4E 00	2.9E 01	4.1E 01	4.8E 01	4.9E 01	4.9E 01	4.9E 01	4.9E 01	4.9E 01	4.9E 01
Kr89	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Xe131m	4.4E 00	9.3E 01	1.4E 02	1.7E 02	1.8E 02	1.8E 02	1.8E 02	1.8E 02	1.8E 02	1.8E 02
Xe133m	1.4E 02	2.9E 03	4.3E 03	5.2E 03	5.5E 03	5.5E 03	5.5E 03	5.5E 03	5.5E 03	5.5E 03
Xe133	9.4E 02	2.0E 04	3.0E 04	3.6E 04	3.8E 04	3.8E 04	3.8E 04	3.8E 04	3.8E 04	3.8E 04
Xe135m	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Xe135	3.2E 01	6.7E 02	9.8E 02	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03
Xe137	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Xe138	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
Total NG	1.1E 03	2.4E 04	3.6E 04	4.4E 04	4.6E 04	4.6E 04	4.6E 04	4.6E 04	4.6E 04	4.6E 04

Table 15.7-22

FUEL-HANDLING ACCIDENT (DESIGN BASIS ANALYSIS) RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusive Area *	1.14	2.90
Low Population Zone *	0.58	1.46

*Applicant to Supply

Table 15.7-23

FUEL-HANDLING ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE REFUELING BUILDING (CURIES)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	5.95E 01	5.87E 01	5.79E 01	5.32E 01	4.24E 01	1.53E 01	2.30E-03
I132	7.93E-03	5.83E-03	4.26E-03	6.47E-04	4.25E-06	6.23E-16	0.
I133	1.45E 01	1.38E 01	1.33E 01	1.02E 01	5.05E 00	2.14E 01	2.85E-13
I134	6.35E-08	2.89E-08	1.29E-08	1.05E-10	2.78E-16	0.	0.
I135	2.04E 00	1.82E 00	1.62E 00	8.09E-01	1.27E-01	3.04E-05	0.
Total	7.60E 01	7.44E 01	7.28E 01	6.42E 01	4.76E 01	1.55E 01	2.30E-03
Kr83m	8.89E-03	4.31E-03	2.07E-03	2.50E-05	1.92E-10	0.	0.
Kr85m	1.01E 01	6.09E 00	3.64E 00	1.64E-01	4.24E-05	0.	0.
Kr85	3.81E 02	2.67E 02	1.86E 02	2.13E 01	6.55E-02	3.12E-13	0.
Kr87	8.40E-04	3.44E-04	1.39E-04	5.94E-07	2.88E-13	0.	0.
Kr88	2.16E 00	1.19E 00	6.45E-01	1.67E-02	9.75E-07	0.	0.
Kr89	0.	0.	0.	0.	0.	0.	0.
Xe131m	4.57E 01	3.20E 01	2.22E 01	2.50E 00	7.41E-03	2.97E-14	0.
Xe133m	5.49E 02	3.80E 02	2.61E 02	2.76E 01	6.90E-02	1.29E-13	0.
Xe133	6.34E 03	4.42E 03	3.06E 03	3.39E 02	9.54E-01	3.07E-12	0.
Xe135m	0.	0.	0.	0.	0.	0.	0.
Xe135	2.57E 02	1.67E 02	1.08E 02	7.83E 00	7.19E-03	1.47E-16	0.
Xe137	0.	0.	0.	0.	0.	0.	0.
Xe138	0.	0.	0.	0.	0.	0.	0.
Total	7.58E 03	5.27E 03	3.64E 03	3.98E 02	1.10E 00	3.54E-12	0.

