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15.0 ACCIDENT ANALYSES

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 Regulatory Guide 1.70. The results of the events are summarized in Table 15.0-1. Each event evaluated is assigned to one of the following applicable categories:

a. 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.

b. 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

c. 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.

d. 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 increasing core reactivity and power level.

e. Increase in Reactor Coolant Inventory:

Increasing coolant inventory could result in excessive moisture carryover to the main turbine, feedwater turbines, etc.

f. 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.

g. Radioactive Release from a Subsystem or Component:

Loss of integrity of a radioactive containment component is postulated.

h. Anticipated Transients Without Scram:

In order to determine the capability of plant design to accommodate 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 exists are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of the following frequency groups:

a. 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."

b. 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."

c. Limiting faults - these are occurrences 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."

d. 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):

a. A release of radioactive material to the environs that exceeds the limits of 10 CFR 20.

b. Reactor operation induced fuel cladding failure.

c. Nuclear system stresses in excess of that allowed for the transient classification by applicable industry codes.

d. 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):

a. Release of radioactivity which results in dose consequences that exceed a small fraction of 10 CFR 100.

b. Fuel damage that would preclude resumption of normal operation after a normal restart.

c. Generation of a condition that results in consequential loss of function of the reactor coolant system.

d. 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):

a. Radioactive material release which results in dose consequences that exceed the guideline values of 10 CFR 100.

b. Failure of fuel cladding which would cause changes in core geometry such that core cooling would be inhibited.

c. Nuclear system stresses in excess of those allowed for the accident classification by applicable industry codes.

d. Containment stresses in excess of those allowed for the accident classification by applicable industry codes when containment is required.

e. 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:

a. A step-by-step sequence of events from initiation to final stabilized condition.

b. The extent to which normally operating plant instrumentation and controls are assumed to function.

c. The extent to which plant and reactor protection systems are required to function.

d. The credit taken for the functioning of normally operating plant systems.

e. The operation of engineered safety systems that is required.

f. 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 categories only. Regulatory Guide 1.7 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 criteria 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 SACF aspects. Even under these postulated events, the plant damage allowances or limits were very much the same as those for normal operation.

Regulatory Guide 1.7 suggests that the transient and accident scenarios should now include "and" (multi-failure) event sequences. The information requested by the format for multi-failure events for initiating occurrence, single equipment failure, and/or operator error analysis is an equipment failure or an operator error, another equipment failure or failures and/or another operator error or errors.

This 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 in this section. 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 additional assumptions and conditions (initial event and SACF and/or SOE), 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 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

Initiating event analysis consists of the following:

a. The undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow), or

b. the undesired starting or stopping of any single component, or

c. the malfunction or maloperation of any single control device, or

d. any single electrical component failure, or

e. 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:

a. Those actions that could be performed by one person.

b. Those actions that would have constituted a correct procedure had the initial decision been correct.

c. 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.

Examples of single operator errors are as follows:

a. An increase in power above the established flow control power limits by control rod withdrawal in the specified sequences.

b. The selection and complete withdrawal of a single control rod out of sequence.

c. An incorrect calibration of an average power range monitor.

d. 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

Single active component failure or single operator failure analysis is as follows:

a. The undesired action or maloperation of a single active component, or

b. Any single operator error where operator errors are defined as in Section 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 (Section 4.4.1.4) for incidents of moderate frequency is verified statistically with consideration given to date, calculation, manufacturing, and operating uncertainties. An acceptable criterion was determined to be that 99.9 percent of the fuel rods in the core would not be expected to experience boiling transition[i]. 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. 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 1. The initial condition assumed for all full power transient MCPR calculations is that the bundle is operating at or above the MCPR limit (1.20). Maintaining MCPR greater than the safety MCPR limit is a sufficient, but not necessary, condition to assure that no fuel damage occurs. This is discussed in Section 4.4, "Thermal and Hydraulic Design."

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.

These correlations are substantiated by fuel rod failure tests and are discussed in Section 4.4, "Thermal and Hydraulic Design," and Section 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 15.0-1. 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 percent) corresponding to 105 percent Nuclear Boiler Rated 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 percent rod line A-D), the lower bound is the zero power line H-J, the right bound is the rated valve position line A-H, and the left bound is either the low pump speed, minimum valve position line D-J or 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 cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria. For instance, if the licensed power is 100 percent nuclear boiler rated (NBR), the power/flow map is truncated by the line B- C 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- M, 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 percent 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 M 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.

15.0.3.3.4 Results

The results of analytical evaluations are provided for each event. In addition critical parameters are shown in Table 15.0-2. From the data in Table 15.0-2 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

This section primarily evaluates the performance of the reactor coolant pressure boundary (RCPB) and the containment system during transients and accidents.

During transients that occur with no release of coolant to the containment only RCPB performance is considered. If release to the containment occurs as in the case of limiting faults, then challenges to the containment are evaluated as well.

15.0.3.5 Radiological Consequences

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

a. The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of worst case bounding the event and determining the adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

b. 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.

Doses resulting from the events in Chapter 15 are determined either manually or by computer code. Time dependent releases are evaluated with the Tact 3S computer code[2]. Instantaneous or "puff" type releases are evaluated by methods based on those presented in Regulatory Guide 1.3. Dose conversion factors and breathing rates are presented in Table 15.0-4.

15.0.4 NUCLEAR SAFETY OPERATIONAL ANALYSIS (NSOA) RELATIONSHIP

Appendix 15A is a comprehensive, total plant, system-level, qualitative failure modes and effects analysis, 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 FOR SECTION 15.0

1. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," November 1973 (NEDO-10959 and NEDE-10958).

2. U.S. Nuclear Regulatory Commission Computer Code Tact 3S, Computer Code for Calculating Radiological Consequences of Time Varying Radioactive Releases, Feb. 1975, Accident Analysis Branch, personal communication.

TABLE 15.0-1

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

1. Thermal Power Level, MWt	
Rated Value	3,579
Analysis Value	3,729
2. Steam Flow, lbs per hr	
Warranted Value	15.40×10^6
Analysis Value (nominal)(1)	16.17×10^6
3. Core Flow, lbs per hr	104×10^6
4. Feedwater Flow Rate, lb per sec	
Rated Value	4,269
Analysis Value (nominal)(1)	4,483
5. Feedwater Temperature, °F	425
6. Vessel Dome Pressure, psig	1,045
7. Vessel Core Pressure, psig	1,056
8. Turbine Bypass Capacity, % NBR	35
9. Core Coolant Inlet Enthalpy, Btu per lb	529.9
10. Turbine Inlet Pressure, psig	960
11. Fuel Lattice	8 x 8
12. Core Average Gap Conductance, Btu/sec-ft^2-°F	.1546
13. Core Leakage Flow, %	11
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	
Nominal (EOC-1)	
Analysis Data	0.132

TABLE 15.0-1 (Continued)

17.	Void Coefficient (-)¢/% Rated Voids	
	Analysis Data for Power Increase Events	14.0
	Analysis Data for Power Decrease Events	4.0
18.	Core Average Rated Void Fraction, %	42.91
19.	Scram Reactivity, $ΔK	
	Analysis Data	Figure 15.0-2
20.	Control Rod Drive Speed, Position versus time	Figure 15.0-3
21.	Jet Pump Ratio, M	2.257
22.	Safety/Relief Valve Capacity, % NBR	
	@ 1,210 psig	111.4
	Manufacturer	Dikker
	Quantity Installed	19
23.	Relief Function Delay, seconds	0.4
24.	Relief Function Response Time Constant, seconds	0.1
25.	Safety Function Delay, seconds	0.0
26.	Safety Function Response Time Constant, seconds	0.2
27.	Set Points for Safety/Relief Valves	
	Safety Function, psig	1,175, 1,185, 1,195, 1,205, 1,215
	Relief Function, psig	1,125, 1,135, 1,145, 1,155
28.	Number of Valve Groupings Simulated	
	Safety Function, No.	5
	Relief Function, No.	4
29.	High Flux Trip, % NBR	
	Analysis set point (122 x 1.042)	127.2
30.	High Pressure Scram Set Point, psig	1,095
31.	Vessel Level Trips, Feet Above Bottom of Separator Skirt Bottom	
	Level 8 - (L8), feet	5.89
	Level 4 - (L4), feet	4.04
	Level 3 - (L3), feet	2.165
	Level 2 - (L2), feet	(-) 1.739

TABLE 15.0-1 (Continued)

32. APRM Thermal Trip, %NBR Analysis Set Point $(114 \times 1.04)^{(1)}$	118.8
33. Recirculation Pump Trip Delay, seconds	0.14
34. Recirculation Pump Trip Inertia Time Constant for Analysis, seconds$^{(2)}$	5
35. Total Steamline Volume, ft^3	3850

NOTES:

1. Actual analysis value is within ±.2%.
2. 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 inertial ($lb\text{-}ft^2$).

n = rated pump speed (rps).

g = gravitational constant (ft/sec^2).

T_o = pump shaft torque (lb-ft).

TABLE 15.0-2

RESULTS SUMMARY OF TRANSIENTS EVENTS APPLICABLE TO BWRs

										Duration of Blowdown	
Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR	Frequency Category(1)	No. of Valves First Blowdown	Duration of Blowdown (sec)
15.1		DECREASE IN CORE COOLANT TEMPERATURE									
15.1.1	15.1-1	Loss of Feedwater Heater, Automatic Flow Control	112	1,045	1,087	1,034	106	0.06	a	0	0
15.1.1	15.1-2	Loss of Feedwater Heater, Manual Flow Control	122	1,059	1,101	1,046	114	0.12	a	0	0
15.1.2	15.1-3	Feedwater Control Failure, Max Demand	157	1,164	1,194	1,158	107	0.08	a	19	5
15.1.3	15.1-4	Pressure Regulator Fail - Open	104	1,138	1,162	1,136	100	(2)	a	10	5
15.1.4		Inadvertent Opening of Safety or Relief Valve	See Text						a		
15.1.6		RHR Shutdown Cooling Malfunction Decreasing Temp	See Text						a		
15.2		INCREASE IN REACTOR PRESSURE									
15.2.1	15.2-1	Pressure Regulation Downscale Failure	158	1,185	1,220	1,180	104	0.07	a	19	7
15.2.2	15.2-2	Generator Load Rejection, Bypass-On	139	1,158	1,187	1,153	100	(2)	a	19	5
15.2.2	15.2-3	Generator Load Rejection, Bypass-Off	259	1,200	1,231	1,194	106	0.10	b	19	8
15.2.3	15.2-4	Turbine Trip, Bypass On	120	1,156	1,184	1,152	100	(2)	a	19	5
15.2.3	15.2-5	Turbine Trip, Bypass Off	212	1,197	1,227	1,192	103	0.06	b	19	7
15.2.4	15.2-6	All MSIV Closure	104	1,172	1,206	1,170	100	(2)	a	19	5
15.2.5	15.2-7	Loss of Condenser Vacuum	120	1,154	1,182	1,150	100	(2)	a	19	5

TABLE 15.0-2 (Continued)

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR	Frequency Category(1)	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.2.6	15.2-8	Loss of Auxiliary Power Transformer	104	1,175	1,190	1,173	100	(2)	a	19	5
15.2.6	15.2-9	Loss of All Grid Connections	111	1,159	1,182	1,153	100	(2)	a	19	17
15.2.7	15.2-10	Loss of All Feedwater Flow	104	1,045	1,087	1,033	100	(2)	a	0	0
15.2.8		Feedwater Piping Break	See Table 15.0-1a, event 15.6.6								
15.2.9		Failure of RHR Shutdown Cooling	See Text								
15.3		DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE									
15.3.1	15.3-1	Trip of One Recirculation Pump Motor	104	1,046	1,087	1,035	100	(2)	a	0	0
15.3.1	15.3-2	Trip of Both Recirculation Pump Motors	104	1,138	1,153	1,136	100	(2)	a	10	5
15.3.2	15.3-3	Fast Closure of One Main Recirc. Valve	104	1,049	1,087	1,037	100	(2)	a	0	0
15.3.2	15.3-4	Fast Closure of Two Main Recirc. Valves	104	1,139	1,151	1,137	100	(2)	a	10	5
15.3.3	15.3-5	Seizure of One Recirculation Pump	104	1,139	1,153	1,137	100	(2)	c	10	5
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					(2)	a		
15.4.3		Control Rod Misoperation	See 15.4.1 and 15.4.2								

TABLE 15.0-2 (Continued)

Section No.	Figure No.	Description	Maximum Neutron Flux (% NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam Line Pressure (psig)	Maximum Core Average Surface Heat Flux (% of Initial)	ΔCPR	Frequency Category(1)	Duration of Blowdown: No. of Valves First Blowdown	Duration of Blowdown: Duration of Blowdown (sec)
15.4.4	15.4-1	Abnormal Startup of Idle Recirculation Loop	100	988	1,001	983	149	(3)	a	0	0
15.4.5	15.4-2	Fast Opening of One Main Recirc Valve	215	973	993	970	131	(3)	a	0	0
15.4.5	15.4-3	Fast Opening of Both Main Recirc Valves	149	972	990	969	123	(3)	a	0	0
15.4.7		Misplaced Bundle Accident	See Text								
15.5		INCREASE IN REACTOR COOLANT INVENTORY									
15.5.1	15.5-1	Inadvertent HPCS Pump Start	104	1,045	1,087	1,033	100	(2)	a	0	0
15.5.3		BWR Transients	See appropriate events in 15.1 and 15.2								

NOTES:

1. a = moderate, b = infrequent, c = limiting fault
2. no significant change
3. not start from full power

TABLE 15.0-3

SUMMARY OF ACCIDENTS

Section	Title	Failed Fuel Pins: GE Calculated Value	Failed Fuel Pins: 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 Outside Containment	101	101
15.7.5	Cask Drop Accident	None	None
15.7.6	Fuel Handling Accident Inside Containment	<124	124
15.8	ATWS	SPECIAL EVENT STILL UNDER NEGOTIATION	

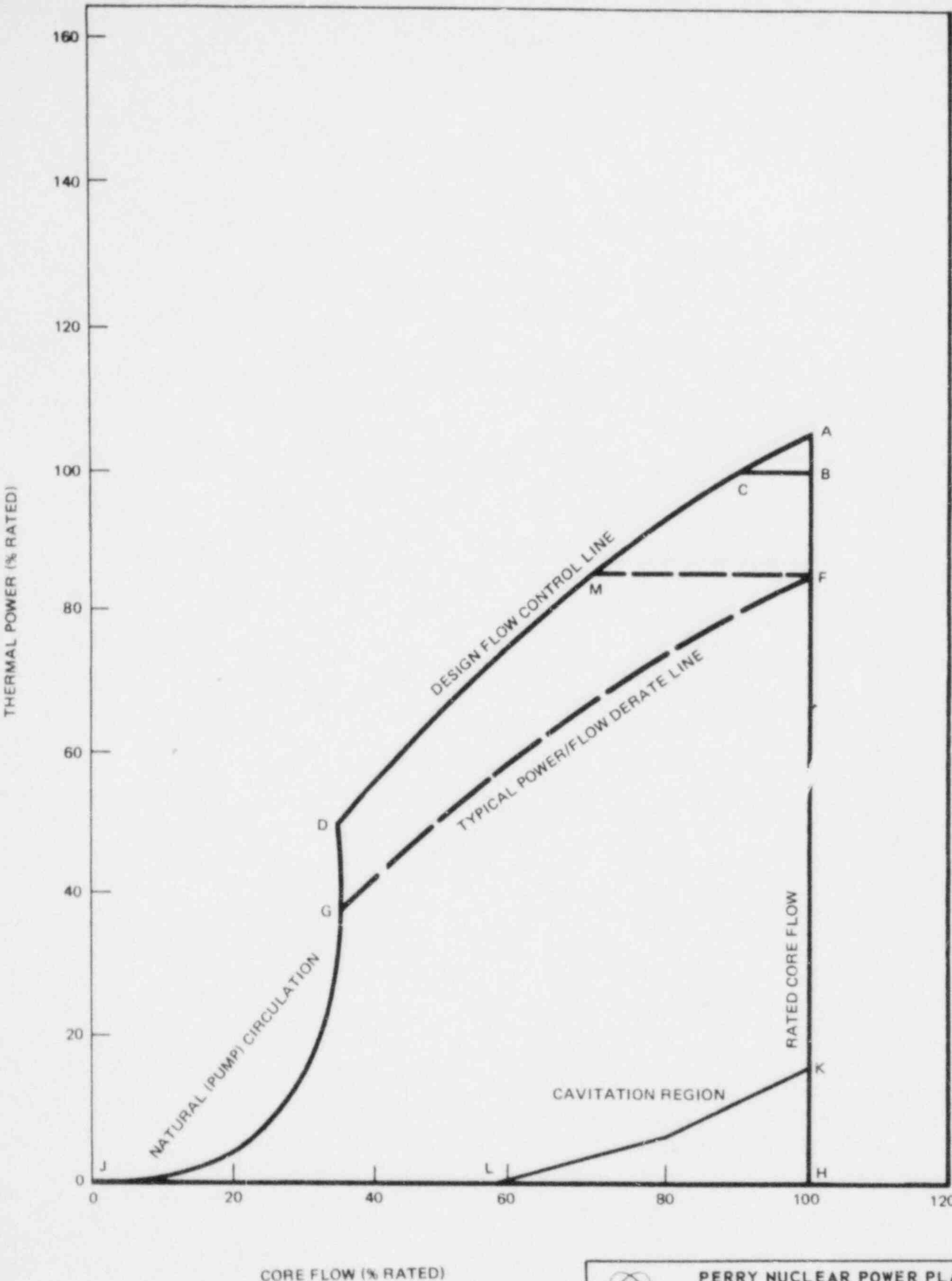
TABLF 15.0-4

DOSE CONVERSION FACTORS

Isotope	Thyroid (Rem/Ci)	Whole Body 0.25xMeV/dis
I-131	1.49+6	8.72-2
I-132	5.35+4	5.13-1
I-133	3.97+5	1.55-1
I-134	2.54+4	5.32-1
I-135	1.24+5	4.21-1
Kr-83m		5.02-6
Kr-85		3.72-2
Kr-85m		5.25-4
Kr-87		1.87-1
Kr-88		4.64-1
Kr-89		5.25-1
Xe-131m		2.92-3
Xe-133m		8.00-3
Xe-133		9.33-3
Xe-135m		9.92-2
Xe-135		5.72-2
Xe-137		4.53-2
Xe-138		2.81-1

Breathing Rates

Time Period (hr)	Breathing Rate (m^3/sec)
0-8	3.47×10^{-4}
8-24	1.75×10^{-4}
24-720	2.32×10^{-4}



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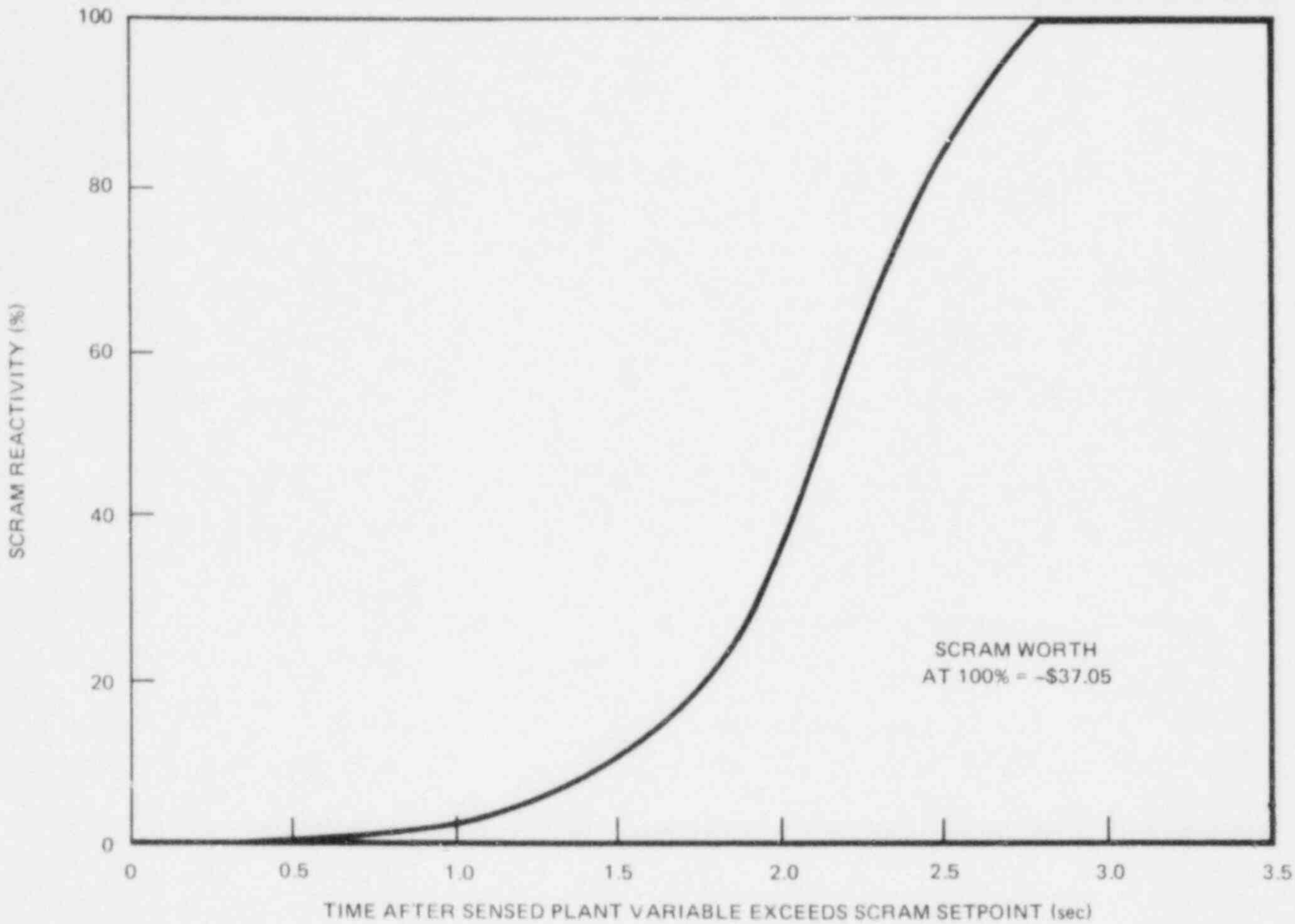
Typical Power/Flow Map

Figure 15.0-1



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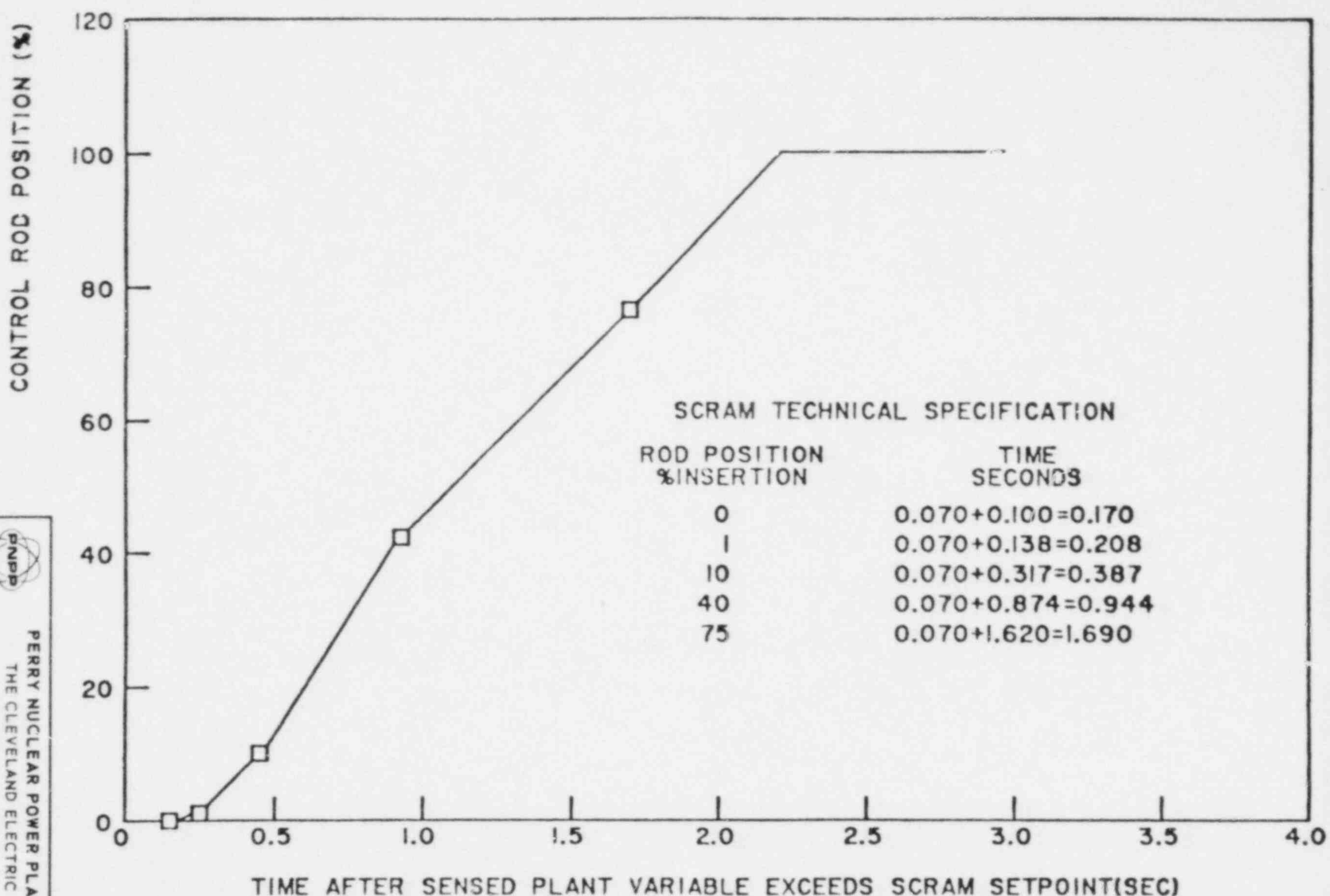
Scram Reactivity Characteristics

Figure 15.0-2



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Scram Time Characteristics

Figure 15.0-3

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:

a. Steam extraction line to heater is closed,

b. Feedwater is bypassed around heater.

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

This event is analyzed under worst case conditions of a 100°F loss and full power. The probability of occurrence of this event is, therefore, regarded as small.

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. Operating procedures describe T-G operation 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 assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The thermal power monitor (TPM) 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 TPM mentioned in Section 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 <u>Core and System Performance</u>

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:

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

b. 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.

c. Four primary system pressure nodes are simulated. The nodes represent the core exit pressure, vessel dome pressure, steam line pressure (at a point representative of the safety/relief valve location) and turbine inlet pressure.

d. 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 steadystate 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.

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

f. 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 15.0-1.

The plant is assumed to be operating at 105 percent of NB rated 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 time volume between the heaters and the spargers are adjusted to reduce the time delays since they are not critical to the calculation of 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 percent NBR (106 percent of initial power), below the flow-referenced APRM thermal power scram setting and core flow is reduced to approximately 80 percent 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 manual mode, no compensation is provided by core flow and thus the power increase is greater than in the automatic mode. A scram on high APRM thermal power may occur. Vessel steam flow increases and the initial system pressure increase is slightly larger. Peak heat flux is 114 percent of its initial value and the average fuel temperature increases 120°F. The increased core inlet subcooling aids core thermal margins and minimum MCPR is 1.08. Therefore, the design basis is satisfied. The transient responses of the key plant variables for this mode of operation are shown in Figure 15.1-2.

If the reactor scrams, water level drops to the low level trip point (L2). This initiates RPT as shown in Table 15.1-2.

This transient is less severe from lower initial power levels for two main reasons: lower initial power levels will have initial values greater than the limiting initial value assumed, and 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:

a. Observe that high feedwater pump trip has terminated the failure event.

b. Switch the feedwater controller from auto to manual control in order to try to regain a correct output signal.

c. 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 system and the high pressure core spray 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 computer model described in Section 15.1.1.3.1 was used to simulate this event.

15.1.2.3.2 Input Parameters and Initial Conditions

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

The void reactivity coefficient used for power increase events is applied since a more negative value conservatively increases the apparent severity of the power increase. End of equilibrium cycle (all rods out) scram characteristics are assumed. The safety-relief valve action is conservatively assumed to occur with higher than nominal set points. The transient is simulated by programming an upper limit failure in the feedwater system such that 130 percent NBR feedwater flow occurs at a system design pressure of 1,065 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 seconds. Scram occurs simultaneously, and limits the neutron flux peak and fuel thermal transient so that no fuel damage occurs. MCPR remains above safety limit and peak fuel center temperature increases less than 77°F. The turbine bypass system opens to limit peak pressure in the steam line near the safety/relief valves to 1,158 psig and the pressure at the bottom of the vessel to about 1,194 psig.

The level will gradually drop to the low level reference point (Level 2), activating the 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 Section 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Section 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 to approximately 130 percent NB rated.

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

15.1.3.1.2 Frequency Classification

This transient disturbance 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 is the sequence of operator actions expected during the course of the event. Once isolation occurs the pressure will increase to a point where the relief valves open. The operator should:

a. Monitor that all rods are in.

b. Monitor reactor water level and pressure.

c. Observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries.

d. Observe that the reactor pressure relief valves open at their set point.

e. Observe that RCIC and HPCS initiate on low-water level.

f. Secure both HPCS and RCIC when reactor pressure and level are under control and it is verified that the initiation is not due to a LOCA.

g. Monitor reactor water level and continue cooldown per the normal procedure.

h. 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 set point. Normal startup and actuation can take up to 30 seconds before effects are realized. If these events occur, they will

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

15.1.3.2.3 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 limit switches on the main steam line 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 for a detailed discussion of this subject.

15.1.3.3 Core and System Performance

15.1.3.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Section 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 output to a high value, which causes the turbine admission valves and the turbine bypass valves to open. 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 percent steam flow was simulated as a worst case since 115 percent is the normal maximum flow limit.

A 5-second isolation valve closure instead of a 3-second closure is assumed when the turbine pressure decreases below the turbine inlet low pressure set point for main steam line isolation initiation. This is within the specification limits of the valve and represents a conservative assumption.

Reactor scram is initiated when the isolation valves reach the 10 percent closed position. This is the maximum travel from the full open position allowed by specification.

This analysis has been performed, unless otherwise noted, with the plant conditions listed in Table 15.0-1.

15.1.3.3.3 Results

Figure 15.1-4 shows graphically how the isolation valve closure stops vessel depressurization and produces a normal shutdown of the isolated reactor.

The main steam line isolation valves automatically close at approximately 28 seconds when pressure at the turbine decreases below 825 psig. Depressurization results in formation of voids in the reactor coolant and causes a rapid decrease in reactor power almost immediately. Reactor vessel isolation limits the duration and severity of the depressurization so that no significant thermal stresses are imposed on the reactor coolant pressure boundary. After the rapid portion of the transient is complete and the isolation effective, the nuclear system safety/relief valves operate intermittently to relieve the pressure rise that results from decay heat generation. No significant reductions in fuel thermal margins occur. Because the rapid portion of the transient results in only momentary depressurization of the nuclear system and because the safety/relief valves need operate only to relieve the pressure increase caused by decay heat, the reactor coolant pressure boundary is not threatened by high internal pressure.

15.1.3.3.4 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 it 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 analysis set to fail at 130 percent we would expect something less than 23 percent 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 set point.

Depressurization rate has a proportional effect upon the voiding action of the core. If it is large enough, the sensed vessel water level trip set point (L8) may be reached initiating scram and turbine and feedwater pump trip early in the transient. Reactor scram will shut down the reactor. Since main turbine is tripped, the depressurization will be terminated.

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 1,162 psig which is below the ASME code limit of 1,375 psig for the reactor coolant pressure boundary. Minimum vessel dome pressure of 803 psig occurs at about 30 seconds.

15.1.3.5 Radiological Consequences

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

event is much less than those consequences identified in Section 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Section 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 analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

15.1.4.1.2 Frequency Classification

This transient disturbance is categorized as an infrequent incident 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 Chapter 15. 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 Section 15.1.1.3.1 was previously used to simulate this event in earlier FSARs. This model is discussed in detail in Reference 1. 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 percent of rated steamflow conditions when a safety/relief valve is inadvertently opened. Manual recirculation flow control is assumed. Flow through the valve at normal plant operating conditions stated above is approximately 7 percent of rated steam flow.

15.1.4.3.3 Qualitative Results

The opening of a safety/relief 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 consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. 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 of the containment is chosen the release will be in accordance in 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 RHR system shutdown cooling.

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 FOR SECTION 15.1

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

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.
37	APRM initiates reactor scram on high thermal power.
53	Wide Range (WR) sensed water level reaches Level 2 (L2) set point.
53	Recirculation pump trip initiated due to Level 2 trip. (Not included in simulaton).
83 (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 a simulated failure of 130% upper limitation feedwater flow.
12.1	L8 vessel level set point initiates reactor scram and trips main turbine and feedwater pumps.
12.2	Recirculation pump trip (RPT) actuated by stop valve position switches.
12.2	Main turbine bypass valves opened due to turbine trip.
13.6	Safety/relief valves open due to high pressure.
18.6	Safety/relief valves close.
37.0	Water level dropped to low water level setpoint (L2).
67.0 (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 maximum limit on steam flow to main turbine.
1.78	Vessel water level (L8) trip initiates reactor scram and main turbine and feedwater turbine trips.
1.78	Turbine trip initiates bypass operation to full flow.
1.79	Main turbine stop valve reaches 90% open position and initiates recirculation pump trip (RPT).
1.88	Turbine stop valves closed. Turbine bypass valves opening to full flow.
2.1	Recirculation pump motor circuit breakers open causing decrease in core flow to natural circulation.
4.7	Group 1 pressure relief valves actuated.
9.7	Group 1 pressure relief valves close.
21.3	Vessel water level reaches L2 setpoint.
27.8	Main steam line isolation on low turbine inlet pressure (825).
32.8	MSIV closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
51.3 (est)	RCIC and HPCS systems flow enters vessel (not simulated).
100 (est)	Group 1 pressure relief valves actuated.

TABLE 15.1-5

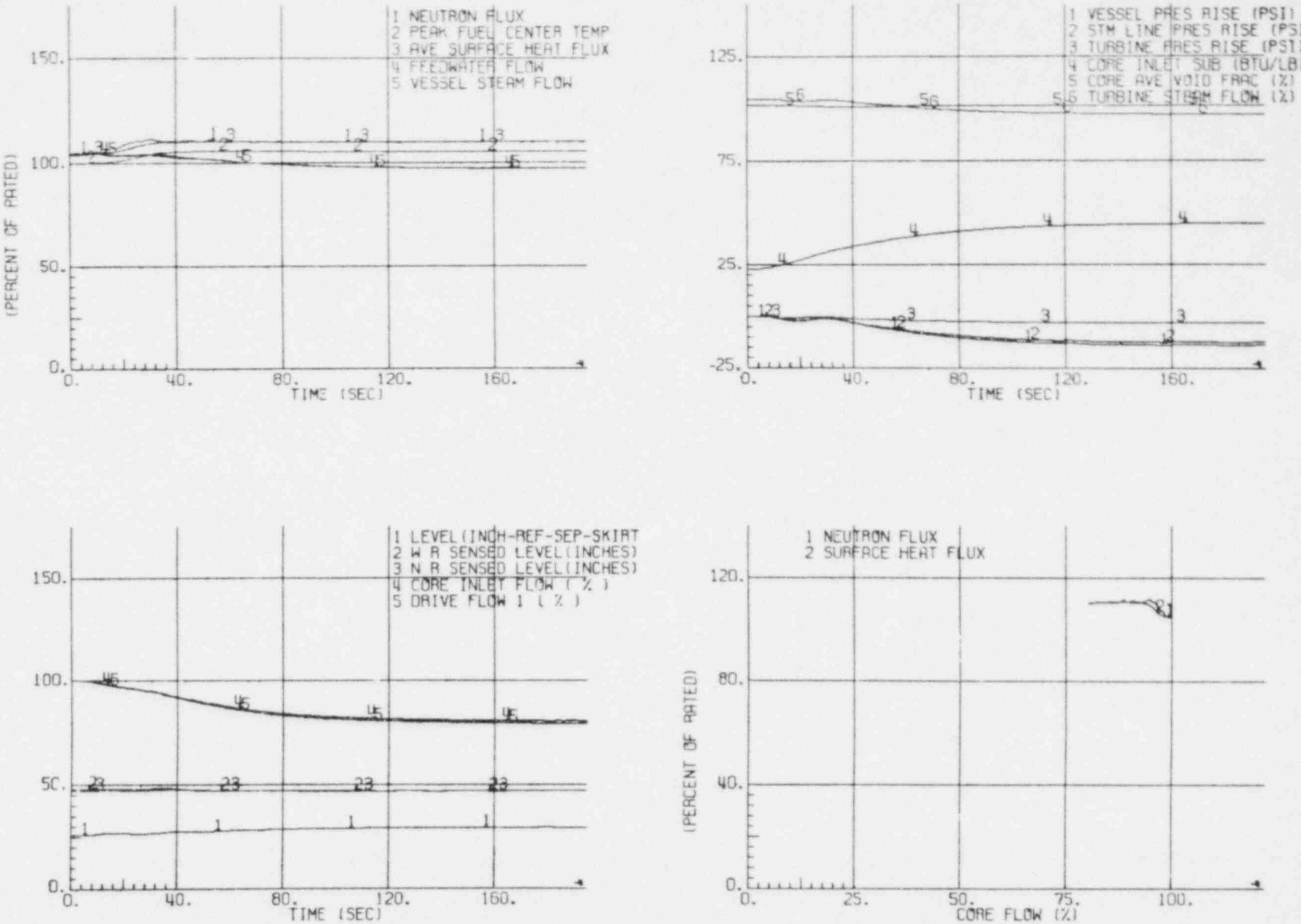
SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

Time-sec	Event
0	Initiate opening of 1 safety/relief 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.



