PILGRIM NUCLEAR POWER STATION

BWR OPERATING STATE F

FSAR TABLE G.5-3

6 OF 6

REVISION [illegible] – JUNE [illegible]

REACTOR VESSEL HEAD ON
REACTOR NOT SHUTDOWN
PRESSURE [illegible]765 psig

SAFETY ACTIONS: SAFETY ACTIONS FOR PLANNED OPERATION (columns 2–16); SAFETY ACTIONS FOR ABNORMAL OPERATIONAL TRANSIENTS (columns 17–23); SAFETY ACTIONS FOR ACCIDENTS (columns 24–37); SAFETY ACTIONS FOR SPECIAL EVENT (columns 38–44, including AUTOMATIC TRIP OF THE RECIRCULATION PUMPS)

PLANT SYSTEMS (columns 45–110; columns 83, 84, 88, 89, 96, 97, 98, 99, 101, 107, 108, 109 marked Aux.)

SAFETY ACTION CODE NUMBER AND COLUMN NUMBER: 1 – 110

Column 1 = APPLICABILITY TO OPERATING STATE. Entries are given as "column: value"; "/" separates values stacked within one cell.

Line	Types of operation and events	Applicable (col. 1)	Safety actions (col.: value)	Plant systems (col.: value)
	PLANNED OPERATION			
1	REFUELING OUTAGE	NO		
2	ACHIEVING CRITICALITY	YES	3: 1-1; 5: 1-4; 7: 1-2/1-4; 8: 1-3/1-4; 9: 1-3; 10: 1-4; 11: 1-3/1-4; 12: 1-4; 13: 1-4; 15: 1-4; 16: 1-2/1-4	46: 5; 48: 7; 49: 8; 51: 11; 52: 13/12; 53: 7L; 54: 8L/10L/11L; 55: 9R; 56: 16; 57: P; 58: 5R/13L/12L; 74: 3; 77: 3; 79: 3; 82: 15L; 83: 15L; 85: 3L; 86: 3L; 103: 13; 104: 13; 106: 16L
3	HEATUP	YES	3: 1-1; 5: 1-2/1-4; 7: 1-2/1-4; 8: 1-3/1-4; 9: 1-3; 10: 1-4; 11: 1-3/1-4; 12: 1-4; 13: 1-4; 15: 1-4; 16: 1-2/1-4	46: 5; 48: 7; 49: 8; 50: 9; 51: 11; 52: 13/12; 53: 5L/7L; 54: 8L/9L/10L/11L; 55: 9R; 56: 16; 58: 13L/12L; 74: 3; 77: 3; 79: 3; 82: 15L; 83: 15L; 85: 3L; 86: 3L; 103: 13; 104: 13; 106: 16L
4	POWER OPERATION	YES	3: 1-1; 4: 1-2/1-4; 5: 1-2/1-4; 6: 1-2/1-4; 7: 1-2/1-4; 8: 1-3/1-4; 10: 1-4; 11: 1-3/1-4; 12: 1-4; 13: 1-4; 15: 1-4; 16: 1-2/1-4	45: 4; 46: 5; 47: 6; 48: 7; 49: 8; 51: 11; 52: 13/6/12; 53: 4L/5L/6L 7L; 54: 8L/10L/11L; 55: 9R; 56: 16; 58: 13L/12L; 74: 3/6; 77: 3; 79: 3; 82: 15L; 83: 15L; 85: 3L; 86: 3L; 103: 13; 104: 13; 106: 16L
5	ACHIEVING SHUTDOWN	YES	3: 1-1; 4: 1-2/1-4; 5: 1-2/1-4; 6: 1-2/1-4; 7: 1-2/1-4; 8: 1-3/1-4; 9: 1-3; 10: 1-4; 11: 1-3/1-4; 12: 1-4; 13: 1-4; 15: 1-4; 16: 1-2/1-4	45: 4; 46: 5; 47: 6; 48: 7; 49: 8; 50: 9; 51: 11; 52: 13/6/12; 53: 4L/5L/6L 7L 12L; 54: 8L/9L/10L/11L; 55: 9R; 56: 16; 58: 13L/12L; 74: 3/6; 77: 3; 79: 3; 82: 15L; 83: 15L; 85: 3L; 86: 3L; 103: 13; 104: 13; 106: 16L
6	COOLDOWN	NO		
7				
8				
9				
10				
11				
	ABNORMAL OPERATIONAL TRANSIENTS			
	NUCLEAR SYSTEM PRESSURE INCREASE			
12	GENERATOR TRIP	YES	17: 2-2/2-3; 18: 2-3	58: 17SF; 64: 18SF; 71: 17SF
13	TURBINE TRIP (WITH BYPASS)	YES	17: 2-2/2-3; 18: 2-3	58: 17SF; 64: 18SF; 71: 17SF
14	ISOLATION OF ALL MAIN STEAM LINES	YES	17: 2-2/2-3; 18: 2-3	58: 17SF; 64: 18SF; 71: 17SF
15	ISOLATION OF ONE MAIN STEAM LINE	YES	17: 2-2/2-3	58: 17SF; 71: 17SF; 74: 17SF
16	LOSS OF VACUUM (TURBINE TRIP WITHOUT BYPASS)	YES	17: 2-2; 18: 2-3	58: 17SF; 64: 18SF; 71: 17SF
	MODERATOR TEMPERATURE DECREASE			
17				
18	LOSS OF FEEDWATER HEATING	YES	17: 2-2	58: 17SF; 71: 17SF; 74: 17SF
19	SHUTDOWN COOLING (RHRS) MALFUNCTION (TEMPERATURE DECREASE)	NO		
20	INADVERTENT PUMP START (TEMPERATURE DECREASE)	YES		
	REACTIVITY INSERTION			
21	CONTROL ROD WITHDRAWAL ERROR	YES	17: 2-2	58: 17SF; 71: 17SF; 74: 17SF
22	FUEL ASSEMBLY INSERTION	NO		
23	CONTROL ROD REMOVAL	NO		
24	CONTROL CURTAIN REMOVAL	NO		
	LINE REFERENCE	1 – 110		
	LOSS OF COOLANT INVENTORY			
25	PRESSURE REGULATOR FAILURE	YES	17: 2-2; 18: 2-3; 20: 2-2/2-3	57: 20SF; 58: 17SF; 64: 18SF; 71: 17SF; 73: 20SF; 89: 73SF(S96) 96; 96: 73SF (S89)
26	INADVERTENT OPENING OF RELIEF VALVE	YES		
27	LOSS OF FEEDWATER FLOW	YES	17: 2-2/2-3; 18: 2-3; 19: 2-2; 20: 2-2	57: 20SF; 58: 17SF; 60: 19SF (S65/92); 64: 18SF; 65: 19SF (S60); 71: 17SF; 73: 20SF; 83: 60-65; 89: 96 / 60-65/92SF / 73SF(S96); 92: 19SF (S60); 96: 107 97 / 99 / 73SF(S89); 97: 107SF; 99: 97SF; 104: 83; 107: 60-65SF
28	TOTAL LOSS OF OFF-SITE AC POWER	YES	17: 2-2; 18: 2-3; 19: 2-2; 21: 2-2	58: 17SF; 60: 19SF (S65/92); 63: 19SF P; 64: 18SF/19SF; 65: 19SF (S60); 66: 19SF P; 67: 19SF P; 71: 17SF; 83: 19L/60-65/67; 88: 21SF 63SF 107SF / 97SF 99SF / 67SF; 89: 66SF 63SF / 60-65/92SF / 67SF / 88SF 99SF; 92: 19SF (S60); 97: 67SF/63SF/107SF; 99: 97SF; 104: 83; 107: 60-65SF/67SF/63SF
	CORE COOLANT FLOW DECREASE			
29	RECIRCULATION FLOW CONTROL FAILURE—DECREASING FLOW	YES		
30	TRIP OF ONE RECIRCULATION PUMP	YES		
31	TRIP OF TWO RECIRCULATION PUMPS	YES		
32	RECIRCULATION PUMP SEIZURE	YES		
	CORE COOLANT FLOW INCREASE			
33	RECIRCULATION FLOW CONTROL FAILURE—INCREASING FLOW	YES	17: 2-2	58: 17SF; 71: 17SF; 74: 17SF
34	STARTUP OF IDLE RECIRCULATION PUMP	YES	17: 2-2	58: 17SF; 71: 17SF; 74: 17SF
	CORE COOLANT TEMPERATURE INCREASE			
35	LOSS OF SHUT DOWN COOLING	NO		
	EXCESS OF COOLANT INVENTORY			
36	FEEDWATER CONTROLLER FAILURE-MAXIMUM DEMAND	YES	17: 2-2; 18: 2-3	58: 17SF; 64: 18SF; 71: 17SF
	ACCIDENTS			
37				
38	CONTROL ROD DROP ACCIDENT	YES	24: 3-2/3-3; 25: [redacted]; 26: [redacted]; 28: 3-2; 30: 3-4; 32: 3-2/3-3; 35: [redacted]; 36: 3-5	57: 25SF; 58: 24SF; 60: 28SF (S92/65); 63: 28SF P / 30SF P; 64: 28SF/35SF; 65: 28SF (S60); 66: 28SF P; 67: 28SF P; 71: 24SF; 73: [redacted]; 74: 24SF; 80: 24SF/25SF/26SF; 82: 26; 83: 28L 60/65/67/82; 88: 67[illegible] 63SF / 73SF(S89) / 100SF 107SF / 97SF 99SF; 89: 67SF / 73SF(S88) 63SF / 88SF [illegible]9SF 66SF / [illegible]; 92: 28SF (S60); 93: 32; 97: 67SF/63SF/107SF; 99: 97SF; 100: 36SF P; 104: 83; 107: 65-60SF/67SF/63SF; 109: 82SF
39	PIPE BREAK INSIDE PRIMARY CONTAINMENT	YES	24: 3-2; 26: 3-1; 27: 3-1; 28: 3-2; 30: 3-4; 31: 3-3; 36: 3-5	57: 26SF; 58: 24SF; 62: 28SF (S67) 28SF P; 63: 30SF / 28SF P; 64: 28SF; 65: 28SF (S66); 66: 28SF / 28SF (S92/65); 67: 28SF (S62) 28SF P; 69: 31; 71: 24SF; 73: 26SF/27SF; 82: 26; 83: 65/28L/62-67/82; 88: 62 SF 63SF / 73SF(S89) 67SF / 91SF 97SF 99SF / 100SF 107SF; 89: 65-66SF / 73SF(S88)66SF / 63SF 62SF 88SF / 92SF 67SF 99SF; 90: 27; 91: 27SF; 92: 28SF; 94: 30 SF / 28 SF / P; 97: 62SF/63SF/67SF/107SF; 99: 97SF; 100: 36SF P; 102: 27SF; 104: 83; 105: 27; 107: 62SF 65 / 67SF / 63SF; 109: 82SF
40	FUEL HANDLING ACCIDENT	YES	27: 3-1; 36: 3-5	79: 27SF; 88: 91SF/100SF; 89: 88SF; 90: 27; 91: 27SF; 100: 36SF P; 102: 27SF; 105: 27
41	PIPE BREAK OUTSIDE PRIMARY CONTAINMENT	YES	24: 3-2; 25: 3-1; 28: 3-2; 33: 3-2; 35: 3-3; 36: 3-5	57: 25SF; 58: 24SF; 62: 28SF (S67); 63: 28SF P; 64: 28SF / 35SF; 65: 28SF ~~(S66)~~; 66: 28SF (S65/92) / 28SF P; 67: 28SF (S62) / 28SFP; 71: 24SF; 73: 25SF P; 83: 28L/65/67; 88: 62 SF 63SF / 73SF(S89) / 100SF 107SF / 97SF 67SF 99SF; 89: 66SF 92SF / 63SF 66-65/92SF / 73SF(S88) / 99SF 67SF 88SF; 92: 28SF; 95: 33; 97: 62 SF/63SF/107SF/67SF; 99: 97SF; 100: 36SF P; 104: 83; 107: 62 SF/63SF/67SF/65
42				
43				
44	SPECIAL EVENT SHUT DOWN FROM OUTSIDE CONTROL ROOM	YES	38: 4-1; 39: 4-2	58: 38; 63: 39 P; 64: 39; 65: 39 P; 71: 38 P; 89: 65/63/96; 96: 63/107/97/99; 97: 63/107; 99: 97; 107: 65/63
45	SPECIAL EVENT—SHUT DOWN WITHOUT CONTROL RODS	YES	42: 5-1	68: 42 P; 89: 96; 96: 68; 109–110: 44
46				
47				
	LINE REFERENCE	1 – 110		