Table 15.7-24

FUEL-HANDLING ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	3.59E-04	2.14E-02	4.24E-02	1.63E-01	4.38E-01	1.13E 00	1.52E 00
I132	4.79E-08	2.47E-06	4.28E-06	8.44E-06	9.18E-06	9.18E-06	9.18E-06
I133	8.71E-05	5.12E-03	1.00E-02	3.53E-02	7.76E-02	1.17E-01	1.19E-01
I134	3.85E-13	1.60E-11	2.32E-11	2.90E-11	2.90E-11	2.90E-11	2.90E-11
I135	1.23E-05	6.97E-04	1.32E-03	3.85E-03	5.98E-03	6.38E-03	6.38E-03
Total	4.58E-04	2.72E-02	5.38E-02	2.02E-01	5.22E-01	1.26E 00	1.65E 00
Kr83m	5.39E-05	2.30E-03	3.41E-03	4.41E-03	4.42E-03	4.42E-03	4.42E-03
Kr85m	6.13E-02	2.88E 00	4.60E 00	7.04E 00	7.15E 00	7.15E 00	7.15E 00
Kr85	2.30E 00	1.16E 02	1.97E 02	3.62E 02	3.84E 02	3.84E 02	3.84E 02
Kr87	5.10E-06	2.03E-04	2.84E-04	3.39E-04	3.39E-04	3.39E-04	3.39E-04
Kr88	1.31E-02	5.90E-01	9.11E-01	1.28E 00	1.29E 00	1.29E 00	1.29E 00
Kr89	0.	0.	0.	0.	0.	0.	0.
Xe131m	2.76E-01	1.39E 01	2.36E 01	4.32E 01	4.57E 01	4.57E 01	4.57E 01
Xe133m	3.32E 00	1.66E 02	2.81E 02	5.06E 02	5.33E 02	5.33E 02	5.33E 02
Xe133	3.83E 01	1.93E 03	3.27E 03	5.95E 03	6.28E 03	6.28E 03	6.28E 03
Xe135m	0.	0.	0.	0.	0.	0.	0.
Xe135	1.55E 00	7.57E 01	1.25E 02	2.07E 02	2.14E 02	2.14E 02	2.14E 02
Xe137	0.	0.	0.	0.	0.	0.	0.
Xe138	0.	0.	0.	0.	0.	0.	0.
Total	4.58E 01	2.31E 03	3.90E 03	7.08E 03	7.47E 03	7.47E 03	7.47E 03

Table 15.7-25

FUEL-HANDLING ACCIDENT (REALISTIC ANALYSIS) RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area *	9.7E-2	4.6E-2
Low Population Zone *	6.4E-2	6.5E-2

*Applicant to Supply

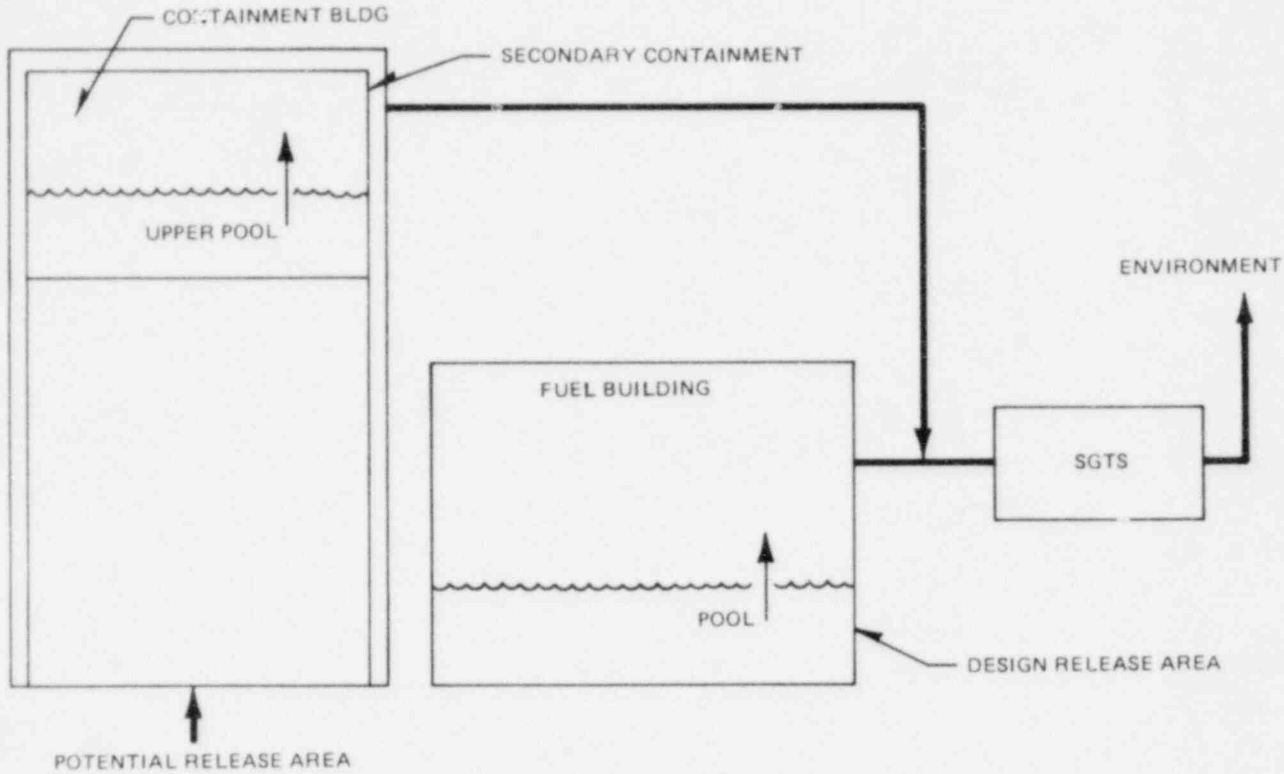

Figure 15.7-1. Leakage Path for Fuel Handling Accident