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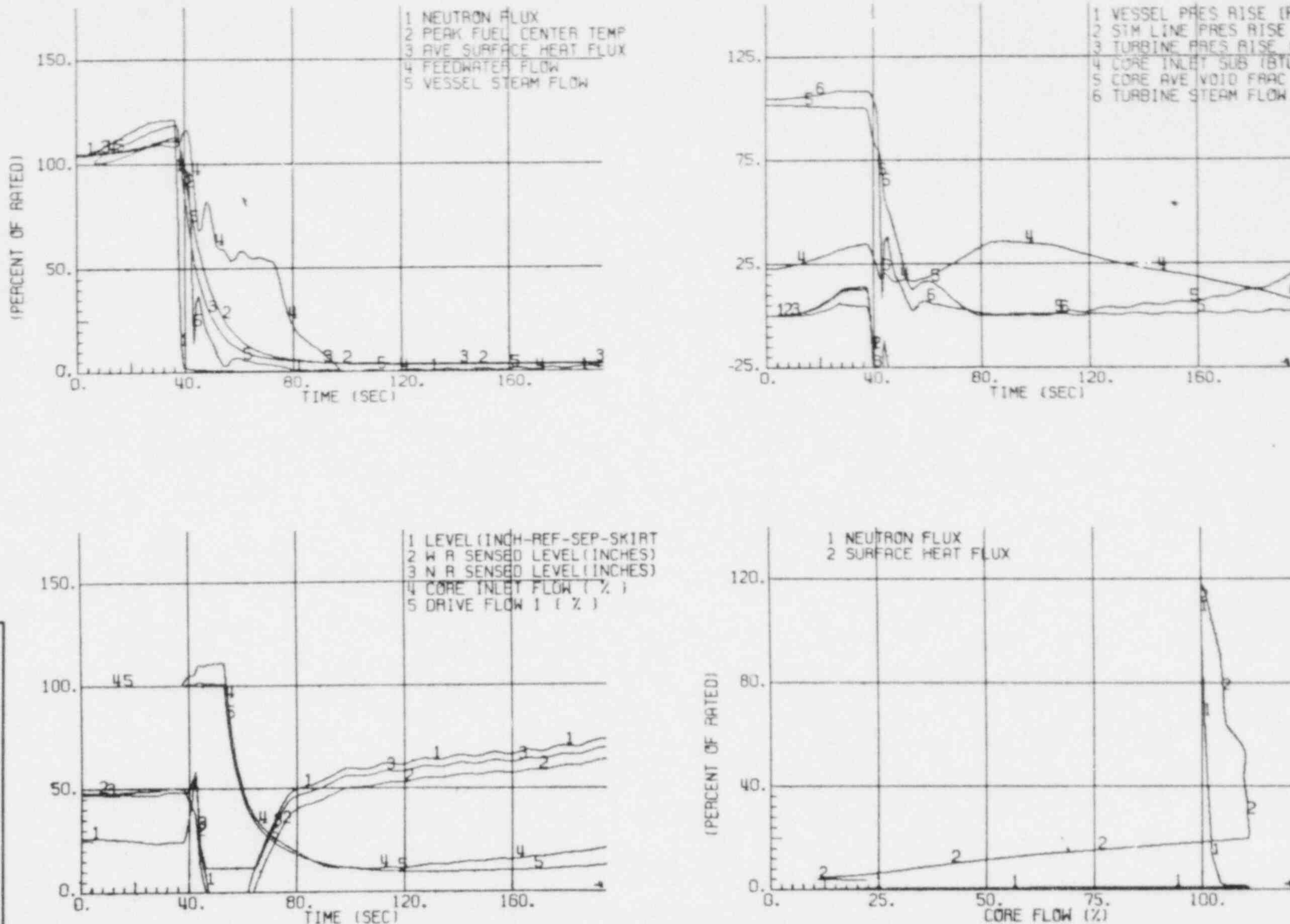
Loss of 100°F Feedwater Heater, Automatic Flow Control

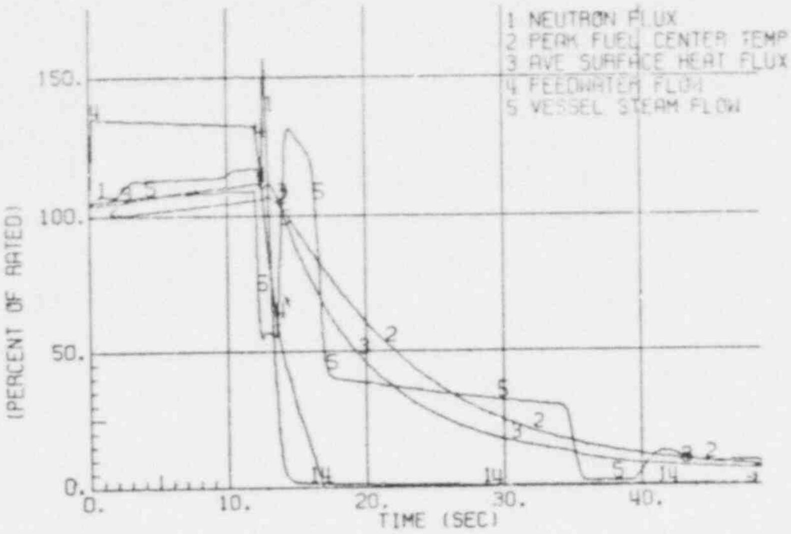
Figure 15.1-1

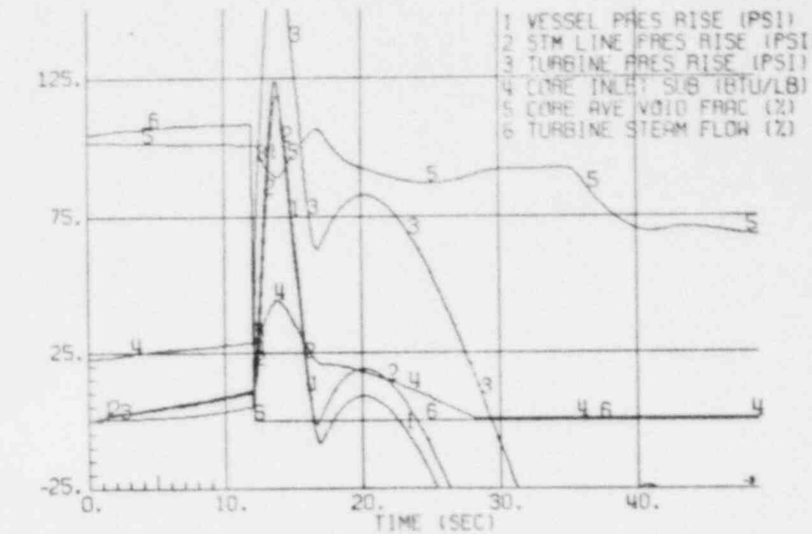


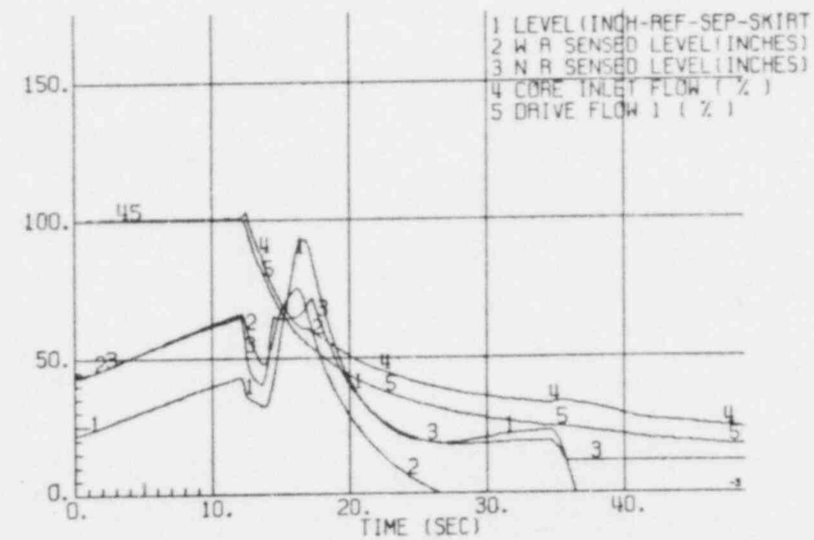
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

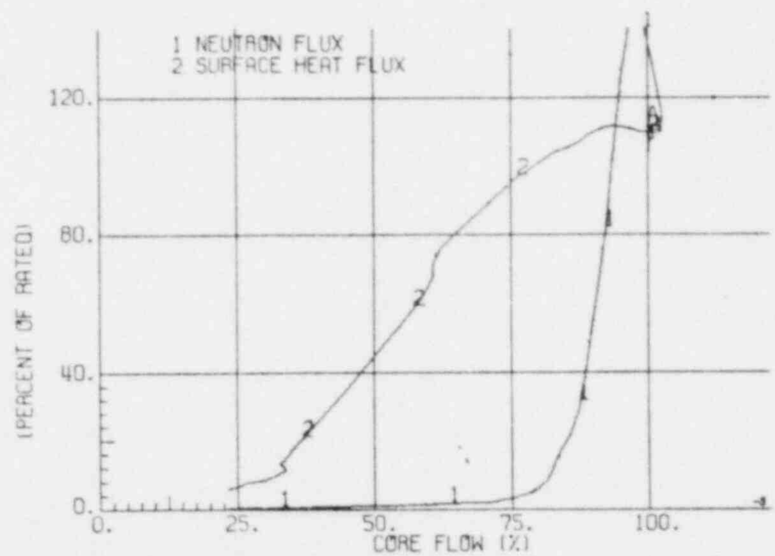
Loss of 100°F Feedwater Heater,
Manual Flow Control

Figure 15.1-2





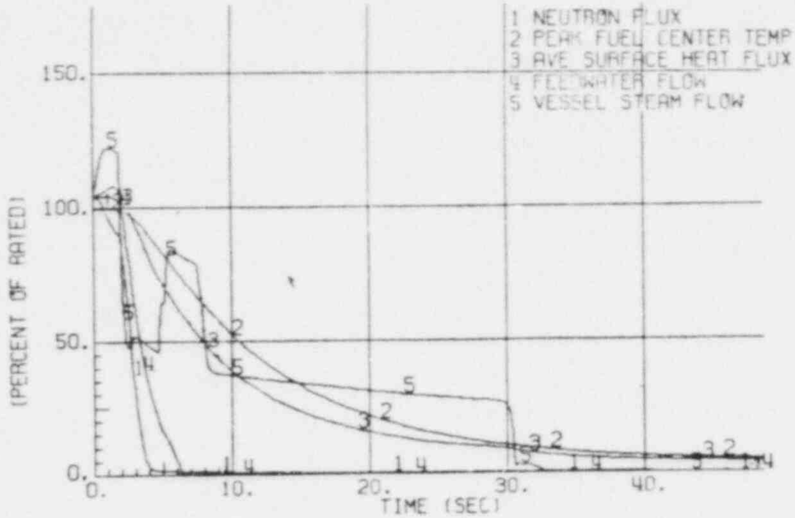


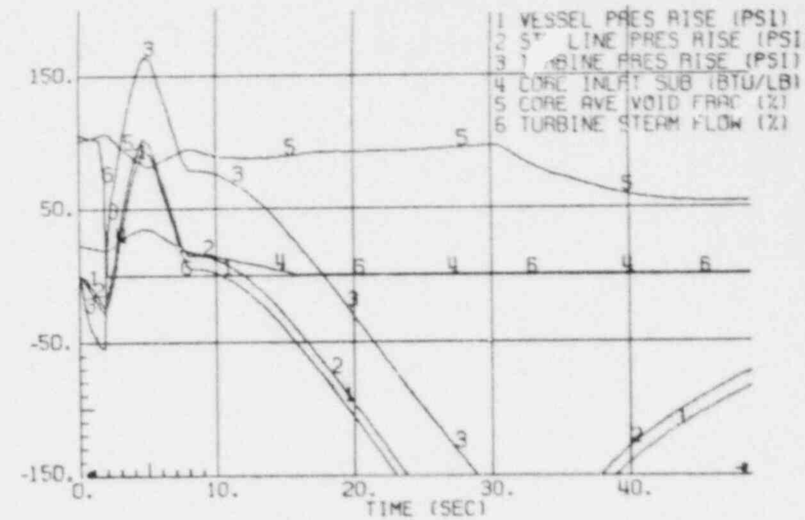


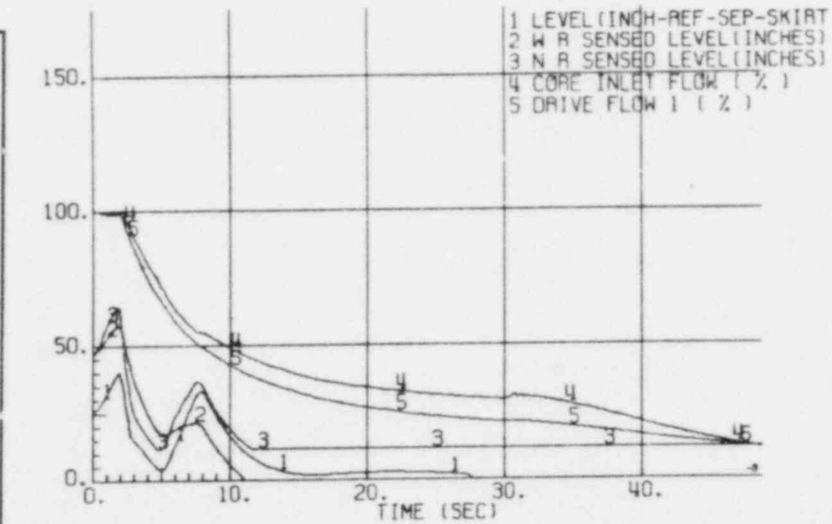
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

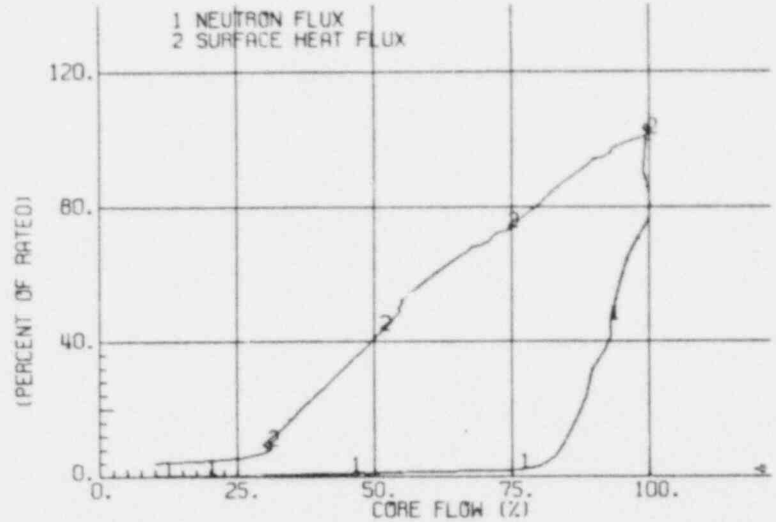
Feedwater Controller Failure,
Maximum Demand with High
Level Turbine Trip

Figure 15.1-3









PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Pressure Regulator Failure -
Open to 130%

Figure 15.1-4

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 set points to create proportional error signals that produce each regulator output. The output of both regulators feeds in a high value gate. The regulator with the highest output controls the main turbine control valves. The lowest pressure set point 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, reactoı scram will be initiated when high neutron flux scram set point 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 Section 15.2.1.1.1 will cause the turbine control valves to close momentarily. 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 set point 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 Section 15.2.1.2.3.2.

15.2.1.2.1.3.2 Pressure Regulation Downscale Failure

The operator should:

a. Monitor that all rods are in.

b. Monitor reactor water level and pressure.

c. Observe turbine coastdown and break vacuum before the loss of steam seals. Check turbine auxiliaries.

d. Observe that the reactor pressure relief valves open at their set point.

e. Monitor reactor water level and continue cooldown per the normal procedure.

f. 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 controls 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 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 control valves will start to close causing reactor pressure to increase, a flux scram trip would be initiated to shut down the reactor. This event is similar to that described in Section 15.2.1.2.1.1. Detailed discussions on this subject can be found 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. Detailed discussions on this subject can be found 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 Section 15.1.1.3.1 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 tabulated in Table 15.0-2.

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, less than 2 seconds or so, 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 set points.

15.2.1.3.3.2 Pressure Regulation Downscale Failure

A pressure regulation downscale failure is simulated at 105 percent NB rated 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 set point, a reactor scram is initiated. The neutron flux increase is limited to 158 percent NB rated by the reactor scram. Peak fuel surface heat flux does not exceed 104 percent 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 set points, 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; therefore, these barriers maintain their integrity and function as designed.

15.2.1.4.2 Pressure Regulation Downscale Failure

Peak pressure at the safety/relief valves reaches 1,180 psig. The peak nuclear system pressure reaches 1,220 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 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 Section 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.

15.2.2 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 initiates a scram trip signal and will cause a sudden reduction in steam flow which results in an increase in system pressure.

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
Mean time between events (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 the bypass yields a frequency of a generator load rejection with bypass failure of 0.0036 event/plant year.

15.2.2.2 <u>Sequence of Events and System Operation</u>

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:

a. Verify proper bypass valve performance.

b. Observe that the feedwater/level controls have maintained the reactor water level at a satisfactory value.

c. Observe that the pressure regulator is controlling reactor pressure at the desired value.

d. Record peak power and pressure.

e. 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 percent NB rated. In addition, recirculation pump trip (RPT) is initiated. Both of these trip signals satisfy 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 set points is assumed to function normally during the time period analyzed.

15.2.2.2.2.2 Generator Load Rejection with Failure of Bypass

The sequence of events for this failure is the same as in Section 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. Turbine control valve 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 Section 15.1.1.3.1 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 15.0-1.

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

The closure characteristics of the turbine control valves 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 seconds.

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, including 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 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 percent NB rated power. Peak neutron flux rises 35 percent above initial 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 259 percent of rated, average surface heat flux reaches 106 percent 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. MCPR stays above 1.10 for this event.

15.2.2.3.4 Consideration of Uncertainties

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

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 worth characteristics). Expected 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 safety/relief valves reaches 1,194 psig. The peak nuclear system pressure reaches 1,231 psig at the bottom of the vessel, well below the nuclear barrier transient pressure limit of 1,375 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 Section 15.2.4.5. Therefore, the radiological exposures noted in Section 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 high level and first stage reheater drain tank high levels, and feedwater heater high levels, high vibrations, operator lock out, 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

Frequency Basis: As discussed in Section 15.2.2.1.2.2, 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:

a. Verify auto transfer of buses supplied by generator to incoming power if automatic transfer does not occur, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Check turbine for proper operation of all auxiliaries during coastdown.

d. Depending on conditions, initiate normal operating procedures for cool-down, or maintain pressure for restart purposes.

e. Put the mode switch in the startup position before the reactor pressure decays to <850 psig.

f. Secure the RCIC operation if auto initiation occurred due to low water level.

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

h. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

i. 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.

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 set points is assumed to function normally during the time period analyzed.

It should be noted that below 40 percent NB rated 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 elimi ~e 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 functional as before and credit is taken for those protection system trips.

15.2.3.2.2.2 Turbine Trip with Failure of the Bypass

This sequence of events is the same as in Section 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.3 The Effect of Single Failures and Operator Errors

15.2.3.2.3.1 Turbine Trips at Power Levels Greater Than 40 Percent 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 single failure criterion.

15.2.3.2.3.2 Turbine Trips at Power Levels Less Than 40 Percent NBR

This sequence is the same as in Section 15.2.3.2.3.1 except RPT and stop valve 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 Section 15.1.1.3.1 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 15.0-1.

Turbine stop valves full stroke closure time is 0.1 second.

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

Reduction in core recirculation flow is initiated by position switches on the main stop valves, which actuate trip circuitry which trips the recirculation pumps. This recirculation pump trip signal is automatically bypassed when the reactor is below 40 percent NB rated power level.

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 percent NB rated 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 120 percent 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 percent NB rated steam flow conditions in Figure 15.2-5.

Peak neutron flux reaches 212 percent of its rated value, and peak fuel center temperature increases approximately 102°F. 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 incidents of moderate frequency. However, the MCPR for this transient is 1.14 which is just above the safety limit for incidents of moderate frequency and, therefore, the design basis is satisfied.

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 percent of rated power, the turbine stop valve closure and turbine control valve closure scrams 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 set points 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 set points and will be significantly below the RCPB transient limit of 1,375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit.

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:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rods-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve's nominal set point.

15.2.3.4 <u>Barrier Performance</u>

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1,184 psig, which is below the ASME code limit of 1,375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1,156 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 safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1,227 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1,375 psig. Peak dome pressure does not exceed 1,197 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in Section 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 Section 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 CLOSURE

15.2.4.1 Identification of Causes and Frequency Classification

15.2.4.1.1 Identification of Causes

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

15.2.4.1.2 Frequency Classification

15.2.4.1.2.1 Closure of All Main Steam Line 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 steam line isolation valve closure, position switches on the valves provide a reactor scram if the valves in three or more main steam lines are less than 90 percent 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 Steam Line 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 done manually. Operator error or equipment malfunction may cause a single MSLIV to be closed inadvertently. If reactor power is greater than about 80 percent when this occurs, a high flux scram or high steam lire flow isolation 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:

a. Observe that all rods have inserted.

b. Observe that the relief valves have opened for reactor pressure control.

c. Check that RCIC/HPCS auto starts on the impending low reactor water level condition.

d. Switch the feedwater controller to the manual position.

e. Initiate operation of the RHR system in the steam condensing mode only.

f. When the reactor vessel level has recovered to a satisfactory level, secure RCIC/HPCS.

g. When the reactor pressure has decayed sufficiently for RHR operation, put it into service per procedure.

h. Before resetting the MSLIV isolation, determine the cause of valve closure.

i. Observe turbine coastdown and break vacuum before the loss of sealing steam. Check T-G auxiliaries for proper operation.

j. Do not reset and open MSLIVs unless conditions warrant and be sure the pressure regulator set point is above vessel pressure.

k. 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 Steam Line Isolation Valves

MSLIV closures initiate a reactor scram trip via position signals to the pro· ion system. Credit is taken for successful operation of the protection syst

The pressure relief system which initiates opening of the relief valves when system pressure exceeds relief valve instrumentation set points 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 Steam Line 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 Failures 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 criterion 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 1,375 psig. The design basis and performance of the pressure relief system is discussed in Section 5.0.

15.2.4.3 Core and System Performance

15.2.4.3.1 Mathematical Model

The computer model described in Section 15.1.1.3.1 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 15.0-1.

The main steam isolation valves close in 3 to 5 seconds. The worst case, the 3 second closure time, is assumed in this analysis.

Position switches on the valves initiate a reactor scram when the valves are less than 90 percent 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 Steam Line Isolation Valves

Figure 15.2-6 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 105 percent of NB rated steam flow. Peak neutron flux and fuel surface heat flux show no increase.

Water level decreases sufficiently to cause a recirculation system trip and initiation of the HPCS and RCIC system at approximately 17.1 seconds. However, there is a delay up to 30 seconds before the water supply enters the vessel. Nevertheless, there is no change in the thermal margins.

15.2.4.3.3.2 Closure of One Main Steam Line Isolation Valve

Only one isolation valve is permitted to be closed at a time for testing purposes to prevent scram. Normal test procedure requires an initial power reduction to approximately 75 to 80 percent of design conditions in order to

avoid high flux scram, high pressure scram, or full isolation from high steam flow in the "live" lines. With a 3 second closure of one main steam isolation valve during 105 percent rated power conditions, the steam flow disturbance raises vessel pressure and reacto power enough to initiate a high neutron flux scram. This transient is considerably milder than closure of all MSIV's 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 set points.

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 Section 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 examples:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve's nominal set point.

15.2.4.4 Barrier Performance

15.2.4.4.1 Closure of All Main Steam Line Isolation Valves

The nuclear system relief valves begin to open at approximately 3 seconds after the start of isolation. The valves close sequentially as the stored heat is dissipated but continue to discharge the decay heat intermittently. Peak pressure at the vessel bottom reaches 1,206 psig, clearly below the pressure limits of the reactor coolant pressure boundary. Peak pressure in the main steam line is 1,170 psig.

15.2.4.4.2 Closure of One Main Steam Line 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 or pressure scram will result. The main turbine bypass system will continue to regulate system pressure via the other three "live" steam lines.

15.2.4.5 Radiological Consequences

15.2.4.5.1 General Observations

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

The above observations (a) through (d) lead to the realization that radiological aspects can involve a broad spectrum of results. For example:

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

b. 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 2 is described below. However, it should be noted, that most transients are like example 1 and the radiological envelope conservatively over-predicts the actual radiological impact by a factor greater than 100.

15.2.4.5.2 Depressurization - Shutdown Evaluation

15.2.4.5.2.1 Fission Product Release from Fuel

While no fuel rods are damaged as a consequence of this event, fission product activity associated with normal coolant activity levels as well as that released from previously defective rods will be released to the suppression pool as a consequence of SRV actuation and vessel depressurization. The release of activity from previously defective rods is based in part upon measurements obtained from operating BWR plants[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 steam line isolation valves. The specific models and assumptions used in the evaluation are described in Reference 2. The activity released to the environs is 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 containment vessel and purge system occurs under average annual meteorological conditions and commences 8 hours after initiation of the event. The annulus exhaust gas treatment system (AEGTS) efficiency for iodine is 99 percent.

15.2.4.5.2.3 Offsite Dose

As noted above, purging of the containment is assumed to occur under average annual meteorological conditions. To simplify the radiological calculation, it is assumed the radiological dose commitment is proportional to the average annual χ/Q value, which is 2.7×10^{-6} sec/m^3. The breathing rate is assumed to be 347 cc/sec and the dose recipient is located at one position for the entire release period. The radiological doses for this event are presented in Table 15.2-8.

15.2.4.5.2.4 Onsite Dose

The onsite radiological consequences of this event are presented in Section 12.2.2.

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 15.2-9.

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 15.2-10 lists the sequence of events for Figure 15.2-7.

15.2.5.2.1.1 Identification of Operator Actions

The operator should:

a. Verify auto transfer of buses supplied by generator to incoming power; if automatic transfer has not occurred, manual transfer must be made.

b. Monitor and maintain reactor water level at required level.

c. Check turbine for proper operation of all auxiliaries during coastdown.

d. Depending on conditions, initiate normal operating procedures for cooldown, or maintain pressure for restart purposes.

e. Put the mode switch in the "startup" position before the reactor pressure decays to <850 psig.

f. Secure the RCIC operation if auto initiation occurred due to low water level.

g. Monitor control rod drive positions and insert both the IRMs and SRMs.

h. Investigate the cause of the trip, make repairs as necessary, and complete the scram report.

i. Cool down 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 15.2-11.

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 Section 15.7.1.

Single failures will not effect 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 Section 15.1.1.3.1 was used to simulate this transient event.

15.2.5.3.2 Input Parameters and Initial Conditions

This analysis was performed with plant conditions tabulated in Table 15.0-1 unless otherwise noted.

Turbine stop valves full stroke closure time is 0.1 second.

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

The analysis presented here is a hypothetical case with a conservative 2 inches Hg per second 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 inches Hg less than the stop valve closure.

15.2.5.3.3 Results

Under this hypothetical 2 inches Hg per second vacuum decay condition, the turbine bypass valve and main steam line isolation valve closure would follow main turbine and feedwater turbine trips about 5 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam line 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 steam line flow. Figure 15.2-7 shows the transient expected for this event. It is assumed that the plant is initially operating at 105 percent of NB rated steam flow conditions. Peak neutron flux reaches 120 percent of NB rated 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 steam line isolation valves and bypass valves. While these are the major events occurring, other resultant actions will include scram (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 15.2-9. 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.

Other 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:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rods-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be 1 to 2 percent higher than the valve's nominal set point.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1,182 psig at the vessel bottom clearly below the reactor coolant pressure boundary transient pressure limit of 1,375 psig. Vessel dome pressure does not exceed 1,154 psig. A comparison of these values to those for turbine trip with bypass failure, at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

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

15.2.6 LOSS OF A-C 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 operator 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 <u>Sequence of Events and Systems Operation</u>

15.2.6.2.1 Sequence of Events

15.2.6.2.1.1 Loss of Auxiliary Power Transformer

Table 15.2-12 lists the sequence of events for Figure 15.2-8.

15.2.6.2.1.2 Loss of All Grid Connections

Table 15.2-13 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 satisfactorily to prevent turbine bearing damage. Also, he should verify proper switching and loading of the emergency diesel generators.

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

a. Following the scram, verify all rods in.

b. Check that diesel generators start and carry the vital loads.

c. Check that relays on the reactor protection system (RPS) drop out.

d. Check that both RCIC and HPCS start when reactor vessel level drops to the initiation point after the relief opens.

e. Break vacuum before the loss of sealing steam occurs.

f. Check T-G auxiliaries during coastdown.

g. When both the reactor pressure and level are under control, secure both HPCS and RCIC as necessary and it has been verified that initiation is not due to a LOCA.

h. Continue cooldown per the normal procedure.

i. 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.

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

a. Recirculation pumps are tripped at a reference time, t=0, with normal coastdown times.

b. Within 8 seconds, 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 percent 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 (L8) set point before 8 seconds.

c. At approximately 28 seconds, the loss of condenser vacuum is expected to reach the bypass valves closure set point and main steam line isolation set point.

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 Section 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 Section 15.1.1.3.1 was used to simulate this 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 tabulated in Table 15.0-1 and under the assumed systems constraints described in Section 15.2.6.2.2.

15.2.6.3.2.2 Loss of All Grid Connections

Same as Section 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 2 seconds scram and main steam line isolation valve closure occur.

Sensed level drops to the RCIC and HPCS initiation set point at approximately 20 seconds 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 Section 15.2.2. Figure 15.2-9 shows graphically the simulated event. Peak neutron flux reaches 111 percent of NB rated power while fuel surface heat flux shows no increase.

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 seconds 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.

Following main steam line isolation the reactor pressure is expected to increase until the safety/relief valve set points 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 set points. The pressure in the vessel bottom is limited to a maximum value of 1,182 psig, well below the vessel pressure limit of 1,375 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 Section 15.2.4.5; therefore, the radiological exposures noted in Section 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 15.2-14 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:

a. Verify all rods in, following the scram.

b. Verify HPCS and RCIC initiation.

c. Verify that the recirculation pumps trip on reactor low low level.

d. Secure HPCS when reactor level and pressure are under control.

e. Continue operation of RCIC until decay heat diminishes to a point where the RHR system can be put into service.

f. Monitor turbine coastdown, break vacuum as necessary.

g. 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 second after this trip to scram the reactor. The low level (L3) scram trip function meets single failure criterion.

Containment isolation, when it occurs, would also initiate a main steam line 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 efforts to shut down the reactor are automatic and designed to satisfy 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 Section 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 15.0-1.

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 seconds. Subcooling decreases causing a reduction in core power level and pressure. As power level is lowered, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure for the first 5 seconds or so. Water level continues to drop until the vessel level (L3) scram trip set point is reached whereupon the reactor is shut down and the recirculation flow is run back. Vessel water level continues to drop to the L2 trip. At this time, the recirculation system is tripped and HPCS and RCIC operation is initiated. MCPR remains consid ly above the safety limit since increases in heat flux are not experience

15.2.7.3.4 rations of Uncertainties

End-of-cycle s teristics are assumed.

This transient is most 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 the simulation of the first 50 seconds 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 Section 15.2.7.2.3.

15.2.7.4 Barrier Performance

Peak pressure in the bottom of the vessel reaches 1,087 psig, which is below the ASME Code limit of 1,375 psig for the RCPB. Vessel dome pressure does not exceed 1,045 psig. 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 Section 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 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:

a. Loss of all offsite a-c power.

b. Utilization of safety shutdown equipment only.

c. Operator involvement only after 10 minutes 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 102 percent rated power when a long-term loss of offsite power occurs, causing multiple safety-relief valve actuation (see Section 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:

a. No RHR valves have failed in the shutdown cooling mode in BWR total operating experience.

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

15.2.9.2 <u>Sequence of Events and System Operation</u>

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-15.

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 Section 15.2.6 (loss of offsite power event with isolation/scram). The operator should do the following:

a. At approximately 10 minutes into the transient, initiate suppression pool cooling (again for purposes of this analysis, it is assumed that only one RHR heat exchanger is available).

b. After establishing RPV water level through HPCS operation, isolate feedwater system. This is conservatively assumed to occur at 10 minutes.

c. Initiate RPV shutdown depressurization by manual actuation of 5 ADS valves.

d. 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.

e. 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 utilized.

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 operator error can make the consequences of this event any worse. See Appendix 15A for a discussion of this subject.

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 minute 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 15.2-16 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 (see 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.

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 200°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 safety/relief valves. 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.

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:

a. The vessel is at 100 psig and saturated conditions;

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

c. There is no offsite power available.

In the event that the RHRS 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 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 (D-C Division 1 and D-C Division 2)
RHR Loop A (Division 1)

HPCS (Division 3)
RCIC (D-C Division 1)
LPCS (Division 1)

Since availability or failure of Division 3 equipment does not effect 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:

a. Division 1 fails, Divisions 2 and 3 functional:

Failed systems	Functional systems
RHR loop A	HPCS
LPCS	ADS
	RHR loops B and C
	RCIC

Assuming the single ailure 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.

b. 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, 14, and 15 show RHR loops A, B and/or C (simplified).

Using the above assumptions and following the depressurization rate shown in Figure 15.2-16 case a or b, the suppression pool temperature is shown in Figure 15.2-17 case a or b.

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 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 Section 15.2.4.5. Therefore, the radiological exposures noted in Section 15.2.4.5 cover the consequences of this event.

15.2.10 LOSS OF INSTRUMENT AIR

15.2.10.1 Event Evaluation

Loss of the instrument air system during normal plant operation could occur as a result of a major line break in the system or as a result of mechanical or electrical failure of the normal air supply from the service air system and the backup instrument air compressor.

15.2.10.2 Analysis of Effects and Consequences

Loss of the instrument air system will result in the shutdown of the reactor due to the closing of the main steam isolation valves. The failure of instrument air will not interfere with the safe shutdown of the reactor since all equipment using instrument air is designed to fail to a position that is consistent with the safe shutdown of the plant.

Air operated equipment that must be available for use in the event of a failure of the instrument air system, is provided with backup accumulators to provide the required air supply.

15.2.10.3 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 is designed. Therefore, these barriers maintain their integrity and function as designed.

15.2.10.4 Radiological Consequences

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

15.2.11 REFERENCES FOR 15.2

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

2. Tschaeche, A. N., "Mark III Suppression Pool Source Terms", 22A6215 (July, 1978).

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

4. Bilanin, W. I., Bodily, R. J., and Cruz, G. A., "The General Electric Mark III Pressure Suppression Containment System Analytical Model (Supplement 1), "September 1975 (NEDO-20533, Suppl. 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 set point and initiates a reactor scram.
2.4	Recirculation pump motors are tripped due to high dome pressure.
2.5	Safety/relief valves open due to high pressure.
6.4	Vessel water level (L8) trip initiates main turbine and feedwater turbine trips.
6.5	Main turbine stop valves closed.
9.6	Safety/relief valves close.
9.8	Group 1 safety/relief valves open again to relieve decay heat.
16.0	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.
1.7	Safety/relief valves open due to high pressure.
4.2	Vessel water level (L8) trip initiates trip of the feedwater turbines.
7.2	Safety/relief valves close.

TABLE 15.2-3

SEQUENCE OF EVENTS FOR FIGURE 15.2-3

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.
0	Fast control valve closure (FCV) initiates scram trip and recirculation pump trip (RPT).
0.07	Turbine control valves closed.
1.3	Safety/relief valves open due to high pressure.
5.7	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.9	Safety/relief valves close.
9.6	Group 1 safety/relief valves open again to relieve decay heat.
15.4	Group 1 safety/relief valves close again.
19.8	Group 1 safety/relief valves open again to relieve decay heat.
26.0	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.
1.8	Safety/relief valves open due to high pressure.
4.3	Vessel water level (L8) trip initiates trip of the feedwater turbines.
7.1	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.
1.3	Safety/relief valves open due to high pressure.
5.6	Vessel water level (L8) trip initiates trip of the feedwater turbines.
8.7	Safety/relief valves close.
9.4	Group 1 safety/relief valves open again to relieve decay heat.
15.2	Group 1 safety/relief valves close again.
19.6	Group 1 safety/relief valves open again to relieve decay heat.
25.8	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 steam line isolation valves (MSLVI).
0.3	MSLIVs reach 90% open.
0.3	MSLIV position trip scram initiated.
2.7	Recirculation pump drive motors are tripped due to high dome pressure.
2.8	Safety/relief valves open due to high pressure.
8.0	Safety/relief valves close.
8.7	Group 1 safety/relief valves open again to relieve decay heat.
14.6	Group 1 safety/relief valves close again.
17.1	Vessel water level reaches L2 setpoint.
17.1	Group 1 safety/relief valves open again to relieve decay heat.
23.6	Group 1 safety/relief valves close again.
29.5	Group 1 safety/relief valves open again to relieve decay heat.
34.8	Group 1 safety/relief valves close again.
44.9	Group 1 safety/relief valves open again to relieve decay heat.
47.1	HPCS and RCIC flow into vessel (not included in simulation).
49.9	Group 1 safety/relief valves close again.

TABLE 15.2-7

CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

I-131	7.3-4
132	1.9-3
133	6.5-4
134	1.8-5
135	3.4-4
Kr-83m	1.4+1
85	5.4-1
85m	7.7+1
87	5.7+0
88	1.0+2
Xe-131m	3.7+0
133m	3.0+1
133	1.4+3
135m	2.3-2
135	4.8+2

TABLE 15.2-8

CLOSURE OF ALL MAIN STEAM ISOLATION VALVES
RADIOLOGICAL EFFECTS

Site boundary (worst location on land receptor - 1298m)

Annual average χ/Q = 2.7-6 sec/m^3

Whole body dose (mRem/event) = 0.24

Thyroid dose (mRem/event) = 0.0014

TABLE 15.2-9

TYPICAL RATES OF DECAY FOR CONDENSER VACUUM

	Cause	Estimated Vacuum Decay Rate
a.	Failure or Isolation of Steam Jet Air Ejectors	<1 inch Hg/minute
b.	Loss of Sealing Steam to Shaft Gland Seals	~1 to 2 inches Hg/minute
c.	Opening of Vacuum Breaker Valves	~2 to 12 inches Hg/minute
d.	Loss of One or More Circulating Water Pumps	~4 to 24 inches Hg/minute

TABLE 15.2-10

SEQUENCE OF EVENTS FOR FIGURE 15.2-7

Time-sec	Event
-3.0 (est)	Initiate simulated loss of condenser vacuum at 2 inches of Hg per second.
0.0 (est)	Low condenser vacuum main turbine trip actuated.
0.0 (est)	Low co[illegible] vacuum feedwater trip actuated.
0.01	Main turbine trip initiates recirculation pump trip (RP) and scram.
1.8	Safety/relief valves open due to high pressure.
5.0	Low condenser vacuum initiates main steam line isolation valve closure.
5.0	Low condenser vacuum initiates bypass valve closure.
6.9	Safety/relief valves close.
8.0	Group 1 safety/relief valves open again to relieve decay heat.
13.0	Group 1 safety/relief valves close again.
13.7	Water level reaches Level 2 set point and initiates HPCS and RCIC.
15.7	Group 1 safety/relief valves open again to relieve decay heat.
22.5	Group 1 safety relief valves close again.
27.5	Group 1 safety/relief valves open again to relieve decay heat.
32.8	Group 1 safety/relief valves close again.
41.7	Group 1 safety/relief valves open again to relieve decay heat.
43.7 (est)	HPCS and RCIC flow enters vessel (not in simulation).
46.7	Group 1 safety/relief valves close again.

TABLE 15.2-11

TRIP SIGNALS ASSOCIATED WITH LOSS OF CONDENSER VACUUM

Vacuum (inches of Hg)	Protective Action Initiated
27 to 28	Protective action initiated.
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 15.2-12

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	Hotwell and condensate booster pumps are tripped.
0	Condenser circulating water pumps tripped.
2.0	Main steam isolation valves close due to loss of power causing a reactor scram.
4.0	Feedwater pump turbines are tripped.
5.1	Safety/relief valves open due to high pressure.
10.1	Safety/relief valves close.
20.1	Vessel water level reaches Level 2 set point.
50.1 (est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-13

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 power load unbalance (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.
0.08	Turbine control valves closed.
0.1	Turbine bypass valves open.
1.7	Safety/relief valves open due to high pressure.
2.0	Main steam isolation valves close due to loss of power.
4.0	Feedwater pumps trip due to MSIV closure.
18.6	Safety/relief valves close.
20.4	Vessel water level reaches Level 2 set point.
28.0	Closure of turbine bypass valves is initiated via low condenser vacuum.
50.4	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-14

SEQUENCE OF EVENTS FOR FIGURE 15.2-10

Time-sec	Event
0	Trip of all feedwater pumps initiated.
3.5	Vessel water level reaches level 4 and initiates recirculation flow runback.
5	Feedwater flow decays to zero.
7.0	Vessel water level (L3) trip initiates scram trip.
14.9	Vessel water level reaches Level 2.
15.1 (est)	Recirculation pumps trip due to Level 2 trip.
44.9 (est)	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.2-15

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Approximate Elapsed Time	Event
0	Reactor is operating 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.
10 min	Suppression pool cooling initiated to prevent overheating from SRV actuation.*
23 min	Controlled depressurization initiated (100°F/hr) using selected safety/relief valves
35 min	Blowdown to approximately 100 psig completed.
65 min	Personnel are sent to open RHR shutdown cooling suction valve; this fails.
70 min	ADS valves are opened to complete blowdown to suppression pool, and RHR pump discharge is redirected from pool to vessel via LPCI line. Alternate shutdown cooling path has now been established.

*See Table 15.2-12 for detailed sequence of events for loss of a-c power transient.

TABLE 15.2-16

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

Initial power corresponding to 102% rated power	
Suppression pool mass, lbm	7,470,400
RHR, KHX value, Btu/sec/°F	440
Initial vessel conditions	
Pressure, psia	1,040
Temperature, °F	549
Initial primary fluid inventory, lbm	566,040
Initial pool temperature, °F	90
Service water temperature, °F	80
Vessel heat capacity, Btu/lbm/°F	0.125
HPCS 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

NOTES FOR FIGURE 15.2-11

ACTIVITY A

Initial pressure	1,040 psia
Initial temperature	549°F

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

a. The reactor is assumed to be operating at 102% rated power;

b. A loss of power transient occurs (see Section 15.2.6);

c. 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	1,040 psia
Initial system temperature	549°F

Operator Actions

During approximately the first 23 minutes, 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 23 minutes 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 system. After approximately 10 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 107°F.

When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode. At this time (35 min), the suppression pool temperature will be 146°F.

NOTES FOR FIGURE 15.2-11 (Continued)

ACTIVITY C1 (Division 1 fails, Division 2 available)

System pressure	approximately 100 psig
System temperature	approximately 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 15.2.9-4. Cold shutdown is achieved approximately 2 hours after transient occurred.

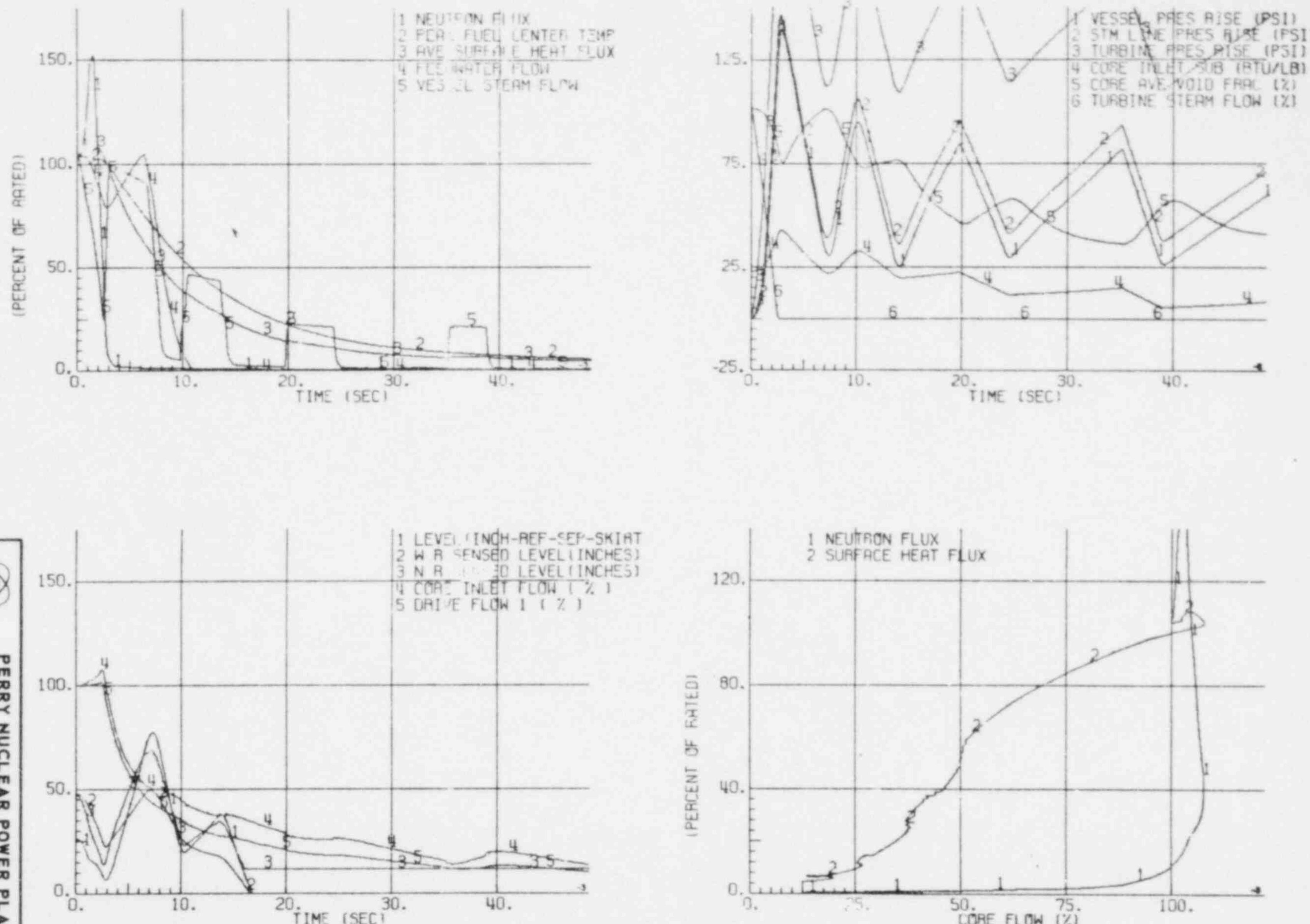
b. Utilizing RHR loops B and C together, water is taken from the suppression pool nd is pumped directly into the reactor vessel. The water passes through the vessel (picking up decay heat) and out the ADS valves returning to the suppression pool as shown in Figure 15.2.9-5. Suppression pool water is then cooled by operation of RHR loop B in the cooling mode (see Figure 15.2.9-6). In this alternate cooling path, RHR loop C is used for injection and RHR loop B for cooling. Cold shutdown is achieved approximately 9.5 hours after transient occurred.

ACTIVITY C2 (Division 2 fails, Division 1 available)

System pressure	approximately 100 psig
System temperature	approximately 340°F

Operator Actions

Utilizing RHR loop A 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.