TABLE G.5-4

SAFETY ACTIONS FOR PLANNED OPERATION

Safety Action	Related Unacceptable Result	Reason Action Required
Radioactive Material Release Control	1-1	To maintain radioactive material release within 10CFR20
Core Coolant Flow Rate Control	1-2 1-4	To limit fuel failure and to remain within the envelope of conditions considered by station safety analysis
Core Power Level Control	1-2 1-4	To limit fuel failure and to remain within the envelope of conditions considered by the station safety analysis
Core Neutron Flux Distribution Control	1-2 1-4	To limit fuel failure and to remain within the envlope of conditions considered by station safety analysis
Reactor Vessel Water Level Control	1-2 1-4	To limit fuel failure and operate only in conditions considered by station safety analysis
Reactor Vessel Pressure Control	1-3 1-4	To limit nuclear process barrier stress and operate only in conditions considered by station safety analysis and so indicate
Nuclear System Temperature Control	1-3	To limit nuclear system process barrier stresses
Nuclear Systems Water Quality Control	1-4	To remain within the conditions considered by station safety analysis
Nuclear System Leakage Control	1-1 1-3 1-4	To limit nuclear system process barrier stresses, to operate only in conditions considered by station safety analysis, and to limit release of radioactive material

TABLE G.5-4 (Cont)

Safety Action	Related Unacceptable Result	Reason Action Required
Core Reactivity Control	1-4	To operate within the conditions considered by station safety analysis
Rod Worth Control	1-4	To operate only in condition considered by station safety analysis
Refueling Restrictions	1-4	To remain within the envelope of conditions by considered station safety analysis
Primary Containment Pressure and Temperature Control	1-4	To remain within the envelope of conditions by the station safety analysis
Stored Fuel Shielding, Cooling, and Reactivity Control	1-2 1-4	To limit fuel failure, to provide adequate shielding of station personnel, and to maintain stored fuel to within the envelope of condition considered by the station safety analysis

TABLE G.5-5

SAFETY ACTIONS FOR TRANSIENTS WITH UNACCEPTABLE RESULTS

Safety Action	Related Unacceptable Result	Reason Action Required
Scram	2-2 2-3	To prevent fuel damage and to limit nuclear system pressure rise resulting from excessive core power level
Pressure Relief	2-3	To prevent excessive nuclear system pressure rise
Core Cooling	2-2	To prevent fuel damage in the event that normal cooling is interrupted
Reactor Vessel Isolation	2-2	To prevent fuel damage by reducing the outflow of steam and water from the reactor vessel, thereby limiting the decrease in reactor vessel water level
Restore AC Power	2-2	To prevent fuel damage by restoring ac power to systems essential to other safety actions

TABLE G.5-6

SAFETY ACTIONS FOR ACCIDENTS

Safety Action	Related Unacceptable Result	Reason Action Required
Scram	3-2 3-3	To prevent excessive fuel cladding temperatures and to prevent excessive nuclear system pressures
Reactor Vessel Isolation	3-1	To limit radiological effects from exceeding the guideline values of 10CFR100
Establish Primary Containment	3-1	To limit radiological effects from exceeding the guideline values of 10CFR100
Establish Secondary Containment	3-1	To limit radiological effects from exceeding the guideline values of 10CFR100
Core Cooling	3-2	To prevent excessive fuel cladding temperatures
Containment Cooling	3-4	To prevent excessive pressure in the primary containment when containment is required
Stop Rod Ejection (passive)	3-3	To prevent excessive nuclear system pressure
Limit Reactivity Insertion Rate	3-2 3-3	To prevent excessive fuel cladding temperatures and nuclear system pressure
Restrict Loss of Reactor Coolant (passive)	3-2	To prevent excessive fuel cladding temperatures
Pressure Relief	3-3	To prevent excessive nuclear system pressure
Control Room Environmental Control	3-5	To limit radiation exposure of station personnel in the control room