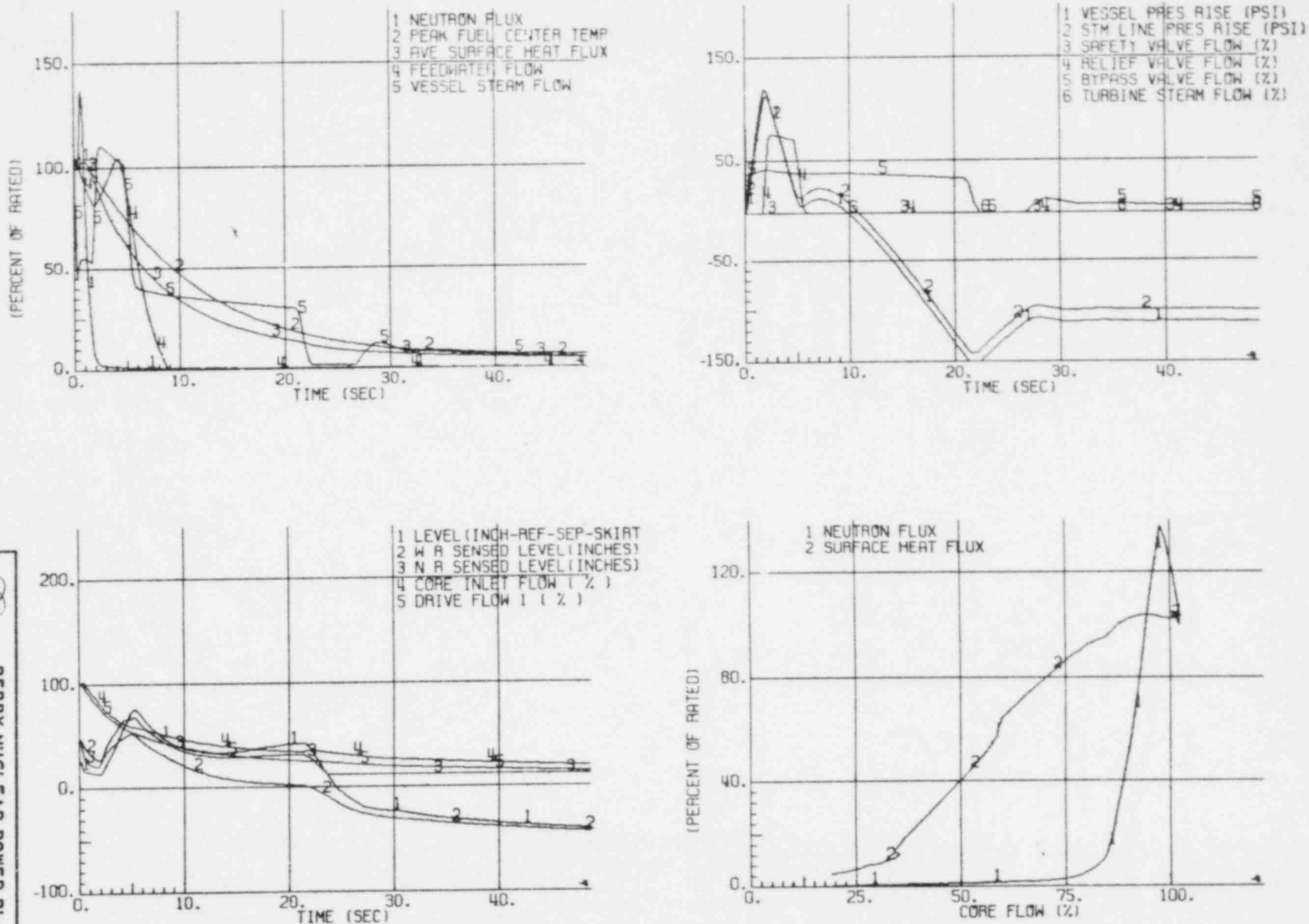


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Results of a Pressure Regulations Downscale Failure

Figure 15.2-1



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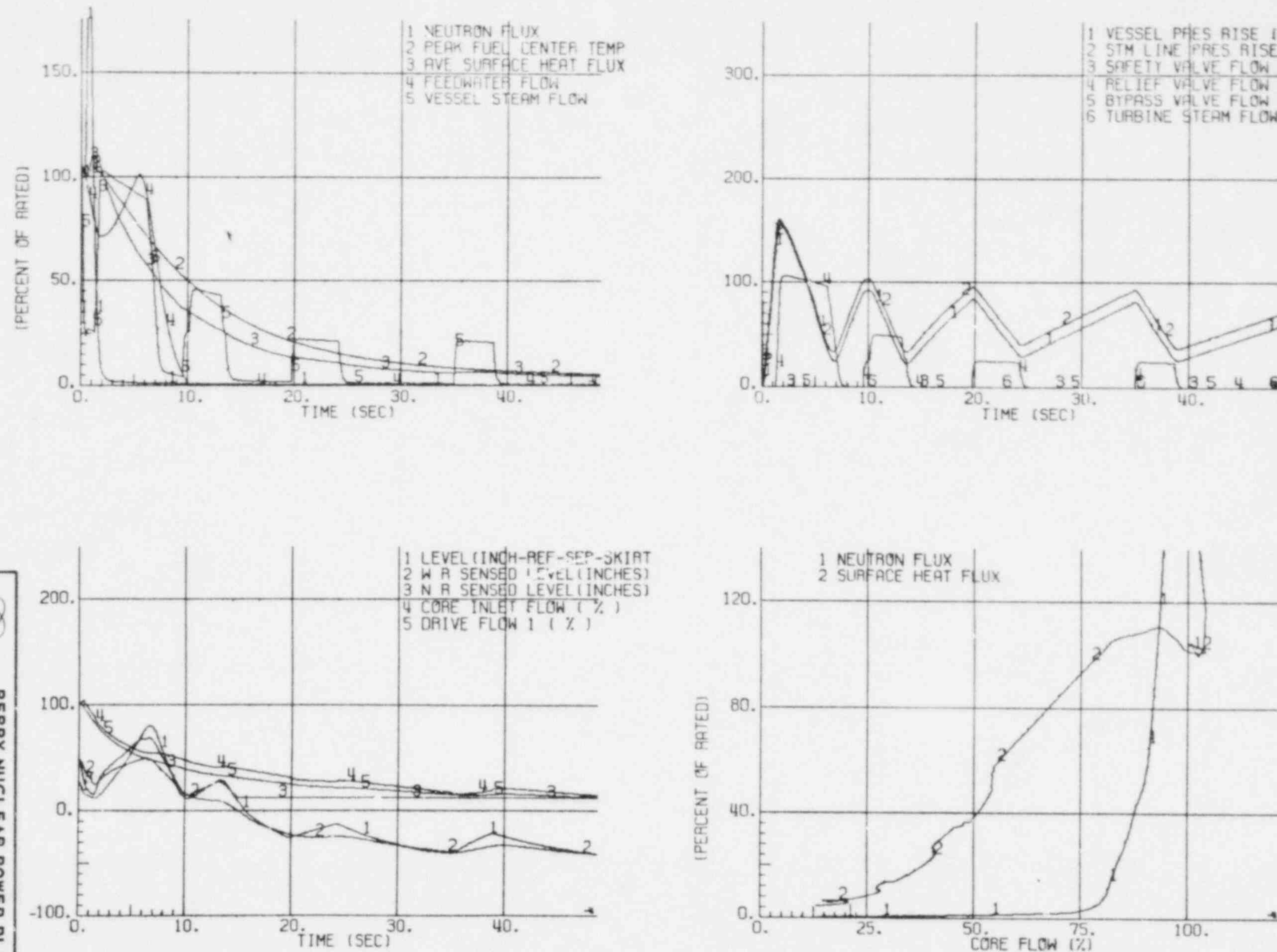
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Generator Load Rejection, Trip Scram, Bypass - On

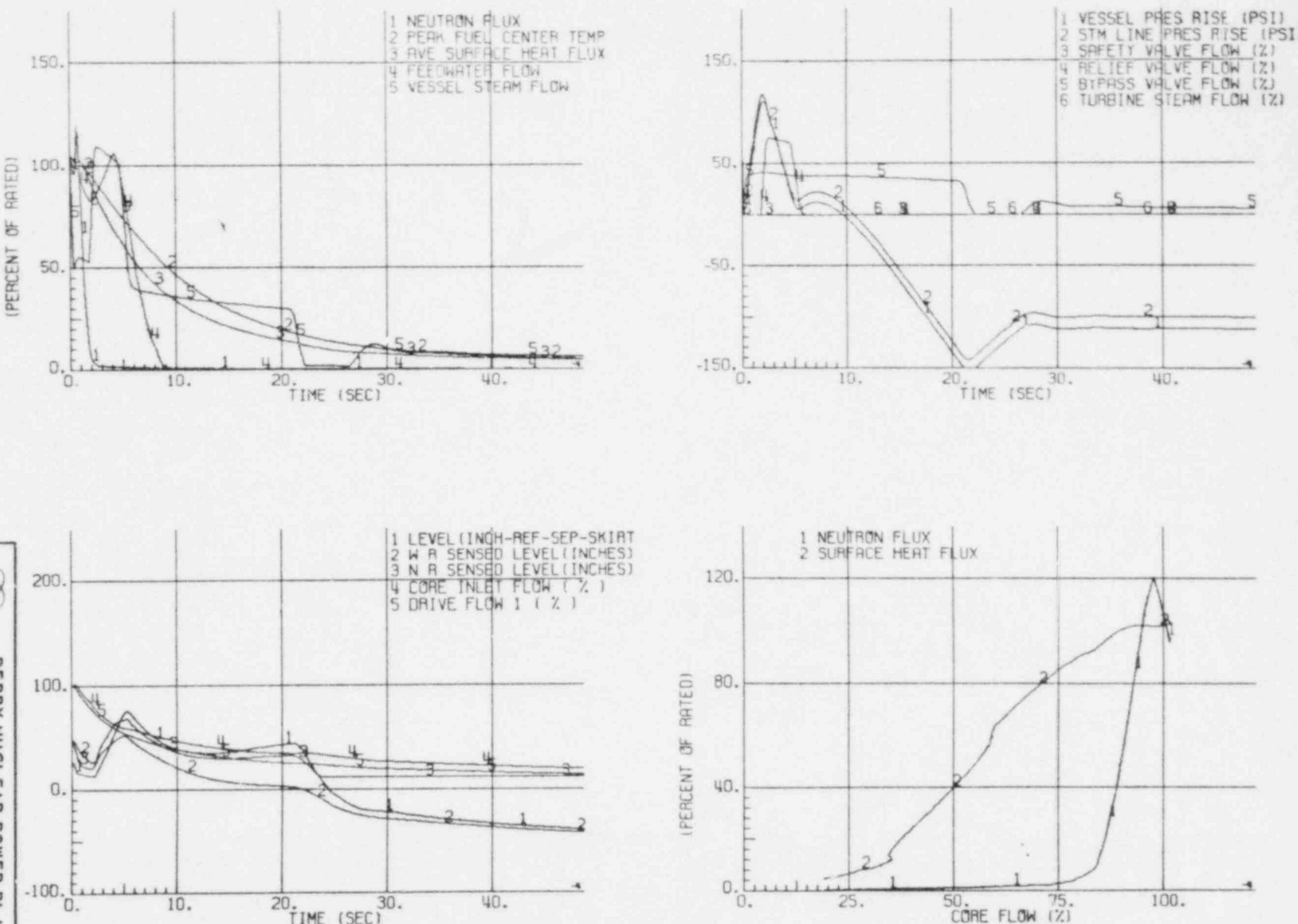
Figure 15.2-2



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Generator Load Rejection,
Trip Scram, Bypass - Off

Figure 15.2-3



PERRY NUCLEAR POWER PLANT
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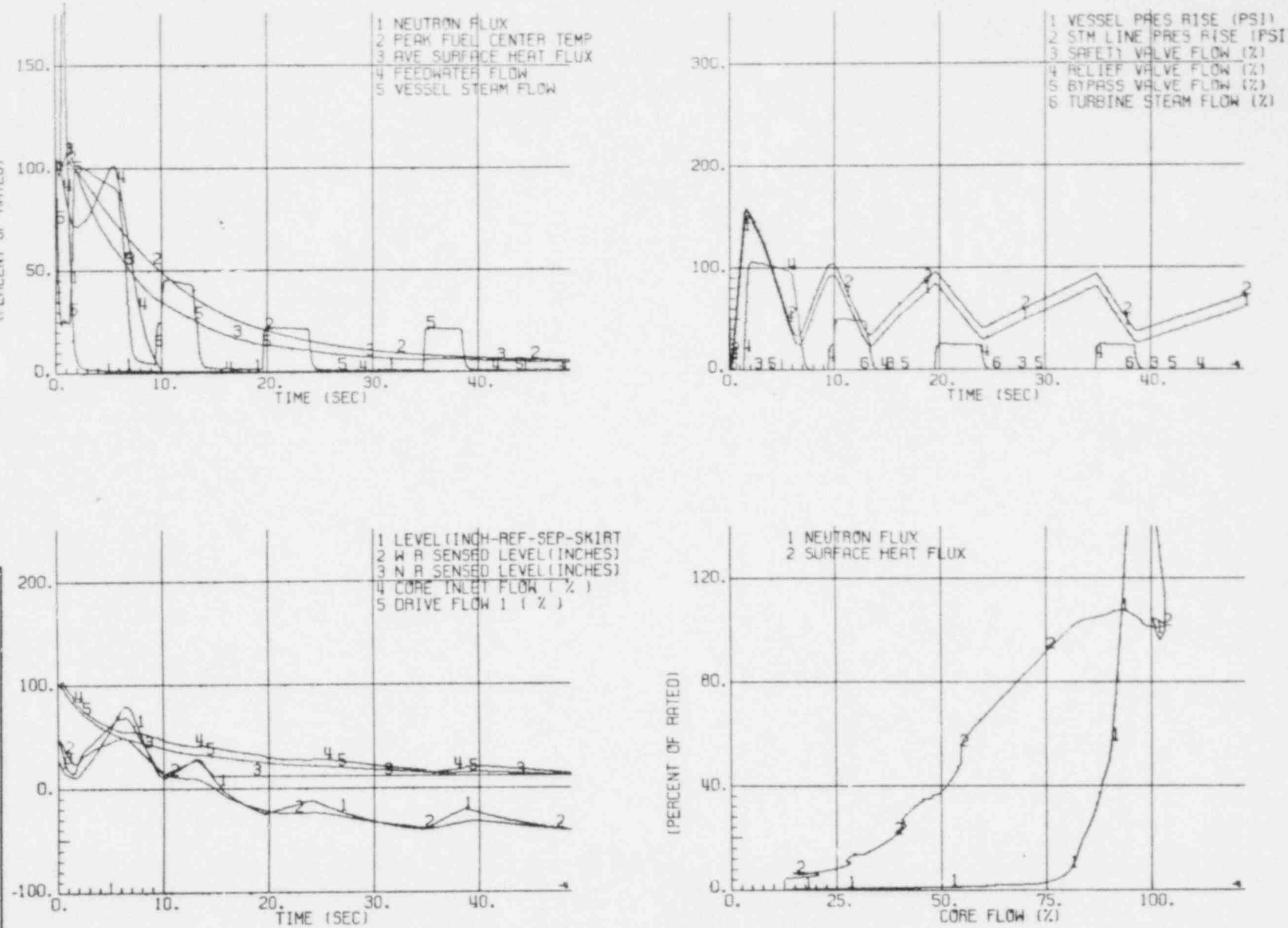
ILLUMINATING COMPANY

Turbine Trip, Trip Scram,
Bypass and RPT - On

Figure 15.2-4



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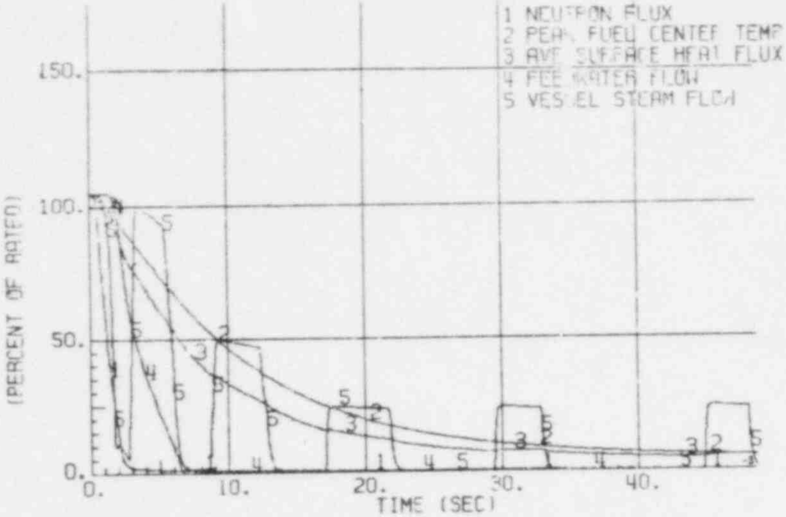
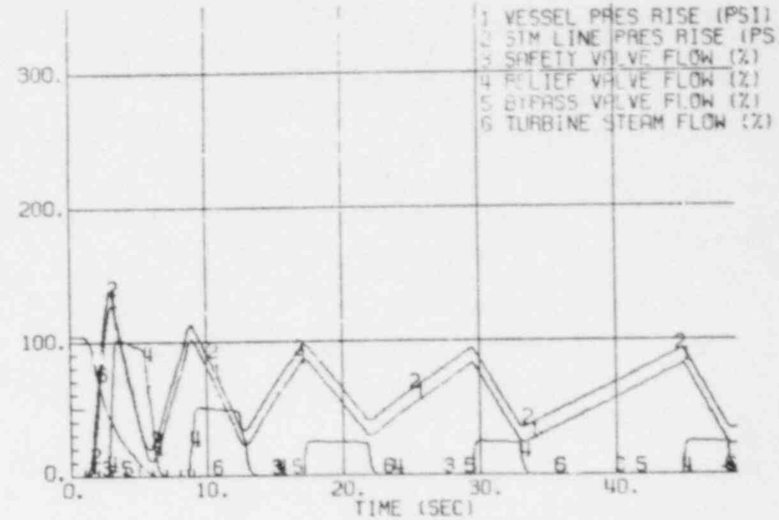
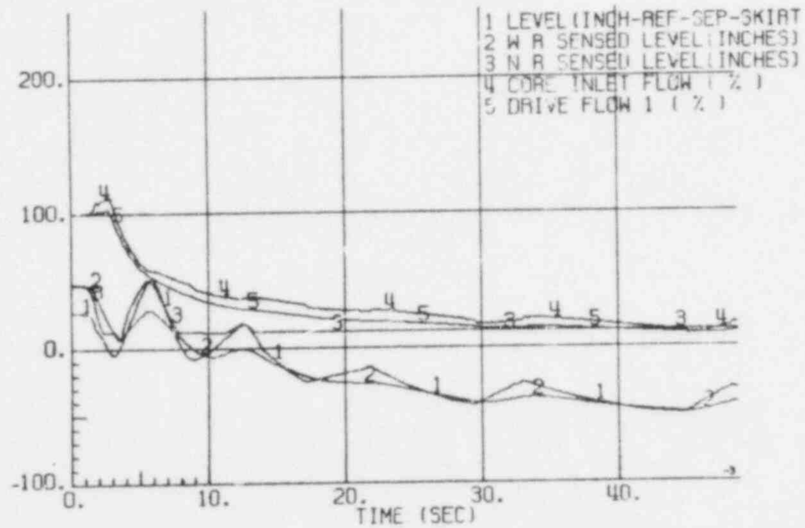
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

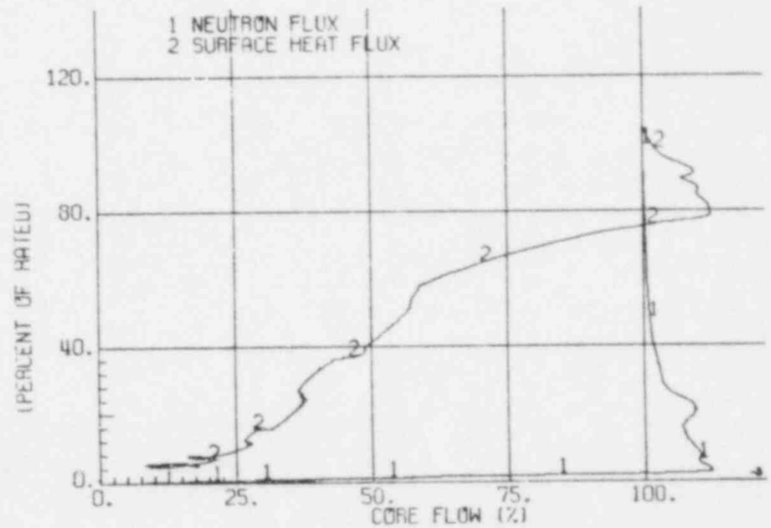
Turbine Trip, Trip Scram, Bypass - Off, RPT - On

Figure 15.2-5





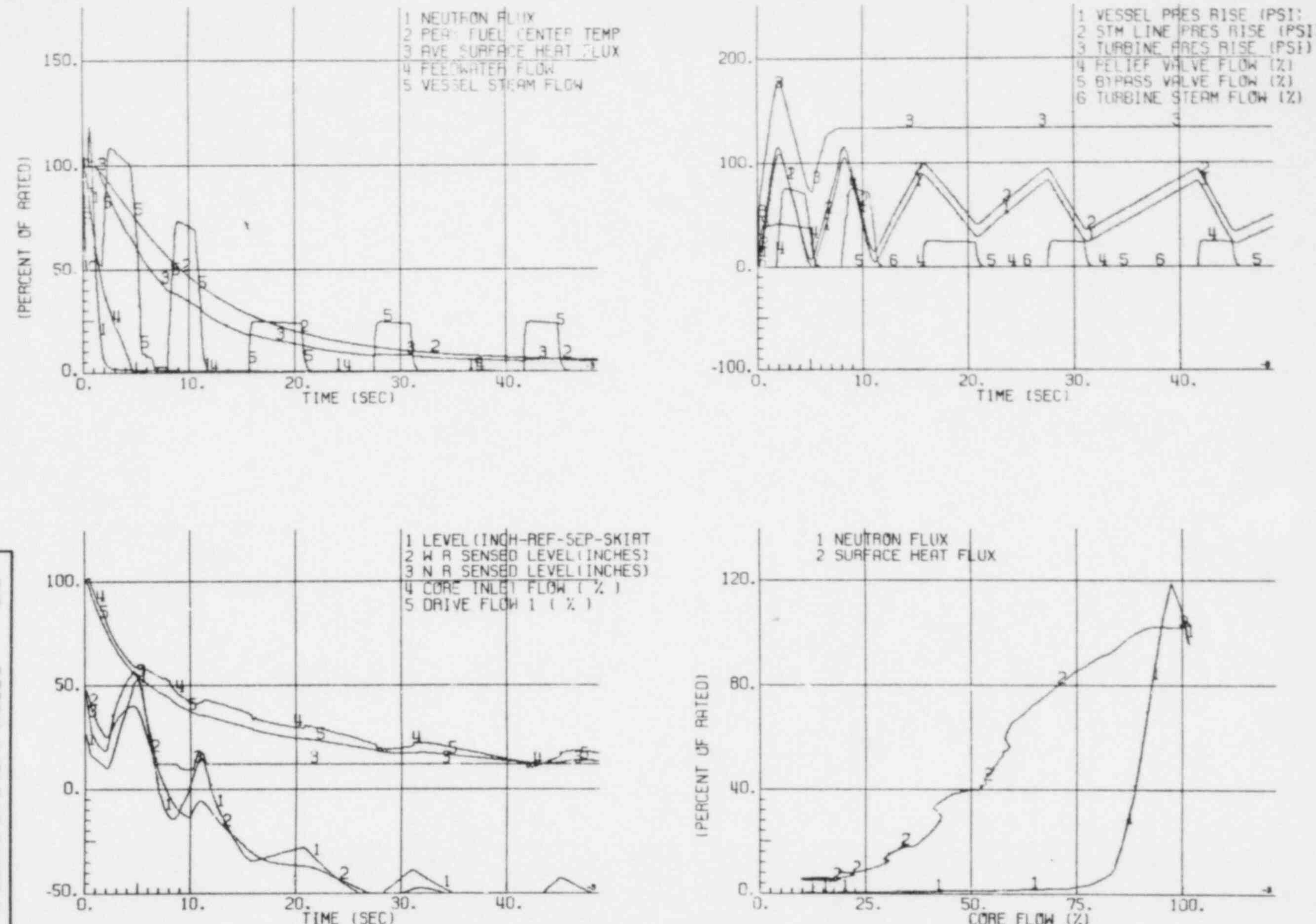




PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Three-Second Closure of All Main Steamline Isolation Valves with Position Switch Scram Trip

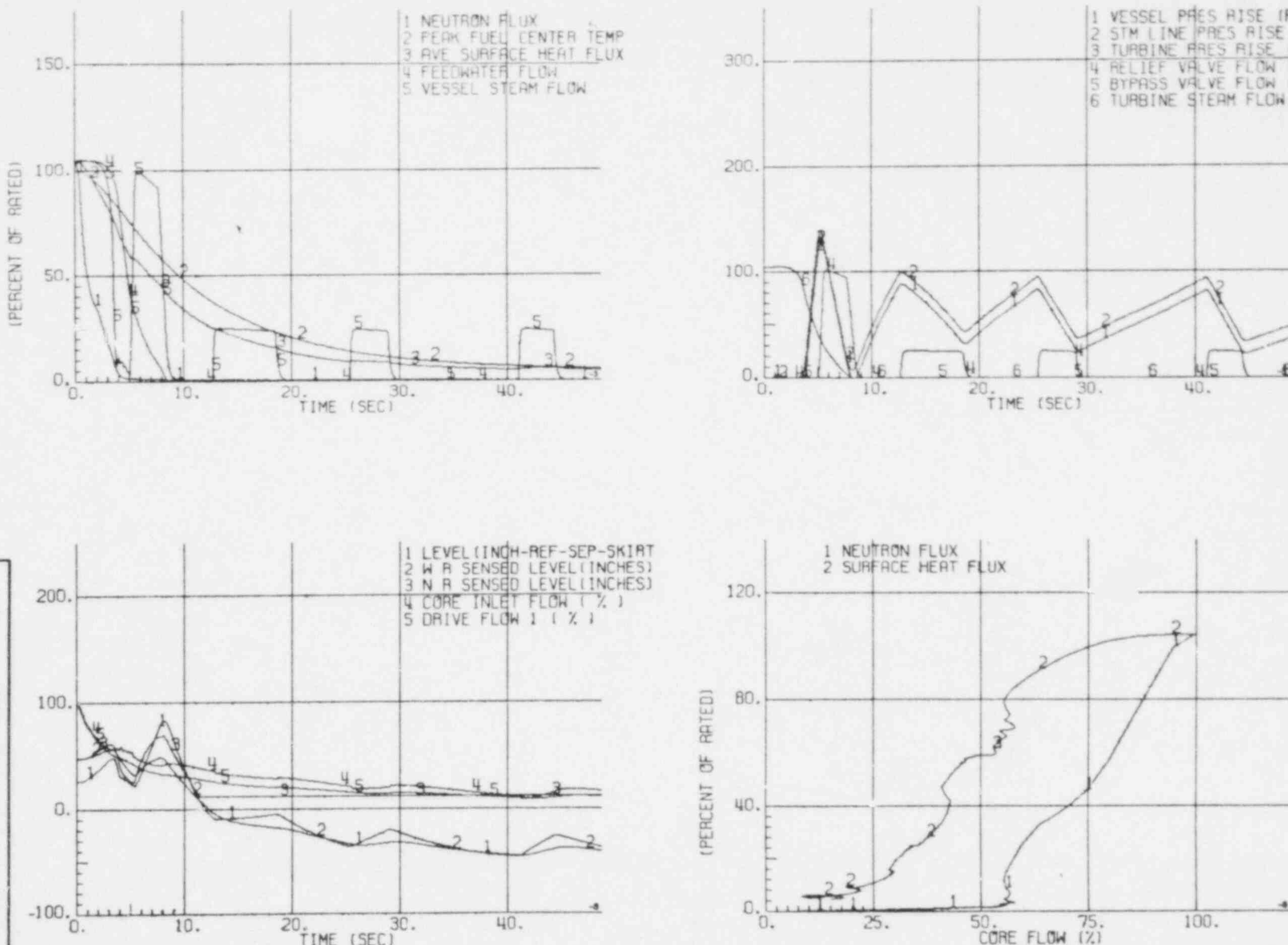
Figure 15.2-6



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Loss of Condenser Vacuum at 2 Inches Per Second

Figure 15.2-7



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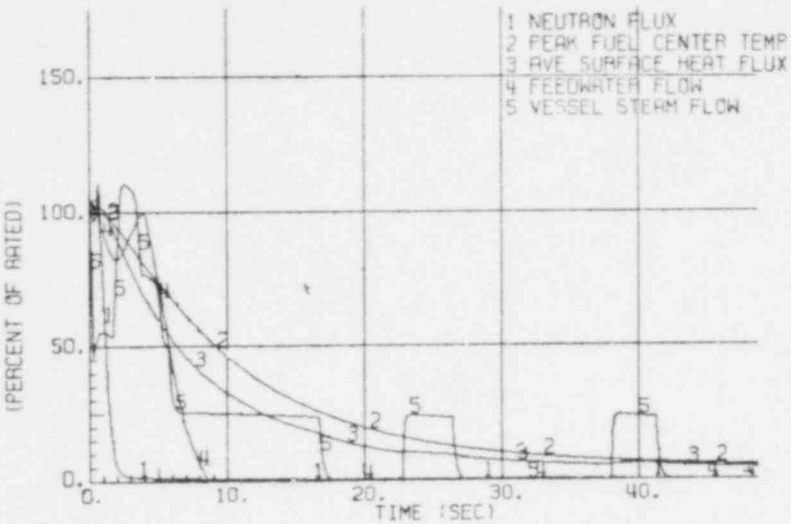
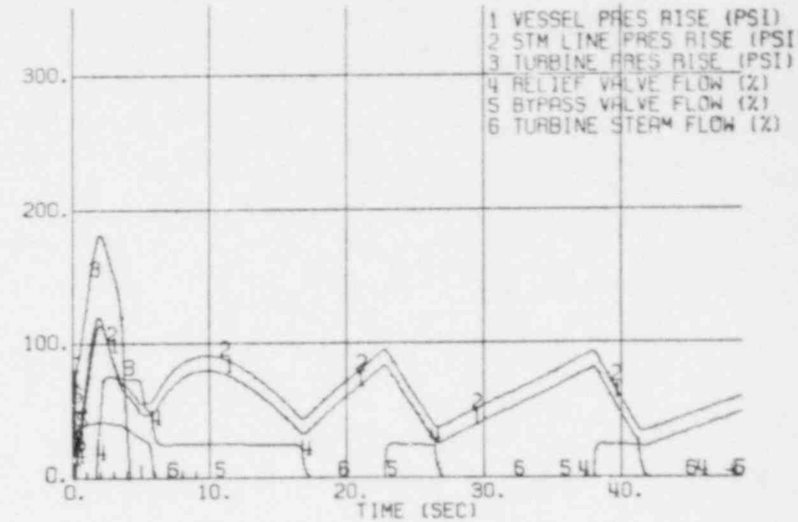
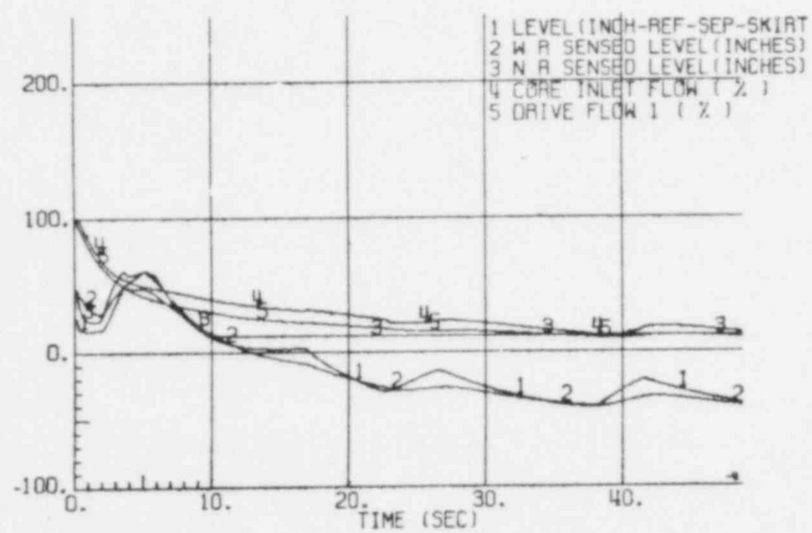
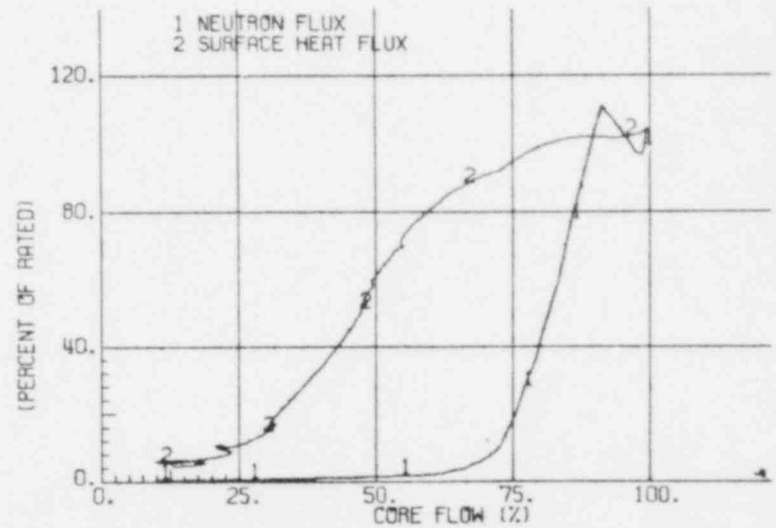
Loss of Auxiliary Power Transformer

Figure 15.2-8







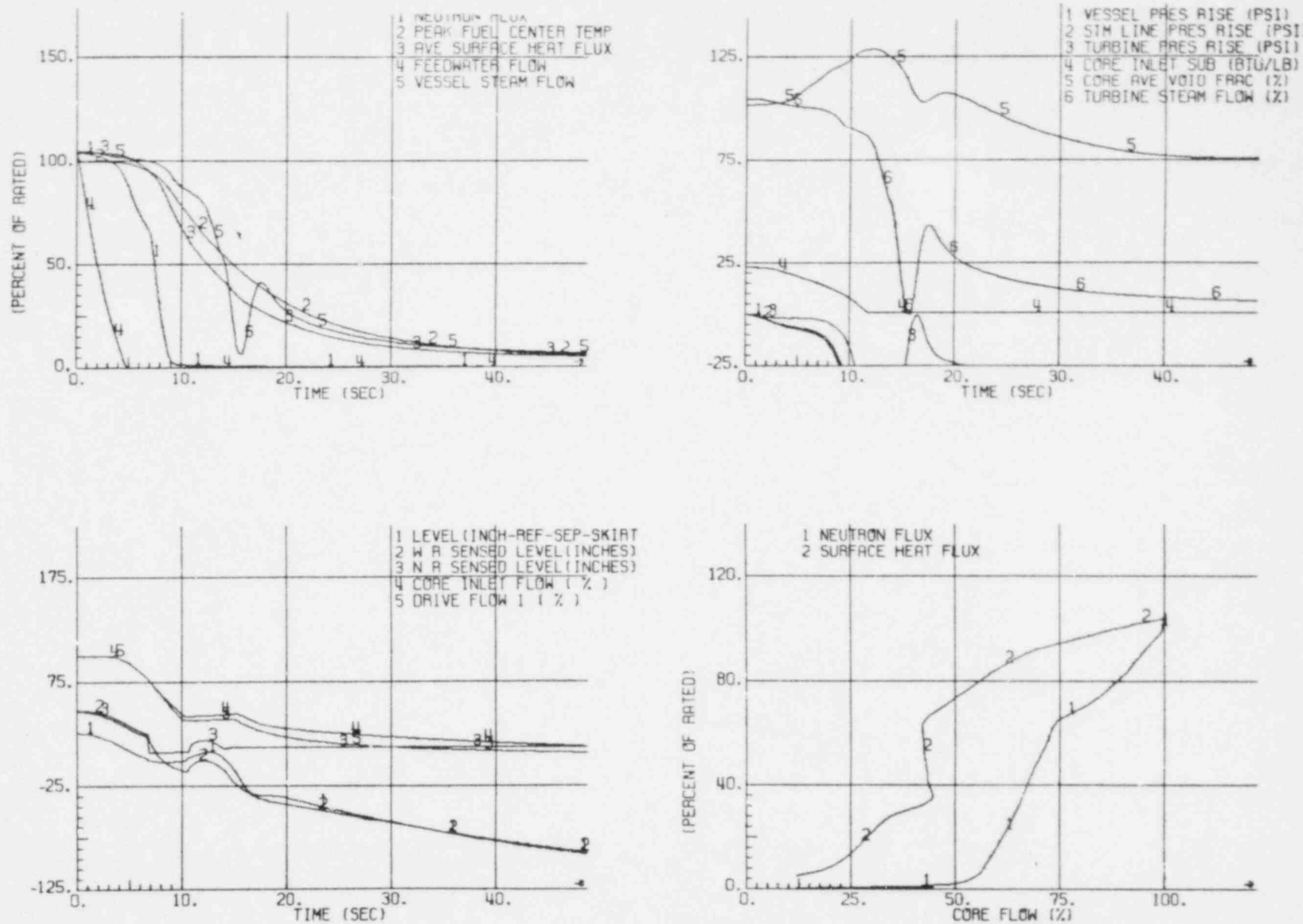


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Loss of All Grid Connections

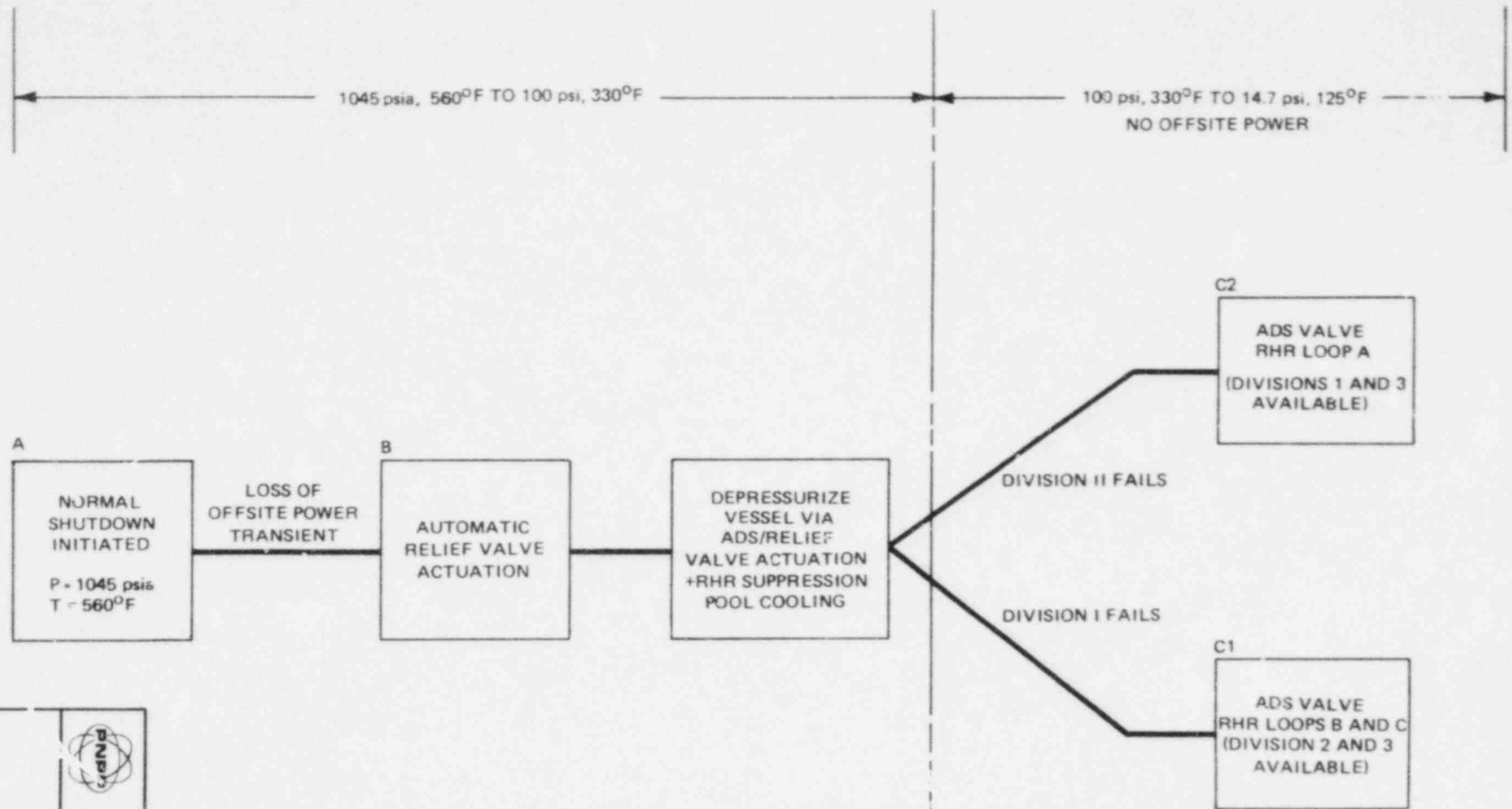
Figure 15.2-9



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Loss of All Feedwater Flow

Figure 15.2-10



PERRY NUCLEAR POWER PLANT
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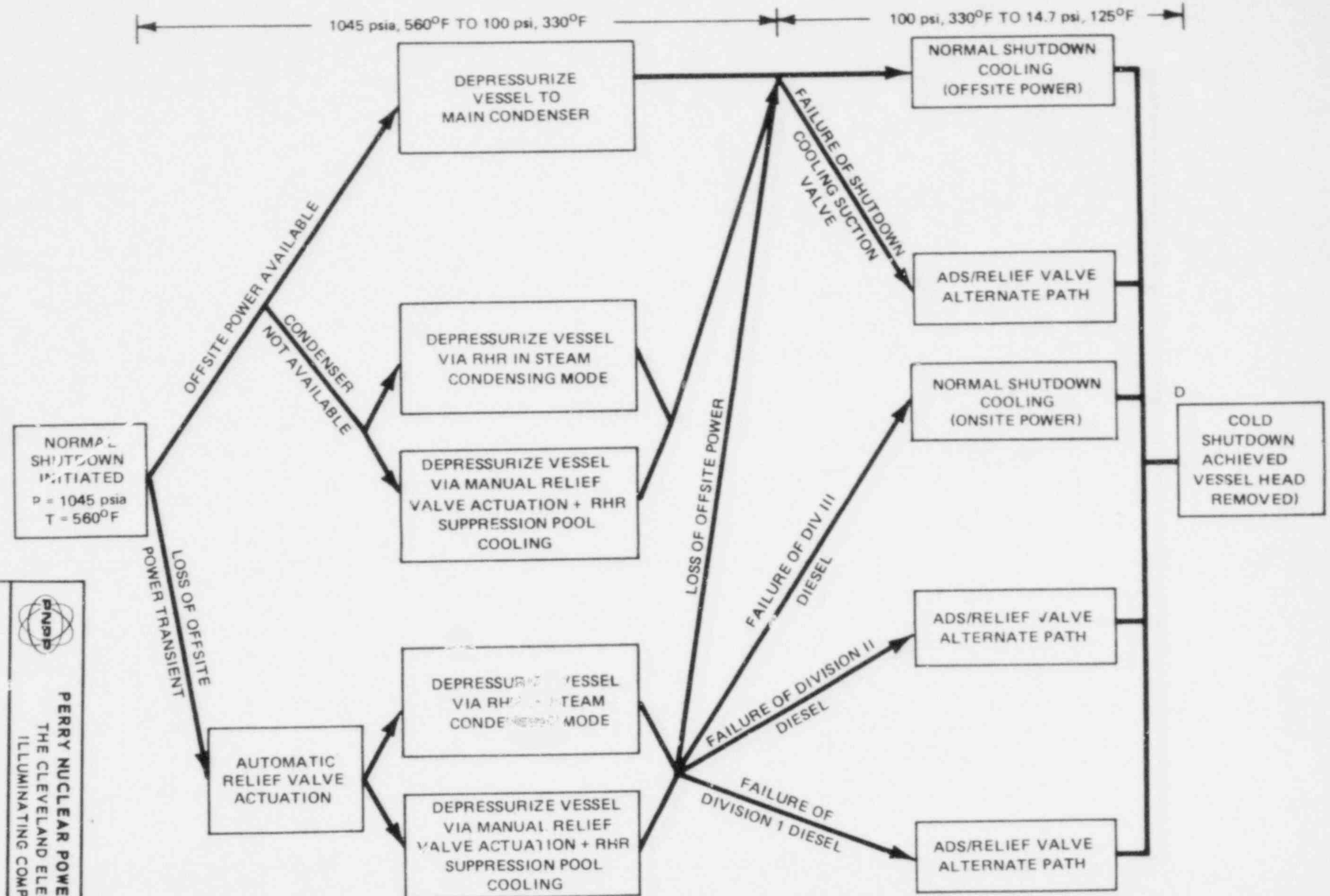
ADS/RHR Cooling Loops