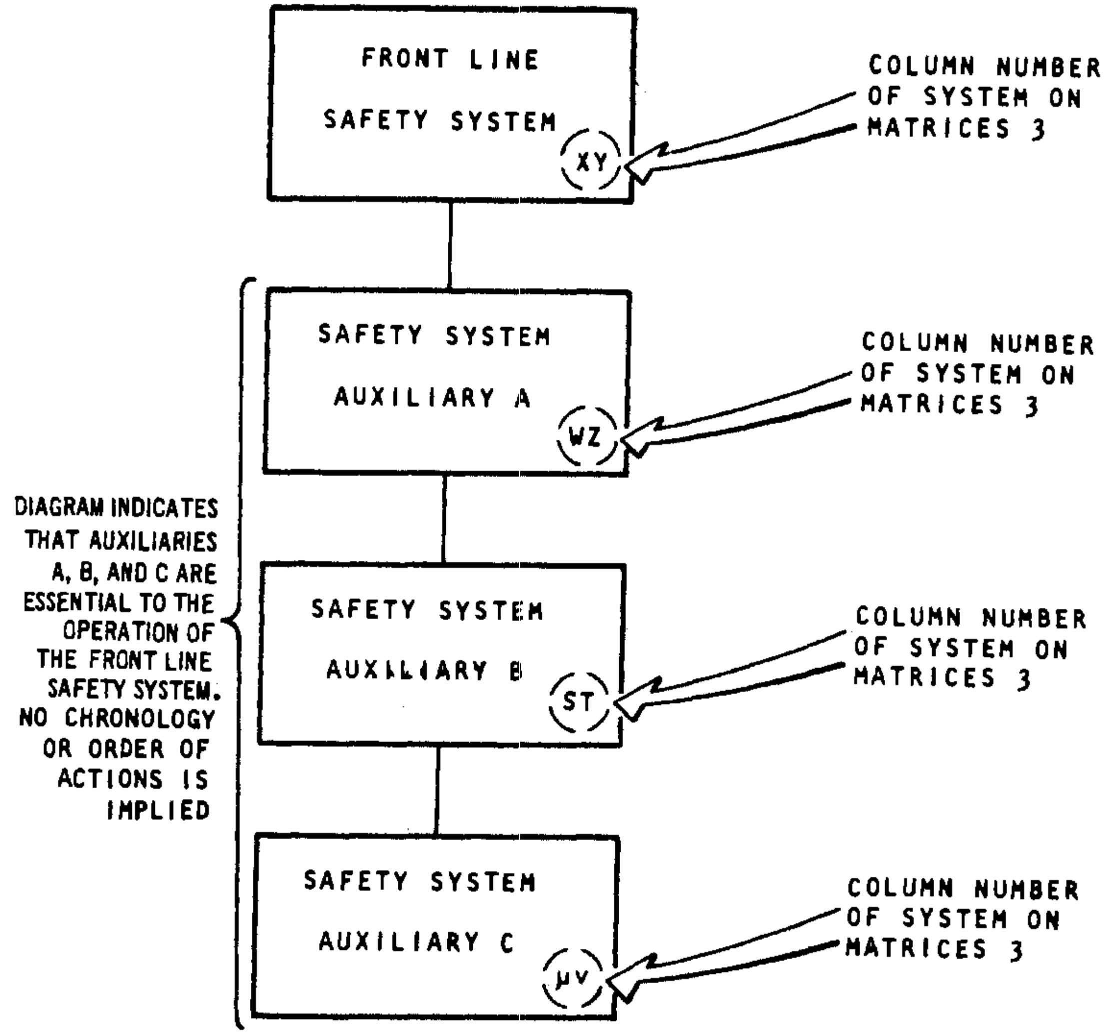
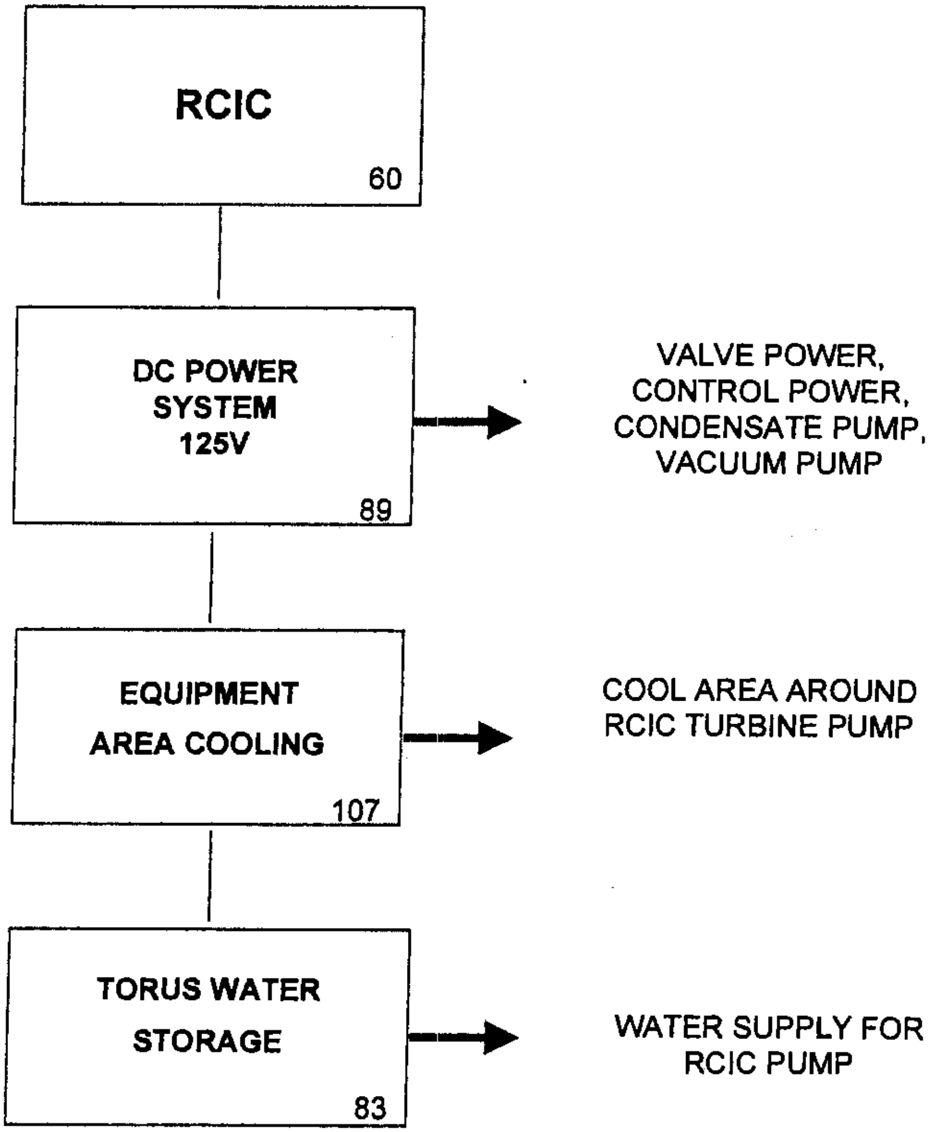


FIGURE G.5-1

SAFETY SYSTEM AUXILIARIES
EXAMPLE OF CONVENTION
USED ON BLOCK DIAGRAMS
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE G.5-2 SHEET 1 OF 1

RCIC AUXILIARIES



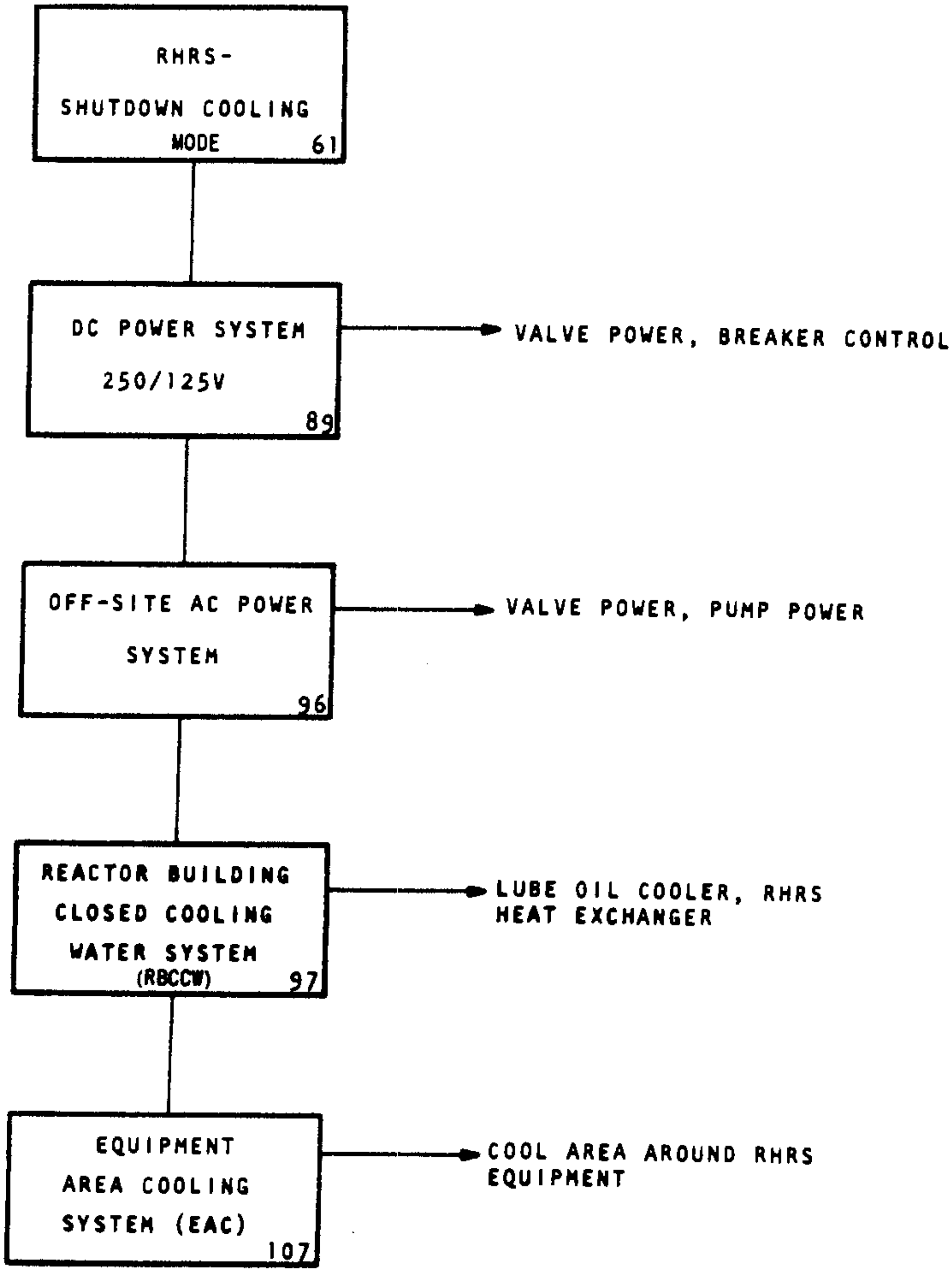


FIGURE G.5-3
RHRS-SHUTDOWN COOLING
MODE AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

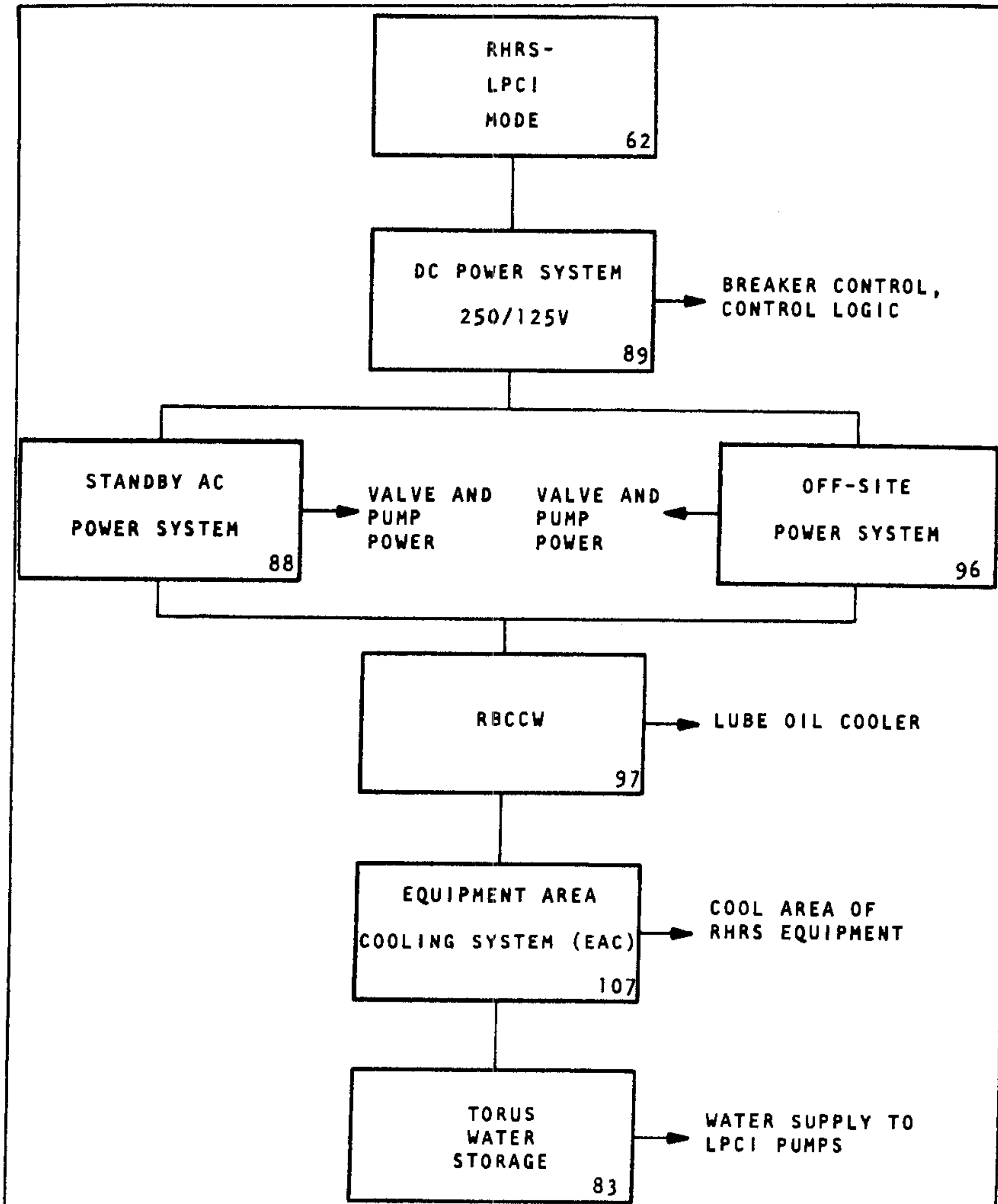


FIGURE G.5-4
RHRS-LPCI MODE AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