Figure 15.2-11



PERRY NUCLEAR POWER PLANT
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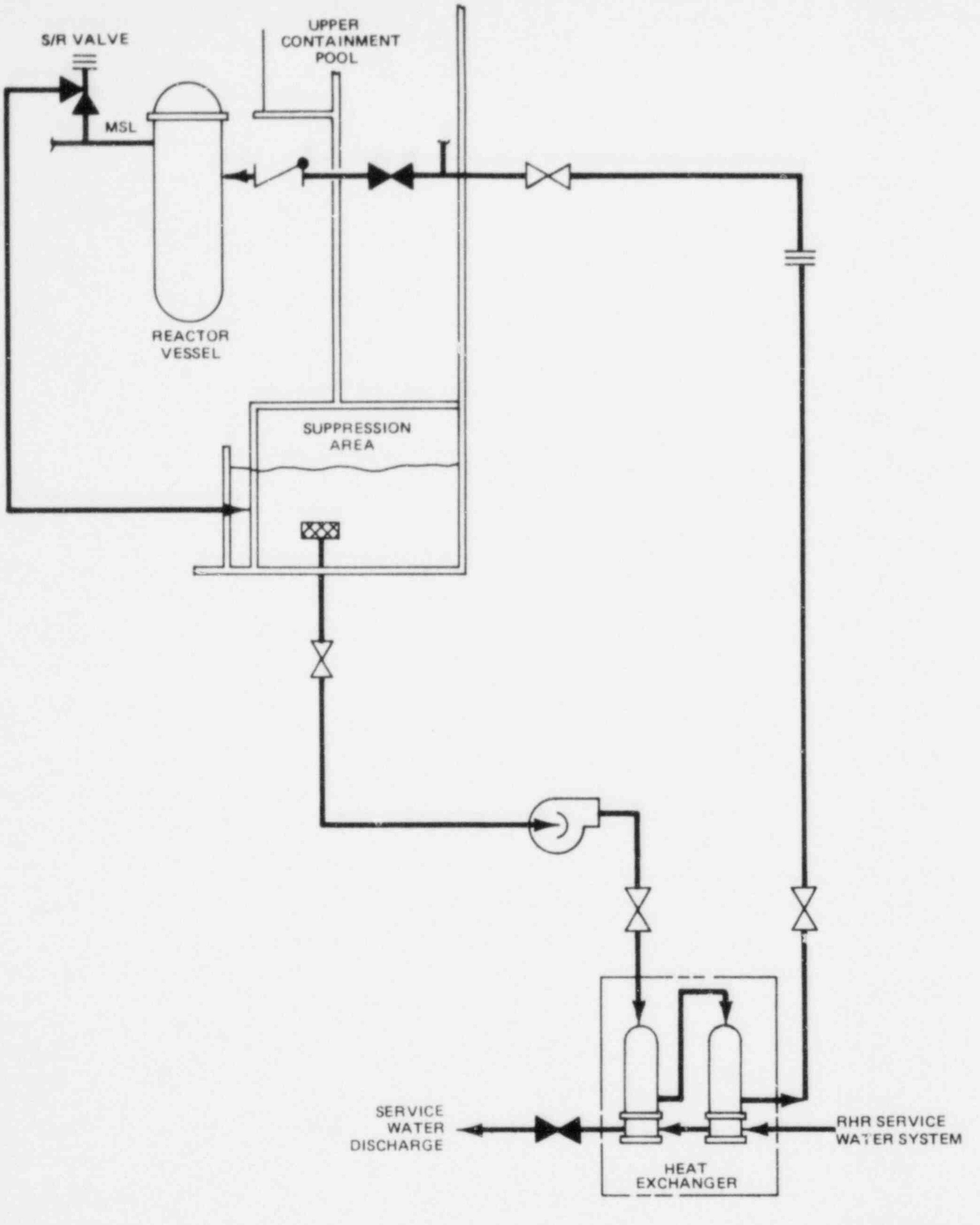
Summary of Paths Available to Achieve Cold Shutdown

Figure 15.2-12



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
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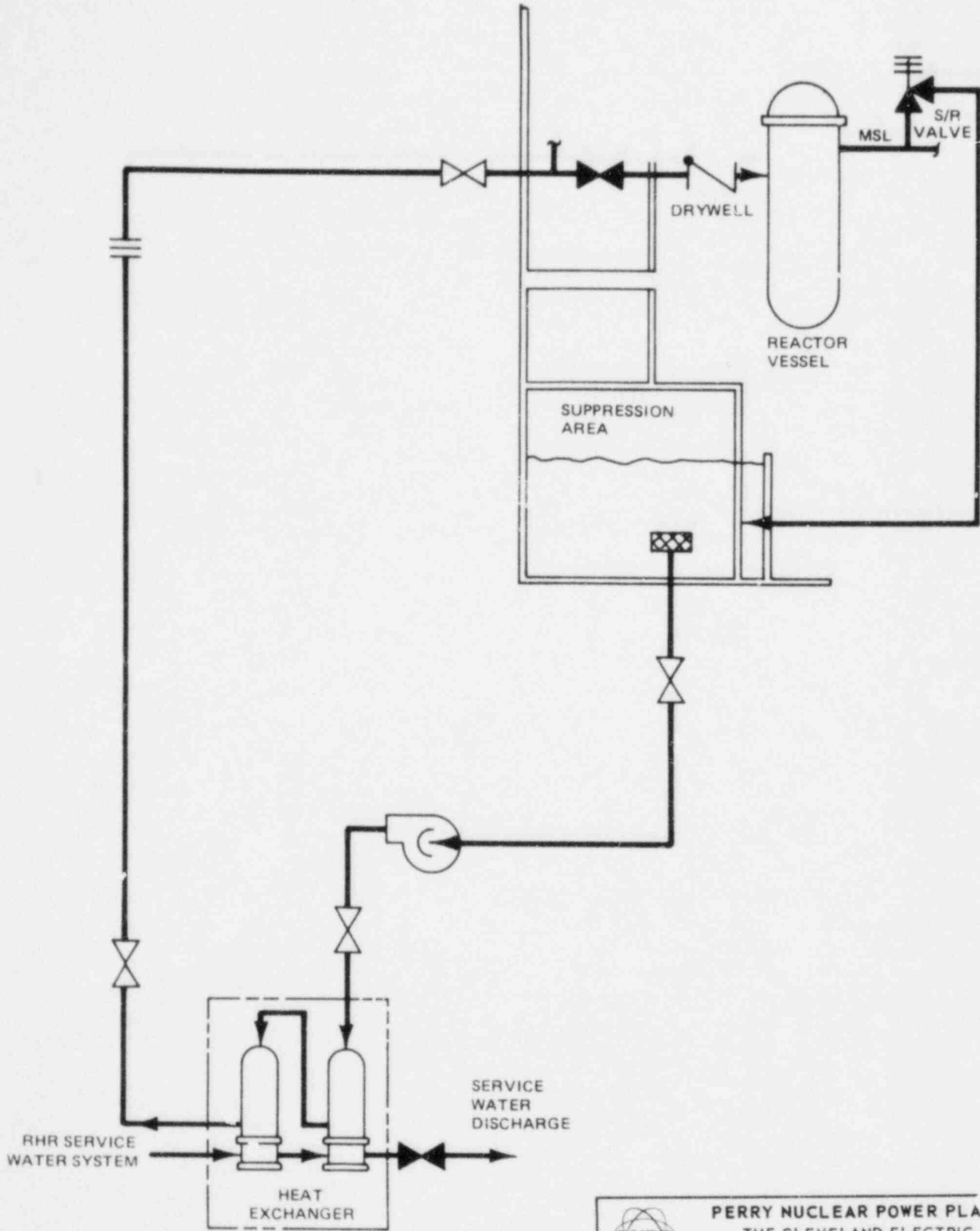
Activity C1 Alternate Shutdown
Cooling Path Utilizing RHR Loop B

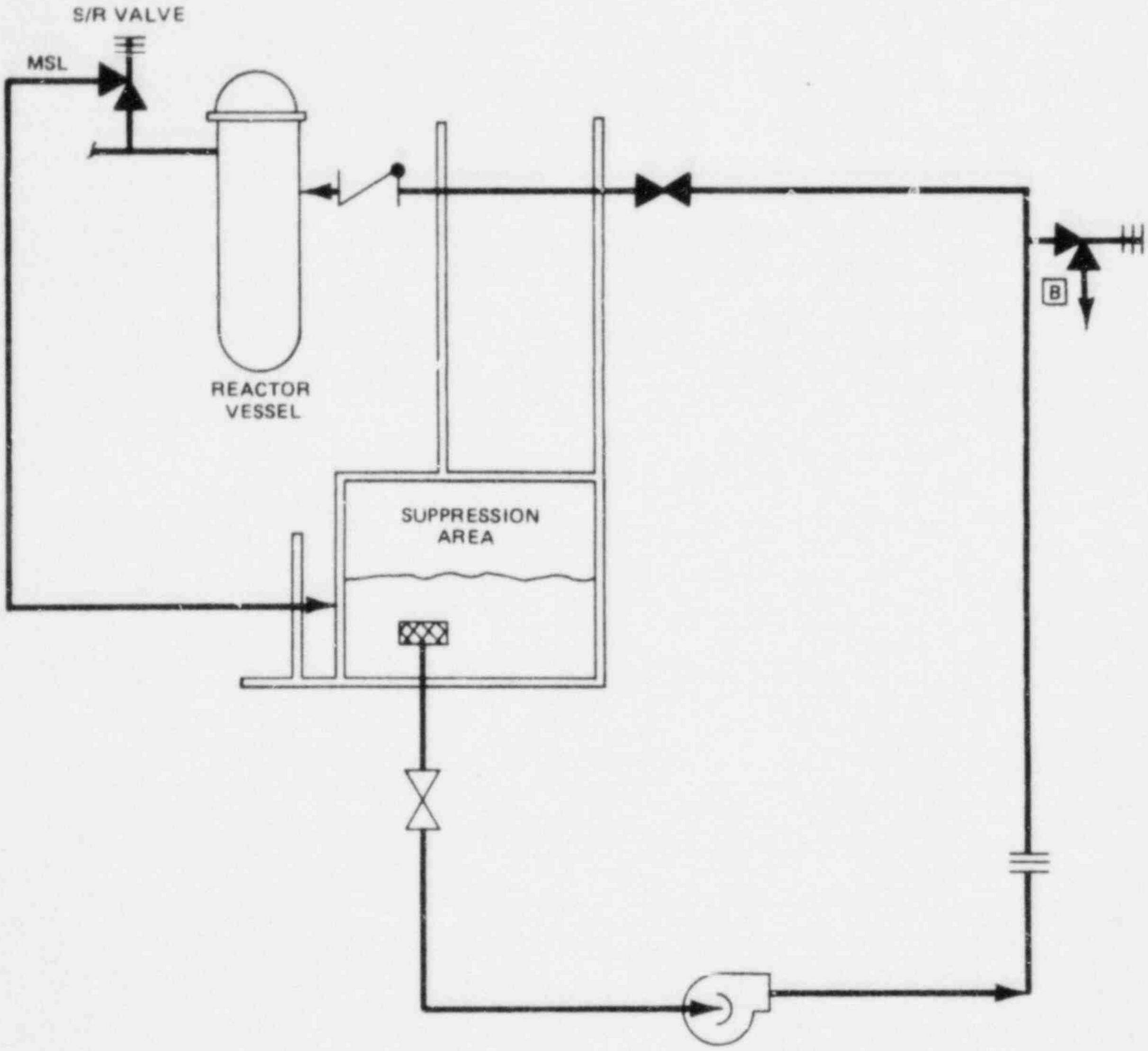
Figure 15.2-13 (Sheet 1 of 2)



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Activity C1 Alternate Shutdown Cooling Path Utilizing RHR Loop B

Figure 15.2-13 (Sheet 2 of 2)



PERRY NUCLEAR POWER PLANT
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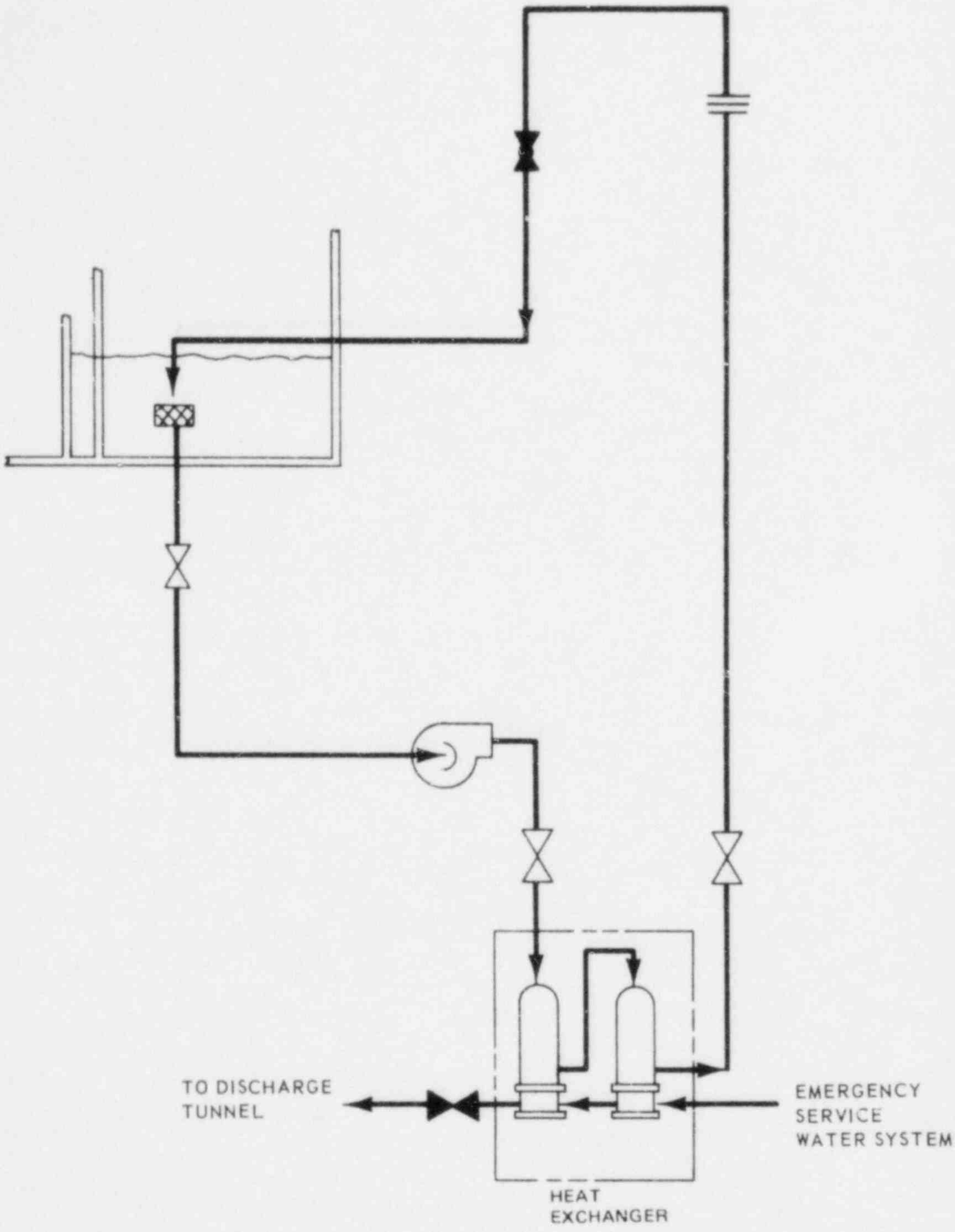
RHR Loop C

Figure 15.2-14



PERRY NUCLEAR POWER PLANT
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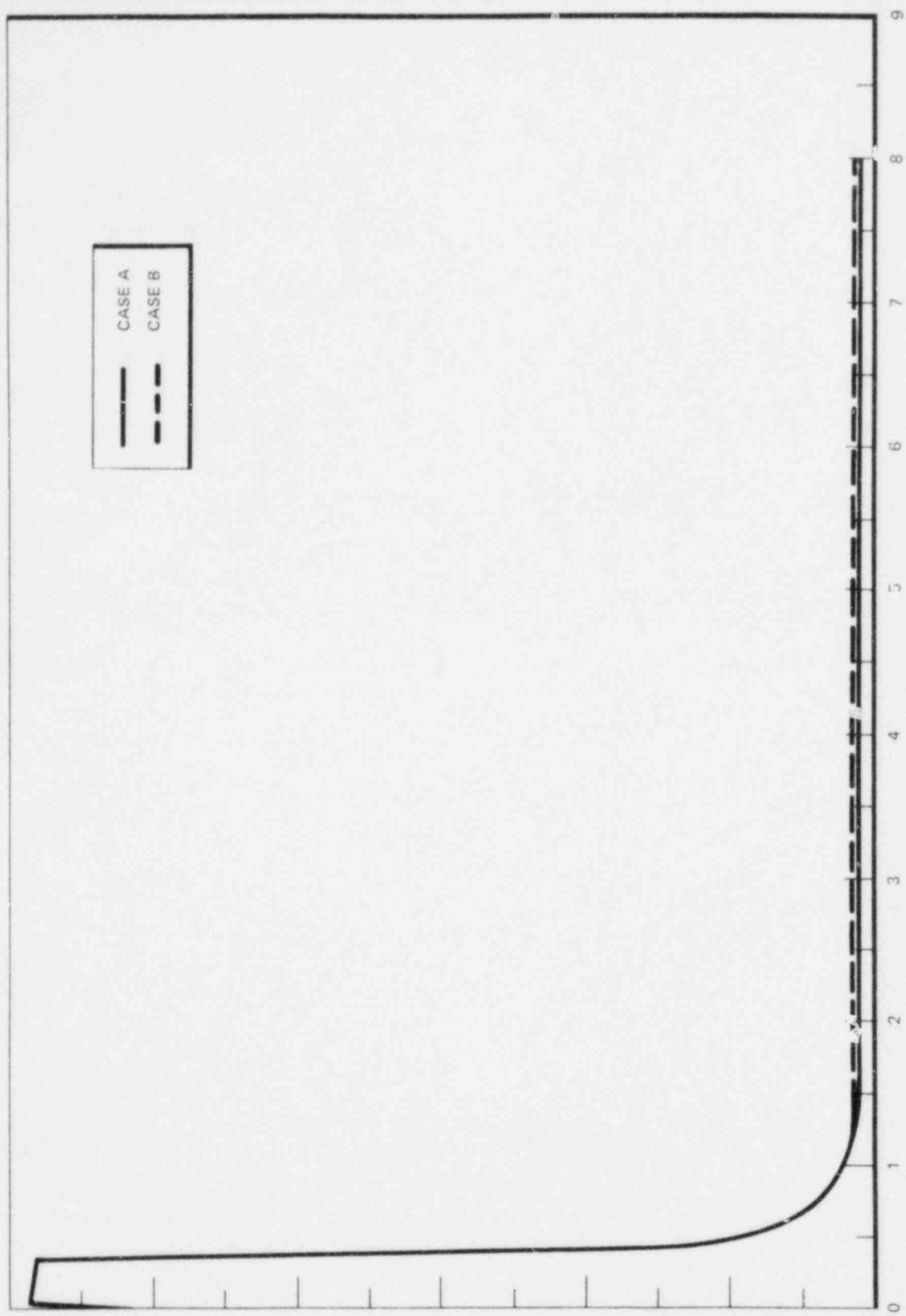
RHR Loop B (Suppression Pool
Cooling Model)

Figure 15.2-15



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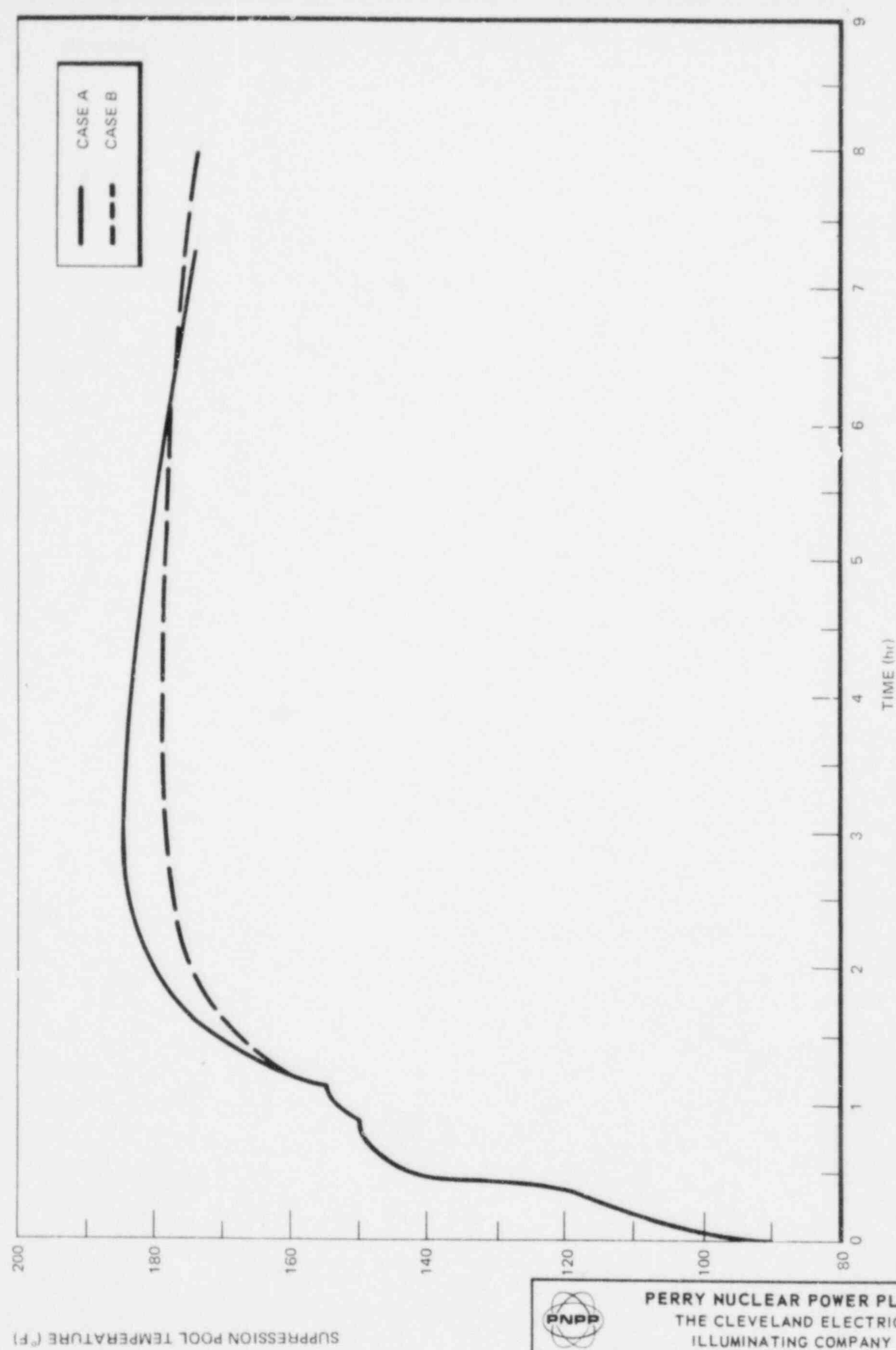
Reactor Vessel Pressure
versus Time

Figure 15.2-16



PERRY NUCLEAR POWER PLANT
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Suppression Pool Temperature
versus Time

Figure 15.2-17

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 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:

a. Reactor vessel water level L2 set point trip.

b. Failure to scram high pressure set point trip.

c. Motor branch circuit over-current protection.

d. Motor overload protection.

e. Suction block or discharge valves not fully open.

f. Auto transfer sequence incomplete (40 sec).

Random tripping will occur in response to:

a. Operator error.

b. Loss of electrical power source to the pumps.

c. 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 conform 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, or by restart of a feedwater pump, 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 per Section 15.3.1.2.2.1, no additional effects of single failures need be discussed. If additional SACF or SOE are assumed (for envelope purposes the other pump is assumed tripped)

then the following two pump trip analysis is provided. Refer to Appendix 15A for specific details.

15.3.1.2.3.2 Trip of Two Recirculation Pumps

Table 15.3-2 lists the vessel level (L^8) 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 specific details.

15.3.1.3 <u>Core and System Performance</u>

15.3.1.3.1 Mathematical Model

The nonlinear, dynamic model described briefly in Section 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 15.0-1.

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 5.6 seconds. However, the ratio of diffuser mass flow to pump mass flow in the active jet pumps increases considerably and produces approximately 130 percent of normal

diffuser flow and 54 percent of rated core flow. MCPR remains above the safety limit and 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 shows graphically 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 steam line 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 drive line 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 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 <u>Barrier Performance</u>

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 peak pressures stay well below the 1,375 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 Section 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.

15.3.2 RECIRCULATION FLOW CONTROL FAILURE - DECREASING FLOW

15.3.2.1 Identification of Causes and Frequency Classification

15.3.2.1.1 Identification of Causes

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 percent per second. 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 capacity 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

Since no scram occurs, no immediate operator action is required. 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. No protection system operation is required.

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 Section 15.3.1.2.3.2. The fast closure of two recirculation valves instead of one would be the envelope case for the additional SCF or SOE. Refer to Appendix 15A for details.

15.3.2.3 Core and System Performance

15.3.2.3.1 Mathematical Model

The nonlinear dynamic model described briefly in Section 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 15.0-1.

The least negative void coefficient in Table 15.0-1 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 percent per second 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 percent per second. Recirculation loop flow is allowed to decrease to approximately 25 percent of rated. 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. It is similar in most respects to the trip of one recirculation pump transient. Design of the hydraulic limit on maximum valve stroking rate is intended to make this transient event less severe than the one pump trip, and 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 Section 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

Figure 15.3-3 indicates a reduction in system pressure and no increases are expected.

15.3.2.4.2 Fast Closure of Two Recirculation Valves

The narrow-range level rises to the high level trip set point 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 1,151 psig, well below the ASME code limit. At approximately 28 seconds, the wide range level falls to the low water level trip set point, causing trip of the recirculation pumps and initiation of HPCS and RCIC system. However, there is a delay of up to 30 seconds 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 Section 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.

15.3.3 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 event. It has been evaluated as having a very mild accident in relation to other design basis accidents such as the LOCA. The analysis has been conducted with consideration to a single or two loop operation.

Refer to Chapter 5.1 for specific mechanical considerations and Chapter 8 for electrical aspects.

The seizure event postulated ordinarily 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 Section 15.3.1.2.3.2.

Refer to 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 Section 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 15.0-1.

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 percent NB rated steamflow. Also, the reactor is assumed to be operating at thermally limited conditions.

The void coefficient is adjusted to the most conservative value, that is, the least negative value in Table 15.0-1.

15.3.3.3.3 Results

Figure 15.3-5 presents 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 3.5 seconds 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 Section 15.2.4.5. Therefore, the radiological exposures noted in Section 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 design basis accident event. It has been evaluated as a very mild accident in relation to other design basis accidents 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 8 for electrical aspects.

This postulated event is bounded by the more limiting case of recirculation pump seizure. Quantitative results for this more limiting case are presented in Section 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 as discussed in Section 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 (L8), 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. 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 of the 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 Section 15.3.1.2.3.2.

Assumption of single equipment failure (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.

Refer to appendix 15A for more details.

15.3.4.3 Core and System Performance

The severity of this pump shaft break event is bounded by the pump seizure event as described in Section 15.3.3. This can be easily demonstrated by consideration of those two events as discussed in subsection below. Since this event is less limiting than the event described in Section 15.3.3 only qualitative evaluation is provided. Therefore no discussion of mathematical model, input parameters, and consideration 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 (see Section 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 Section 15.2.4.5. Therefore, the radiological exposures noted in Section 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.
5.6	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.
4.1	Vessel water level (L8) trip initiates scram, turbine trip and feedwater pump trip.
4.2	Turbine trip initiates bypass operation.
6.6	Safety/relief valves open due to high pressure.
11.6	Safety/relief valves close.
25.0	Vessel water level (L2) set point reached.
55	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.
30	Core flow and power approach new equilibrium conditions.

TABLE 15.3-4

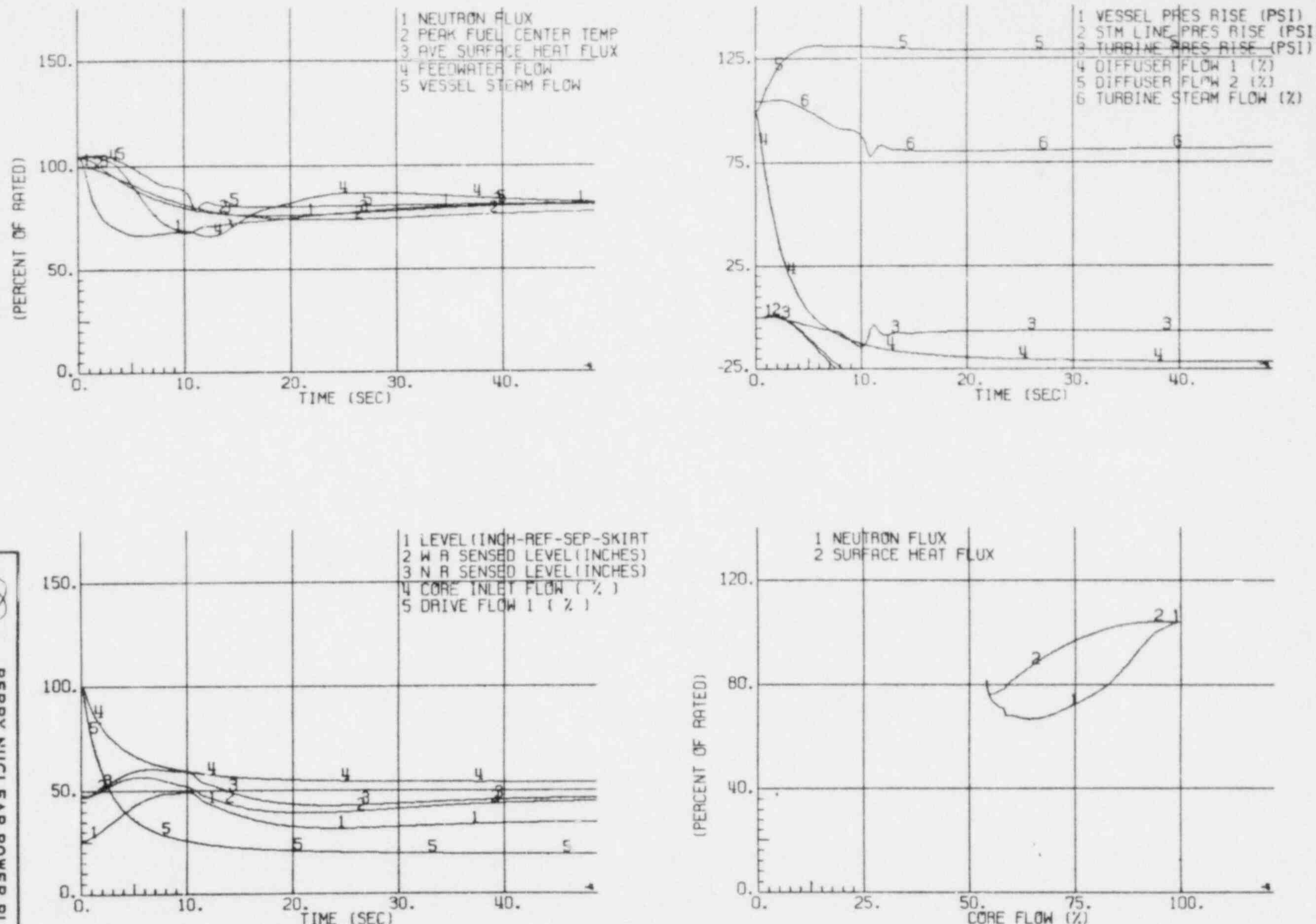
SEQUENCE OF EVENTS FOR FIGURE 15.3-4

Time-sec.	Event
0	Initiate fast closure of both main recirculation valves.
6.5	Vessel level (L8) trip initiates scram and turbine trip.
6.5	Feedwater pumps tripped off.
6.6	Turbine trip initiates bypass operation.
28.1	Vessel water level reaches Level 2 set point.
58.1	HPCS and RCIC flow enters vessel (not simulated).

TABLE 15.3-5

SEQUENCE OF EVENTS FOR FIGURE 15.3-5

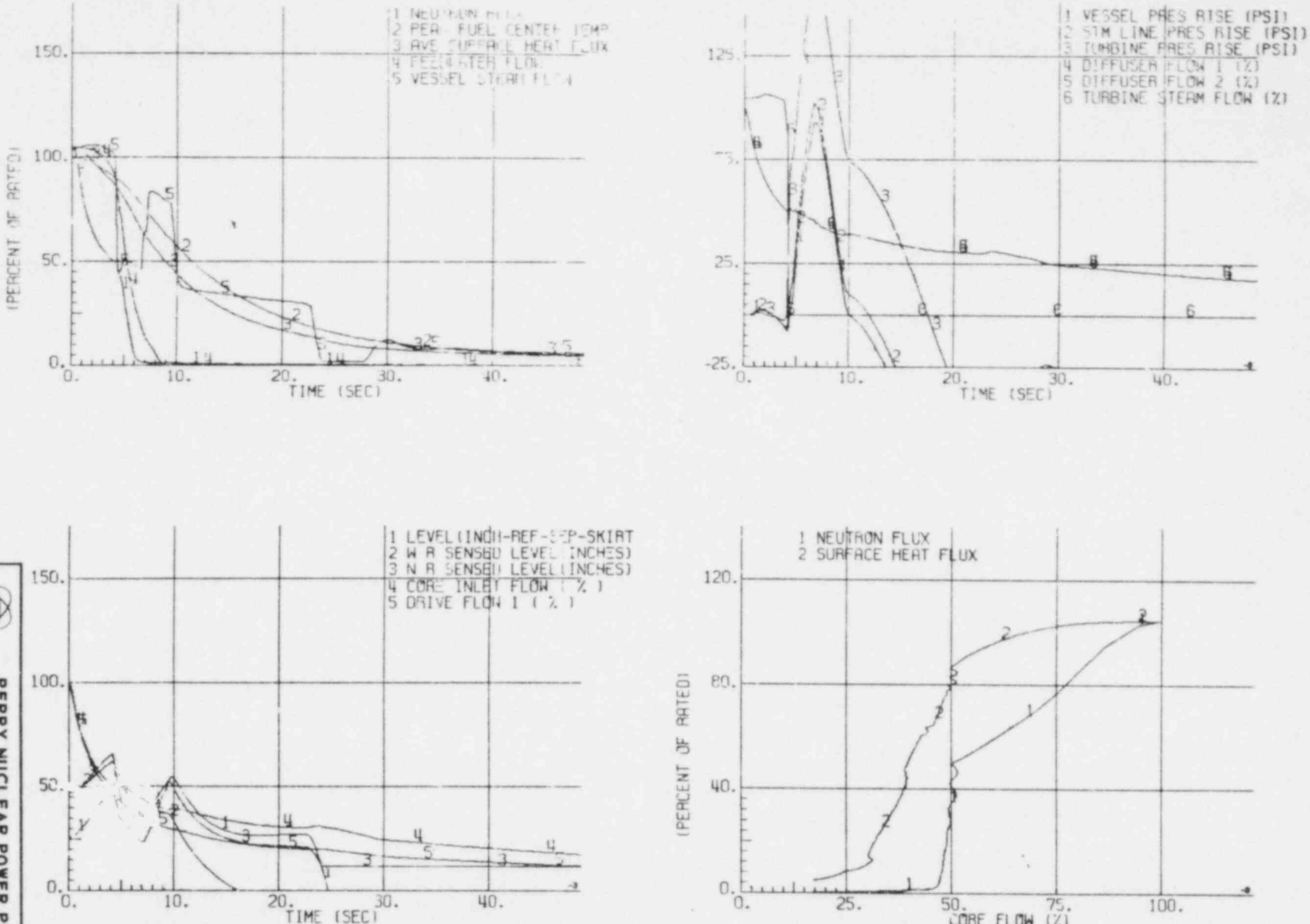
Time-sec.	Event
0	Single pump seizure was initiated.
0.6	Jet pump diffuser flow reverses in seized loop.
3.4	Vessel level (L8) trip initiates scram.
3.4	Vessel level (L8) trip initiates turbine trip.
3.4	Feedwater pumps are tripped off.
3.5	Turbine trip initiates bypass operation.
3.5	Turbine trip initiates recirculation pumps trip.
6.1	Safety/relief valves open due to high pressure.
11.3	Safety/relief valves close.
23.2	Main bypass valves close to regain pressure regulator control.
24.5	Vessel water level reaches Level 2 setpoint.
54.5 (est)	HPCS/RCIC flow enters the vessel (not simulated).



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Trip of One Recirculation
Pump Motor

Figure 15.3-1

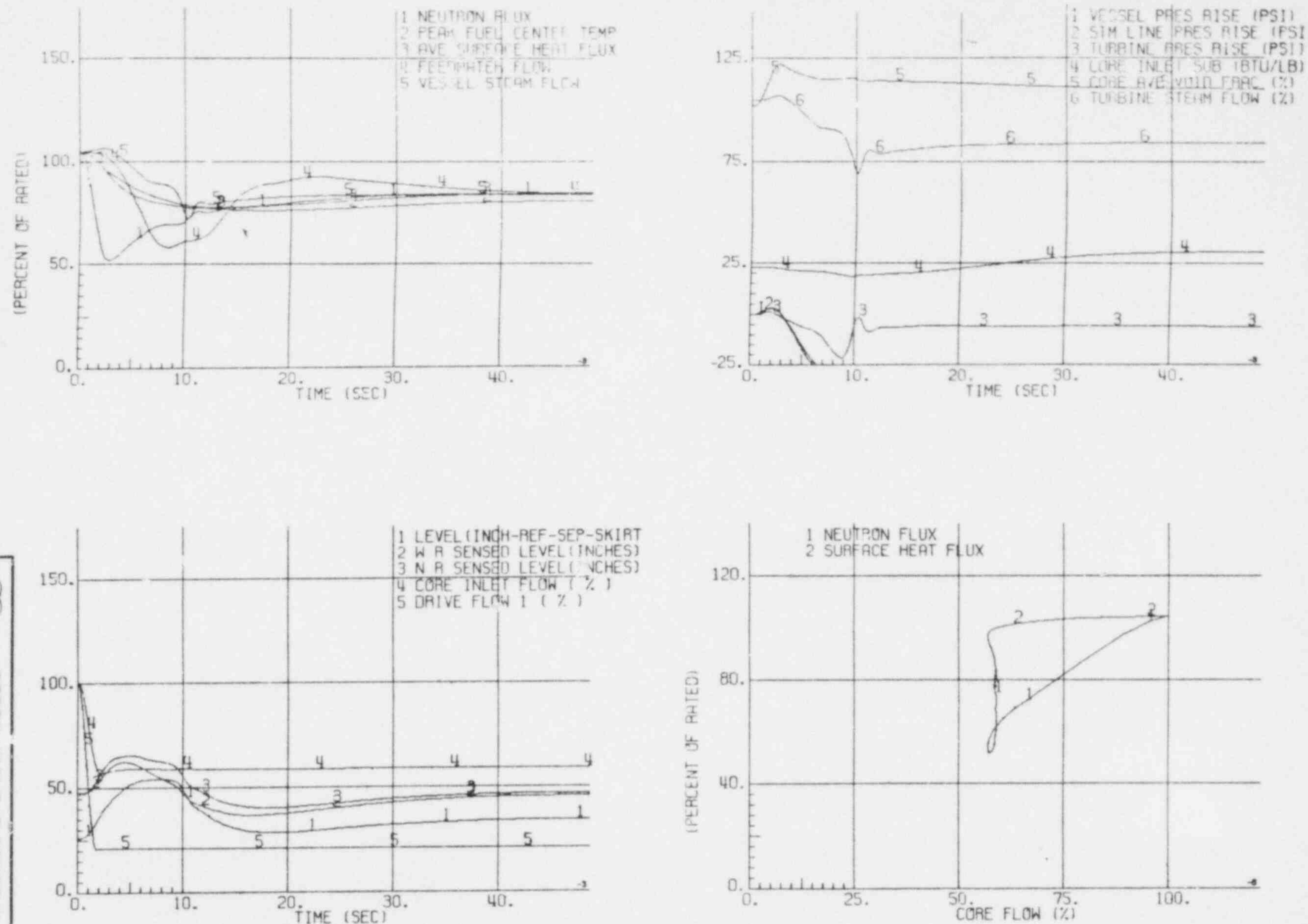


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Trip of Both Recirculation Pump Motors

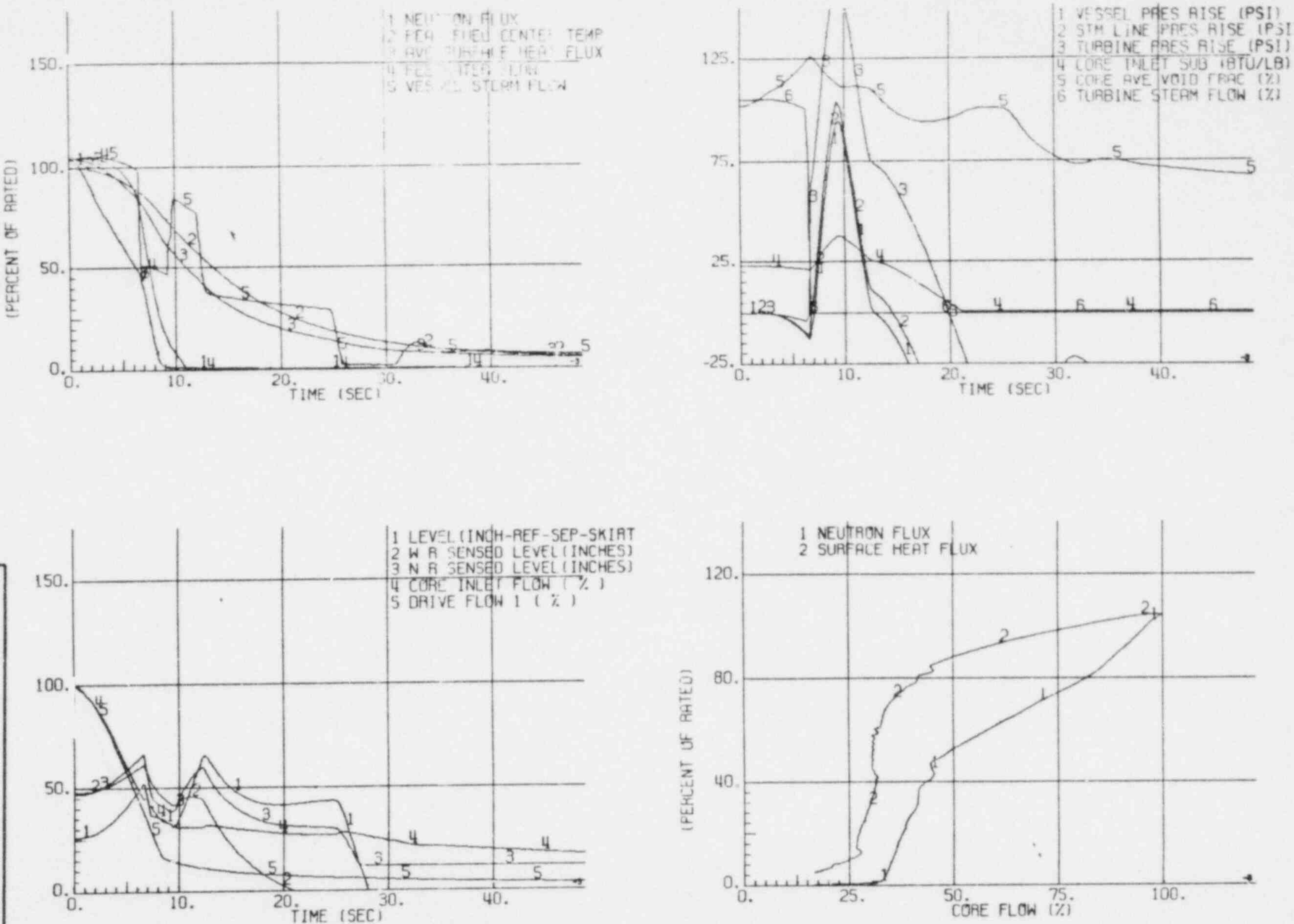
Figure 15.3-2



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Fast Closure of One Main Recirculation Valve at 60% Per Second

Figure 15.3-3

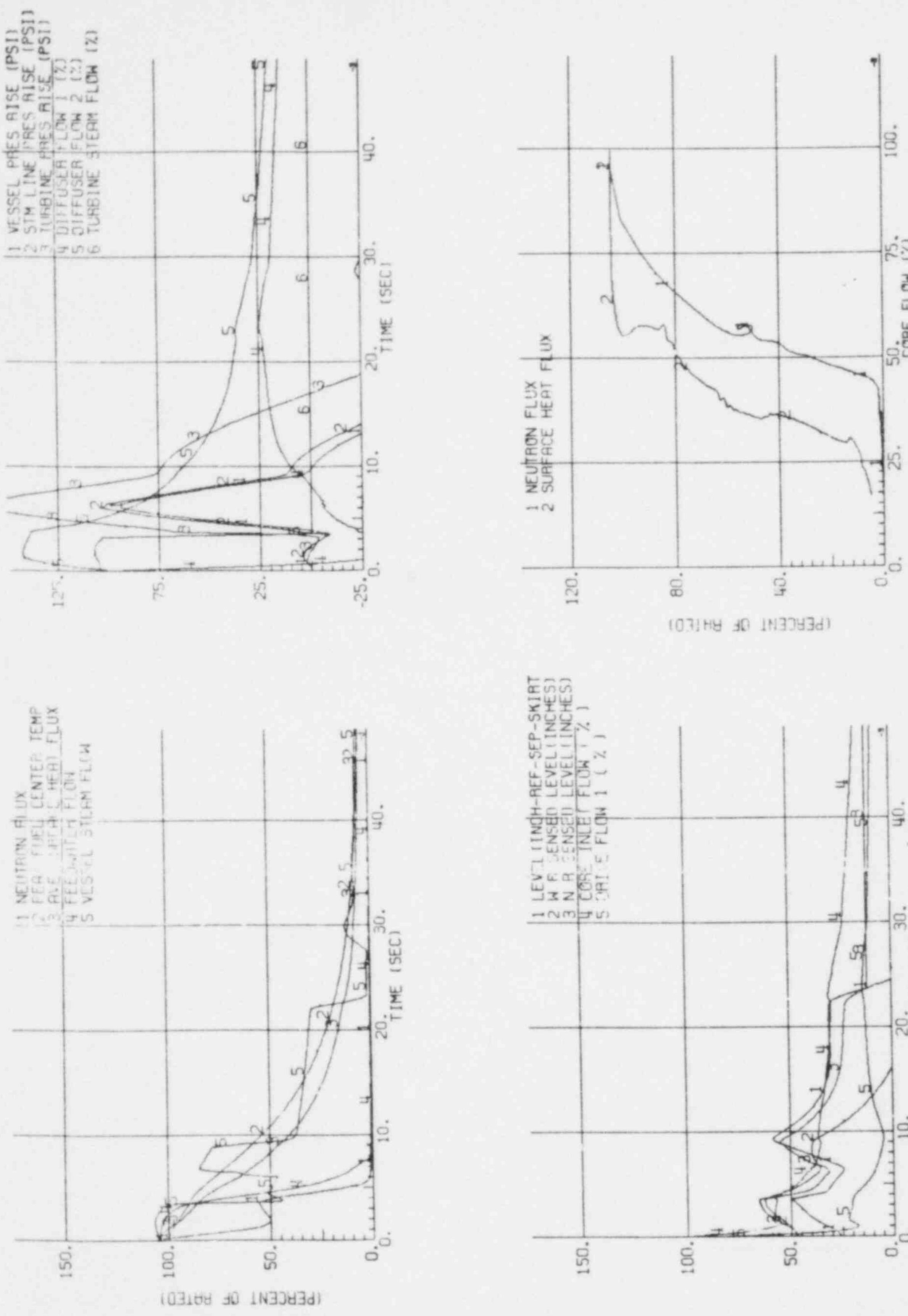


PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Fast Closure of Both Main Recirculation Valves at 11% Per Second

Figure 15.3-4



PNPP

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Seizure of One Recirculation Pump

Figure 15.3-5

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 inadvertant 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 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. This precludes any hazardous condition.

15.4.1.1.2.5 Identification of Operator Actions

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

15.4.1.1.2.6 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 equipment failure (SEF) or single operator error (SOE). The necessary safety actions are taken (e.g., rod block or scram) automatically prior to limit violation. Refer to Appendix 15A for details.

15.4.1.1.3 Core and System Performances

Since the probability 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 Section 4.3.2 for a description of the methods and results of

the shutdown margin analysis.) Additional reactivity insertion is precluded by interlocks. (See Section 7.6) As a result, no radioactive material is ever released from the fuel making it unnecessary to assess any radiological consequences.

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.

15.4.1.1.4 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 Rod Withdrawal During Reactor Startup

15.4.1.2.1 Identification of Causes and Frequency Classification

The probability of the initial causes or error of this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further single failures postulated for this event is even considerably 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 non-acknowledgement 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

Control rod withdrawal errors are not considered credible in the startup and low power ranges. The RCIS prevents the operator from selecting and withdrawing an out-of-sequence control rod.

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 percent to 75 percent control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75 percent rod density to the preset power level. Since only in-sequence control rods can be withdrawn in the 100 percent - 75 percent control rod density and control rods are withdrawn in the banked position mode from the 75 percent control rod density point to the preset power level, there is no basis for the continuous control rod withdrawal error in the startup and low power range. See Section 15.4.2 for description of continuous control rod withdrawal above the preset power level. 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 SEF or SOE, the necessary safety actions are automatically taken (e.g., rod blocks) prior to any limit violation. Refer to 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

An 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 considered 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 continuously until the RWL inhibits further withdrawal. The RWL utilizes rod position indications of the selected rod as input.

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[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 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 Section 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 gangs improperly 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 RWE's initiated from rated conditions is less than 4 percent 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 Sections 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:

a. Adjust rod pattern as necessary for new power level following idle loop start.

b. Determine that the idle recirculation pump suction and discharge block valves are open and that the flow control valve in the idle loop is at minimum position and, if not, place them in this configuration.

c. Readjust flow of the running loop downward to less than half of the rated flow.

d. Determine that the temperature difference between the two loops is no more than 50°F apart.

e. Start the idle loop pump and adjust flow to match the adjacent loop flow. Monitor reactor power.

f. 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 Section 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 15.0-1.

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 85 percent of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 33 percent 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. The motor approaches synchronous speed in approximately 3 seconds 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 81 percent of rated before decreasing after the cold water washed out of the loop at about 19 seconds. 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 neturon 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. See 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 or neutron flux controller can cause an increase in the core coolant flow rate. Failure within 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:

a. Transfer flow control to manual and reduce flow to minimum.

b. 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 been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

a. Observe that all rods are in.

b. Check the reactor water level and maintain above low level (L2) trip to prevent MSLIVs from isolating.

c. Switch the reactor mode switch to the "startup" position.

d. Continue to maintain vacuum and turbine seals.

e. Transfer the recirculation flow controller to the manual position and reduce set point to zero.

f. Survey maintenance requirements and complete the scram report.

g. Monitor the turbine coastdown and auxiliary systems.

h. Establish a restart of the reactor per the normal procedure.

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

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 in view of the fact that 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 Section 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 tabulated in Table 15.0-1.

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 percent NB rated power and 33 percent 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 percent per second.

Maximum stroking rate of a single recirculation loop valve for a loop controller failure is limited by hydraulics to 30 percent per second.

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 percent per second.

The rapid increase in core flow causes a sharp rise in neutron flux initiating a reactor scram at approximately 1.4 seconds. The peak neutron flux reached was 215 percent of NB rated value, while the accompanying average fuel surface heat flux reaches 71 percent of NB rated at approximately 2.3 seconds. MCPR remains considerably above the safety limit and the average fuel temperature increases only 104°F. Reactor pressure is discussed in Section 15.4.5.4.

15.4.5.3.3.2 Fast Opening of Two Recirculation Valves

Figure 15.4-3 illustrates the failure where both recirculation loop main valves are opened at a maximum stroking rate of 11 percent per second. It is very similar to the above transient. Flux scram occurs at approximately 1.9 seconds, peaking at 149 percent of NB rated while the average surface heat

flux reaches 66 percent of NB rated at approximately 2.6 seconds. MCPR remains considerably above the safety limit and average fuel center temperature increases 78°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 as shown in 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 as shown in 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 initial 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 be overlooked and also put in an incorrect location. Third, the misplaced bundles would have to be overlooked during the core verification performed following initial core loading.

15.4.7.1.2 Frequency of Occurrence

This 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 infrequent incident based on the following data.

Expected Frequency: .004 events/operating cycle

The above number is based upon past experience. The only misloading events that have occurred in the past were in reload cores where only two errors are necessary. Therefore, the frequency of occurrence for initial cores is even lower since three errors must occur concurrently.

15.4.7.2 Sequence of Events and Systems Operation

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

Fuel loading errors, undetected by in-core instrumentation following fueling operations, may result in undetected reductions in thermal margins during power operations. No detection is assumed, and therefore, no corrective operator action or automatic protection system functioning occurs.

15.4.7.2.1 Effect of Single Failure and Operator Errors

This analysis already represents the worst case (i.e., operation of a misplaced bundle with three SEF or SOE) and there are no further operator errors which can make the event results any worse. It is felt that this section is not applicable to this event. Refer to Appendix 15A for further details.

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.

15.4.7.3.2 Input Parameters and Initial Conditions

The initial core consists of bundles with average enrichments that are high, medium, or low with correspondingly different gadolinia concentrations. The fuel bundle loading error with the severest consequences occurs at beginning-of-cycle (BOC) when a low-enriched bundle (which should be loaded at the periphery) is interchanged with a high-enriched bundle located adjacent to a LPRM and predicted to have the highest LHGR and/or lowest CPR in the core. After the loading error is made and has gone undetected, it is assumed for purposes of conservatism that the operator uses a control pattern that places the limiting bundle in the four bundle array containing the misplaced bundle, on design thermal limits as recorded by the LPRM.

As a result of loading the low-enriched bundle in an improper location, the reading of the adjacent LPRM decreases. Consequently, because there are no instruments in the 3 mirror images of this four-bundle-array, the operator believes these arrays are operating at the same power as the instrumented one, when in fact they are not (since no loading error occurred in these quadrants). As a result of placing the instrumented array on limits, the 3 mirror-image arrays exceed the design limit. By replacing the high-enriched bundle with the greatest power peaking, by the low-enriched bundle, it is assured that the difference in power peaking between the instrumented and the non-instrumented arrays is maximum, or rather, that the ΔCPR and ΔLHGR is the upper bound for this error.

Other input parameters assumed are given in Table 15.4-7 and Figure 15.4-4.

15.4.7.3.3 Results

Results of analyzing the worst fuel bundle loading error are reported in Table 15.4-8. As can be seen, MCPR remains well above the point where boiling transition would be expected to occur, and the MLHGR does not exceed the 1 percent plastic strain limit for the clad. Therefore, no fuel damage occurs as a result of this event.

15.4.7.3.4 Considerations of Uncertainties

In order to assure the conservatism of this analysis, major input parameters are taken as a worst case, i.e., the bundle is placed in location with the highest LHGR and/or the lowest CPR in the core and the bundle is operating on design thermal limits. This assures that the minimum CPR and maximum LHGR are conservatively bounded for the error.

15.4.7.4 Barrier Performance

An evaluation of the barrier performance was not made for this event since it is a 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 control rod drive system assemblies. The control rod drive 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 is the result of a postulated event in which a high worth control rod, within the constraints of the banked position RCIS, 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 control rod drive position. This results in the removal of large negative reactivity from the core and results in a localized power excursion.

A more detailed discussion is given in Reference 3.

15.4.9.1.2 Frequency of 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 control rod drop accident (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 would 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 rod control and information system (RCIS) limits the worth of any control rod which could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100 to 75 percent rod density range, and from the 75 percent rod density point to the preset power level the RCIS will only allow bank position mode rod withdrawals or insertions.

The RCIS used redundant input to provide absolute assurance on control rod drive position. If either of the diverse input 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 RCIS and APRM scram. The RCIS is designed as a redundant system network and therefore together provide 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 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 control rod drop accident are described in detail in References 3, 4, and 5. They are considered to provide a realistic yet conservative assessment of the associated consequences. The bounding analyses are presented in Reference 6. Compliance checks are made to verify that the maximum rod worth does not exceed 1 percent Δk.

If this criteria is not met, then the bounding analyses is performed. The rod worths are determined using the BWR simulator. Detailed evaluations, if necessary, are made using the methods described in References 3, 4, and 5.

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 percent 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 percent 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 for any plant condition. The data presented in Section 15.4.9.3.3 show the maximum 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 calories per gram design limit.

The results of the compliance check calculation, as shown in the 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 calories per gram peak fuel enthalpy[(3), (4), (5)]. The conclusion is that the 280 calories per gram design limit is not exceeded and the assumed failure of 770 pins 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:

a. 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 10 CFR 100 guidelines. This analysis is referred to as the "Design Basis Analysis".

b. The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This analysis is referred to as the "Realistic Analysis."

A schematic of the leakage path is shown in Figure 15.4-5.

15.4.9.5.1 Design Basis Analysis

The design basis analysis is based on the NRC's Standard Review Plan 15.4.9[7]. The specific models, assumptions and the program used for computer evaluation are described in Reference 8. 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 2,804°C) is estimated to be 0.0077.

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

A maximum equilibrium inventory of fission products in the core is based on 1,000 days of continuous operation at 3,758 MWt. No delay time is considered between departure from the above power condition and the initiation of the accident.

15.4.9.5.1.2 Fission Product Transport to the Environment

The transport pathway is shown in Figure 15.4-5 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 percent of the noble gases and 10 percent of the iodines are assumed to be carried to the condenser before MSLIV closure is complete.

Of the activity reaching the condenser, 100 percent of the noble gases and 10 percent 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 percent 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.

15.4.9.5.1.3 Results

The calculated exposures from the design basis analysis are presented in Table 15.4-14 and are well within the guidelines of 10 CFR 100.

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 and assumptions used for this evaluation are described in Section 15.0.3.5. Specific values of parameters used in the evaluation are presented in Table 15.4-10.

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:

a. The reactor has been operating at design power for 1,000 days until 30 minutes 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 minutes of the departure from design power. The 30 minute 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.

b. An average of 1.8 percent of the noble gas activity and 0.32 percent 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[8].

c. 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:

a. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and 5 percent steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5 percent 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 steam lines 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 steam line isolation valves to achieve full closure.

b. The main steam line isolation valves are assumed to receive an automatic closure signal 0.5 seconds after detection of high radiation in the main steam lines and to be fully closed at 5 seconds from the receipt of the steam line radiation monitors. The total amount of fission product activity transported to the condenser before the steam lines are isolated is, therefore, governed by the 5.5 seconds isolation time and the conditions in a. above.

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

d. The mass ratio of the halogen concentration in steam, to that of the water is assumed to be 2 percent.

e. Fission product plate-out is neglected in the reactor vessel, main steam lines, turbine and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent 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 15.4-15.

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

a. The leak rate out of the condenser is 0.5 percent of the combined condenser and turbine free volume 1.26 $\times 10^5$ ft^3 per day.

b. The activity released from the condenser becomes airborne in the turbine building. All activity leaking from the condenser is assumed to leak directly to the environment without mixing in the turbine building volume.

15.4.9.5.2.3 Results

The calculated offsite exposures for the realistic analysis are presented in Table 15.4-16 and demonstrate the wide margin of conservatism in the design basis analysis.

15.4.10 REFERENCES FOR SECTION 15.4

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

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

3. R. C. Stirn, et al., "Rod Drop Accident Analysis fo- Large BWRs", March 1972 (NEDO-10527).

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

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

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

7. N. R. Horton, W.A. Williams, K.W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors," March 1969, (APED-5756).

8. USNRC Standard Review Plan, NUREG-75/037, Washington, D.C., Rev. 1.

TABLE 15.4-1

SEQUENCE OF EVENTS - RWE IN POWER RANGE

Elapsed Time	
0	Core is assumed to be operating at rated conditions.
0	Operator selects and withdraws the maximum worth single control rod or rod gang.
~1 sec	The total core power and the local power in the vicinity of the control rod increase.
~6 sec	The RWL mode blocks withdrawal.
~25 sec	Reactor core stabilizes at slightly higher core power level.
~45 sec	Operator re-inserts control rod to reduce core power level.
~60 sec	Core stabilizes at rated conditions.

TABLE 15.4-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR ROD WITHDRAWAL TRANSIENT

	Rated Power
Core power, MWt	3,579
Average core exposure, MWd/t	5,000
Xenon state	No xenon
Average linear heat generation rate, kW/ft	5.927
Maximum linear heat generation rate, kW/ft	13.41
Location of maximum LHGR bundle	(21,30)
Minimum CPR	1.24
Location of minimum CPR bundle	(21,38)
Maximum worth control rod	(25,36)
Core coolant flow rate, lb/hr x 10^6	104
Core coolant inlet enthalpy, Btu/lb.	527.7
Core average steam volume fraction	0.5755
Reactor coolant pressure, average, psia	1,055
Control rod pattern for ganged withdrawal	Figure 15.4-6
Results for rod withdrawal transient	Figure 15.4-7

TABLE 15.4-3

SEQUENCE OF EVENTS FOR FIGURE 15.4-1

Time-sec	Event
0	Start pump motor.
0.73	Jet pump diffuser flows on started pump side become positive.
3.1	Pump motor at full speed and drive flow at about 21% of rated.
17.6 (est)	Last of cold water leaves recirculation drive loop.
18.0 (est)	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.
1.4	Reactor APRM high flux scram trip initiated.
3.4 (est)	Turbine control valves start to close upon falling turbine pressure.
12.0 (est)	Turbine control valves closed. Turbine pressure below pressure regulator set points.
>100	Reactor variables settle into new steady state.
14.0 (est)	Vessel water level reaches Level 2 (L2) setpoint.
14.0 (est)	Recirculation pump drive motors trip due to L2.

TABLE 15.4-5

SEQUENCE OF EVENTS FOR FIGURE 15.4-3

Time-Sec	Event
0	Initiate failure of master controller.
1.9	Reactor APRM high flux scram trip initiated.
3.7 (est)	Turbine control valves start to close upon falling turbine pressure.
9.0 (est)	Turbine control valves closed. Turbine pressure below pressure regulator set points.
>100 (est)	Reactor variables settle into new steady state.
9.0	Vessel water level drops to Level 2 (L2) setpoint.
9.0	Recirculation pump drive motors trip due to L2.

TABLE 15.4-6

SEQUENCE OF EVENTS FOR MISPLACED BUNDLE ACCIDENT

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

TABLE 15.4-7

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR FUEL BUNDLE LOADING ERROR

1.	Power, % rated	100
2.	Flow, % rated	100
3.	MCPR operating limit	1.23
4.	MLHGR operating limit, kW/ft	13.4
5.	Average core exposure, MWd/t	0.0
6.	Location of minimum CPR bundle	(15,40)
7.	Location of maximum LHGR bundle	(15,40)
8.	Control Rod Pattern	Figure 15.4-4

NOTE: Core conditions are assumed to be normal for a hot, operating core at BOC.

TABLE 15.4-8

MISPLACED BUNDLE ANALYSIS

Bundle (15,40) Replaced with Natural U Bundle

MCPR Operating Limit	MCPR With Misplaced Bundle	ΔCPR
1.23	1.116	-0.114

MLHGR Operating Limit	MLHGR With Misplaced Bundle	ΔMLHGR
13.4 kW/ft	14.30 kW/ft	0.90 kW/ft

TABLE 15.4-9

SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time	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 second	Reactor goes on a positive period and the initial power increase is terminated by the Doppler coefficient.
<1 second	APRM 120% power signal scrams reactor.
<5 seconds	Scram terminates accident.

TABLE 15.4-10

INPUT PARAMETERS AND INITIAL CONDITIONS
FOR ROD WORTH COMPLIANCE CALCULATION

1.	Reactor power, % rated	1.0
2.	Reactor flow, % rated	100
3.	Core average exposure, MWd/t	8.0
4.	Control rod fraction	~0.50
5.	Average fuel temperature, °C	286
6.	Average moderator temperature, °C	286
7.	Xenon state	No xenon

TABLE 15.4-11

INCREMENT WORTH OF THE MOST REACTIVE ROD USING BPWS[1]

Core Condition	Control Rod Group[2]	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase In keff
BOC-1	7	4	(22,31)	0→8	0.0034
Sequence A	7	8	(22,31)	0→12	0.0046
Rod groups 1-4 withdrawn	7	12	(22,31)	0→48	0.0052
Rod groups 5,6,8,9,10 fully inserted	7	48	(22,31)	0→48	0.0037

NOTES:

1. 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

2. For definition of rod groups, see Reference 5.

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	3758 MWt	3758 MWt
B. Burnup	NA	NA
C. Fuel damaged	770 rods	770 rods
D. Release of activity by nuclide	Table 15.4-13	Table 15.4-15
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident.	NA	NA
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	NA
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
III. Dispersion Data		
A. Boundary and LPZ distances (m)	863/4,002	863/4,002
B. X/Q's (sec/m^3) for time intervals of:		
(1) 0-2 hr - SB/LPZ	6.7-4/8.2-5	6.7-4/8.2-5
(2) 2-8 hr - LPZ	8.2-5	8.2-5

TABLE 15.4-12 (Cont'd)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Disperson Data (Cont'd)		
(3) 8-24 hr - LPZ	5.2-5	5.2-5
(4) 1-4 days - LPZ	1.9-5	1.9-5
(5) 4-30 days - LPZ	4.7-6	4.7-6
IV. Dose Data		
A. Method of dose calculation	15.0.3.5	15.0.3.5
B. Dose conversion assumptions	15.0.3.5	15.0.3.5
C. Peak activity concentrations in condenser	Table 15.4-13	Table 15.4-15
D. Doses	Table 15.4-14	Table 15.4-16

TABLE 15.4-13

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

Isotope	Activity Curies
I-131	1.6+3
I-132	2.7+3
I-133	2.5+3
I-134	4.6+3
I-135	3.7+3
Kr-83m	2.7+4
Kr-85	1.3+3
Kr-85m	6.4+4
Kr-87	1.4+5
Kr-88	1.9+5
Kr-89	2.5+5
Xe-131m	1.1+3
Xe-133m	4.2+4
Xe-133	2.6+5
Xe-135m	7.1+4
Xe-135	2.5+4
Xe-137	3.7+5
Xe-138	3.8+5

TABLE 15.4-14

CONTROL ROD DROP ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (REM)	Inhalation Dose (REM)
Exclusion area (863 Meters)	6.95-2	7.57-1
Low population zone (4,002 Meters)	2.02-2	9.58-1

TABLE 15.4-15

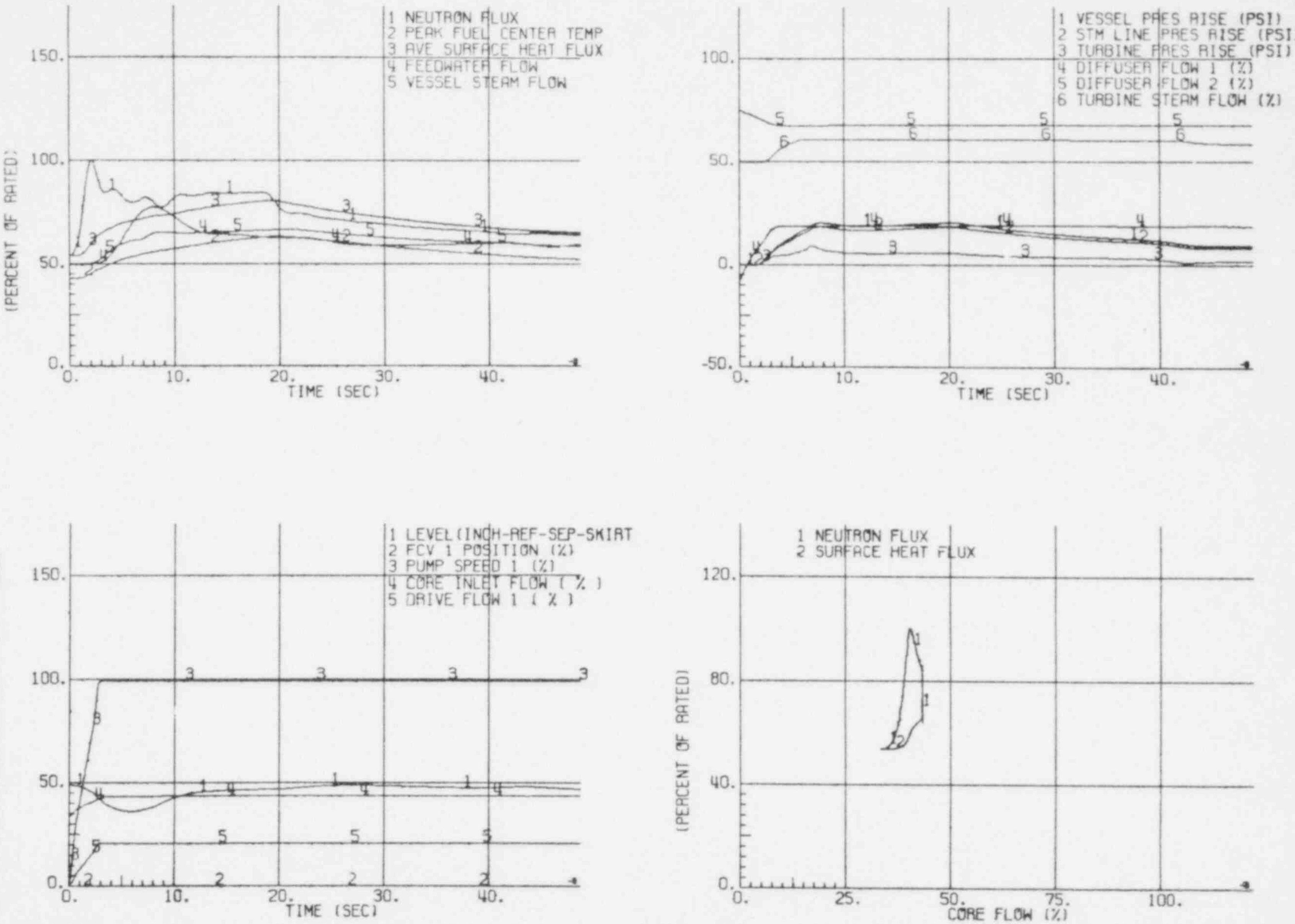
CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (CURIES)

Isotope	Activity Curies
I-131	1.9+2
I-132	3.4+1
I-133	9.2+1
I-134	3.2+1
I-135	7.5+1
Kr-83m	8.3+2
Kr-85	3.7+3
Kr-85m	4.7+3
Kr-87	4.1+3
Kr-88	9.2+3
Kr-89	3.2+0
Xe-131m	4.1+2
Xe-133m	5.8+3
Xe-133	6.0+4
Xe-135m	1.5+2
Xe-135	9.9+3
Xe-137	8.0+0
Xe-138	1.1+3

TABLE 15.4-16

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (REM)	Inhalation Dose (REM)
Exclusion area (863 Meters)	1.47-3	3.16-2
Low population zone (4,002 Meters)	7.90-4	4.98-2



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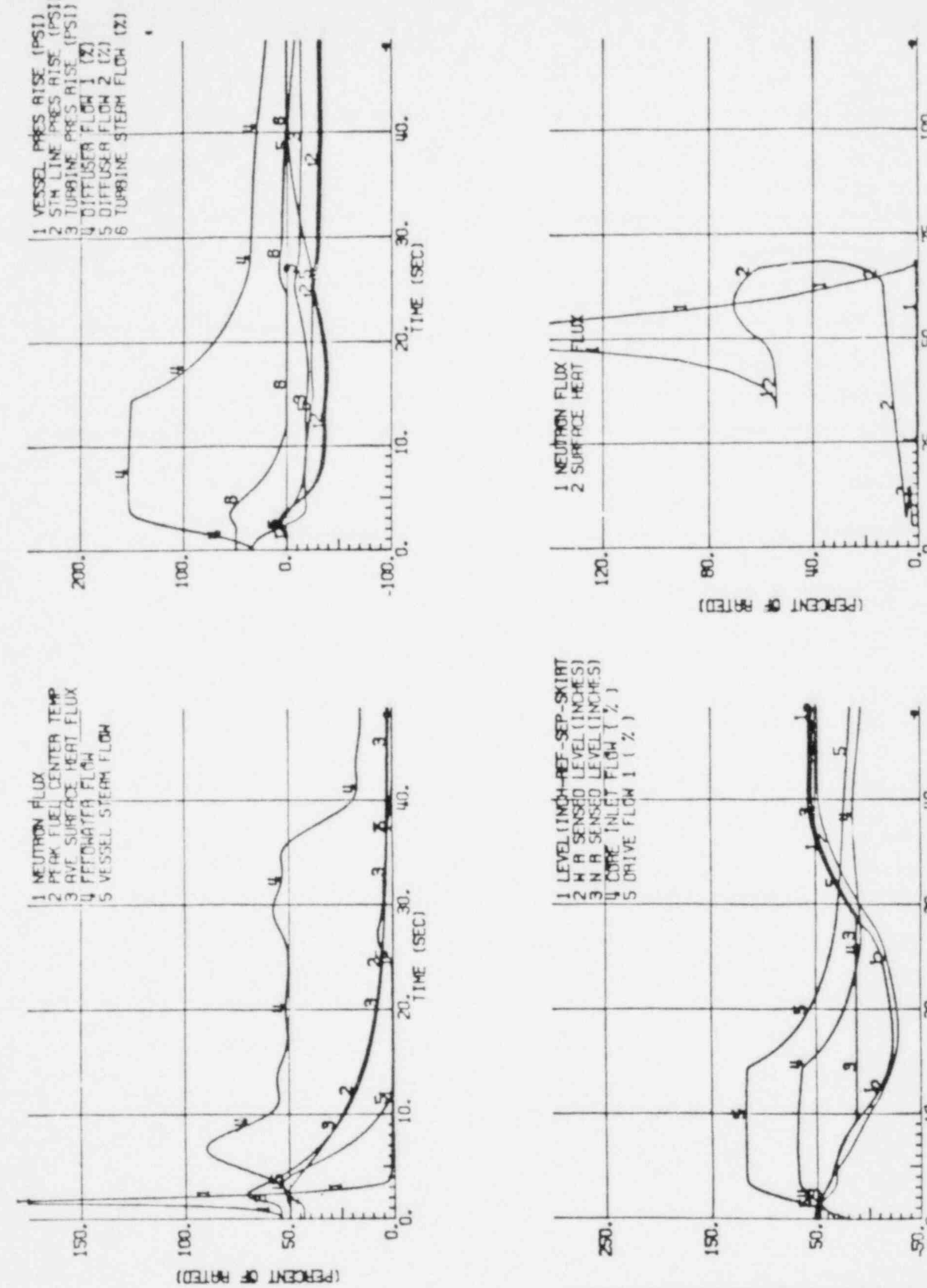
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Abnormal Startup of Idle
Recirculation Loop Pump

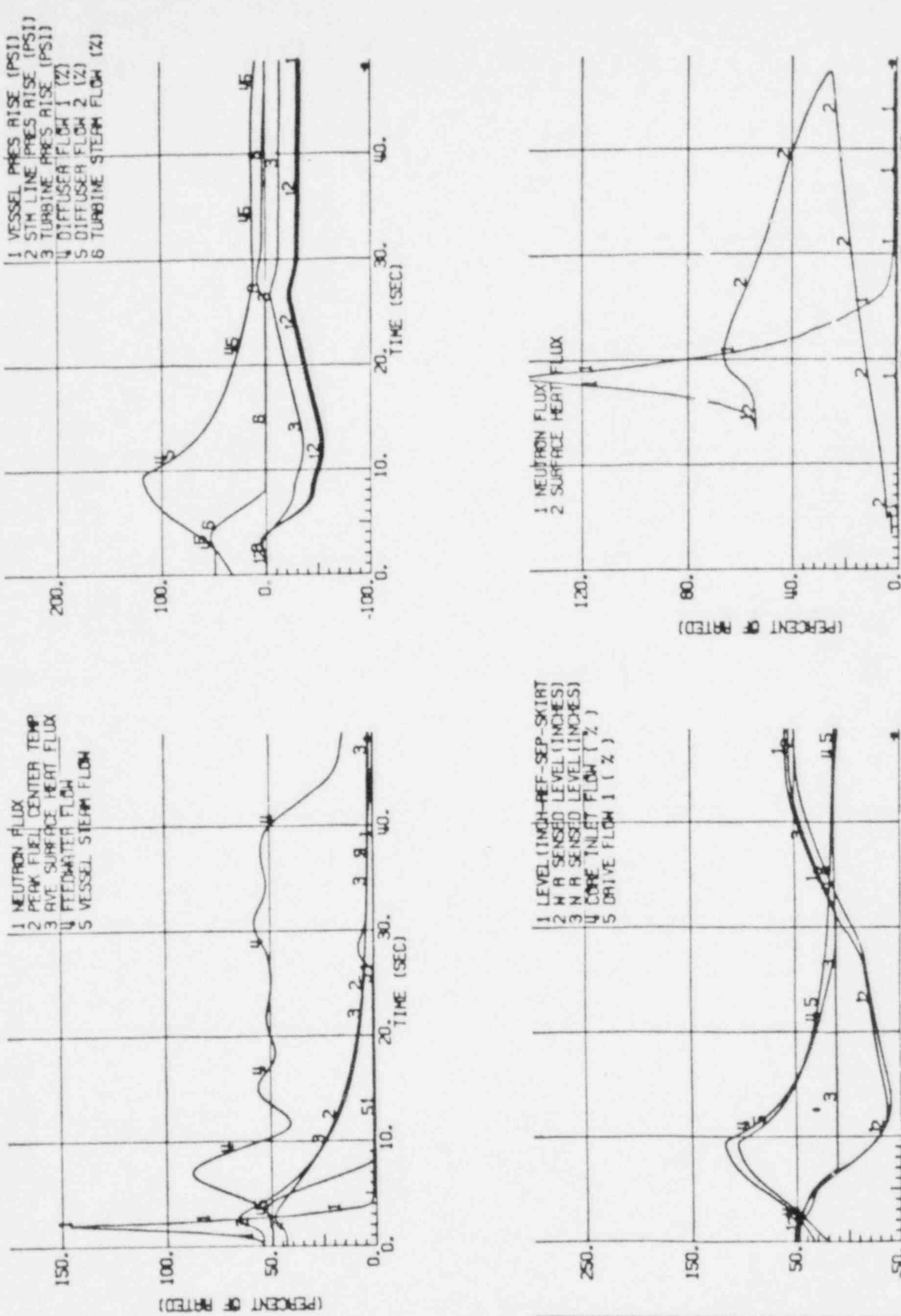
Figure 15.4-1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Fast Opening of One Main
Recirculation Loop Valve at
30% Per Second

Figure 15.4-2



PERRY NUCLEAR POWER PLANT
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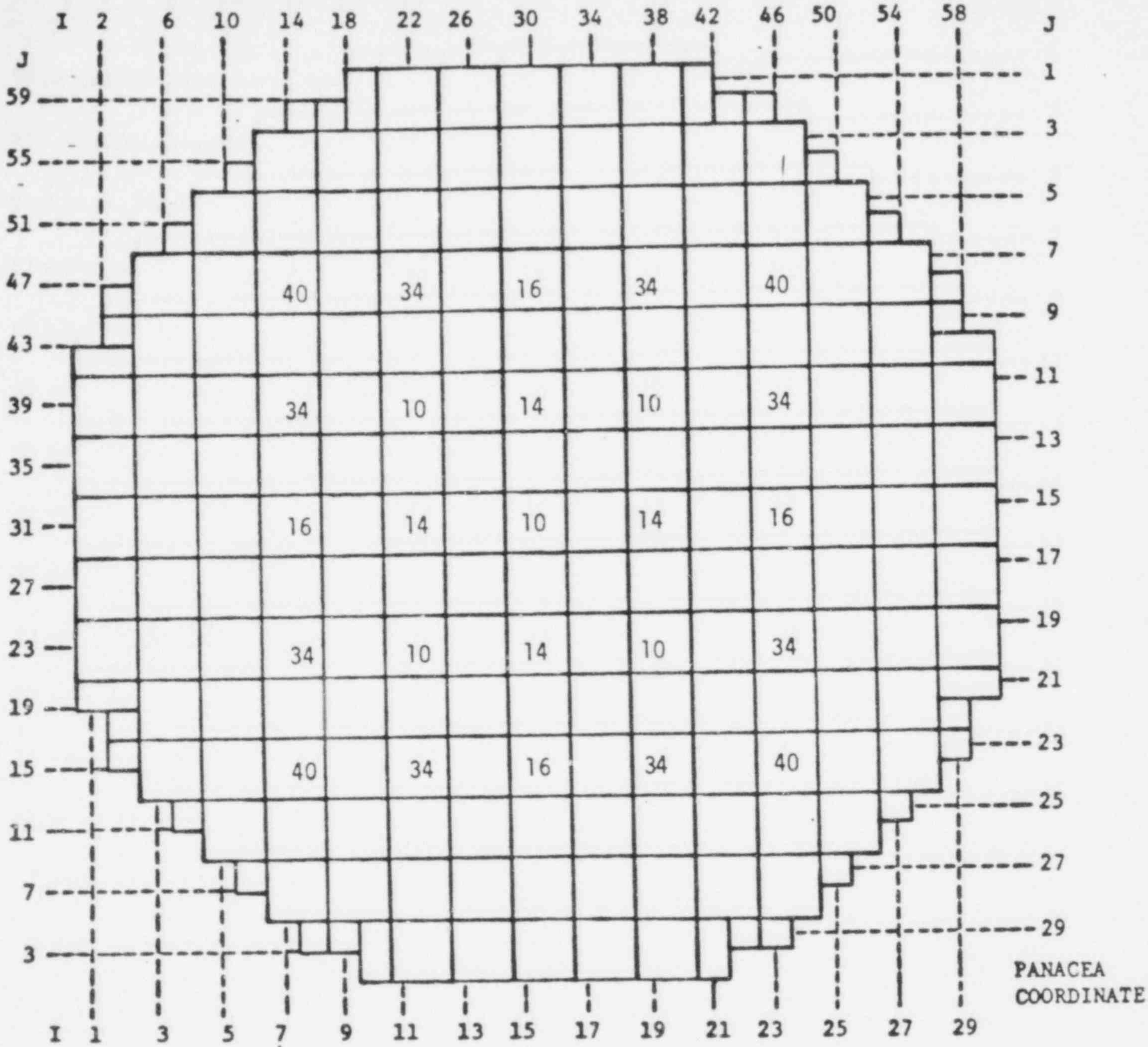
ILLUMINATING COMPANY

Fast Opening of Both Main Recirculation Loop Valves at 11% Per Second

Figure 15.4-3



DESIGN FACTORS		LOCATIONS		
		I	J	K
AXIAL PEAKING	1.22			4
RADIAL PEAKING	1.3261			
GROSS PEAKING				
MCPR	1.3174	11	8	24
MLHGR	12.1134	11	8	11

EXPOSURE	200 MWd/t
KEFF	1.00187
ROD SEQUENCE	A

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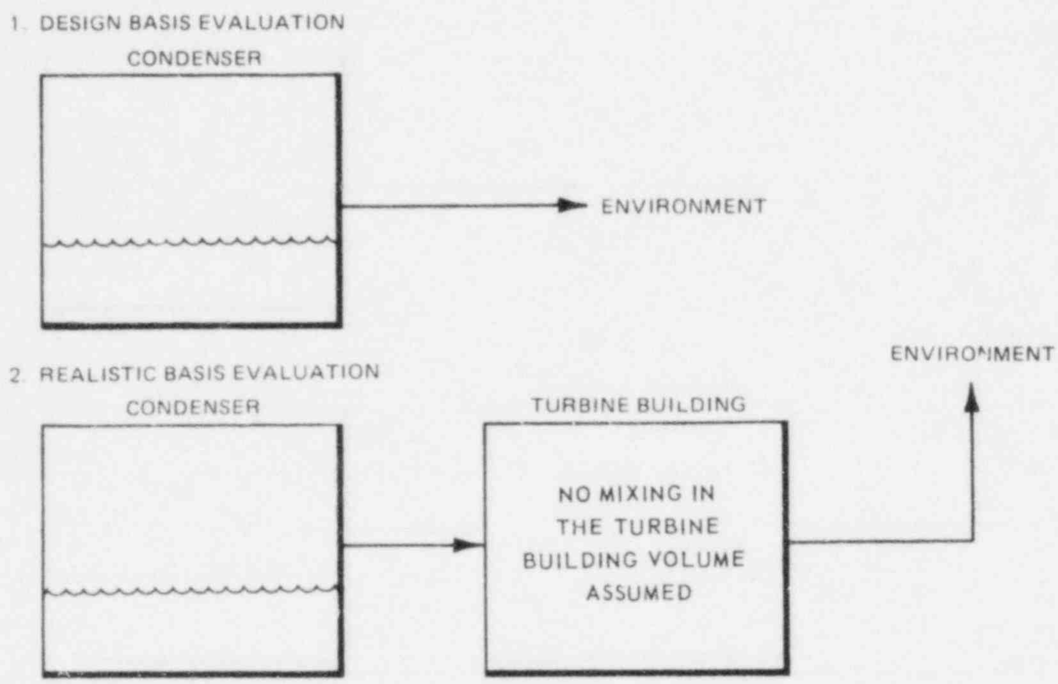
Critical Rod Pattern for
Misplaced Bundle Accident 0.2 GWd/t

Figure 15.4-4



PERRY NUCLEAR POWER PLANT
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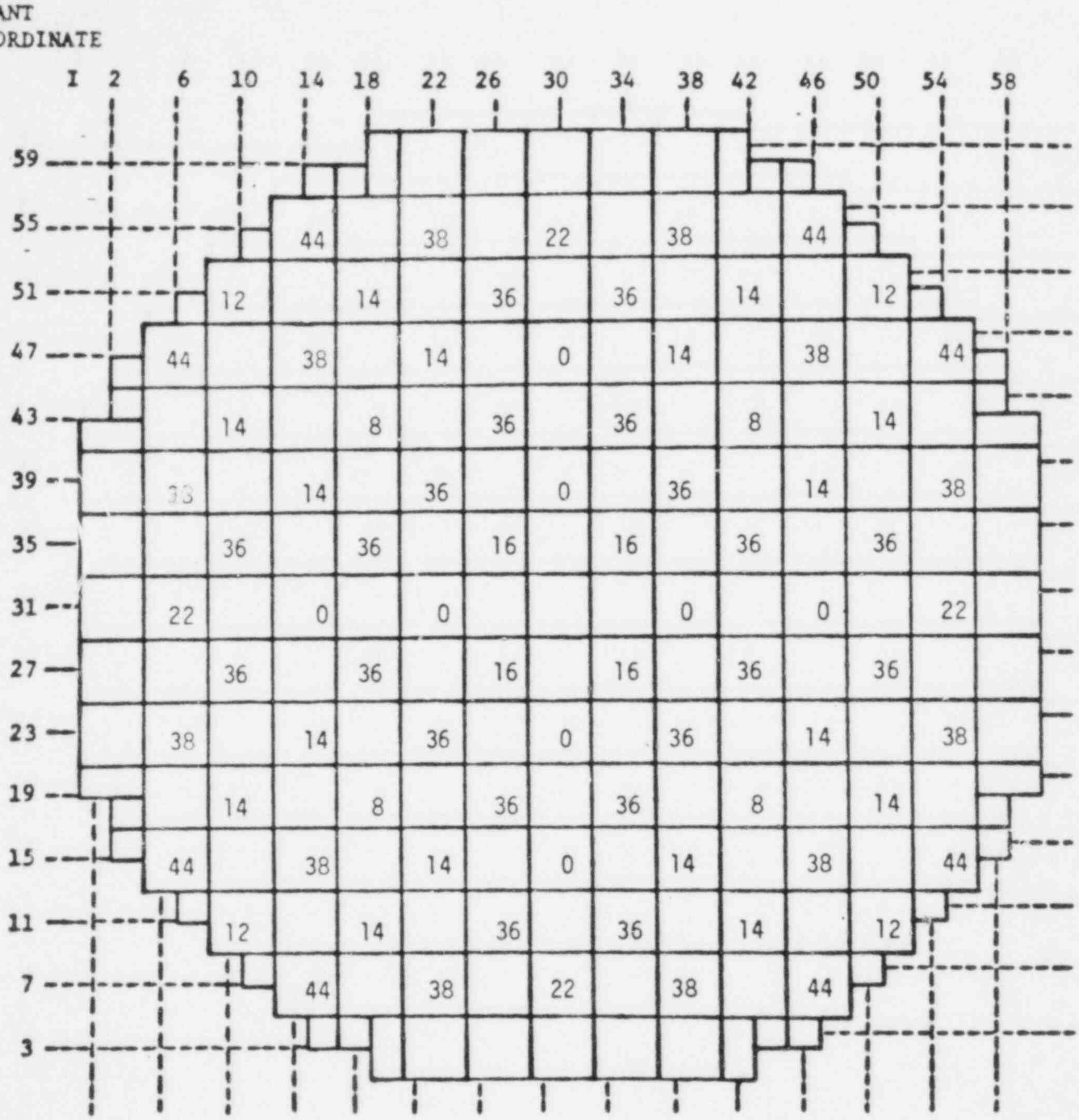
Leakage Path Model for Rod Drop Accident

Figure 15.4-5



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PERRY NUCLEAR POWER PLANT
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Rod Pattern for RWE Analysis 1/4
Core Geometry (Rated Power Case)

Figure 15.4-6

SINGLE ROD				GANGED RODS		
FEET OUT	POWER (MWt)	MCPR	MLHGR (kW/ft)	POWER (MWt)	MCPR	MLHGR (kW/ft)
0	3579.0	1.2395	13.41	3493.1	1.2550	13.01
1				3548.7	1.2286	13.41
2	3598.9	1.2096	13.58	3637.3	1.1838	14.51
3				3748.9	1.1317	16.74
4	3641.9	1.1566	15.43	3847.8	1.0875	18.40
6	3679.4	1.1141	16.17	3972.2	1.0299	19.06
8	3695.6	1.0962	15.25			
9				4048.0	1.0067	17.92
10	3698.8	1.0968	14.90			
12				4048.0	1.0135	17.41

MAXIMUM ALLOWABLE WITHDRAWAL (ΔCPR $\leq$ 0.13)

12.0 FOR SINGLE RODS

2.5 FOR GANGED RODS

PNPP PERRY NUCLEAR POWER PLANT
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Summary of Results for Rod Withdrawal Error

Figure 15.4-7

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 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 pressurization. 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 Sections 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 Section 15.2.2.3.1 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 15.0-1.