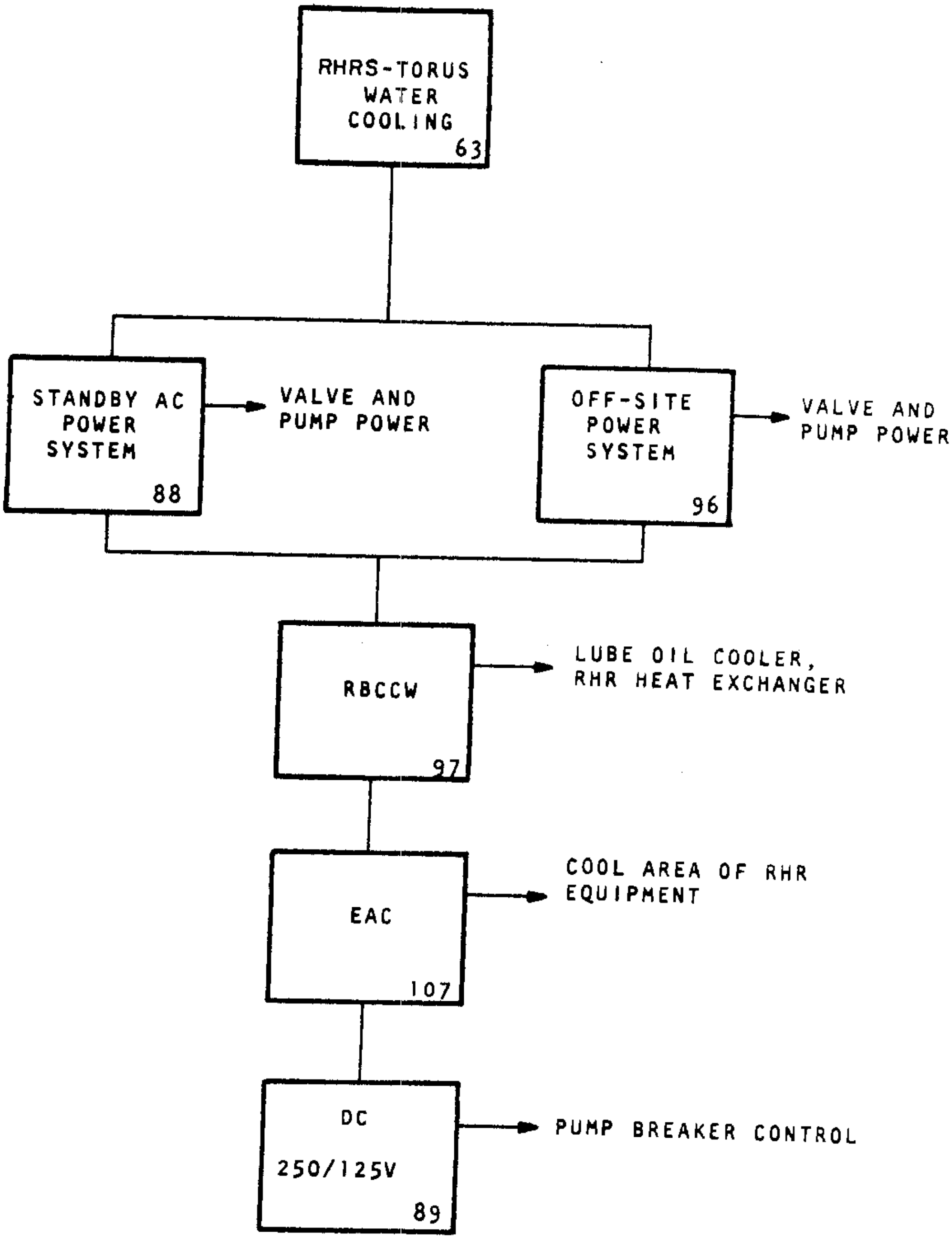


FIGURE G.5-5
RHRS-TORUS COOLING MODE AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

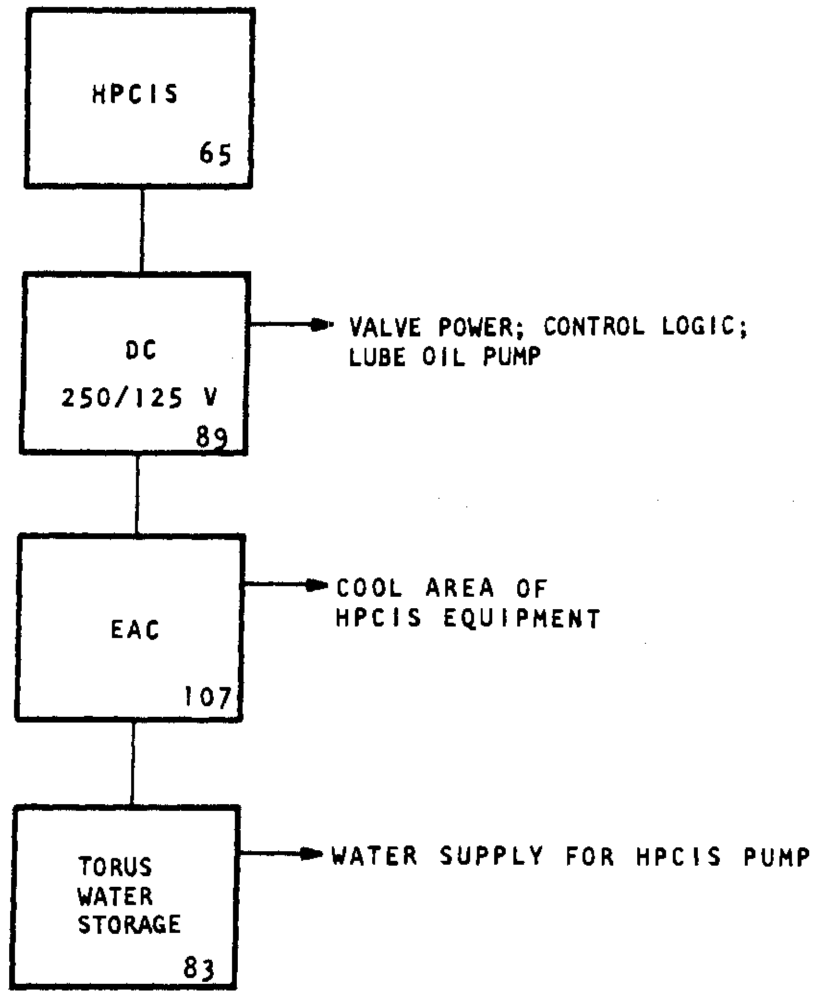


FIGURE G.5-6
HPCIS AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

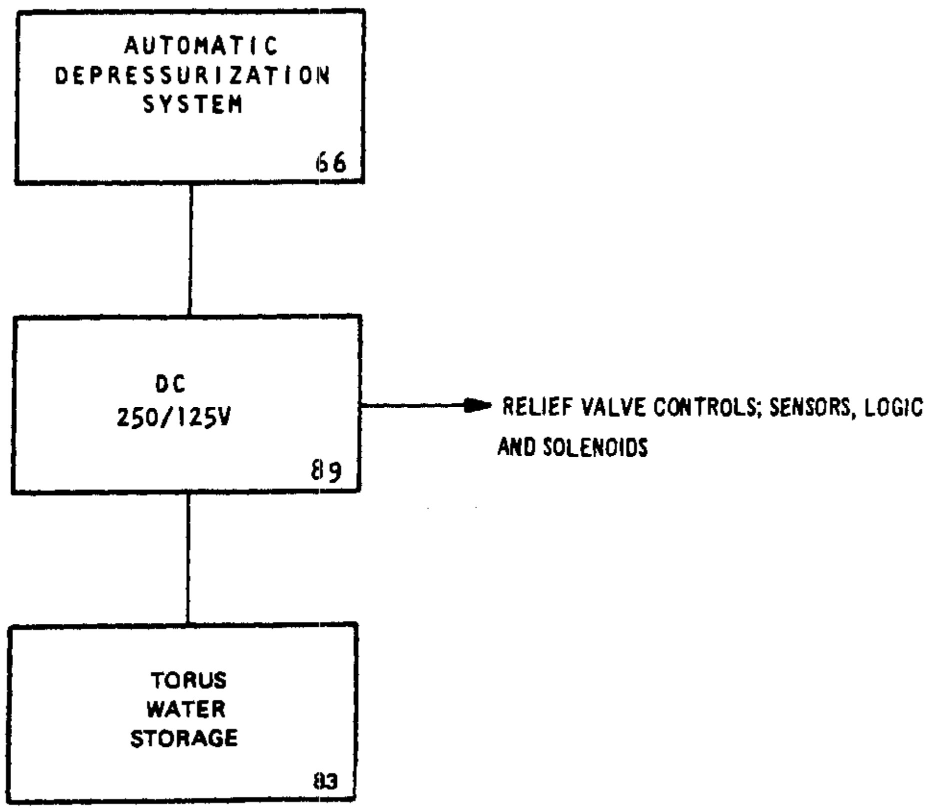


FIGURE G. 5-7
AUTOMATIC DEPRESSURIZATION
SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