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 seconds the full HPCS flow is established at approximately 5.1 percent of the rated feedwater flow rate. This flow is nearly 102 percent 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 depresssurization 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 worst conditions so that any deviations in the actual plant parameters will produce a less severe transient.

15.5.1.4 <u>Barrier Performance</u>

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 CHEMICAL VOLUME CONTROL SYSTEM MALFUNCTION (OR OPERATOR ERROR)

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

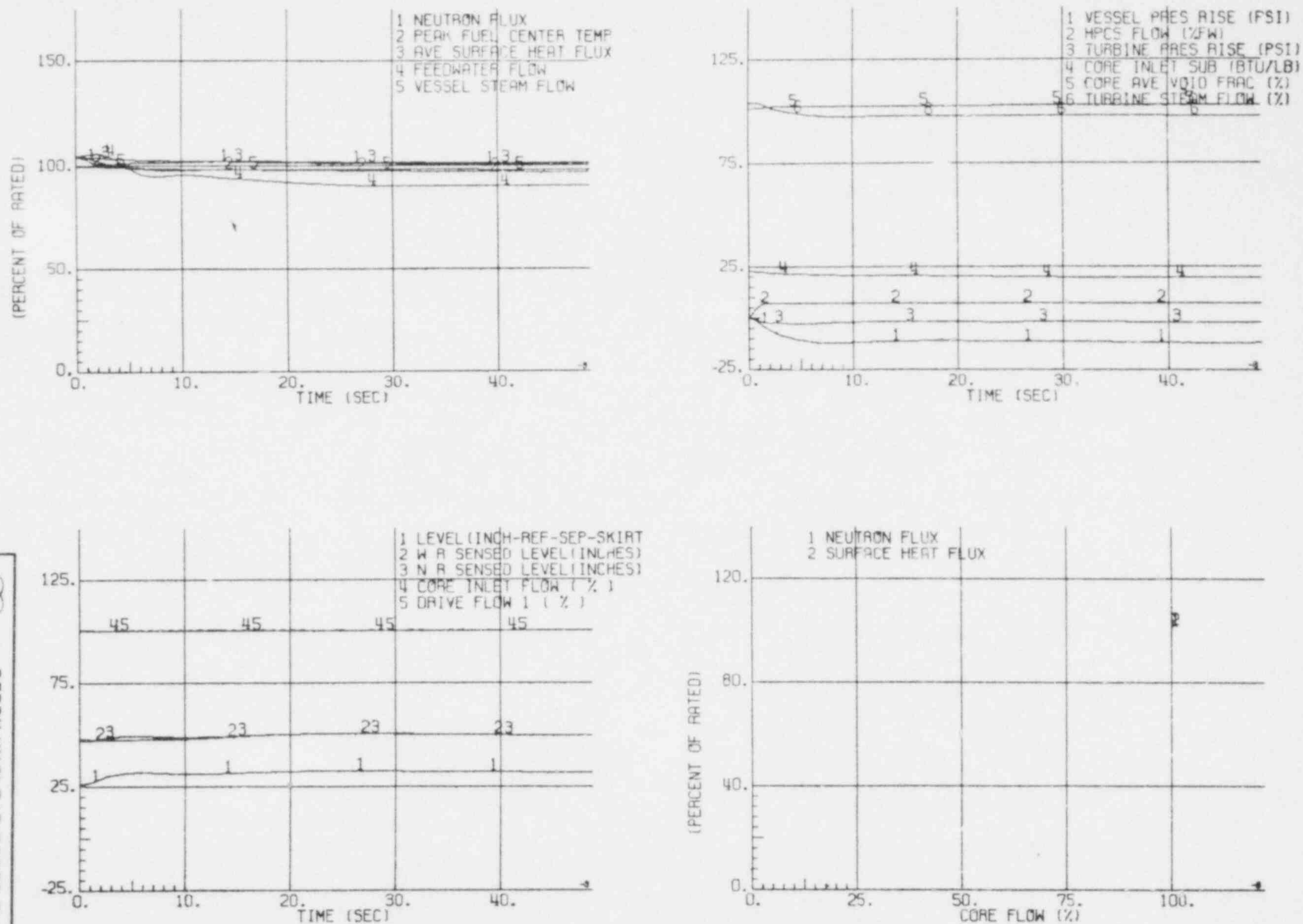
15.5.3 BWR TRANSIENTS WHICH INCREASE REACTOR COOLANT INVENTORY

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.
5	Depressurization effect stabilized.



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Inadvertent Startup of HPCS

Figure 15.5-1

15.6 DECREASE IN REACTOR COOLANT INVENTORY

15.6.1 INADVERTENT SAFETY RELIEF VALVE OPENING

This event is discussed and analyzed in Section 15.1.4.

15.6.2 INSTRUMENT LINE PIPE BREAK

This event involves the postulation of a small steam or liquid line pipe break inside containment. In order to bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where immediate detection is not automatic or apparent.

Obviously, this event is far less limiting than the postulated events in Sections 15.6.4, 15.6.5, and 15.6.6.

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

15.6.2.1 Identification of Causes and Frequency Classification

15.6.2.1.1 Identification of Causes

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

15.6.2.1.1.1 Event Description

A circumferential rupture of an instrument line which is connected to the primary coolant system is postulated to occur outside the drywell but inside the containment structure. This failure results in the release of primary

system coolant to containment until the reactor is depressurized. This event could conceivably occur also in the drywell. However, the associated effects would not be as significant as those from a failure in the containment.

15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.6.2.2 Sequence of Events and Systems Operation

15.6.2.2.1 Sequence of Events

The sequence of events for this accident is shown in Table 15.6-1.

15.6.2.2.1.1 Identification of Operator Actions

The operator should isolate the affected instrument line. Depending on which line is broken, the operator should determine whether to continue plant operation until a scheduled shutdown can be made or to proceed with an immediate, orderly plant shutdown.

As a result of increased radiation, temperature, humidity, fluid, and noise levels within the containment, operator action can be initiated by any one or any combination of the following:

a. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, steam flow, and steam pressure.

b. By annunciation of the control function, either high or low in the main control room.

c. By a half-channel scram if rupture occurred on a reactor protection system instrument line.

d. By a general increase in the area radiation monitor readings.

e. By an increase in the ventilation process radiation monitor readings.

f. By increases in area temperature monitor readings in the containment.

g. Leak detection system actuations.

Upon receiving one or more of the above signals and having made the decision to shutdown the plant, the operator should proceed to shutdown the RPV in an orderly manner.

15.6.2.2.2 System Operation

Normal plant instrumentation and controls are assumed to be fully operational during the entire plant transient to ensure positive identification of the break and safe shutdown of the plant. Minimum reactor and plant protection system operations are assumed for the analysis, e.g., minimum ECCS flow, and pool cooling capability. As a consequence of the accident, the reactor is scrammed and the reactor vessel cooled and depressurized over a 6 hour period.

15.6.2.2.3 The Effect of Single Failures and Operator Errors

The initiating event is handled by a protection sequence which can accommodate additional single failures. See Appendix 15A for a more detailed discussion of this subject.

15.6.2.3 <u>Core and System Performance</u>

15.6.2.3.1 Qualitative Summary - Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Sections 15.6.4, 15.6.5, and 15.6.6. Consequently instrument line

breaks are considered to be bounded specifically by the steam line break, Section 15.6.4. Details of this calculation, including those pertinent to core and system performance are discussed in detail in Section 15.6.4.3.

Since instrument line breaks result in a slower rate of coolant loss and are bounded by the calculations referenced above, the results presented here are qualitative rather than quantitative. Since the rate of coolant loss is slow, an orderly reactor system depressurization follows reactor scram and the primary system is cooled down and maintained without ECCS actuation. No fuel damage or core uncovery occurs as a result of this accident.

15.6.2.3.2 Quantitative Results

Instrument line breaks, because of their small size, are substantially less limiting from a core and system performance standpoint than the steam line break outside containment. Similarly, instrument line breaks are considered within the spectrum considered in ECCS performance calculations discussed in detail in Section 6.3.3.

Therefore, all information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steam line break) event may be found in Section 6.3.

15.6.2.3.3 Considerations of Uncertainties

The approach toward conservatively analyzing this event is discussed in detail for a more limiting case in Section 6.3.

15.6.2.4 Barrier Performance

15.6.2.4.1 General

The release of primary coolant through the orificed instrument line could result in an increase in containment pressure and the potential of isolation of the normal ventilation system.

The following assumptions and conditions are the basis for the mass loss during the 5 hour reactor shutdown period of this event:

a. Shutdown and depressurization initiated at 10 minutes after break occurs.

b. Normal depressurization and cooldown of reactor pressure vessel.

c. Line contains a 1/4-inch diameter flow restricting orifice inside the drywell.

d. Moody critical blowdown flow model[1] is applicable and flow is critical at the orifice.

The total integrated mass of fluid released into the containment via the break during the blowdown is 25,000 pounds. Of this total, 6,000 pounds flash to steam. Release of this mass of coolant results in a containment pressure which is well below the design pressure.

15.6.2.4.2 Containment Effects

Following the postulated failure of an instrument line in the containment, the containment pressure will rise due to the release of primary system fluid and will continue until the reactor is depressurized. The containment pressure increase is evaluated based on the calculated mass release. The calculation is based on the assumptions outlined above and includes the heat losses to the containment structures that will occur.

15.6.2.5 Radiological Consequences

15.6.2.5.1 Design Basis Analysis

While the NRC has developed a standard review plan[2] for this event, a specific regulatory guide calculation method has not been issued to specify unique design basis assumptions. For this reason, only the realistic bases will be provided.

15.6.2.5.2 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 evaluation are described in Section 15.0.3.5. Specific values of parameters used in the evaluation are presented in Table 15.6-2.

15.6.2.5.2.1 Fission Product Release from Fuel

The quantity of activity released as a consequence of reactor scram and vessel depressurization is based in part on measurements during plant shutdowns[3]. These measurements have been used to develop an empirical model which predicts, during the depressurization transient, I-131 releases of 0.42 Ci/bundle for a 50 percent probability value to 2.14 Ci/bundle for the 95 percent probability value. For the purpose of this evaluation, the 95th percentile values are used. The release of other iodine isotopes is considered to be proportional to the fission yields, that is

$$I_{132} = \frac{(2.14)(F_r I_{132})}{F_r I_{131}}$$

The activity airborne in the break location structure is presented in Table 15.6-3.

15.6.2.5.2.2 Fission Product Release to the Environment

The fission product activity released to the environment as a result of this accident is based upon the methods and assumptions outlined below:

a. The failure of an instrument line results in a relatively small release of activity to the containment over a blowdown period of approximately 5 hours. Therefore, it is conservatively assumed that the total iodine activity airborne inside containment presented in Table 15.6-4 is instantaneously released to the environs through the containment purge exhaust system.

b. Charcoal filter efficiency for the containment purge exhaust system is conservatively assumed to be 90 percent for iodine.

15.6.2.5.2.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.6-5.

15.6.3 STEAM GENERATOR TUBE FAILURE

This section is not applicable to the direct cycle BWR. This is a PWR-related event.

15.6.4 STEAM SYSTEM PIPING BREAK OUTSIDE CONTAINMENT

This event involves the postulation of a large steam line pipe break outside containment. It is assumed that the largest steam line, 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 steam line failures outside containment.

15.6.4.1 Identification of Causes and Frequency Classification

15.6.4.1.1 Identification of Causes

A main steam line break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to restrictive seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steam line rupture, the failure of a main steam line 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 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 severence of one of the four main steam lines. The sequence of events and approximate time required to reach the event is given in Table 15.6-6.

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 steam lines 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, 7.2, 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 analyzi g this event. The ECCS aspects are covered in Section 6.3. The break dete tion 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. Refer to 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.

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:

a. The reactor is operating at the power level associated with maximum mass release.

b. Nuclear system pressure is 1,060 psia and remains constant during closure.

c. An instantaneous circumferential break of the main steam line occurs.

d. Isolation valves start to close at 0.5 second on high flow signal and are fully closed at 5.5 seconds.

e. The Moody critical flow model[1] is applicable.

f. Level rise time is conservatively assumed to be 1.0 second. Mixture quality is conservatively taken to be a constant 7.0 (steam weight percentage) during mixture flow.

Initially only steam will issue from the broken end of the steam line. The flow in each line is limited by critical flow at the limiter to a maximum of 200 percent 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 steam line break is 141,687 pounds of which 127,376 pounds is liquid and 14,311 pounds is steam.

15.6.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. 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 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

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

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

15.6.4.5.1 Design Basis Analysis

The design basis analysis is based on NRC Standard Review Plan 15.6.4 and NRC Regulatory Guide 1.5. The specific models, assumptions and the program used for computer evaluation are described in Section 15.0.3.5. Specific values of parameters used in the evaluation are presented in Table 15.6-9.

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 steam lines prior to the break. This level of activity is consistent with an offgas release rate of 100 μCi per second - MWt after 30 minutes delay (~375,800 μCi per second). The iodine concentration in the reactor coolant is then given by (μCi per gram):

I-131	7.5 - 2
I-132	9.7 - 1
I-133	5.5 - 1
I-134	1.8 + 0
I-135	8.9 - 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 seconds results in a discharge of 14,311 pounds of steam and 127,376 pounds 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 15.6-7.

15.6.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-8 and are a small fraction of the guidelines of 10 CFR 100.

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 the program used for computer evaluation are described in Section 15.0.3.5. Specific values of parameters used in the evaluation are presented in Table 15.6-9.

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 percent of the iodine activity contained by an equivalent mass of primary coolant.

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

a. The amount of coolant discharged is that calculated in the analysis of the nuclear system transient.

b. The concentrations of biologically significant radionuclides contained in the primary coolant is given by μCi per gram as follows:

I-131	2.0 - 2
I-132	2.6 - 1
I-133	1.5 - 1
I-134	4.8 - 1
I-135	2.4 - 1

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 percent to 2 percent. For the purpose of this evaluation the conservative assumption is made that the activity per pound of steam is equal to 2 percent of the activity per pound of reactor water.

c. The noble gas discharge rate, after 30 minutes holdup, is assumed to be 0.1 Ci per second, 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.

d. 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 15.6-10.

15.6.4.5.2.3 Results

The calculated exposures for this event are presented in Table 15.6-11. As noted, these values are a small fraction of 10 CFR 100.

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 coincident with an SSE earthquake.

The event has been analyzed quantitatively in Sections 3.6, 6.2, 6.3, 7.3, and 8.3. Therefore, the following discussion provides only new information not presented in the subject sections. All other information is covered by cross-referencing.

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

15.6.5.1 <u>Identification of Causes and Frequency Classification</u>

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 emergency 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 in Table 6.3-1 for core system performance and Table 6.2-9 for barrier (containment) performance.

Following the pipe break and sc , the low-low water level (level 2) or high drywell pressure signal will initiate RCIC and LPCS systems at time 0 plus approximately 30 seconds, and the MSLIV will begin closing on the low-low-low level (level 1) signal.

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 seconds, determine plant condition 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 check that the emergency service water system has been automatically initiated. 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 steam lines upstream of the 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, 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 further details.

15.6.5.3 <u>Core and System Performance</u>

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 6.3-2.

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 occurring. Therefore, any postulated loss of coolant accident does not result in exceeding the containment design limit. For details and results of the analyses, see Sections 3.8, 3.9, and 6.2.

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. 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 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis."

b. 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 the NRC Standard Review Plan[2] 15.6.5 and Regulatory Guides 1.3 Rev. 3 and 1.7 Rev. 2. The specific models, assumptions, and computer code used to evaluate this event based on the above criteria are presented in Section 15.0.3.5 Specific values of parameters used in this evaluation are presented in Table 15.6-12.

15.6.5.5.1.1 Fission Product Release from Fuel

It is assumed that 100 percent of the noble gases and 50 percent of the iodine are released from an equilibrium core operating at a power level of 3,758 MWt for 1,000 days prior to the accident. While not specifically stated in Reg. Guide 1.3 the assumed release of 100 percent of the core noble gas activity and 50 percent of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (see Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100 percent of the noble gases and 50 percent of the iodine become airborne. The remaining 50 percent 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 15.6-13.

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the containment by several different mechanisms to the environment.

a. Containment leakage

The design basis leak rate of the primary containment and its penetrations (excluding the main steam lines) is 0.2 percent per day for the duration of the accident. 96 percent of this leakage is to the shield building annulus and from there to the environment via the AEGTS. Credit is taken for mixing and holdup within the shield building annulus. The remaining 4 percent of the leakage is assumed to be released directly to the environment. The shield building exhaust flow is 1,950 scfm and the mixing volume is 50 percent of the annulus volume.

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

c. 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 required is estimated to be 287 hours after the accident at a rate of 50 scfm. The purge would be processed by AEGTS prior to release to the environment.

d. Leakage from the main steam isolation valve (MSIV) leakage control system (LCS).

The LCS routes any leakage through the MSIVs to the AEGTS. Assuming the MSIVs leak 25.0 cfh per valve, leakage past the inboard MSIVs is conservatively assumed to begin immediately after the accident. The airborne fission products are assumed to be uniformly mixed in the annulus air volumes.

Fission product release to the environment based on the above assumptions is given in Table 15.6-14.

15.6.5.5.1.3 Control Room Habitability

The integrated thyroid dose and external whole body dose to control room personnel were calculated using methods and assumptions listed below:

a. The Tact 3S computer code is used to calculate the integrated unprotected control room doses.

b. The above doses are appropriately reduced by an iodine protection factor (IPF) and the geometry reduction factor (GF) in accordance with the methods presented in Reference 4. In order to account for the high degree of leaktightness and relatively slow buildup of noble gas activity within the control room, a noble gas protection factor (NPF) is also applied. The INHEC computer code[5] is used to compare the unprotected noble gas dose outside the control room to the dose inside the control room. The ratio of these two quantities is the NPF.

c. Control room personnel are assumed to breathe at a rate of 3.47×10^{-4} m^3/sec for the duration of the accident.

d. The control room shield walls have a minimum thickness of 2 feet of concrete. Assuming an average gamma energy of 1.5 MeV results in a dose reduction factor of 0.0069 from the passing cloud dose.

e. Meteorological dispersion data and other pertinent control room parameters are presented in Table 15.6-12.

Based on these methods and assumptions, the resulting doses to control room personnel are presented in Table 15.6-15.

15.6.5.5.1.4 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-15 and are well within the guidelines of 10 CFR 100.

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 Section 15.0.3.5. Specific values of parameters used in the evaluation are presented in Table 15.6-12.

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 various 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 in μCi/gm is:

I-131	2.0 - 2
I-132	2.6 - 1
I-133	1.5 - 2
I-134	4.8 - 1
I-135	2.4 - 1

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

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[3] 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 percent of the time will the release be greater) the probability of the I-131 release is calculated to be 2.14 Ci/bundle and Xe-133 to be 11.54 Ci/bundle. Other iodine and noble gas isotopes are determined in accordance with the method presented in Section 15.6.2.5.2.1 (Table 15.6-16).

The combined airborne activity from the flashing and suppression pool source terms is presented in Table 15.6-17. 100 percent of the noble gas released from the reactor pressure vessel is assumed to remain airborne in the containment.

15.6.5.5.2.2 Fission Product Transport to the Environment

The leak rate from the containment to the environment is 0.2 percent per day. Of this amount, 4.0 percent is assumed to be released directly to the environment and the remainder is released to the shield building annulus. Release from the shield building annulus to the environment is via a 99 percent iodine efficient AEGTS.

15.6.5.5.2.3 Results

The calculated radiological exposures for this event are presented in Table 15.6-18 and as shown are a small fraction of 10 CFR 100.

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 evaluation relative to this type of

occurrence. 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, "Emergency Core Cooling Systems." Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is covered by cross-referencing to appropriate topics in Section 6.3.

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.

15.6.6.2 Sequence of Events and Systems Operation

15.6.6.2.1 Sequence of Events

The sequence of events is shown in Table 15.6-19.

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 for informational purposes:

a. Determine that a line break has occurred and evacuate the area of the turbine building.

b. The operator is not required to take any action to prevent primary reactor system mass loss, but should ensure that the reactor is shutdown and that RCIC and/or HPCS are operating normally.

c. Implement site radiation incident procedures.

d. If possible, shutdown the feedwater system and deenergize any electrical equipment which may be damaged by water from the feedwater system in the turbine building.

e. 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.

f. When the reactor pressure has decreased below 150 psia, initiate RHR in the shutdown cooling mode to continue cooling 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 loss of coolant accident break spectrum considered in detail in Section 6.3. The general single-failure analysis for loss of coolant accidents is discussed in detail in Section 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 detailed description of analysis.

15.6.6.3 Core and System Performance

15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this section is considered to be a conservative, envelope assessment of the consequences of the postulated failure (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 6.3-2.

15.6.6.3.2 Qualitative Results

The feedwater line break outside the containment is less limiting than either of the steam line breaks outside the containment (analysis presented in Sections 6.3 and 15.6.4) or the feedwater line break inside the containment (analysis presented in sections 6.3.3 and 15.6.5). It is far less limiting than the design basis accident (the recirculation line break analysis presented in Sections 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 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 Section 15.6.4. The feedwater system piping break is less severe than the main steam line break.

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 an engineered but still conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are described in Reference 6. Specific values of parameters used in the evaluation are presented in Table 15.6-20. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-3.

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 off-gas release rate of 100,000 μCi/sec at 30 minutes delay and is 0.02 (2 percent 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.

The total integrated mass of coolant leaving the break is 1.454×10^6 lbs of condensate. For the purposes of this evaluation, the conservative assumption is made that the activity of iodine per pound of steam is equal to 2 percent of the activity per pound of water.

Taking no credit for holdup, decay or plateout during transport through the turbine building, the release of activity to the environment is presented in Table 15.6-21. 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 15.6-22 and are a small fraction of 10 CFR 100 guidelines.

15.6.7 REFERENCES FOR SECTION 15.6

1. Moody, F. J., "Maximum Two-Phase Vessel Blowdown From Pipes," ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. USNRC Standard Review, Plan NUREG-75/087.

3. Brutschy, F. J., G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup," August 1972, (NEDO-10585).

4. Murphy, K. G. and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference.

5. Gilbert Associates, Inc., "INHEC - Inhalation and Whole Body Doses Using the AEC Safety Guide Methods," June 1, 1973 (GAI-TR-101P).

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

TABLE 15.6-1

SEQUENCE OF EVENTS FOR INSTRUMENT LINE BREAK

Time	Event
0	Instrument line fails.
0-10 min	Identification of break.
10 min	Activate RHR and initiate orderly shutdown.
	Containment spray initiated by high containment pressure (9 psig)
5 hours	Reactor vessel depressurized and break flow terminated.

TABLE 15.6-2

INSTRUMENT 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	None	NA
B. Burn-up	None	NA
C. Fuel damaged	None	None
D. Release of activity by nuclide	None	Table 15.6-3
E. Iodine fractions		
(1) Organic	None	0
(2) Elemental	None	1
(3) Particulate	None	0
F. Reactor coolant activity before the accident	None	15.6.2.5.1.1
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate (%/day)	None	NA
B. Secondary containment leak rate (%/day)	None	NA
C. Valve movement times	None	NA
D. Adsorption and filtration efficiencies		
(1) Organic iodine	None	90
(2) Elemental iodine	None	90
(3) Particulate iodine	None	90
(4) Particulate fission products	None	NA
E. Recirculation system parameters		
(1) Flow rate	None	NA
(2) Mixing efficiency	None	NA
(3) Filter efficiency	None	NA
F. Containment spray parameters (flow rate, drop size, etc.)	None	NA
G. Containment volumes	None	NA
H. All other pertinent data and assumptions	None	None

TABLE 15.6-2 (Cont'd)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	None	863/4002
B. χ/Q's (sec/m^3) for time intervals of		
(1) 0-2 hr - SB/LPZ	None	6.7-4/8.2-5
(2) 2-8 hr - LPZ	None	8.2-5
(3) 8-24 hr - LPZ	None	5.2-5
(4) 1-4 days - LPZ	None	1.9-5
(5) 4-30 days - LPZ	None	4.7-6
IV. Dose Data		
A. Method of dose calculation	NA	15.0.3.5
B. Dose conversion assumptions	NA	15.0.3.5
C. Peak activity concentrations in containment	NA	15.6.2.3
D. Doses	NA	15.6.2.5

TABLE 15.6-3

INSTRUMENT LINE FAILURE (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN INSTRUMENT LINE BREAK STRUCTURE (CURIES)

Isotope	Activity
I-131	7.46 + 1
I-132	1.15 + 2
I-133	1.79 + 2
I-134	1.97 + 2
I-135	1.70 + 2

TABLE 15.6-4

INSTRUMENT LINE FAILURE (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	7.46 + 0
I-132	1.15 + 1
I-133	1.79 + 1
I-134	1.97 + 1
I-135	1.70 + 1

TABLE 15.6-5

INSTRUMENT LINE FAILURE RADIOLOGICAL EFFECTS
(REALISTIC ANALYSIS)

	Inhalation Dose (Rem)
Exclusion area (863 Meters)	1.99 + 0
Low population zone (4,002 Meters)	2.44 - 1

TABLE 15.6-6

SEQUENCE OF EVENTS FOR STEAM LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	Guillotine break of one main steam line outside primary containment.
~0.5	High steam line flow signal initiates closure of main steam line isolation valve.
<1.0	Reactor begins scram.
≤5.5	Main steam line 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 1,100 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 1,100 psi.
600	Operator initiates ADS or manually controls relief valves. Vessel depressurizes rapidly.
763 (see Section 6.3.3)	Low pressure ECCS systems initiated. Reactor fuel uncovered partially.
(see Section 6.3.3)	Core effectively reflooded and clad temperature heatup terminated. No fuel rod failure.

TABLE 15.6-7

STEAM LINE BREAK ACCIDENT
(DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	4.83 + 0
I-132	6.24 + 1
I-133	3.54 + 1
I-134	1.16 + 1
I-135	5.72 + 1
Kr-83m	7.59 - 2
Kr-85m	1.33 - 1
Kr-85	5.19 - 4
Kr-87	4.14 - 1
Kr-88	4.26 - 1
Kr-89	1.77 + 0
Xe-131m	4.23 - 4
Xe-133m	6.33 - 3
Xe-133	1.78 - 1
Xe-135m	5.19 - 1
Xe-135	4.80 - 1
Xe-137	2.34 + 0
Xe-138	1.77 + 0

TABLE 15.6-8

STEAM LINE BREAK ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion area (863 Meters)	8.38 - 2	8.04 + 0
Low population zone (4,002 Meters)	1.03 - 2	9.84 - 1

TABLE 15.6-9

STEAM 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. Burn-up	NA	NA
	C. Fuel damaged	None	None
	D. Release of activity by nuclide	Table 15.6-8	Table 15.6-10
	E. Iodine fractions		
	(1) Organic	0	0
	(2) Elemental	1	1
	(3) Particulate	0	0
	F. Reactor coolant activity before the accident	15.6.4.5.1.1.1	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		
	(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 15.6-9 (Cont'd)

		Design Basis Assumptions	Realistic Basis Assumptions
III.	Dispersion Data		
A.	Boundary and LPZ distance (m)	863/4002	863/4002
B.	χ/Q's for total dose - SB/LPZ	6.7-4/8.2-5	6.7-4/8.2-5
IV.	Dose Data		
A.	Method of dose calculation	15.0.3.5	15.0.3.5
B.	Dose conversion assumptions	15.0.3.5	15.0.3.5
C.	Peak activity concentrations in containment	NA	NA
D.	Doses	Table 15.6-9	Table 15.6-11

TABLE 15.6-10

STEAM LINE BREAK ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.23 + 0
I-132	1.67 + 1
I-133	9.65 + 0
I-134	3.09 + 1
I-135	1.54 + 1
Kr-83m	2.53 - 2
Kr-85m	4.43 - 2
Kr-85	1.73 - 4
Kr-87	1.38 - 1
Kr-88	1.42 - 1
Kr-89	5.89 - 1
Xe-131m	1.41 - 4
Xe-133m	2.11 - 3
Xe-133	5.92 - 2
Xe-135m	1.73 - 1
Xe-135	1.60 - 1
Xe-137	7.80 - 1
Xe-138	5.89 - 1

TABLE 15.6-11

STEAM LINE BREAK ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion area (863 Meters)	2.26 - 2	2.17 + 0
Low population zone (4,002 Meters)	2.76 - 3	2.65 - 1

TABLE 15.6-12

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	3758 MWt	3758 MWt
B. Burn-up	NA	NA
C. Fuel damaged	100%	0
D. Release of activity by nuclide	Table 15.6-14	Table 15.6-18
E. Iodine fractions		
(1) Organic	0.04	0.01
(2) Elemental	0.91	0.99
(3) Particulate	0.05	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)	15.6.5.5.1.2	15.6.5.5.2.2
B. Secondary containment leak rate (%/day)	15.6.5.5.1.2	15.6.5.5.2.2
C. Valve movement times	NA	NA
D. Adsorption and filtration efficiencies		
(1) Organic iodine	99	99
(2) Elemental iodine	99	99
(3) Particulate iodine	99	99
(4) Particulate fission products	99	99
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 15.6-12 (Cont'd)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	863/4002	863/4002
B. χ/Q's for total dose - SB/LPZ		
(1) 0-2 hr - SB/LPZ	6.7-4/8.2-5	6.7-4/8.2-5
(2) 2-8 hr - LPZ	8.2-5	8.2-5
(3) 8-24 hr - LPZ	5.2-5	5.2-5
(4) 1-4 days - LPZ	1.9-5	1.9-5
(5) 4-30 days - LPZ	4.7-6	4.7-6
C. Control room χ/Q's for time intervals of		
(1) 0-8 hrs	5.0-3	
(2) 8-24 hrs	2.8-3	
(3) 1-4 days	1.4-3	
(4) 4-30 days	2.9-4	
IV. Dose Data		
A. Method of dose calculation		
B. Dose conversion assumptions		
C. Peak activity concentrations in containment	Table 15.6-13	Table 15.6-17
D. Doses	Table 15.6-15	Table 15.6-18
V. Control Room		
A. Volume (ft^3)	294,492.	
B. Control room inleakage (cfm)	30.0	
C. Recirculation system		
(1) Flow rate (cfm)	30,000.	
(2) Filter efficiency for iodine (%)	95.	

TABLE 15.6-14

LOSS OF COOLANT ACCIDENT
(DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT
(CURIES)

Isotope	Activity
I-131	2.34 + 4
I-132	4.40 + 2
I-133	6.19 + 3
I-134	2.72 + 2
I-135	1.77 + 3
Kr-83m	2.08 + 3
Kr-85m	1.90 + 4
Kr-85	8.51 + 4
Kr-87	6.80 + 3
Kr-88	2.65 + 4
Kr-89	7.06 + 1
Xe-131m	2.58 + 4
Xe-133m	3.19 + 4
Xe-133	2.93 + 6
Xe-135m	3.49 + 2
Xe-135	1.82 + 5
Xe-137	1.23 + 2
Xe-138	8.21 + 3

TABLE 15.6-15

LOSS OF COOLANT ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

a. Without MSIV Leakage

		Whole Body Gamma Dose (Rem)	Inhalation Dose (Rem)
1. Offsite Doses			
Exclusion area (863 Meters)		3.33 + 0	1.01 + 2
Low population zone (4,002 Meters)		2.31 + 0	1.42 + 2
2. Control Room Doses (0-30 days)			
	β-Skin[(1)] Dose (Rem)		
	1.50 + 1	2.27 + 0	1.32 + 1

b. With 100 SCFH MSIV Leakage Starting at T = 0

		Whole Body Gamma Dose (Rem)	Inhalation Dose (Rem)
1. Offsite Doses			
Exclusion area (863 Meters)		5.68 + 0	1.01 + 2
Low population zone (4,002 Meters)		4.07 + 0	1.42 + 2
2. Control Room Doses (0-30 days)			
	β-Skin[(1)] Dose (Rem)		
	2.82 + 1	3.78 + 0	1.32 + 1

NOTE:

1. Unprotected beta skin dose; no credit is taken for any reduction afforded by clothing.

TABLE 15.6-15

LOSS OF COOLANT ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

a. Without MSIV Leakage

	β-Skin(1) Dose (Rem)	Whole Body Gamma Dose (Rem)	Inhalation Dose (Rem)
1. Offsite Doses			
Exclusion area (863 Meters)		3.33 + 0	1.01 + 2
Low population zone (4,002 Meters)		2.31 + 0	1.42 + 2
2. Control Room Doses (0-30 days)	1.50 + 1	2.27 + 0	1.32 + 1

b. With 100 SCFH MSIV Leakage Starting at T = 0

	β-Skin(1) Dose (Rem)	Whole Body Gamma Dose (Rem)	Inhalation Dose (Rem)
1. Offsite Doses			
Exclusion area (863 Meters)		5.68 + 0	1.01 + 2
Low population zone (4,002 Meters)		4.07 + 0	1.42 + 2
2. Control Room Doses (0-30 days)	2.82 + 1	3.78 + 0	1.32 + 1

NOTE:

1. Unprotected beta skin dose; no credit is taken for any reduction afforded by clothing.

TABLE 15.6-16

ISOTOPIC SPIKING ACTIVITY

Isotope Name	The 95th Cumulative Probability Spiking Activity (Ci/Bundle)
I-131	2.14
I-132	3.21
I-133	5.03
I-134	5.44
I-135	4.79
Kr-83m	9.04 - 1
Kr-85m	2.23 + 0
Kr-85	4.90 - 1
Kr-87	4.33 + 0
Kr-88	6.12 + 0
Kr-89	7.96 + 0
Xe-131m	6.60 - 2
Xe-133m	3.26 - 1
Xe-133	1.16 + 1
Xe-135m	1.80 + 0
Xe-135	1.10 + 1
Xe-137	1.05 + 1
Xe-138	1.06 + 1

TABLE 15.6-17

LOSS OF COOLANT ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONTAINMENT (CURIES)

Isotope	Activity
I-131	3.2 + 2
I-132	4.8 + 2
I-133	7.6 + 2
I-134	8.2 + 2
I-135	7.2 + 2
Kr-83m	6.8 + 2
Kr-85m	1.7 + 3
Kr-85	3.7 + 2
Kr-87	3.2 + 3
Kr-88	4.6 + 3
Kr-89	6.0 + 3
Xe-131m	4.9 + 1
Xe-133m	2.1 + 2
Xe-133	8.6 + 3
Xe-135m	1.3 + 3
Xe-135	8.2 + 3
Xe-137	7.8 + 3
Xe-138	8.0 + 3

TABLE 15.6-18

LOSS OF COOLANT ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion area (863 Meters)	1.44 - 4	1.48 - 3
Low population zone (4,002 Meters)	1.01 - 4	2.06 - 3

TABLE 15.6-19

SEQUFNCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time-sec	Event
0	One feedwater line breaks.
0+	Feedwater line check valves isolate the reactor from the break.
<30	At low water level, the reactor would scram. At low low-water level, RCIC would initiate, HPCS would initiate and recirculation pumps would trip. MSLIV will close at water level 1.
~2 min	The safety relief valves would open and close and maintain the reactor vessel pressure at approximately 1100 psig.
1 to 2 hours	Normal reactor cooldown procedure established.

TABLE 15.6-20

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. Burn-up	NA	NA
C. Fuel damaged	NA	None
D. Release of activity by nuclide	NA	Table 15.6-21
E. Iodine fractions		
(1) Organic	NA	0
(2) Elemental	NA	1
(3) Particulate	NA	0
F. Reactor coolant activity before the accident	NA	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		
(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 15.6-20 (Cont'd)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Data		
A. Boundary and LPZ distance (m)	863/4002	863/4002
B. χ/Q's for total dose - SB/LPZ	6.7-4/8.2-5	6.7-4/8.2-5
IV. Dose Data		
A. Method of dose calculation	NA	15.0.3.5
B. Dose conversion assumptions	NA	15.0.3.5
C. Peak activity concentrations in containment	NA	NA
D. Doses	NA	Table 15.6-22

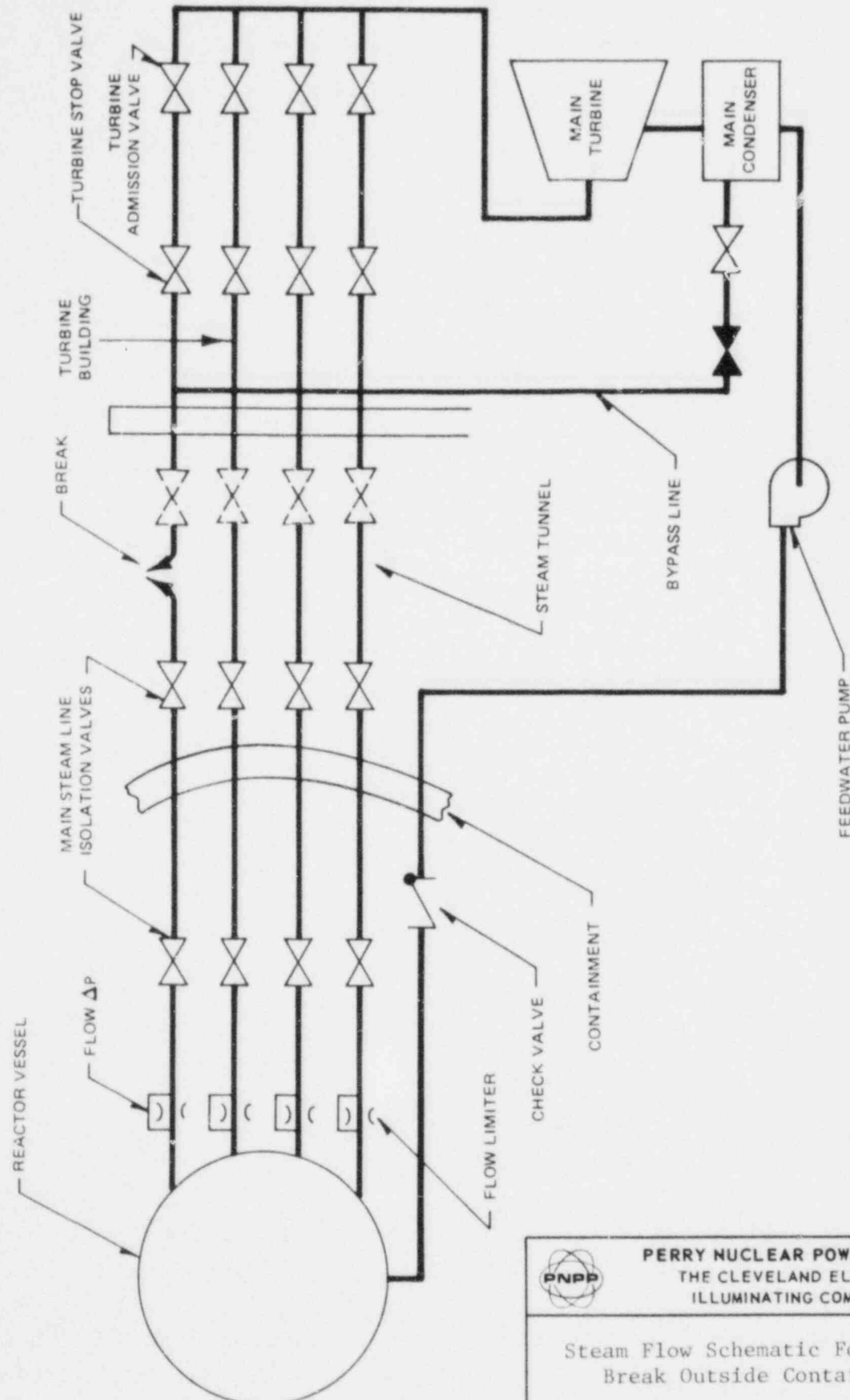
TABLE 15.6-21

FEEDWATER LINE BREAK
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	2.64 - 1
I-132	3.42 + 0
I-133	1.98 + 0
I-134	6.33 + 0
I-135	3.17 + 0

FEEDWATER LINE BREAK
RADIOLOGICAL EFFECTS

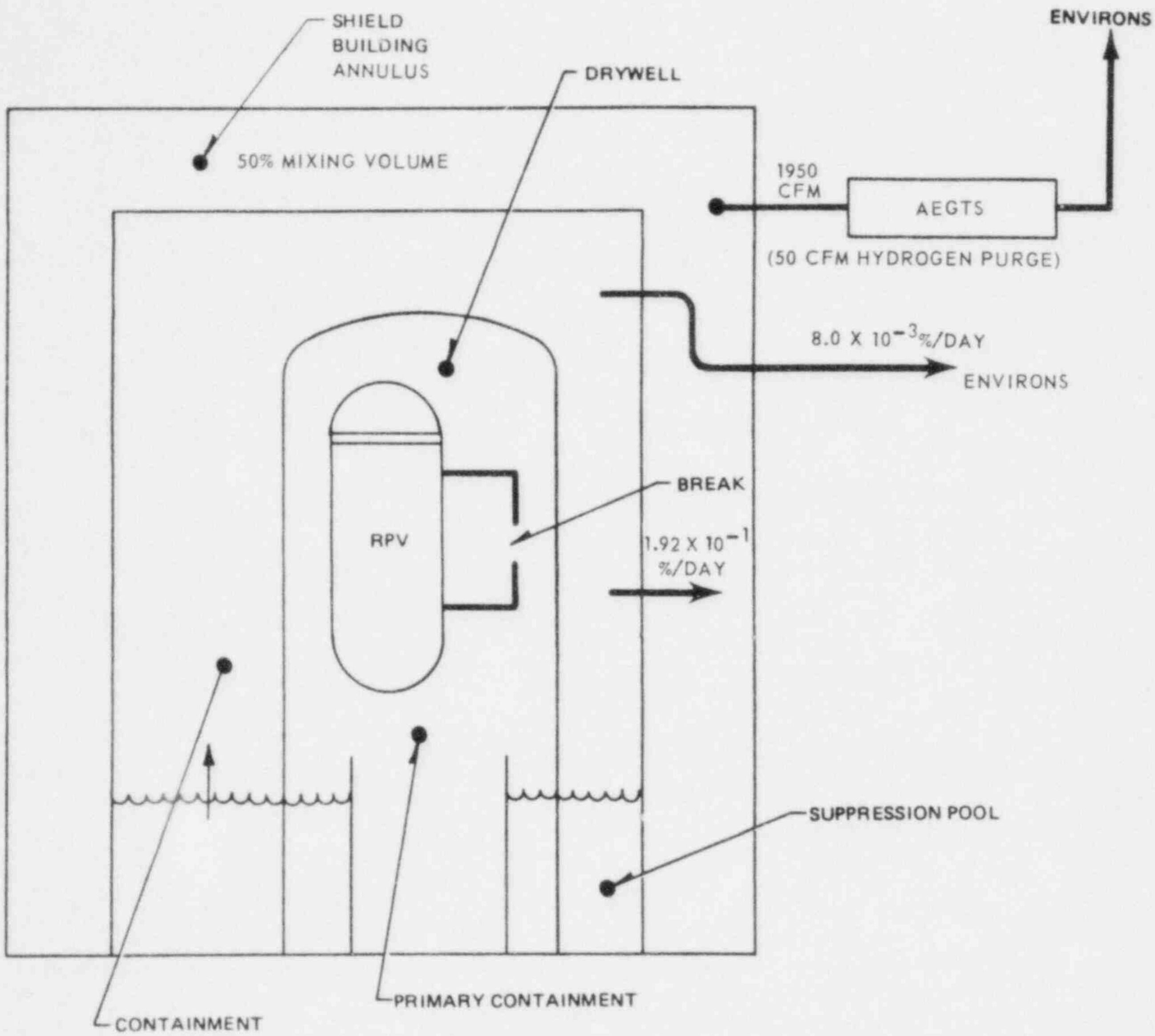
	Inhalation Dose (Rem)
Exclusion area (863 Meters)	4.46 - 1
Low population zone (4,002 Meters)	5.46 - 2



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Steam Flow Schematic For Steam
Break Outside Containment

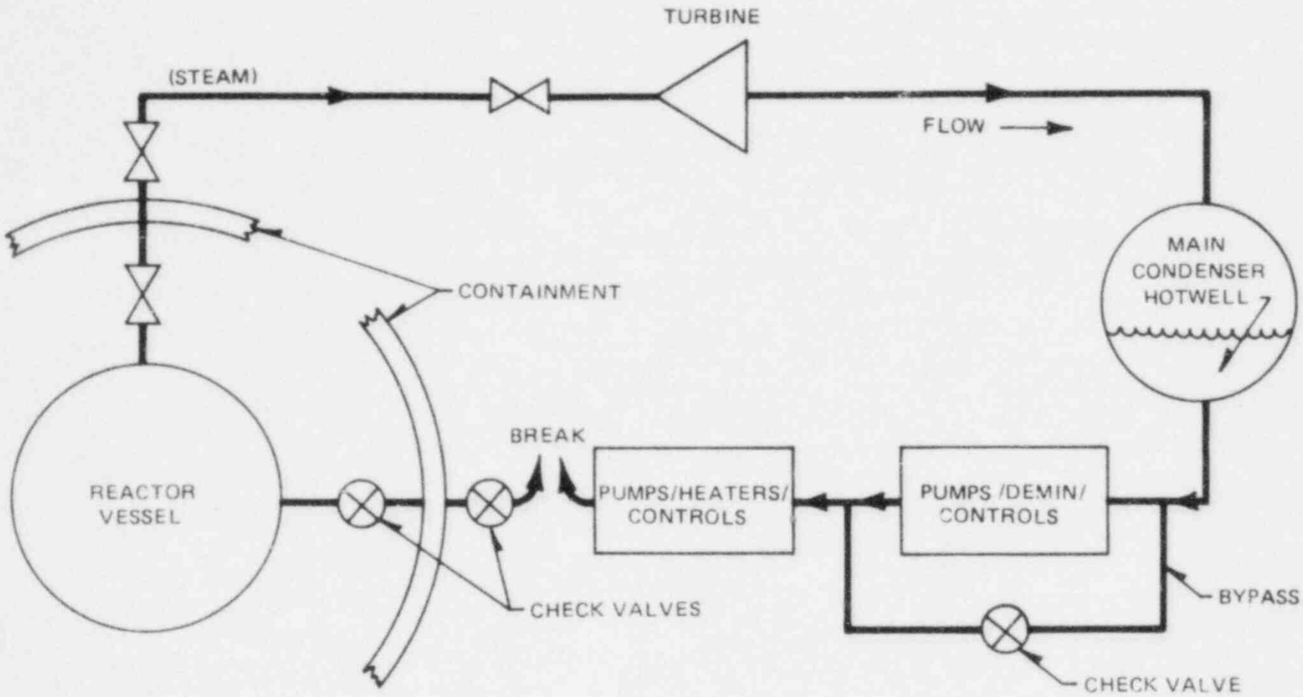
Figure 15.6-1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Leakage Flow for Loca

Figure 15.6-2



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Leakage Path for Feedwater Line Break Outside Containment

Figure 15.6-3

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:

a. Main condenser gas treatment system failure.

b. Malfunction of main turbine gland sealing system.

c. Failure of air ejector lines.

15.7.1.1 Main Condenser Off-gas 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 off-gas treatment system are:

a. A seismic occurrence exceeding the seismic capabilities of the equipment

b. A hydrogen detonation which ruptures the system pressure boundary.

c. A fire in the filter assemblies .

d. Failure of adjacent equipment which could subsequently compromise off-gas equipment.

The seismic event is considered to be the most probable and is the only conceivable event which could cause significant system damage.

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 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 seismic event more severe than the design requirements 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 probable 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 minutes.

15.7.1.1.2.3 Systems Operation

In analyzing the postulated off-gas 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:

a. Capability to detect the failure itself as indicated by an alarmed increase in radioactivity levels seen by the area radiation monitoring system, an alarmed loss of flow in the off-gas system, and an alarmed increase in activity at the vent release.

b. Capability to isolate the system and shutdown the reactor.

c. 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.

The seismic event which is assumed to occur beyond the present plant design basis for non-safety equipment will cause the tripping of the 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.

However, for conservatism, the SJAE will be assumed to continue pumping process gas for 30 minutes.

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 Section 15.2.5.

15.7.1.1.4 Barrier Performance

The postulated failure is the rupture of the off-gas 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 Section 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 the seismic accident:

a. The first analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

b. The second is based on assumptions considered to provide a realistic 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 during normal operation of the off-gas system.

a. Off-gas condenser - 100 percent retained and continuously washed out with condensate

b. Water separator - (included with off-gas condenser)

c. Holdup pipe - 60 percent retained and continuously washed out with condensate

d. Cooler condenser - 100 percent retained and continuously washed out with condensate

e. Moisture separator - (included with cooler condenser)

f. Prefilter - 100 percent retained, element changed annually

g. Dessicant dryer - 100 percent retained, dessicant replaced approximately once every five years

h. Charcoal adsorbers - 100 percent retained

i. After filter - 100 percent retained, element changed annually

Components not listed are assumed to have zero retention of solid daughter products.

Both analyses assume that the SJAE continues to pump the process gas out of a break near the failed component for 30 minutes after the accident. The release rates for breaks at the SJAE exit and holdup pipe exit are given in Tables 15.7-2 and 15.7-3.

15.7.1.1.5.2 Design Basis Analysis

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:

a. 2 SCFM air inleakage.

b. 100,000 μCi/sec noble gas after 30 minutes delay for a period of 11 months, followed by 1 month of 350,000 μCi/sec at 30 minutes.

15.7.1.1.5.2.1.2 Assumptions

Depending upon the assumptions as to radionuclide release fractions for each equipment piece, the assumed single failure of any one of several equipment pieces could be controlling with respect to dose consequences. The assumed release fractions for the design basis analysis are found in Table 15.7-4. The basis for the assumptions for failure of those equipment pieces which are expected to have the worst dose consequences are as follows:

a. Charcoal Adsorbers and Desiccant Vessel

   Because these vessels are designed with thick walls for detonation resistance, the only credible failure that could result in loss of carbon is a vessel nozzle failure due to excessive nozzle loads during the seismic event. Assuming the vessel supports fail along with the nozzle failure, it is expected that no more than 10-15 percent of the carbon would be displaced from the vessel. This percentage of the carbon is assumed to be from the top of the first bed and therefore would contain virtually all of the activity stored in the beds. Because iodine is strongly bonded to the charcoal, it is not expected to be removed by exposure to the air. However, the conservative assumption is made that 1 percent of the iodine activity contained in the adsorber tank is released to the vault containing the off-gas equipment. Additionally, the conservative assumption is made that 1 percent of the solid daughters retained in the charcoal is released.

   It is further assumed that 10 percent of the noble gas activity is released from a failed vessel because of the small fraction of carbon exposed to the air. Measurements made at KRB indicate that off-gas is about 30 percent 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.

b. Prefilters

Because of the design features of the prefilter vessel, approximately 24 inch diameter, 4 feet height, 350 psig design pressure, 1/2 inch 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, one percent release of particulate activity is assumed.

c. Holdup pipe

Pipe rupture and depressurization of the pipe is considered. For the design basis analysis, 100 percent of the noble gases and all of the remaining solid daughters after the 60 percent washout are assumed to be released.

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 to the environment through the building ventilation system based on the release fractions given in Table 15.7-4. The inventory of activities in each equipment piece before the assumed failure is presented in Table 12A-4 in Appendix 12A. The release rates from letting the SJAE continue to pump are given in Tables 15.7-2 and 15.7-3, depending on where the worst failure occurs.

15.7.1.1.5.2.3 Results

Dose consequences due to failure of the worst single component [charcoal adsorber] piece and assuming the SJAE continues to pump for 30 minutes after the break is presented in Table 15.7-5.

15.7.1.1.5.3 Realistic Analysis

The realistic analysis is based on an engineered 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 15.7-6.

15.7.1.1.5.3.1 Fission Product Release

15.7.1.1.5.3.1.1 Initial Conditions

The activity in the off-gas system is based on the following normal operating conditions:

a. 30 scfm air inleakage.

b. 100,000 μCi/sec noble gas after 30 minutes delay.

The activity stored in the various equipment pieces before failure is given in Table 11A-1 in Appendix 11A.

15.7.1.1.5.3.1.2 Assumptions

The assumed release fractions for the realistic analysis are found in Table 15.7-4. The basis for the assumptions for failure of those pieces of equipment which could have the worst dose consequence are as follows:

a. Charcoal Adsorbers and Desiccant Vessel

Same as for the design basis analysis except for the solid daughters. There is no reason to believe that any of the solid daughter products formed and retained within the micropore structure of the carbon will be released. Hence, no such release is assumed for the realistic analysis.

b. 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 is conservatively taken as 20 percent. The model used assumes retention and washout of 60 percent of the particulate daughters for the calculation of the holdup pipe inventory.

c. Prefilter

Same as for design basis analysis.

15.7.1.1.5.3.2 Fission Product Transport to the Environment

The release of activity to the environment is determined by applying the release fractions in Table 15.7-4 to the inventories in Table 12A-1 in Appendix 12A. The release rates from letting the SJAE continue to pump for 30 minutes are given in Tables 15.7-2 and 15.7-3, depending on where the worst failure occurs.

15.7.1.1.5.3.3 Results

The calculated exposures for the realistic analysis are presented in Table 15.7-7 resulting from failure of the worst single equipment piece and letting the SJAE continue to pump for 30 minutes after the assumed failure.

15.7.1.2 <u>Malfunction of Main Turbine Gland Sealing System</u>

15.7.1.2.1 Identification of Causes and Frequency Classification

15.7.1.2.1.1 Identification of Causes

Plausible malfunctions of the turbine gland sealing system include the failure of the steam seal evaporator and its backup steam supply, failure of the steam packing exhauster fan and excessive pressure in the steam seal header.

15.7.1.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.2.2 Sequence of Events and System Operation Sequence of Events

15.7.1.2.2.1 Identification of Operator Actions

It is assumed that the system fails near the condenser. This results in activity normally processed by the off-gas treatment system being discharged directly to the turbine building and subsequently through the ventilation system to the environment.

The operator should initiate normal shutdown of the reactor to reduce the gaseous activity being discharged. A loss of main condenser vacuum will result in a turbine trip and reactor shutdown.

15.7.1.2.2.2 System Operation

15.7.1.2.2.3 The Effects of Single Failures and Operator Errors

See Appendix 15A for further details.

15.7.1.2.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply systems (NSSS). It will, of course, lead to decoupling of the NSSS with power-conversion system.

The tripping of the main turbine via main condenser signals will result in an anticipated operational transient examined earlier in Chapter 15.

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.1.2.4 Barrier Analysis

This release occurs outside the containment hence does not involve any barrier integrity aspects. However, a discussion of the release of the radioactivity to the environment is presented in order to assess the radiological impact relative to applicable safety limits.

15.7.1.2.5 Radiological Consequences

15.7.1.2.5.1 General

Failure of the steam seal evaporator and its backup steam supply would result in air leakage through the low pressure shaft seals to the condenser and in the discharge of a small amount of contaminated steam from the high pressure shaft seals to the steam packing exhauster. The loss of seal steam to the low pressure seals requires that the turbine be shut down to prevent excessive cooling of the turbine shaft. The small amount of contaminated steam that would be discharged to the atmosphere during the short period before the turbine shutdown is assumed to be inconsequential.

Failure of the steam packing exhauster fan results in the escape of clean steam from the high pressure and low pressure shaft seals. The most undesirable result of operating in this condition is that some condensate from the escaping seal steam could leak into the lube oil system.

Excessive pressure in the sealing steam header as a result of a malfunction of the seal steam evaporator or the backup steam supply valve is prevented by a relief valve so that there is no detrimental effect on the operation of the shaft seals.

15.7.1.3 Failure of Main Turbine Steam Air Ejector Lines

15.7.1.3.1 Identification of Causes and Frequency Classification

15.7.1.3.1.1 Identification of Cause

Those events which could cause a malfunction failure in the main turbine steam air ejector sealing system are:

a. Failure of the steam line to the air ejectors.

b. Failure of the air ejector suction line.

c. Failure of the air ejector discharge line to the off-gas system.

In each of these failures it is assumed that the worst case condition exists and that the failure is in a section of line common to both air ejectors so as to negate the use of the standby air ejector.

15.7.1.3.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.1.3.2 Sequence of Events and System Operation

15.7.1.3.2.1 Sequence of Events

15.7.1.3.2.2 Identification of Operator Actions

Failure of the steamline to the air ejectors would result in the loss of condenser vacuum and the discharge of radioactive steam to the atmosphere. The high air activity would result in an alarm on the atmospheric radiation monitors and the loss of condenser vacuum would result in a turbine trip and a reactor scram.

Failure of the air ejector suction line would result in the loss of condenser vacuum which would result in a turbine trip and a reactor scram.

Failure of the air ejector discharge line to the off-gas system would result in the discharge of radioactive gas into the atmosphere. This failure would result in a "loss-of-flow to the off-gas system" after which the operator should initiate a normal shutdown of the reactor to reduce the amount of gaseous activity being discharged to the atmosphere.

15.7.1.3.2.3 System Operation

15.7.1.3.2.4 The Effects of Single Failures and Operator Errors

See Appendix 15A for further details.

15.7.1.3.3 Core and System Performance

The failure of this power-conversion system does not directly affect the nuclear steam supply systems (NSSS). It will, of course, lead to decoupling of the NSSS with power-conversion system.

The tripping of the main turbine via main condenser signals will result in an anticipated operational transient examined earlier in Chapter 15.

This failure has no applicable effect on the core or the NSSS safety performance.

15.7.1.3.4 Barrier Analysis

This release occurs outside the containment hence does not involve any barrier integrity aspects. However, a discussion of the release of the radioactivity to the environment is presented in order to assess the radiological impact relative to applicable safety limits.

15.7.1.3.5 Radiological Consequences

15.7.1.3.5.1 Fission Product Release

Of the three lines considered to fail in Section 15.7.1.3.1.1, the most severe radiological consequences offsite would be due to failure of the air ejector discharge line. The assumptions used in calculating the amount of gaseous radioactive materials released from this break are given as follows:

a. Loss of flow in the off-gas system will be indicated by an alarm in the control room. It is conservatively assumed that it takes 15 minutes after the break for the operators to shutdown the plant.

b. During this period, the noble gas activity is conservatively assumed to be released from the break at the same rate it is released from the reactor vessel (i.e., no credit is taken for decay of the isotopes while in transit from the reactor to the point of the break).

c. The iodine activity released from the break is based on assuming 2 percent carryover from the reactor water to the steam and a mass loss of approximately 1,725 pounds through the break before termination of the accident.

d. No credit is taken for plateout of the radioiodine.

e. It is assumed that an equilibrium coolant concentration consistent with an off-gas release rate of 100,000 μCi/second after 30 minutes exists prior to the accident.

15.7.1.3.5.2 Fission Product Transport to the Environment

The following assumptions are used in calculating the amount of activity released to the environs:

a. It is conservatively assumed that all of the iodine and noble gas activity released from the break is instantaneously released to the

environment via the off-gas building ventilation system where it is treated by a series of roughing, HEPA, and charcoal filters.

b. The charcoal filter efficiency is assumed to be 95 percent for the removal of iodine.

c. All other assumptions relating to this event are tabulated in Table 15.7-8. The activity released to the environment is presented in Table 15.7-9.

15.7.1.3.5.3 Results

The calculated exposures for this analysis are presented in Table 15.7-10 and are a very small fraction of 10 CFR 100 guidelines.

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM FAILURES (RELEASE TO ATMOSPHERE)

15.7.2.1 Identification of Causes and Frequency Classification

The liquid radwaste treatment systems are generally classified as quality Group D and non-seisimic. Radioactive releases considered include rupture of radwaste tanks, equipment malfunction, or small leaks in the system process lines that transport liquid radwaste. The limiting case considered is the simultaneous failure of the non-seismic, quality Group D components in the system.

15.7.2.2 Sequence of Events and Systems Operations

The sequence of events and systems operations is as follows:

a. Event begins - postulated simultaneous failure of system component occurs.

b. Area radiation alarms alert plant personnel. No credit for any operator action is considered in evaluating this event.

15.7.2.3 Core and System Performance

This event has no effect on the core or NSSS safety functions.

15.7.2.4 Barrier Performance

This release occurs outside the containment, hence does not involve any barrier integrity aspects.

15.7.2.5 Radiological Consequences

The assumptions used to evaluate the simultaneous failure of the non-seismic Group D components in the liquid radwaste system are given in Table 15.7-11, and the radioactive inventory in the system is listed in Table 15.7-12.

In calculating the gaseous releases, 100 percent of the radioactive inventory in each piece of failed equipment is assumed to be released to the building. Of this amount, 10 percent of the iodine and 1 percent of the particulate activity is conservatively assumed airborne in the building atmosphere and available for release to the environment.

The airborne activity is instantaneously released to the environment via the radwaste building ventilation system. However, no credit for filtering is taken in this analysis.

Offsite doses resulting from the design basis event are presented in Table 15.7-13.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

15.7.3.1 Identification of Causes and Frequency Classification

It is considered highly improbable for significant cracks to develop in the Seismic Category 1 safety class buildings containing radioactive waste

materials and equally improbable for the Seismic Category I tanks to fail. However, it is postulated that an unspecified event causes a failure of a tank in the radwaste building and subsequent failure of the radwaste building.

This failure is classified as a limiting fault.

15.7.3.2 Sequence of Events and Systems Operations

The sequence of events and systems operations is as follows:

a. Failure occurs - contents of failed component is released into the radwaste building.

b. Area radiation alarms alert plant personnel.

c. Operator action begins - for the evaluation of this failure no credit is taken for operator action and it is assumed that liquid leaks from the building into the ground.

15.7.3.3 Core and System Performance

The failure of these radwaste components does not directly affect the NSSS.

15.7.3.4 Barrier Performance

This event does not involve any containment barrier integrity.

15.7.3.5 Radiological Consequences

The following methods and assumptions are applied in the analysis of the offsite exposures resulting from the release of liquids to the groundwater from a failure in the liquid radwaste system:

a. For each piece of failed equipment, it is conservatively assumed that 80 percent of the design capacity is immediately released from the building, i.e., no credit is taken for retention of any of the released

liquids in the Seismic Category I radwaste building. The radioactive inventory in the system is listed in Table 15.7-12.

b. After the liquids leave the building they enter the porous concrete mat and are mixed with clean, non-contaminated groundwater.

c. The plant operating procedures require that upon indication of a seismic event or high radiation levels in the radwaste building, the service and backup underdrain pumps are manually tripped with a positive, safety-related cutoff switch. In addition, in-line type radiation monitors located directly in the underdrain system effluent discharge line to the gravity discharge manholes will alarm in the control room and automatically stop the service and backup underdrain pumps upon detection of high radioactivity (see Section 11.5). The radwaste building is then inspected to determine whether a gross failure of any components housing radioactive liquids has occurred. If no failure has occurred, the underdrain pumps are reactivated. If failure is discovered, the pumps will not be reactivated until it can be determined that contaminated water has not entered the underdrain system. If radioactivity has been released to the underdrain system, the pumps will not be reactivated, and the groundwater will be allowed to rise to the gravity drain discharge system (Elevation 590.0').

   The time required for this level to be achieved is approximately 23 days. During this time credit is taken for radiological decay of the released radioisotopes. The quantity of clean groundwater available for dilution at this time is conservatively calculated to be 240,000 ft^3.

d. This decayed and diluted mixture then drains via the gravity drain system to the emergency service water pumphouse bay area at a rate of 80 gpm (maximum design groundwater inflow).

e. The isotopic concentrations are then further reduced by mixing with the minimum emergency service water flow of 25,000 gpm. No credit is assumed

for any dilution with the non-contaminated water in the emergency service water pumphouse bay.

f. The emergency service water pump would normally discharge to Lake Erie via the plant discharge tunnel where the radioactive liquids would be well mixed (diluted) with the non-contaminated lake water. In the event of a collapse of blockage of the non-seismic portion of the discharge tunnel piping, however, the emergency service water system will discharge via a standpipe to the yard outside of the auxiliary building. At this location a grass swale is provided to carry the flow from the auxiliary building area, between the cooling towers, to the minor stream diversion on the east side of the plant. This water then flows in the stream diversion over the sediment control dam and ultimately enters Lake Erie at the shoreline. If this path were used by the effluents following the postulated accident, dilution of the radioactive liquids would occur in the minor stream diversion and in Lake Erie with the non-contaminated lake water. In calculating the resultant individual exposures, however, it was conservatively assumed not to take credit for the dilution mechanisms available in either of the discharge paths, i.e., the discharge tunnel or the grass swale.

g. No credit is taken for any settling or plating out of the radioisotopes.

h. The dose conversion factors for the isotopes considered are taken from Reference 2.

i. For the purposes of calculating the average fraction of MPC the total release of the radioisotopes into the lake is averaged over a one year period.

j. The resultant ingestion exposure is calculated for an individual drinking potentially contaminated water for a period of one year at a rate of 1,200 cc/day. The isotopic concentrations in this water are conservatively assumed to be the concentrations calculated at the discharge of the emergency service water pipe, corrected for radiological decay.

k. Even though there will be mixing of the released effluents with the clean lake water, the actual amount is difficult to quantify. Many factors such as lake conditions, currents, and temperatures influence this process and, therefore, as stated above, all of the resultant doses and isotopic concentrations given in the tables are taken at the discharge of the emergency service water pipe.

l. After the initial isotopic concentrations are calculated at the discharge of the emergency service water pipe, no credit is taken for any further dilution that will be afforded by the incoming 80 gpm of clean groundwater or the continuing 25,000 gpm of emergency service water flow.

The result exposures from liquid releases to the groundwater is presented in Table 15.7-14.

The individual isotopic concentrations and fraction of maximum permissible concentrations (FMPC) for the radionuclides released by a postulated failure of the waste collector tank are given in Table 15.7-15. A summary of the total isotopic concentration and total FMPC for all of the other components postulated to fail is given in Table 15.7-16. The results presented in both of these tables include instantaneous and average annual values taken at the discharge of the emergency service water pipe.

As indicated by these results the annual concentrations yield values, even at the discharge from the emergency service water pipe, are well below the 10 CFR 20 MPC limits for unrestricted areas (10 CFR 20, Appendix B, Table II, Column 2). Likewise, the resultant annual exposures are a small fraction of acceptable limits for this type of event. The instantaneous fraction of MPC at the discharge of the emergency water service pipe also remains less than 1.0 for all of the postulated component failures.

15.7.4 FUEL HANDLING ACCIDENT OUTSIDE CONTAINMENT

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.

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 unchanneled spent fuel in the spent fuel racks in the fuel handling building. The sequence of events which is assumed to occur is as follows:

Event	Approximate Elapsed Time
a. 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.	0
b. Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
c. Gases pass from the water to the fuel handling building.	0

	Event	Approximate Elapsed Time
d.	The fuel handling building ventilation system high radiation alarm alerts plant personnel.	<1 Min
e.	Operator actions begin.	<5 Min

15.7.4.2.2 Identification of Operator Actions

The operator actions are as follows:

a. Initiate the evacuation of the fuel handling building and the locking of the building doors.

b. The fuel handling supervisor should give instructions to go immediately to the radiation protection personnel decontamination area.

c. The fuel handling supervisor should make the operations shift supervisor aware of the accident.

d. The shift supervisor should initiate action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the fuel handling building.

e. Determine if the fuel handling area ventilation system (FHAVS) is operating.

f. The shift supervisor should post the appropriate radiological control signs at the entrance of the fuel handling building.

g. Before entry to the fuel handling 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 operation of the fuel handling area ventilation 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 FHAVS is designed to single failure criteria and safety requirements.

Refer to Sections 7.6 and 9.4 and to Appendix 15A for further 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 percent uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of 76 percent fuel, 19 percent cladding, and 5 percent 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. Conservation 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 listed below:

a. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

b. 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, the fuel grapple, or the grapple cable breaks.

c. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

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 assumed height of 10 ft (actual height is 8 ft), results in an impact velocity of 25.4 ft/sec.

The kinetic energy acquired by the falling fuel assembly is less than 8,000 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 63 percent.

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 88 ft-lb (approximately 2 percent 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. In calculating the activity release, however, it is conservatively assumed that one rod fails on the 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	63 percent
Second impact	35 percent
Third impact	2 percent (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 0.63 x 8,000 or 5,040 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent 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 8 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 failure of all the remaining rods of the 2 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 struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 5{,}040 \quad \times \quad \dfrac{19}{19+5}}{250} = 8$$

Thus, during the first impact, fuel rod failures are as follows:

Dropped assembly	62 rods (bending)
Struck assemblies	16 tie rods (bending)
Struck assemblies	8 rods (compression)
	86 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:

$$\frac{\frac{0.35}{2} \times 8,000 \times \frac{19}{19+5}}{250} = 5$$

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	16 tie rods (bending)
Struck assemblies	5 rods (compression)
	21 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	86 rods
Second impact	21 rods
Third impact	1 rods
	108 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 Section 15.7.4.5.

15.7.4.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. 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 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

b. The second analysis is based on assumptions considered to provide a realistic 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 1,000 days of continuous operation at 3,758 MWt. A 24 hour period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based on NRC Standard Review Plan 15.7.4 and NRC Regulatory Guide 1.25. Specific values of parameters used in the evaluation are presented in Table 15.7-17.

15.7.4.5.1.1 Fission Product Release from Fuel

The following conditions are assumed applicable for this event:

a. The fuel rod gap activity is assumed to consist of 10% of the total halogen and noble gas activity in the rods at the time of the accident, except for Kr-85 which is assumed to be 30%.

b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be available for release.

c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.

Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-18.

15.7.4.5.1.2 Fission Product Transport to the Environment

In accordance with the criteria presented in Regulatory Guides 1.25 and 1.52 it is assumed that the airborne activity in the fuel handling building (Table 15.7-18) is released to the environment over a 2 hour period via a 95 percent iodine efficient FHAVS. The total activity released to the environment is presented in Table 15.7-19.

15.7.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-20 and are well within the guidelines of 10 CFR 100.

15.7.4.5.2 Realistic Analysis

The realistic analysis is based on a realistic but still conservative assessment of this accident. Specific values of parameters used in the evaluation are presented in Table 15.7-17.

15.7.4.5.2.1 Fission Product Release from Fuel

Fission product release estimates for the fuel handling accident are based on the following assumptions:

a. An average of 1.8 percent of the noble gas activity and 0.32 percent 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[(3)].

b. Because of the negligible particulate activity available for release from the fuel plena, none of the solid fission products are assumed to be released.

c. It is conservatively assumed that 100 percent of the noble gas plenum activity and 1.0 percent of the halogen plenum activity in the damaged fuel rods is released from the spent fuel pool to the fuel handling building atmosphere.

Based on the above conditions the activity airborne in the fuel handling building is presented in Table 15.7-21.

15.7.4.5.2.2 Fission Product Transport to the Environment

It is conservatively assumed that all activity released to the fuel handling building is released to the environment in the first two hours after the accident via a 95 percent iodine efficient FHAUS. Based on these assumptions, the activity released to the environment is shown in Table 15.7-22.

15.7.4.5.2.3 Results

The calculated exposure for the realistic analysis are presented in Table 15.7-23 and are well below the guidelines set forth in 10 CFR 100.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

15.7.5.1 Cask Drop from Transport Vehicle

In the unlikely event that the fuel cask falls from the transport vehicle, the maximum height which the cask will drop should be in general less than 10 ft. Since the cask is designed to withstand a 30 ft drop onto a non-yielding surface without failure, the fall from the transport vehicle will cause no failure of the cask.

15.7.5.2 Cask Drop from Crane

The Mark III type containment desi .n includes a separate fuel handling building. The spent fuel storage pools in th .s building are arranged so that the overhead crane which handles the cask cannot possibly move the cask above the spent fuel storage pool. This precludes the possibility of the cask falling on the stored spent fuel bundles. Also, the pools are arranged so that a rupture of the cask loading pool floor will not drain water from the spent fuel storage pool. The cask loading area design and operating procedures are specifically formulated so that a cask drop will not result in failure of the cask. The cask is designed to sustain a free fall in air of 30 ft to an unyielding surface without failure.

15.7.6 FUEL HANDLING ACCIDENT INSIDE CONTAINMENT

15.7.6.1 Identification of Causes and Frequency Classification

15.7.6.1.1 Identification of Causes

Various mechanisms for fuel failure during refueling have been investigated. The refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system is able to initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products to the containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

15.7.6.1.2 Frequency Classification

This event has been categorized as a limiting fault.

15.7.6.2 Sequence of Events and Systems Operations

15.7.6.2.1 Sequence of Events

The sequence of events which is assumed to occur is as follows:

	Event	Approximate Elapsed Time
a.	Channeled fuel bundle being removed from reactor vessel by crane. Fuel bundle is dropped from maximum height allowed by the refueling equipment. Fuel bundle strikes core.	0
b.	Some rods in both dropped and struck bundles fail releasing radioactive gases to pool water.	0
c.	Gases pass from water immediately to building	0
d.	Containment vessel and drywell purge ventilation system isolates due to high radiation signal.	20 sec.
e.	Evacuation alarm in reactor building manually initiated by control room operator.	< 1 min.
f.	Operator action begin	< 5 min.

15.7.6.2.2 Identification of Operator Actions

The operator actions are as follows:

a. The first action of the men involved would be to immediately evacuate the reactor building.

b. The fuel handling supervisor should instruct his men to go immediately to the radiation prote [illegible] [illegible]sonnel decontamination area.

c. The fuel handling supervisor should make the operations shift supervisor aware of the drop accident.

d. The shift supervisor is to immediately determine if the normal ventilation system has isolated and the annulus exhaust gas treatment system is in operation.

e. The shift supervisor should initiate action to determine the extent of radiolytic gas release by measuring the radiation levels in the vicinity or close to the reactor building.

f. As soon as possible, an environmental study made to determine if the annulus exhaust gas treatment system is performing per design.

g. The shift supervisor is to have posted to the entrance of the reactor building, "HIGH RADIATION AREA" signs.

h. Before entry is made to the reactor building after the accident, a careful study of conditions, radiation levels, etc., is to be made.

15.7.6.3 Core and System Performance

The methods used for this evaluation are assumed to be identical to those presented in Section 15.7.4.3.

15.7.6.4 Radiological Consequences

Two separate radiological analyses are provided for this accident:

a. 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 10 CFR 100 guidelines. This analysis is referred to as the "design basis analysis".

b. The second analysis is based on assumptions considered to provide a realistic conservative estimate of radiological consequences. This anaysis 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 1,000 days of continuous operations at 3,758 MWt. A 24 hour period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hours following initiation of reactor shutdown. Specific values for parameters used in the analysis are provided in Table 15.7-24.

15.7.6.4.1 Design Basis Analysis

15.7.6.4.1.1 Fission Product Release from Fuel

The fission product activity released from the fuel damaged as a result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.1.1. A total of 124 fuel rods fail as a result of this accident.

15.7.6.4.1.2 Fission Product Activity Airborne in the Reactor Building

The following assumptions and initial conditions are used in calculating the fission product activity released to the reactor building.

a. The iodine gap inventory is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

b. The minimum water depth between the top of the damaged fuel rods and the containment pool surface is 23 feet.

c. The pool decontamination factors for the inorganic and organic species of iodine are 133 and 1, respectively, giving an overall effective decontamination factor of 100 (i.e., 99 percent of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75 percent inorganic and 25 percent organic species.

d. The retention of noble gases in the pool is negligible (i.e., decontamination factor of 1).

e. The effects of plateout and fallout are neglected.

Based on these assumptions, the activity released from the pool to the reactor building is listed in Table 15.7-25.

15.7.6.4.1.3 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the fission products released to the environs:

a. The containment vessel and drywell purge system are in operation at the time the accident occurs. These systems are described in Section 9.4. It is conservatively assumed that isolation of the containment vessel and drywell purge system will occur 20 seconds after the release of fission product activity from the containment pool due to a high radiation signal in this system.

b. All activity entering the containment pool air exhaust duct system during the first 20 seconds after the accident is assumed to be released to the environs via the containment vessel and drywell purge exhaust system filter as a "puff" release.

c. No credit is taken for filtering iodine.

d. The activity remaining in containment is released to the environs (via the annulus exhaust gas treatment system) based on assumptions identical to those presented in Section 15.6.5.

Based on these assumptions, the activity released to the environment during the first 20 seconds following the release of activity from the containment pool is presented in Table 15.7-26.

15.7.6.4.1.4 Results

Based on these assumptions, the integrated whole body doses and integrated thyroid dose at the exclusion boundary and low population zone are summarized in Table 15.7-27. The doses at these distances are well below the 10 CFR 100 limits.

15.7.6.4.2 Realistic Analysis

15.7.6.4.2.1 Fission Product Release from Fuel

The fission product activity released from the fuel damaged a result of a fuel handling accident is calculated using the methods outlined in Section 15.7.4.5.2.1. As a result of this accident, 124 fuel rods are assumed to fail.

15.7.6.4.2.2 Fission Product Activity Released to Containment

The following assumptions and initial conditions are used in calculating the fission products released to the containment:

a. The fission product activity released to the containment will be in proportion to the removal efficiency of the water in the upper containment 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 containment.

b. The iodine activity in the fuel rod plena is composed of inorganic species (99.75 percent) and organic species (0.25 percent).

c. The removal efficiency of the water for iodines can be defined in terms of the partition factor, for which values between 10^3 and 10^8 have been experimentally determined to be applicable for the conditions under investigation. A partition factor of 10^2 for the iodines has been

conservatively assumed for this accident. Thus, the computed inhalation exposures will be overestimated by a factor of from 10 to 10^6.

d. The effects of plateout and fallout are neglected.

Based on these assumptions, the activities released from the pool to the refueling building are shown in Table 15.7-28.

15.7.6.4.2.3 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the fission products released to the environs:

a. The containment vessel and drywell purge system are in operation at the time the accident occurs. These systems are described in Section 9.4. It is conservatively assumed that isolation of the containment vessel and drywell purge system will occur 20 seconds after the release of fission product activity from the containment pool due to a high radiation signal in this system.

b. All activity entering the containment pool air exhaust duct system during the first 20 seconds after the accident is assumed to be released to the environs via the containment vessel and drywell purge exhaust system filter as a "puff" release.

c. No credit is taken for filtering iodine.

d. The activity remaining in containment is released to the environs (via the annulus exhaust gas treatment system) based on assumptions identical to those presented in Section 15.6.5.

Based on these assumptions, the activity released to the environment during the first 20 seconds following the release of activity from the containment pool is presented in Table 15.7-29.

15.7.6.4.2.4 Results

Based on these assumptions, the integrated whole body doses and integrated thyroid dose at the exclusion boundary and low population zone are summarized in Table 15.7-30. The doses at these distances are well below the 10 CFR 100 limits.

15.7.7 REFERENCES FOR SECTION 15.7

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

2. Bunch, F. D., "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides", IDO-12054, AEC Research and Development Report, TID-4500, August 1966.