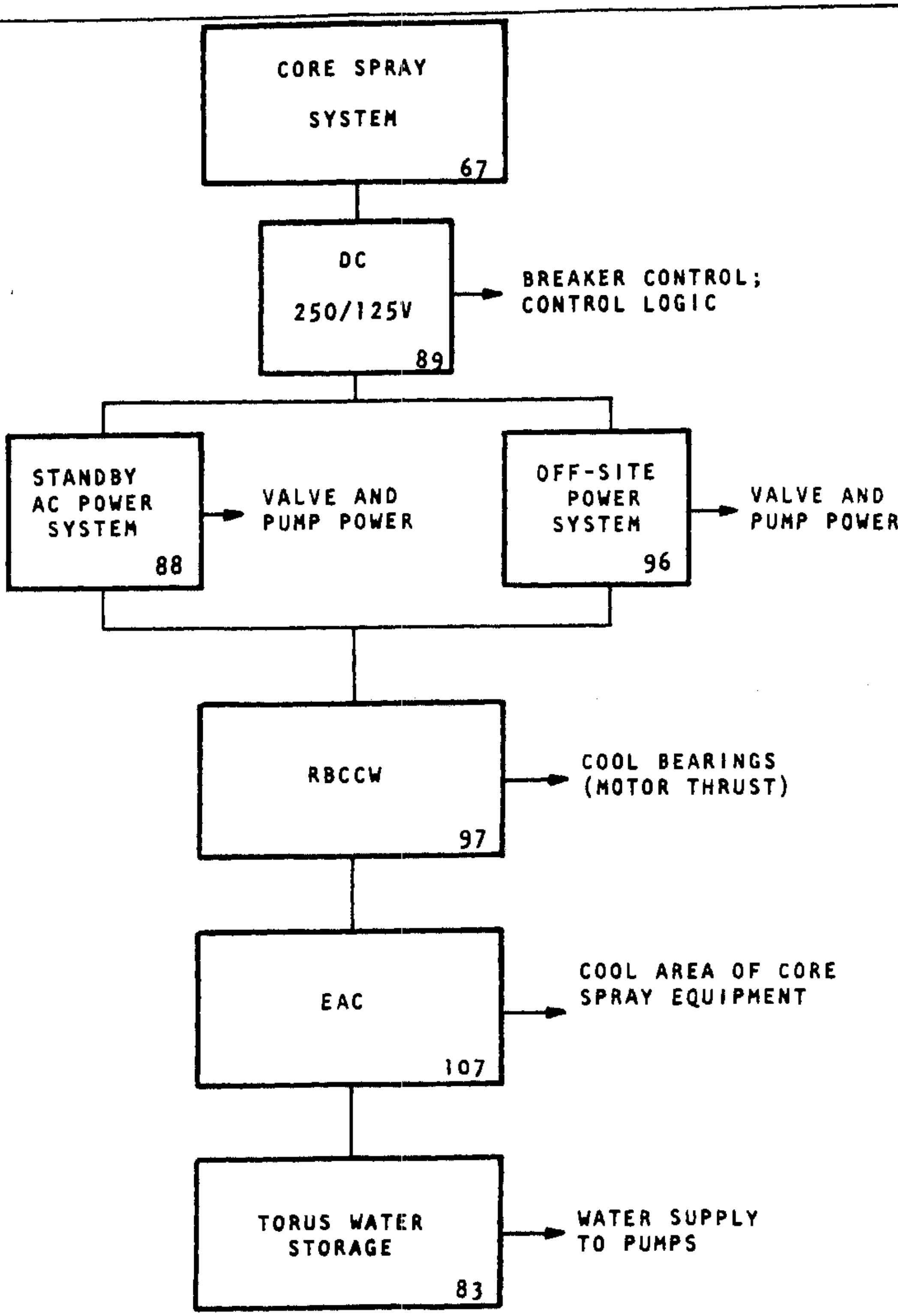


FIGURE G.5-8
CORE SPRAY SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

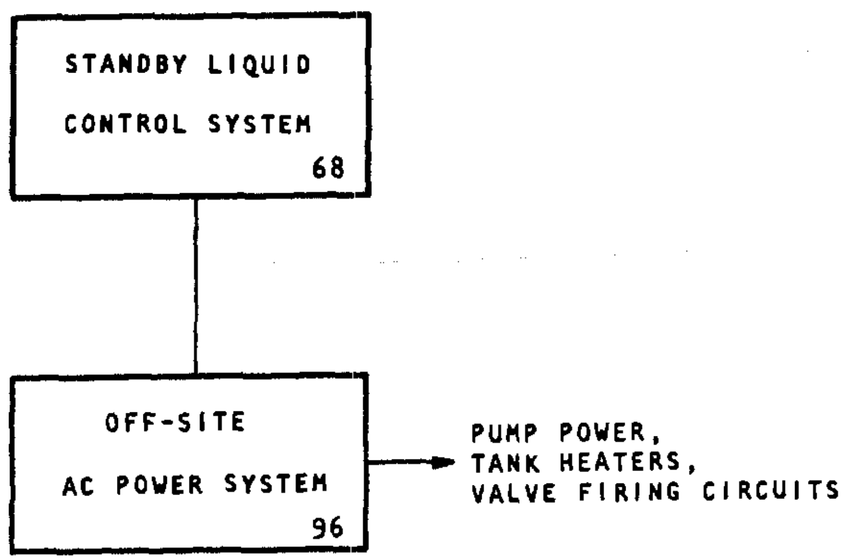


FIGURE G.5-9
STANDBY LIQUID CONTROL SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

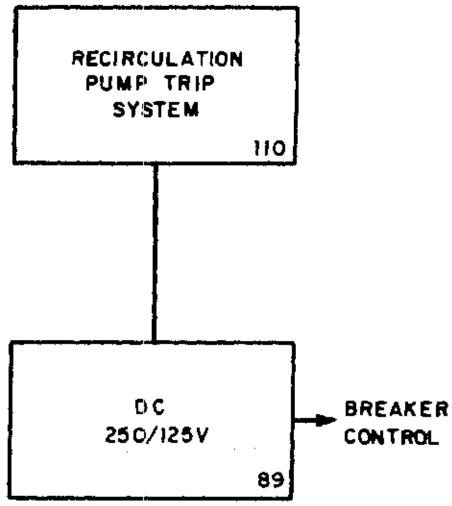


FIGURE G.5-9A
RECIRCULATION PUMP TRIP SYSTEM AUXILIARES

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

REVISION 2-JULY 1983

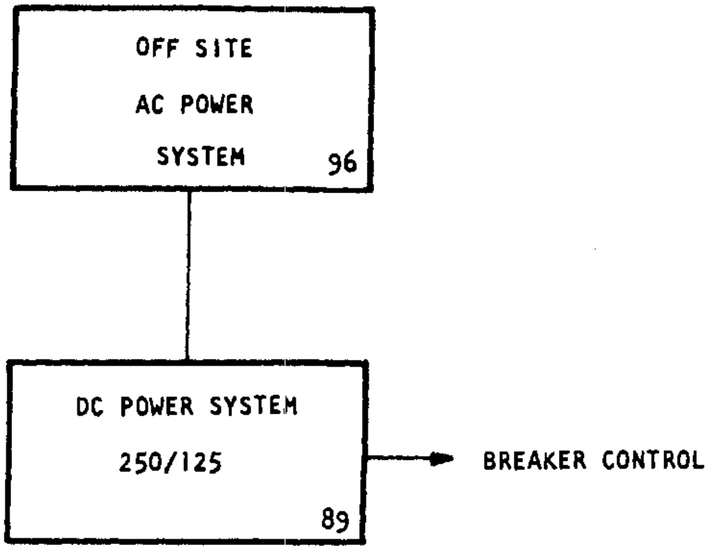


FIGURE G.5-10
OFFSITE AC POWER
SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