3. N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor," March 1969, (APED5756).

TABLE 15.7-1

PROBABLE 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

RECHAR SYSTEM SJAE PIPE BREAK ACTIVITY RELEASE RATE

INPUT: 100,000 μCi/sec Mix

Isotope	μCi/sec	Isotope	μCi/sec	Isotope	μCi/sec
T-3	1.128+0	SE-88	2.307-3	ZR-95	4.084-5
N2-13	3.691+3	BR-88	6.702+0	NB-95m	8.958-7
A-13m	9.577+0	KR-88	2.231+4	NB-95	4.039-5
NO-13	5.922-2	RB-88	2.873-1	ZR-97	3.208-4
C-14	1.037-1	SE-89	4.997-7	NB-97m	1.221-1
N2-16	8.270+6	BR-89	2.942+0	NB-97	3.986-3
A-16m	3.623+4	KR-89	2.356+5	ZR-99	9.221-2
NO-16	8.801+1	RB-89	3.572+0	NB-99m	2.746-2
N2-17	1.373+3	SR-89	2.533-3	NB-99	1.583-1
A-17m	7.679+0	Y-89m	1.297-8	MO-99	3.149-4
NO-17	3.422-2	BR-90	4.051-1	TC-99m	1.685-1
F-18	4.952+0	KR-90	5.912+5	TC-99	2.382-8
O-19	7.677+2	RB-90m	3.748+0	MO-101	1.654-2
NA-24	2.294-3	RB-90	4.749-1	TC-101	2.114-1
P-32	2.288-5	SR-90	1.442-4	MO-102	1.744-2
CR-51	5.852-4	Y-90m	4.208-9	TC-102m	1.815-1
MN-54	4.566-5	Y-90	1.361-6	TC-102	5.549-5
MN-56	5.873-2	BR-91	1.702-3	TC-103	1.310-1
CO-58	5.822-3	KR-91	5.157+5	RU-103	2.225-5
FE-59	9.343-5	RB-91	1.576+2	RH-103m	2.442-3
CO-60	5.511-4	SR-91	7.977-2	MO-104	2.996-2
NI-65	3.524-4	Y-91m	1.855-3	TC-104	7.260-2
ZN-65	2.263-6	Y-91	2.067-5	MO-105	1.962-2
ZN-69m	3.442-5	BR-92	6.636-8	TC-105	4.045-2
AS-83	1.306-2	KR-92	2.316+4	RU-105	1.996-4
SE-83m	7.043-3	RB-92	2.364+2	RH-105m	6.637-3
SE-83	4.631-4	SR-92	2.785-1	RH-105	2.874-5
BR-83	2.583+0	Y-92	9.591-4	TC-106	1.621-2
KR-83m	4.449+3	KR-93	1.809+3	RU-106	1.579-6
AS-84	8.704-3	RB-93	4.965+1	AG-110m	6.876-5
SE-84	1.056-2	SR-93	8.862-1	SB-129	1.207-4
BR-84m	9.329-2	Y-93	6.444-4	TE-129m	5.988-6
BR-84	5.602+0	ZR-93	6.726-11	TE-129	7.822-4
AS-85	2.242-3	NB-93m	7.577-12	I-129	1.574-8
SE-85m	1.378-2	KR-94	7.887-6	SB-131	4.986-3
SE-85	1.680-2	RB-94	2.165-1	TE-131m	2.522-5
BR-85	5.481+0	SR-94	2.759-1	TE-131	5.295-3
KR-85m	6.625+3	Y-94	1.670-2	I-131	1.878+0
KR-85	1.680+1	KR-95	2.787-2	XE-131m	3.481+1
AS-87	8.716-5	RB-95	1.176-3	TE-132	2.010-4
SE-87	3.002-2	SR-95	2.931-1	I-132	2.399+1
BR-87	7.509+0	Y-95	2.907-2	SB-133	2.915-2
KR-87	2.287+4				

TABLE 15.7-2 (Continued)

Isotope	μCi/sec	Isotope	μCi/sec	Isotope	μCi/sec
TE-133m	3.983-3	CS-137	1.493-4	LA-142	3.687-3
TE-133	1.170-2	BA-137m	1.706-4	XE-143	3.099-4
I-133m	3.476-1	XE-138	1.332+5	CS-143	1.462-1
I-133	1.378+1	CS-138m	1.037-2	BA-143	3.556-1
XE-133m	3.430+2	CS-138	9.669-1	LA-143	2.127-2
XE-133	8.360+3	XE-139	6.076+5	CE-143	1.830-4
TE-134	8.721-3	CS-139	1.597+1	PR-143	5.178-5
I-134m	1.901+0	BA-139	1.078-1	XE-144	5.810+0
I-134	4.388+1	XE-140	5.723+5	CS-144	1.937-1
XE-134m	2.155-1	CS-140	1.445+2	BA-144	4.804-1
I-135	2.207+1	BA-140	8.676-3	LA-144	1.787-1
XE-135m	3.965+4	LA-140	6.328-5	CE-144	2.418-5
XE-135	2.486+4	XE-141	1.447+4	ND-147	2.155-5
CS-135m	2.390-5	CS-141	6.001+1	P-147m	5.395-6
CS-135	1.922-9	BA-141	4.697-1	ND-149	5.058-4
TE-136	5.217-2	LA-141	1.373-3	P-149m	2.339-5
I-136m	7.170+0	CE-141	4.086-5	W-187	3.523-3
I-136	1.086+1	XE-142	1.086+3	NP-239	2.440-1
I-137	9.789+0	CS-142	3.082+1		
XE-137	2.755+5	BA-142	7.785-1	Total	1.141+7

TABLE 15.7-3

RECHAR SYSTEM HOLDUP PIPE BREAK ACTIVITY RELEASE RATE

INPUT: 100,000 μCi/sec Mix

Isotope	μCi/sec	Isotope	μCi/sec	Isotope	μCi/sec
T-3	1.893-2	SE-88	0.	ZR-95	2.841-15
NO-13	2.108-3	BR-88	2.760-11	NB-95m	6.113-17
A-13m	1.070-1	KR-88	2.144+4	NB-95	2.758-15
NO-13	2.319-12	RB-88	1.923+3	ZR-97	2.179-14
C-14	3.488-2	SE-89	0.	NB-97m	1.769-14
NO-16	3.874-14	BR-89	1.157-34	NB-97	3.089-13
A-16m	3.320-18	KR-89	4.045+4	ZR-99	0.
NO-16	2.826-29	RB-89	1.016+4	NB-99m	2.072-13
NO-17	2.484-32	SR-89	3.465-1	NB-99	5.507-22
A-17m	2.718-36	Y-89m	5.511-28	MO-99	2.258-14
NO-17	0.	BR-90	0.	TC-99m	1.133-11
F-18	2.821-2	KR-90	1.908+1	TC-99	1.627-18
O-19	5.452-5	RB-90m	5.417+2	MO-101	7.727-13
NA-24	1.557-13	RB-90	3.442+3	TC-101	9.883-12
P-32	1.562-15	SR-90	1.572-3	MO-102	7.224-13
CR-51	3.996-14	Y-90m	2.792-19	TC-102m	3.571-12
MN-54	3.119-15	Y-90	8.503-7	TC-102	0.
MN-56	3.870-12	BR-91	0.	TC-103	1.152-14
CO-58	3.976-13	KR-91	4.464-11	RU-103	1.598-15
FE-59	6.381-15	RB-91	1.960+1	RH-103m	1.511-13
CO-60	3.764-14	SR-91	2.814+0	MO-104	6.388-14
NI-65	2.320-14	Y-91m	6.607-2	TC-104	3.712-12
ZN-65	1.546-16	Y-91	2.820-5	MO-105	2.816-15
ZN-69m	2.335-15	BR-92	0.	TC-105	1.427-12
AS-83	1.758-23	KR-92	0.	RU-105	3.224-14
SE-83m	5.018-15	RB-92	1.541-31	RH-105m	7.955-17
SE-83	2.691-14	SR-92	4.758-5	RH-105	2.075-15
BR-83	1.257-2	Y-92	4.682-7	TC-106	1.376-16
KR-83m	4.205+3	KR-93	0.	RU-106	1.037-16
AS-84	7.202-38	RB-93	2.757-28	AG-110m	4.696-15
SE-84	1.375-13	SR-93	3.335-7	SB-129	8.072-15
BR-84m	1.873-4	Y-93	1.645-9	TE-129m	4.090-16
BR-84	2.380-2	ZR-93	1.148-20	TE-129	4.958-14
AS-85	0.	NB-93m	5.175-22	I-129	7.961-11
SE-85m	2.328-20	KR-94	0.	SB-131	2.675-13
SE-85	2.277-16	RB-94	0.	TE-131m	1.746-15
BR-85	4.016-3	SR-94	2.628-13	TE-131	3.121-13
KR-85m	6.446+3	Y-94	1.384-12	I-131	9.498-3
KR-85	1.669+1	KR-95	0.	XE-131m	3.446+1
AS-87	0.	RB-95	0.	TE-132	1.371-14
SE-87	3.154-38	SR-95	5.534-17	I-132	1.165-1
BR-87	9.678-5	Y-95	1.531-12	SB-133	1.973-13
KR-87	2.112+4				

TABLE 15.7-3 (Continued)

Isotope	μCi/sec	Isotope	μCi/sec	Isotope	μCi/sec
TE-133m	2.606-13	CS-137	1.822-2	LA-142	3.552-8
TE-133	5.813-13	BA-137m	1.830-3	XE-143	0.
IM-133	1.532-19	XE-138	8.926+4	CS-143	0.
I-133	6.941-2	CS-138m	1.046-13	BA-143	6.023-22
XE-133m	3.391+2	CS-138	5.981+3	LA-143	1.221-12
XE-133	8.274+3	XE-139	1.376+2	CE-143	1.431-14
TE-134	5.219-13	CS-139	6.160+3	PR-143	3.537-15
IM-134	2.060-3	BA-139	1.939+2	XE-144	0.
I-134	2.000-1	XE-140	1.334-5	CS-144	0.
XE-134m	0.	CS-140	9.379+1	BA-144	8.949-22
I-135	1.101-1	BA-140	2.898-1	LA-144	1.356-13
XE-135m	2.733+4	LA-140	2.492-4	CE-144	1.896-15
XE-135	2.470+4	XE-141	0.	ND-147	1.472-15
CS-135m	1.470-15	CS-141	3.358-8	P-147m	3.685-16
CS-135	4.303-8	BA-141	7.230-5	ND-149	3.275-14
TE-136	4.662-19	LA-141	7.350-7	P-149m	1.620-15
I-136m	2.613-5	CE-141	3.300-11	W-187	2.397-13
I-136	1.096-3	XE-142	0.	NP-239	1.664-11
I-137	6.587-8	CS-142	0.		
XE-137	6.439-4	BA-142	1.270-7	Total	3.388+5

TABLE 15.7-4

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	NA
Catalytic Recombiner	1.00/1.00	1.00/1.00	N/A
Off-gas 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	N/A
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	0.01/0.0	0.01/0.01
Afterfilter	1.00/1.00	0.01/0.01	N/A

TABLE 15.7-5

GASEOUS RADWASTE SYSTEM FAILURE
CHARCOAL ADSORBER VESSEL
(DESIGN BASIS ANALYSIS)
OFF-SITE RADIOLOGICAL EFFECTS (REM)

Distance (m)	Bone	Liver	Whole Body	Thyroid	Kidney	Lung	GI
863	2.6	1.6-1	6.0-1	1.9-5	4.4-2	1.2+1	2.9
4,002	3.1-01	2.0-03	7.4-02	2.2-06	5.4-03	1.4	0.35

TABLE 15.7-6

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 postulated accidents		
A. Power level	NA	NA
B. Burn-up	NA	NA
C. Fuel damage	None	None
D. Inventory of activity by nuclide	App. 12A	App. 12A
E. Iodine fractions		
(1) Organic	0	0
(2) Elemental	1.0	1.0
(3) Particulate	0	0
F. Reactor coolant activity before the accident	NA	NA
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. Isolation valve closure time (sec)	NA	NA
D. Adsorption and filteration efficiencies:	NA	NA
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA

TABLE 15.7-6 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(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		
III. Dispersion data		
A. Boundary and LPZ distances (m)	863/4002	863/4002
B. χ/Q's for SB/LPZ (sec/m^3)	6.7-4/8.2-5	3.8-5/3.8-6
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

TABLE 15.7-7

GASEOUS RADWASTE SYSTEM FAILURE
CHARCOAL ADSORBER VESSEL
(REALISTIC ANALYSIS)
OFFSITE RADIOLOGICAL EFFECTS (REM)

Distance (m)	Bone	Liver	Whole Body	Thyroid	Kidney	Lung	GI
863	0	0	1.4-2	1.9-6	0	0	0
4,002	0	0	1.4-2	6.9-8	0	0	0

TABLE 15.7-8

FAILURE OF THE AIR EJECTOR LINE - 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	None	NA
	B. Burn-up	None	NA
	C. Fuel damage	None	None
	D. Inventory of activity by nuclide	None	Table 15.7-9
	E. Iodine fractions		
	(1) Organic	None	0
	(2) Elemental	None	1
	(3) Particulate	None	0
	F. Reactor coolant activity before the accident	None	Section 15.6.4.5.2.2
II.	Data and assumptions used to estimate activity released		
	A. Containment leak rate (% day)	None	NA
	B. Secondary containment leak rate (% day)	None	NA
	C. Isolation valve closure time (sec)	None	NA
	D. Adsorption and filtration efficiencies:		
	(1) Organic iodine	None	NA
	(2) Elemental iodine	None	NA

TABLE 15.7-8 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
(3) Particulate Iodine	None	NA
(4) Particulate fission products	None	NA
E. Recirculation system parameters		
(1) Flow rate	None	NA
(2) Mixing Efficiency	None	NA
(3) Filter Efficiency	None	NA
F. Containment spray parameters (flow rate, drop size, etc.)	None	NA
G. Containment volumes	None	NA
H. All other pertinent data and assumptions	None	NA
III. Dispersion data		
A. Boundary and LPZ distances (m)	None	863/4,002
B. χ/Q's for total dose - SB/LPZ	None	6.7-4/8.2-5
IV. Dose Data		
A. Method of dose calculation	None	15.0.3.5
B. Dose conversion assumptions	None	15.0.3.5
C. Peak activity concentrations in containment	None	NA
D. Doses	None	Table 15.7-10

TABLE 15.7-9

FAILURE OF AIR EJECTOR LINE
(REALISTIC ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.56-5
I-132	2.03-4
I-133	1.17-4
I-134	3.75-4
I-135	1.88-4
Kr-83m	3.79+0
Kr-85m	2.59-2
Kr-85	6.64+0
Kr-87	2.07+1
Kr-88	2.12+1
Kr-89	8.83+1
Xe-131m	2.12-2
Xe-133m	3.17-1
Xe-133	8.87+0
Xe-135m	2.59+1
Xe-135	2.39+1
Xe-137	1.17+2
Xe-138	8.83+1

TABLE 15.7-10

FAILURE OF AIR EJECTOR LINE
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area (863 Meters)	6.33-2	3.18-5
Low Population Zone (4,002 Meters)	7.75-3	3.89-6

TABLE 15.7-11

LIQUID WASTE SYSTEM FAILURE 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	None	NA
	B. Burn-up	None	NA
	C. Fuel damage	None	None
	D. Inventory of activity by nuclide	None	Table 15.7-12
	E. Iodine fractions		
	(1) Organic	None	0
	(2) Elemental	None	1
	(3) Particulate	None	0
	F. Reactor coolant activity before the accident based on	None	.1 Ci/sec @ 30 min
II.	Data and assumptions used to estimate estimate activity released		
	A. Primary containment leak rate (% day)	None	NA
	B. Secondary containment leak rate (% day)	None	NA
	C. Isolation valve closure time (sec)	None	NA
	D. Adsorption and filtration efficiencies - radwaste bldg.		
	(1) Organic iodine	None	NA
	(2) Elemental iodine	None	NA

TABLE 15.7-11 (Continued)

		Design Basis Assumptions	Realistic Basis Assumptions
	(3) Particulate iodine	None	NA
	(4) Particulate fission products	None	NA
E.	Recirculation system parameters		
	(1) Flow rate	None	NA
	(2) Mixing efficiency	None	NA
	(3) Filter efficiency	None	NA
F.	Containment spray parameters (flow rate, drop size, etc.)	None	NA
G.	Containment volumes	None	NA
H.	All other pertinent data and assumptions	None	NA
III.	Dispersion data		
A.	Boundary and LPZ distances (m)	None	863/4,002
B.	χ/Q's for total dose - SB/LPZ	None	6.7-4/8.2-5
IV.	Dose Data		
A.	Method of dose calculation	None	15.0.3.5
B.	Dose conversion assumptions	None	15.0.3.5[1]
C.	Peak activity concentrations in containment	None	NA
D.	Doses	None	Table 15.7-13

NOTE:

1. The dose conversion factors for the other isotopes considered are taken from "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides" by D. F. Bunch, IDO-12054, AEC Research and Development Report, TID-4500, August 1966.

TABLE 15.7-12

INVENTORY ACTIVITIES FOR LIQUID RADWASTE SYSTEM COMPONENTS(1)
(microcuries)

Isotope	Waste Collector Tank	Waste Sample Tank	Floor Drains Collector Tank	Floor Drains Sample Tank	Chemical Waste Tank	Concentrated Waste Tank	Chemical Waste Distillate Tank	Spent Resin Tank	Cond. Filter Backwash Rec. Tank	Cond. Filter Backwash Settling Tank	RWCU F/D Backwash Settling Tank	Fuel Pool F/D Backwash Settling Tank
F-18	3.17+4	1.48+4	1.12+3	[illegible]	1.17+4	0.0	3.51-3					
Na-24	2.55+4	4.58+3	4.46+3	[illegible]	7.99+3	6.33+4	4.81-1		5.60+4	8.27+4	2.77+6	3.30+4
P-32	4.17+2	8.30+1	7.22+1	[illegible]	8.29+1	8.24+2	8.02-3		4.30+3	1.60+4	1.11+6	1.40+4
Cr-51	1.05+4	2.10+3	1.82+3	[illegible]	2.13+3	2.06+4	2.11-1		1.20+5	4.63+5	4.64+7	6.20+5
Mn-54	8.54+2	1.70+2	1.45+2	[illegible]	1.64+2	1.64+3	1.64-2		1.00+4	3.99+4	6.94+6	1.30+5
Mn-56	1.62+5	1.83+4	2.86+4		1.79+5	6.14+5	9.61-1		2.30+5	2.30+5	1.10+7	3.40+4
Co-58	1.06+5	2.13+4	1.83+4	[illegible]	2.14+4	2.08+5	2.13+0		1.20+6	4.73+6	6.97+8	1.10+7
Co-60	1.07+4	2.14+3	1.85+4	3.42+2	2.14+3	2.08+4	2.14-1		1.30+5	5.20+5	9.21+7	1.60+6
Fe-59	1.69+3	3.39+2	2.88+2	5.25+1	3.29+2	3.19+3	3.27-2		1.90+4	7.43+4	9.47+6	1.40+5
Ni-65	9.69+2	1.09+2	1.69+2	3.37+0	1.04+3	3.53+3	5.47-3		1.40+3	1.40+3	6.50+4	2.00+2
Zn-65	4.26+1	8.53+0	7.38+0	1.36+0	8.25+0	8.32+1	8.25-4		5.10+2	2.03+3	3.50+5	5.90+3
Zn-69m	3.72+2	6.63+1	6.38+1	3.89+0	1.20+2	9.34+2	6.92-3		7.80+2	1.11+3	3.78+4	4.40+1
I-131	2.79+5	2.76+3	4.55+4	3.91+2	1.84+7	1.80+6	1.76+3	5.32+6			4.50+8	
I-134	2.81+5	7.29+2	4.75+4	2.10+1	1.79+6	2.21+6	3.10-2	3.27+4			2.40+7	
Sr-89	6.57+4	6.57+2	1.13+4	1.04+2	1.02+6	1.02+7	1.10+2	7.62+6			4.25+8	
Cs-134	3.63+3	3.36+1	6.20+2	5.76+0	8.93+4	9.07+5	8.93+0	9.78+5			3.33+7	
Cs-136	2.30+3	2.28+1	3.99+2	3.42+0	1.35+4	1.33+5	1.32+0	7.51+4			6.32+6	
W-187	4.59+4	[illegible]	7.89+3	6.75+2	1.21+4	1.05+5	8.86-1		1.40+5	2.63+5	7.15+6	9.80+5
Sr-90	5.12+3	5.12+1	8.84+2	8.09+0	1.31+5	1.32+6	1.31+1	1.50+6			4.93+7	
Y-90	6.19+2	7.38+0	1.08+2	2.39+0	8.02+2	7.79+4	1.54+0	1.21+6			4.48+7	
Sr-92	4.09+5	2.38+3	7.09+4	7.31+1	5.16+5	1.82+6	3.19+0	1.12+5			3.00+7	
Y-92	4.00+5	3.89+3	6.88+4	1.48+2	8.40+4	1.67+6	1.05+1	1.69+5			6.00+6	
Mo-99	4.34+5	4.24+3	7.47+4	5.02+2	6.54+5	6.27+6	5.84+1	2.88+6			2.17+8	
Tc-99m	2.44+6	1.99+4	4.24+5	1.07+3	1.79+6	1.32+7	9.54+1	1.53+6			2.86+8	
Ru-103	4.24+2	4.22+0	7.33+1	6.61-1	5.65+3	5.65+4	5.60-1	3.89+4			2.51+6	
Rh-103m	3.83+2	4.00+0	6.63+1	6.12-1	1.79+3.	4.93+4	5.38-1	1.96+4			2.21+6	
Ru-106	5.77+1	5.76-1	9.89+0	9.20-2	1.37+3	1.38+4	1.37-1	1.42+4			5.22+6	
Ag-110m	1.28+4	2.56+2	2.24+2	4.05+1	2.47+2	2.46+3	2.47-2		1.50+4	5.98+4	1.04+7	1.80+5
Te-132	9.60+5	9.40+3	1.66+5	1.15+3	1.63+6	1.55+7	1.48+2	7.31+6			5.80+8	
I-132	1.20+6	1.08+4	2.05+5	1.14+3	2.26+6	1.78+7	1.54+2	4.25+5			3.27+8	
I-135	1.06+6	8.36+3	1.76+5	3.45+2	6.08+6	3.74+5	1.97+2	6.96+5			8.30+7	
Cs-137	5.35+3	5.35+1	9.11+2	6.05+0	1.37+5	1.38+6	1.37+1	1.54+6			5.12+7	
Ba-140	1.91+5	1.90+3	3.34+4	2.82+2	1.11+6	1.10+7	1.07+2	5.86+6			5.02+8	
La-140	3.50+4	4.12+2	6.10+3	1.22+2	1.36+4	1.04+6	1.98+1	1.21+6			4.48+8	
Ce-143	6.02+2	5.73+0	1.04+2	5.25-1	5.79+2	5.27+3	4.56-2	1.97+3			1.37+5	
Pr-143	8.32+2	8.30+0	1.44+2	1.47+0	4.84+3	4.88+4	4.74-1	2.76+4			2.32+6	
Ce-144	7.68+2	7.68+0	1.32+2	1.22+0	1.79+4	1.80+5	1.79+0	1.81+5			6.91+6	
Nd-147	2.89+2	2.88+0	4.99+1	4.24-1	1.42+3	1.40+4	1.38-1	7.69+3			6.92+5	
Pm-147	3.01-1	3.22-3	1.82-2	1.32-2	2.51-1	4.61+0	7.11-5	4.69+2			2.03+4	
Np-239	4.61+6	4.47+5	7.88+5	5.02+3	6.20+6	5.86+5	5.43+2	2.57+7			1.95+9	

TABLE 15.7-12 (Continued)

Isotope	Waste Collector Tank	Waste Sample Tank	Floor Drains Collector Tank	Floor Drains Sample Tank	Chemical Waste Tank	Concentrated Waste Tank	Chemical Waste Distillate Tank	Spent Resin Tank	Cond. Filter Backwash Rec. Tank	Cond. Filter Backwash Settling Tank	RWCU F/D Backwash Settling Tank	Fuel Pool F/D Backwash Settling Tank
Pu-239	1.53+4	1.53+2	3.22-2	1.00+3	4.56-1	5.52+0	6.62-5	1.15+7			5.65+3	
Sr-91	7.26+5	6.16+3	1.26+5	2.92+2	5.21+5	3.76+6	2.38+1	6.97+5			6.73+7	
Y-91m	4.26+5	3.92+3	7.41+4	1.69+2	1.23+5	2.21+6	1.57+1	6.70+4			1.47+6	
Y-91	1.83+4	1.90+2	7.74+2	9.19+2	5.47+3	6.31+4	7.09-1	4.96+6			7.33+6	
Zr-95	8.71+2	8.70+0	1.44+2	1.36+0	1.50+4	1.49+5	1.49+0	1.16+5			6.11+6	
Nb-95	9.19+2	9.17+0	1.58+2	1.45+0	1.19+4	1.15+5	1.14+0	1.47+5			7.63+6	
Zr-97	4.46+2	4.07+0	7.75+1	2.68-1	3.35+2	2.78+3	2.15-2	7.73+2			5.74+4	
Nb-97	4.26+2	4.18+0	7.39+1	2.55-1	9.69+1	2.55+3	7.03-3	9.10+1			1.20+3	
Te-129m	8.67+2	8.64+0	1.44+2	1.33+0	1.08+4	1.07+5	1.08+0	6.96+4			4.70+6	
Te-129	5.16+2	5.41+0	8.57+1	8.16-1	2.00+3	6.09+4	6.87-1	1.63+2			2.53+5	
I-133	1.29+6	1.19+4	2.16+5	8.71+2	1.56+6	1.32+7	1.08+2	2.60+6			1.83+8	
Ce-141	1.50+5	1.50+3	6.06+2	9.33+3	2.82+4	2.85+5	2.84+0	4.39+7			1.23+7	

NOTE:

1. Activity in detergent tank is negligible.

TABLE 15.7-13

RADIOLOGICAL DOSES AT THE EXCLUSION BOUNDARY FROM GASEOUS RELEASES FOR FAILURES IN THE QUALITY GROUP D PORTION OF THE LIQUID RADWASTE SYSTEM

	Component	Bone Dose (mrem)	Thyroid Dose (mrem)	Whole Body Dose (mrem)
1.	Waste collector tank	0.52	26.18	5.70
2.	Waste sample tank	(1)	0.208	(1)
3.	Floor drains collector tank	(1)	4.406	(1)
4.	Floor drains sample tank	(1)	(1)	0.39
5.	Chemical water distillate tank	(1)	0.052	(1)
6.	Condensate filter backwash receiving tank	(1)	(1)	(1)
7.	Condensate filter backwash settling tank	(1)	(1)	(1)
8.	Fuel pool F/D backwash settling tank	(1)	(1)	(1)
9.	Remainder of system(2)	0.15	7.48	1.63

NOTES:

1. Doses are less than 10^{-4} mrem and considered negligible.

2. Based on releasing the total volume (approximately 9,800 gallons) of the liquid radwaste system excluding the tanks. It is conservatively assumed that the activity associated with this volume is at the same concentration as in the waste collector tank.

TABLE 15.7-14

ANNUAL INTEGRATED
RADIOLOGICAL DOSES FROM INGESTION OF
WATER FOR FAILURES IN THE QUALITY GROUP D
PORTION OF THE LIQUID RADWASTE SYSTEM

	Component	Whole Body Exposure (mrem)(2)	Thyroid Exposure (mrem)(2)
1.	Waste collector tank	6.0+0	1.4+0
2.	Waste sample tank	6.3-2	1.4-2
3.	Floor drains collector tank	9.6-1	2.3-1
4.	Floor drains sample tank	4.3-2	1.9-3
5.	Chemical waste distillate tank	1.4-2	8.7-3
6.	Condensate filter backwash receiving tank	1.6-1	-
7.	Condensate filter backwash settling tank	1.2-1	-
8.	Waste sludge settling tank	2.1-1	-
9.	Remainder of system(1)	1.7+0	-

(1) Based on releasing the total volume (approximately 9800 gallons) of the liquid radwaste system excluding the tanks. It is conservatively assumed that the activity associated with this volume is at the same concentrations as in the waste collector tank.

(2) Based on the isotopic concentrations calculated to be present at the discharge of the emergency service water pipe.

TABLE 15.7-15

RADIONUCLIDE CONCENTRATIONS AT DISCHARGE OF EMERGENCY SERVICE WATER PIPE FOR FAILURE OF WASTE COLLECTOR TANK

Isotope	MPC	Instantaneous Concentration (μCi/cc)	Instantaneous FMPC	Average Concentration (μCi/cc)	Average FMPC
F-18	5.0-4	(1)	(1)	(1)	(1)
Na-24	3.0-5	(1)	(1)	(1)	(1)
P-32	2.0-5	5.17-11	2.59-6	2.93-12	1.47-7
Cr-51	2.0-3	2.23-9	1.12-6	2.45-10	1.12-6
Mn-54	1.0-4	3.06-10	3.06-6	2.10-10	2.31-7
Mn-56	1.0-4	(1)	(1)	(1)	(1)
Co-58	9.0-5	3.19-8	3.54-4	8.74-9	9.70-5
Co-60	3.0-5	4.00-9	1.33-4	3.73-9	1.24-4
Fe-59	5.0-5	4.47-10	8.94-6	8.05-11	1.61-6
Ni-65	1.0-4	(1)	(1)	(1)	(1)
Zn-65	1.0-4	1.50-11	1.50-7	9.33-12	9.33-8
Zn-69m	6.0-5	(1)	(1)	(1)	(1)
I-131	3.0-7	5.11-8	1.70-1	1.63-9	5.42-3
I-134	2.0-5	(1)	(1)	(1)	(1)
Sr-89	3.0-6	6.33-8	2.11-2	1.27-8	4.24-3
Cs-134	9.0-6	4.69-9	5.21-4	3.99-9	4.43-4
Cs-136	6.0-5	8.88-10	1.48-5	4.56-11	7.61-7
W-187	6.0-5	1.94-15	3.23-11	7.64.18	1.27-13
Sr-90	3.0-7	6.74-9	2.25-2	6.66-9	2.22-2
Y-90	2.0-5	2.06-12	1.03-7	2.16-14	1.54-9
Sr-92	6.0-5	(1)	(1)	(1)	(1)
Y-92	6.0-5	(1)	(1)	(1)	(1)
Mo-99	4.0-5	1.82-9	4.55-5	2.00-11	5.01-7
Tc-99m	3.0-3	(1)	(1)	(1)	(1)
Ru-103	8.0-5	3.75-10	4.69-6	5.93-11	7.41-7
Rh-103m	1.0-2	(1)	(1)	(1)	(1)
Ag-110m	3.0-5	4.53-9	1.51-4	2.85-9	9.51-5
Te-132	2.0-5	9.43-9	4.72-4	1.22-10	6.09-6
I-132	8.0-6	(1)	(1)	(1)	(1)
I-135	4.0-6	(1)	(1)	(1)	(1)
Cs-137	2.0-5	7.05-9	3.53-4	6.97-9	3.49-4
Ba-140	2.0-5	7.25-8	3.63-3	3.67-9	1.84-4
La-140	2.0-5	3.46-12	1.73-7	2.29-14	1.15-0
Ce-143	4.0-5	7.33-15	1.83-10	3.99-17	9.96-13
Pr-143	5.0-5	3.39-10	6.78-6	1.82-11	3.64-7
Ce-144	1.0-5	9.57-10	9.57-5	6.33-10	6.33-5
Nd-147	6.0-5	8.99-11	1.50-6	3.93-12	6.56-8
Pm-147	2.0-4	3.90-13	1.95-9	3.42-13	1.71-9
Np-239	1.0-4	6.87-9	6.87-5	6.38-11	6.38-7
Pu-239	5.0-6	2.02-8	4.04-3	2.02-8	4.04-3

TABLE 15.7-15 (Continued)

Isotope	MPC	Instantaneous Concentration (μCi/cc)	Instantaneous FMPC	Average Concentration (μCi/cc)	Average FMPC
Sr-91	5.0-5	(1)	(1)	(1)	(1)
Y-91m	3.0-3	(1)	(1)	(1)	(1)
Y-91	3.0-5	1.84-8	6.13-4	4.21-9	1.40-4
Zr-95	6.0-5	9.01-10	1.50-5	2.30-10	3.82-6
Nb-95	1.0-4	7.68-10	7.68-6	1.06-10	1.06-6
Zr-97	2.0-5	(1)	(1)	(1)	(1)
Nb-97	9.0-4	(1)	(1)	(1)	(1)
Te-129m	2.0-5	7.17-10	3.59-5	9.68-11	4.85-6
Te-129	8.0-4	(1)	(1)	(1)	(1)
I-133	1.0-6	1.12-14	1.12-8	6.52-18	6.52-12
Ce-141	9.0-5	1.21-7	1.34-3	1.04-11	1.15-7
Ru-106	1.0-5	7.28-11	7.28-6	5.26-11	5.26-6
Total		4.32-7	2.26-1	7.68-8	3.75-2

NOTE:

(1) Concentration at emergency service water pipe discharge is less than 10^{-15} μCi/cc.

TABLE 15.7-16

SUMMARY OF TOTAL CONCENTRATION AND TOTAL FMPC AT DISCHARGE OF EMERGENCY SERVICE WATER PIPE FOR FAILURES OF QUALITY GROUP D EQUIPMENT

	Component	Instantaneous Concentration (μCi/cc)	Average Concentration	Instantaneous FMPC	Average FMPC
1.	Waste collector tank	4.32-7	7.67-8	2.26-1	3.75-2
2.	Waste sample tank	1.24-8	3.29-9	2.35-3	4.24-4
3.	Floor drains collector tank	5.34-8	1.46-8	3.65-2	5.83-3
4.	Floor drains sample tank	1.14-8	2.01-9	7.10-4	3.21-4
5.	Chemical waste distillate tank	5.20-10	7.92-11	1.16-3	9.96-5
6.	Condensate filter backwash receiving tank	7.15-7	1.38-7	4.55-3	3.05-3
7.	Condensate filter backwash setting tank	2.15-7	1.43-7	7.08-3	4.66-2
8.	Waste sludge settling tank	3.34-7	2.37-7	1.32-2	4.73-3
9.	Remainder of system	1.21-7	2.15-8	6.33-2	1.00-2

TABLE 15.7-17

FUEL HANDLING 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	3758 MWt	3758 MWt
	B.	Radial peaking factor	1.5	1.0
	C.	Fuel damage	108 rods	108 rods
	D.	Release of activity by nuclide	Section 15.7.4.5.1.1	Section 15.7.4.5.2.1
	E.	Iodine fractions		
		(1) Organic	0.0025	0.0025
		(2) Elemental	0.9975	0.9975
		(3) Particulate	0	0
II.	Data and assumptions used to estimate estimate activity released			
	A.	Fuel handling building leak rate	100%/2 hr	100%/2 hr
	B.	Adsorption and filtration efficiencies		
		(1) Organic iodine	95%	95%
		(2) Elemental iodine	95%	95%
		(3) Particulate iodine	95%	95%
	C.	All other pertinent data and assumptions	None	None

TABLE 15.7-17 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion data		
A. Boundary and LPZ distances (m)	863/4002	863/4002
B. χ/Q's for time intervals	6.7-4/8.2-5	6.7-4/8.2-5
IV. Dose Data		
A. Method of dose calculation	15.0.3.5	15.0.3.5
B. Dose conversion assumptions	15.0.3.5	15.0.3.5
C. Peak activity concentrations in fuel handling building	Table 15.7-18	Table 15.7-21
D. Doses	Table 15.7-20	Table 15.7-23

TABLE 15.7-18

FUEL HANDLING ACCIDENT
(DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN THE FUEL HANDLING BUILDING (CURIES)

Isotope	Activity
I-131	3.0+2
I-132	4.5+2
I-133	2.2+2
I-134	2.9-4
I-135	6.0+1
Kr-83m	2.1+1
Kr-85m	3.1+2
Kr-85	7.8+2
Kr-87	6.0-2
Kr-88	1.0+1
Kr-89	1.3-2
Xe-131m	2.2+2
Xe-133m	7.4+3
Xe-133	4.8+1
Xe-135m	9.6+2
Xe-135	1.4+4
Xe-137	2.1-2
Xe-138	2.1-2

TABLE 15.7-19

FUEL HANDLING ACCIDENT
(DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.5+1
I-132	2.2+1
I-133	1.2+1
I-134	1.5-5
I-135	3.0+0
Kr-83m	2.1+1
Kr-85m	3.1+2
Kr-85	7.8+2
Kr-87	6.0-2
Kr-88	1.0+2
Kr-89	1.3-2
Xe-131m	2.2+2
Xe-133m	7.4+3
Xe-133	4.8+4
Xe-135m	9.6+2
Xe-135	1.4+4
Xe-137	2.1-2
Xe-138	2.1-2

TABLE 15.7-20

FUEL HANDLING ACCIDENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area (863 Meters)	9.89-1	6.59+0
Low Population Zone (4,002 Meters)	1.21-1	8.06-1

TABLE 15.7-21

FUEL HANDLING ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN FUEL HANDLING BUILDING (CURIES)

Isotope	Activity
I-131	2.5+1
I-132	4.0+0
I-133	5.9+0
I-134	1.6-6
I-135	8.9-1
Kr-83m	4.7-1
Kr-85	5.2+2
Kr-85m	1.7+1
Kr-87	1.7-3
Kr-88	3.8+0
Kr-89	8.2-5
Xe-131m	5.8+1
Xe-133m	7.4+2
Xe-133	8.0+3
Xe-135m	1.7+0
Xe-135	2.9+3
Xe-137	6.8-5
Xe-138	1.8-4

TABLE 15.7-22

FUEL HANDLING ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.3+0
I-132	1.9-1
I-133	3.0-1
I-134	8.0-8
I-135	4.5-2
Kr-83m	4.7-1
Kr-85	5.2+2
Kr-85m	1.7+1
Kr-87	1.7-3
Kr-88	3.8+0
Kr-89	8.2-5
Xe-131m	5.8+1
Xe-133m	7.4+2
Xe-133	8.0+3
Xe-135m	1.7+0
Xe-135	2.9+3
Xe-137	6.8-5
Xe-138	1.8-4

TABLE 15.7-23

FUEL HANDLING ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area (863 Meters)	1.80-1	4.57-1
Low Population Zone (4,002 Meters)	2.19-2	5.59-2

TABLE 15.7-24

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT - PARAMETERS TO BE 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	3579 MWt	3579 MWt
B.	Burn-up		
C.	Fuel damage	124 rods	124 rods
D.	Release of activity to containment pool by nuclide, per failed rod	10% iodine 30% Kr-85	15.7.4.5.2.1
E.	Iodine fractions - organic	.0025	.0025
	inorganic	.9975	.9975
F.	Activity airborne in containment	Table 15.7-25	Table 15.7-28
II.	Data and assumptions used to estimate activity released		
A.	Primary containment leak rate	15.6.5.5.1.2	15.6.5.5.2.2
B.	Secondary containment leak rate	15.6.5.5.1.2	15.6.5.5.2.2
C.	Isolation valve closure times	NA	NA
D.	Filtration efficiencies	NA	NA
E.	Recirculation system parameters (flow rates vs. time, mixing factor, etc.)	NA	NA
F.	Containment spray parameters (flow rate, drop size, etc.)	NA	NA
G.	Containment volumes	NA	NA

TABLE 15.7-24 (Continued)

		Design Basis Assumptions	Realistic Basis Assumptions
	H. All other pertinent data and assumptions	15.7.6	15.7.6
	I. Adsorption efficiencies	15.7.6.4.1.3	15.7.6.4.2.3
	J. Activity released to environment	Table 15.7-26	Table 15.7-29
III. Dispersion Data			
	A. Boundary and LPZ distances (m)	863/4002	863/4002
	B. χ/Q's (for time intervals of 0.2-hr - SB/LPZ	6.7-4/8.2-5	6.7-4/8.2-5
IV. Dose Data			
	A. Method of dose calculation	15.0.3.5	15.0.3.5
	B. Dose conversion assumptions	15.0.3.5	15.0.3.5
	C. Peak (or f(t)) concentrations in containment	NA	NA
	D. Doses	Table 15.7-27	Table 15.7-30

TABLE 15.7-25

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN THE CONTAINMENT BUILDING (CURIES)

Isotope	Activity
I-131	3.4+2
I-132	5.2+2
I-133	2.6+2
I-134	3.3-1
I-135	6.9+1
Kr-83m	2.5+1
Kr-85m	3.6+2
Kr-85	9.0+2
Kr-87	7.0-2
Kr-88	1.2+2
Kr-89	1.5-2
Xe-131m	2.6+2
Xe-133m	8.5+3
Xe-133	5.5+1
Xe-135m	1.1+3
Xe-135	1.6+4
Xe-137	2.5-2
Xe-138	2.5-2

TABLE 15.7-26

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.5+2
I-132	2.2+2
I-133	1.1+2
I-134	1.5-4
I-135	2.9+1
Kr-83m	1.1+1
Kr-85	3.9+2
Kr-85m	1.6+2
Kr-87	3.1-2
Kr-88	5.0+1
Kr-89	6.4-3
Xe-131m	1.1+2
Xe-133m	3.7+3
Xe-133	2.5+4
Xe-135m	4.8+2
Xe-135	6.9+3
Xe-137	1.1-2
Xe-138	1.1-2

TABLE 15.7-27

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area (863 Meters)	5.97-1	6.54+1
Low Population Zone (4,002 Meters)	7.30-2	8.00+0

TABLE 15.7-28

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN CONTAINMENT BUILDING (CURIES)

Isotope	Activity
I-131	2.8+1
I-132	4.5+0
I-133	6.8+0
I-134	1.8-6
I-135	1.0+0
Kr-83m	5.4-1
Kr-85m	2.0+1
Kr-85	6.0+2
Kr-87	2.0-3
Kr-88	4.4+0
Kr-89	9.5-5
Xe-131m	6.6+1
Xe-133m	8.5+2
Xe-133	9.2+3
Xe-135m	2.0+0
Xe-135	3.3+3
Xe-137	7.9+5
Xe-138	1.2-4

TABLE 15.7-29

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (CURIES)

Isotope	Activity
I-131	1.2+1
I-132	2.0+0
I-133	2.9+0
I-134	8.0-7
I-135	4.4-1
Kr-83m	2.3-1
Kr-85m	8.5+0
Kr-85	2.6+2
Kr-87	8.5-4
Kr-88	2.0+0
Kr-89	4.1-5
Xe-131m	2.8+1
Xe-133m	3.7+2
Xe-133	3.9+3
Xe-135m	8.5-1
Xe-135	1.5+3
Xe-137	3.4-5
Xe-138	9.1-5

TABLE 15.7-30

FUEL HANDLING ACCIDENT INSIDE CONTAINMENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (Rem)	Inhalation Dose (Rem)
Exclusion Area (863 Meters)	9.13-2	4.54+0
Low Population Zone (4,002 Meters)	1.12-2	5.56-1

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

15.8.1 CAPABILITIES OF PRESENT BWR DESIGN TO ACCOMMODATE ATWS

The Nuclear Regulatory Commission is considering how to reduce the future risk to the public resulting from the postulated event of no reactor scram during an anticipated transient, i.e., an ATWS. The probability of an ATWS has been assessed to be significantly less than the probability of a design basis event. Because it is so extremely remote, the NRC will require some specific changes to the plant hardware rather than treat ATWS as a design basis event like other Chapter 15 events. These hardware changes in conjunction with existing plant systems will act to prevent or mitigate an ATWS. For example, the Recirculation Pump Trip will quickly reduce reactor power following an ATWS. If the Reactor Protection System (APS) should fail to cause a scram during a transient, the Alternate Rod Insertion System will provide the needed scram with electrical equipment that has a diverse design from the RPS. As a backup to these two highly reliable scram systems, the Standby Liquid Control System can be used to inject boron into the reactor and, thus, achieve subcriticality. These systems provide prevention and mitigation for an ATWS as described in the GE ATWS mitigation report, NEDO-24222, Vol. 2.

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PERRY NUCLEAR POWER PLANT
UNITS 1 & 2

FINAL
SAFETY
ANALYSIS
REPORT

THE CLEVELAND ELECTRIC ILLUMINATING CO.

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