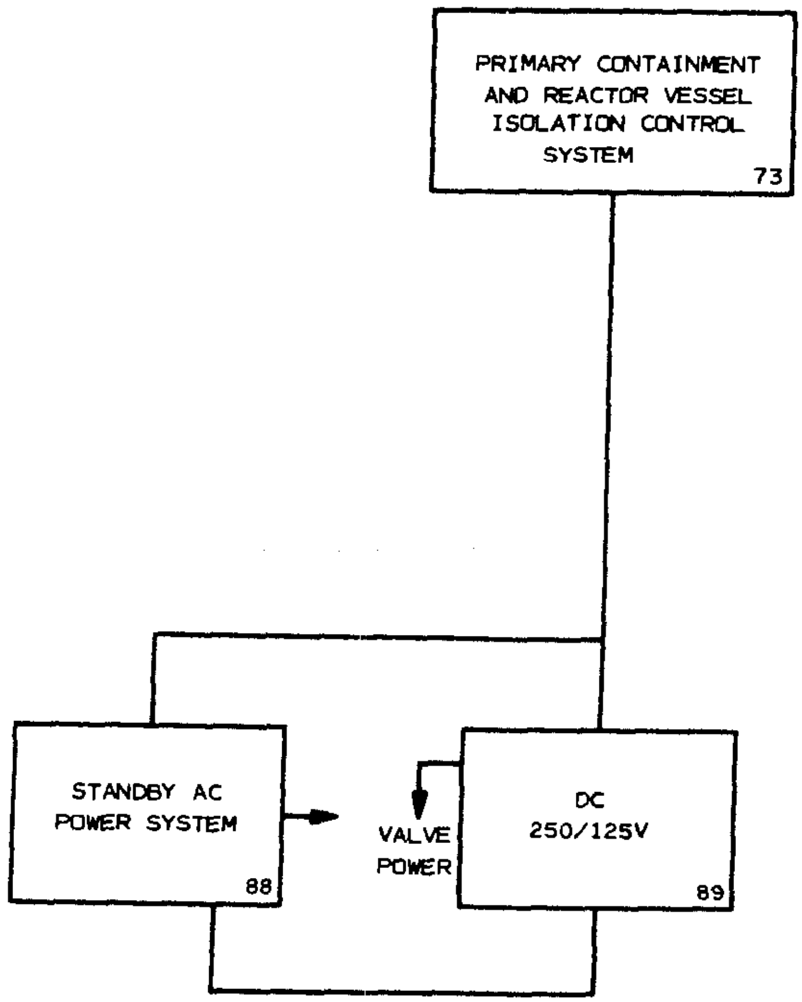


FIGURE G. 5-11
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

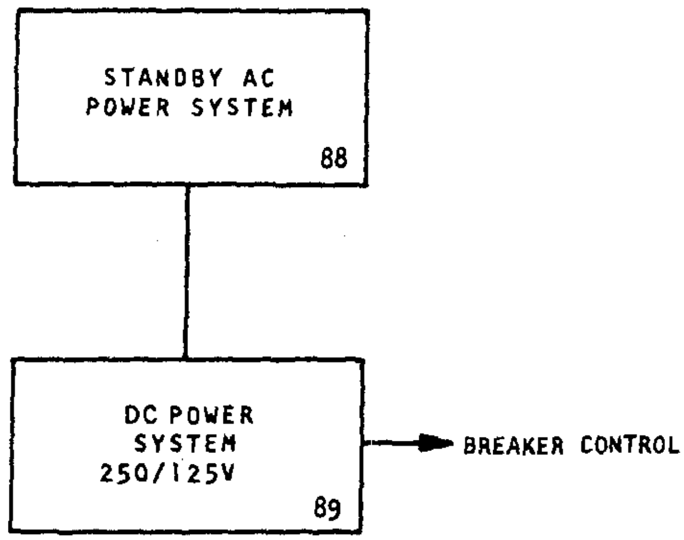


FIGURE 6.5-12
STANDBY AC POWER SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

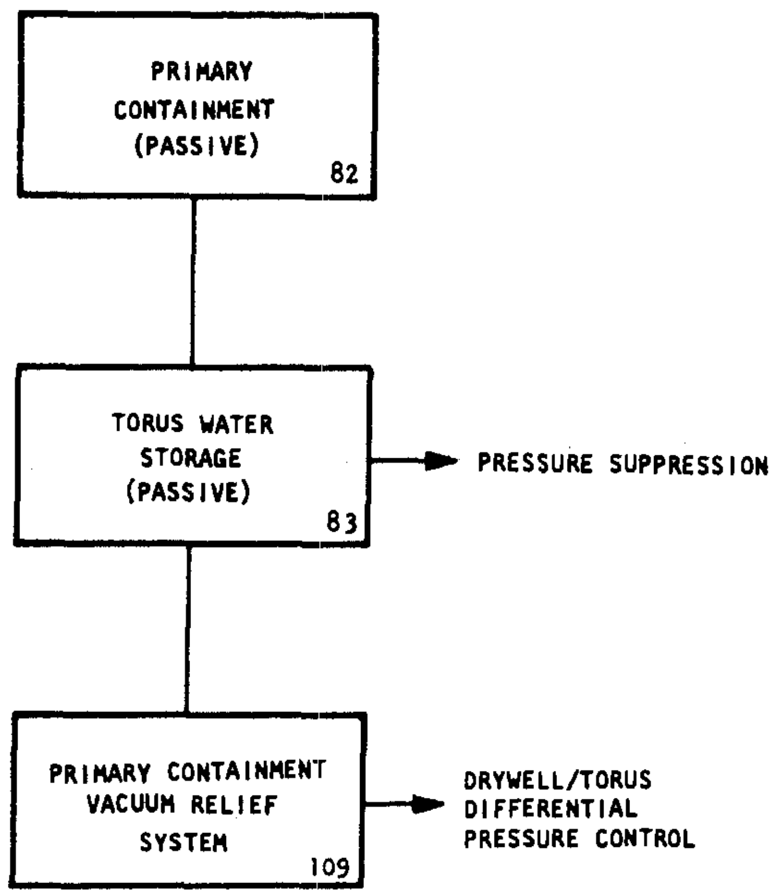


FIGURE G.5-13
PRIMARY CONTAINMENT AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

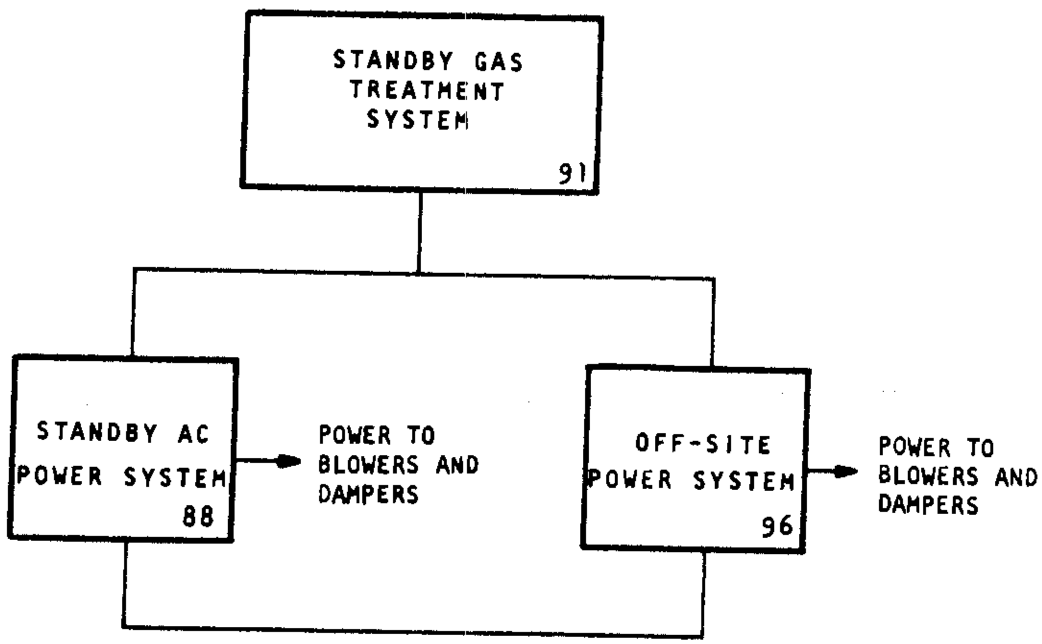


FIGURE G.5-14
STANDBY GAS TREATMENT SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

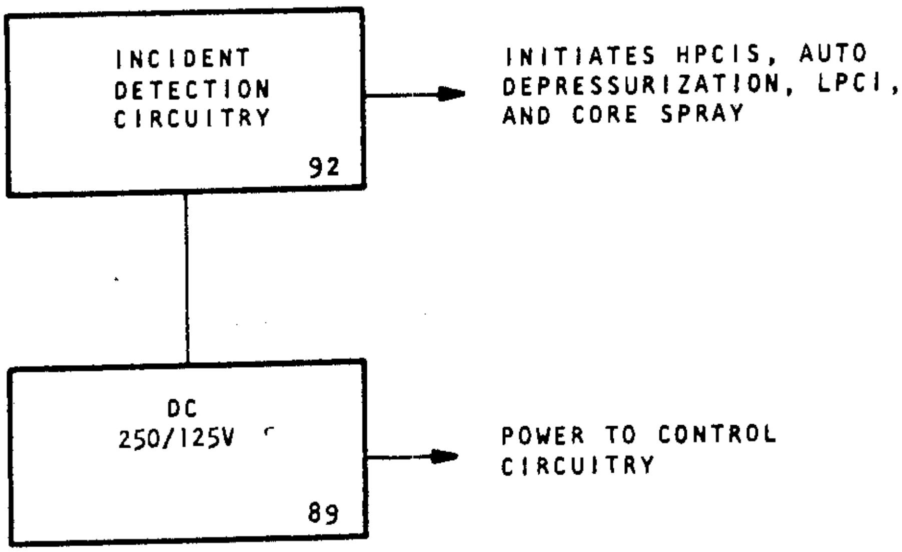


FIGURE G. 5-15
INCIDENT DETECTION CIRCUITRY AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

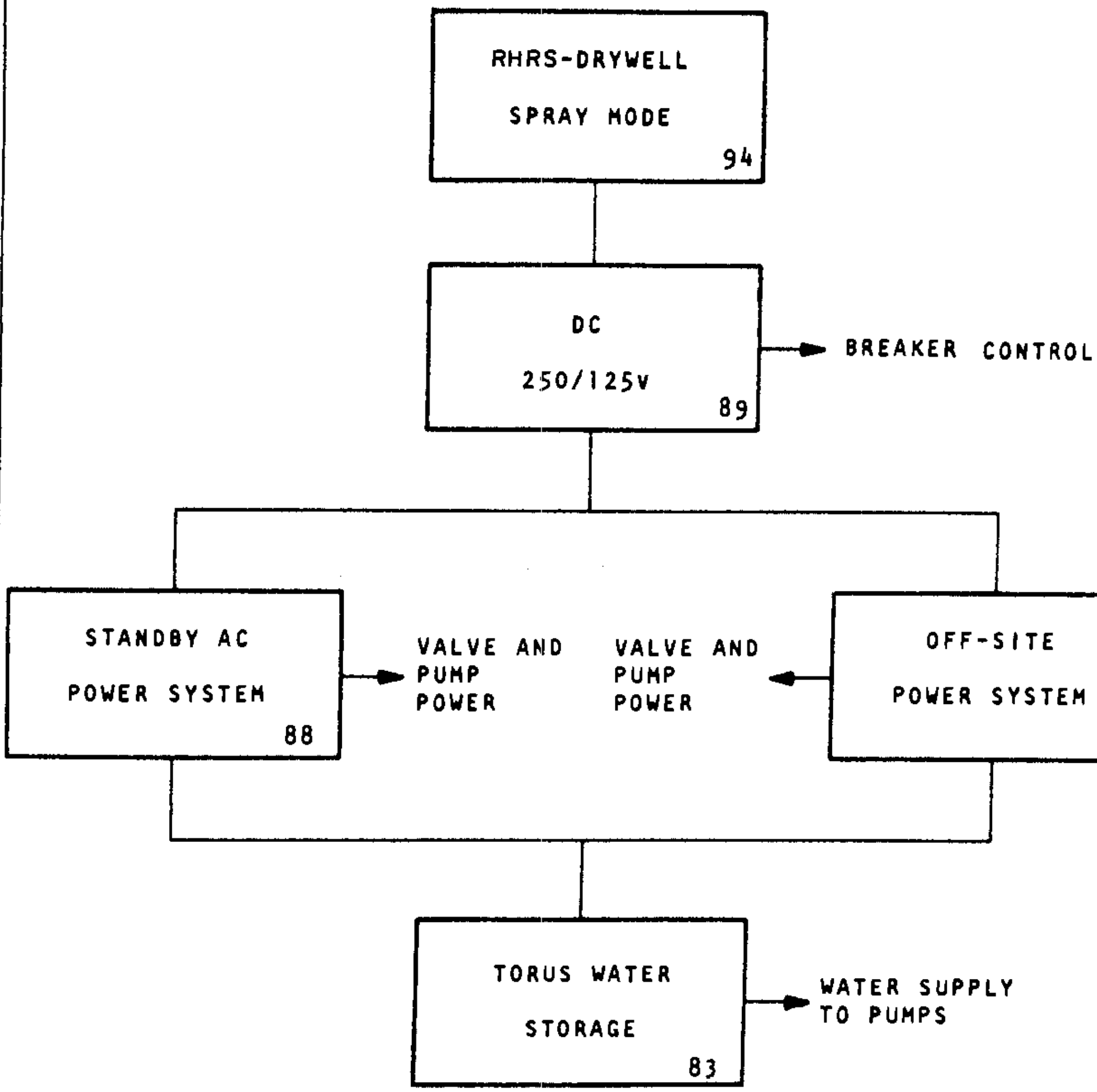


FIGURE G.5-16
RHRS-DRYWELL COOLING
MODE AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

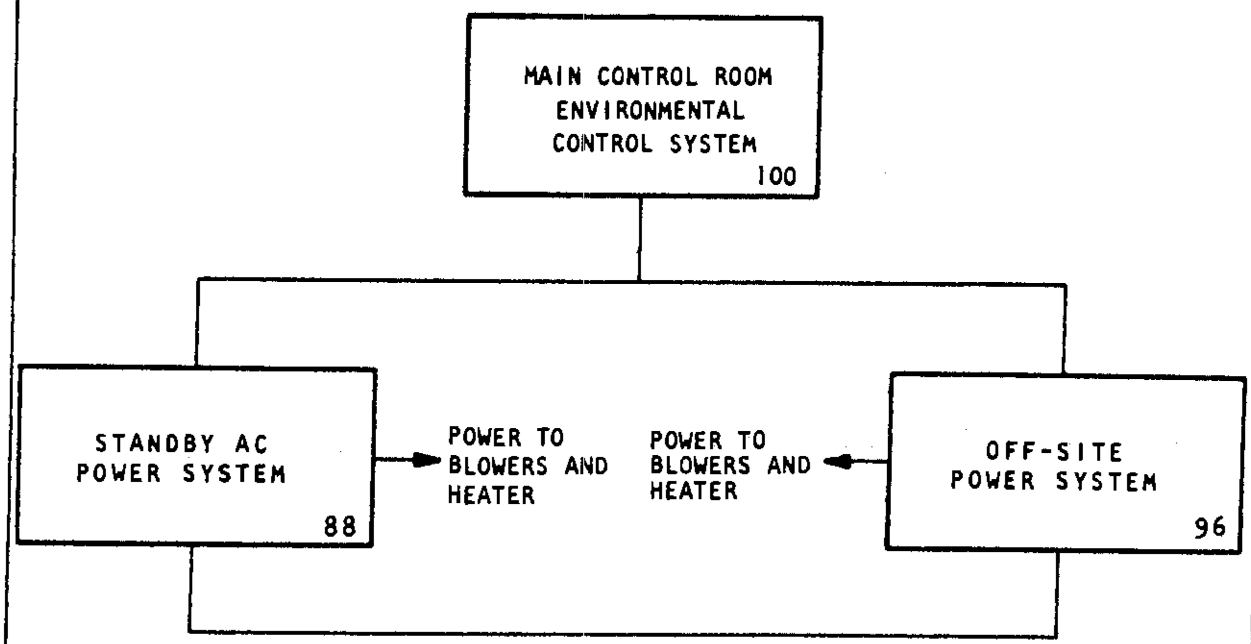


FIGURE G.5-17
MAIN CONTROL ROOM
ENVIRONMENTAL CONTROL
SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

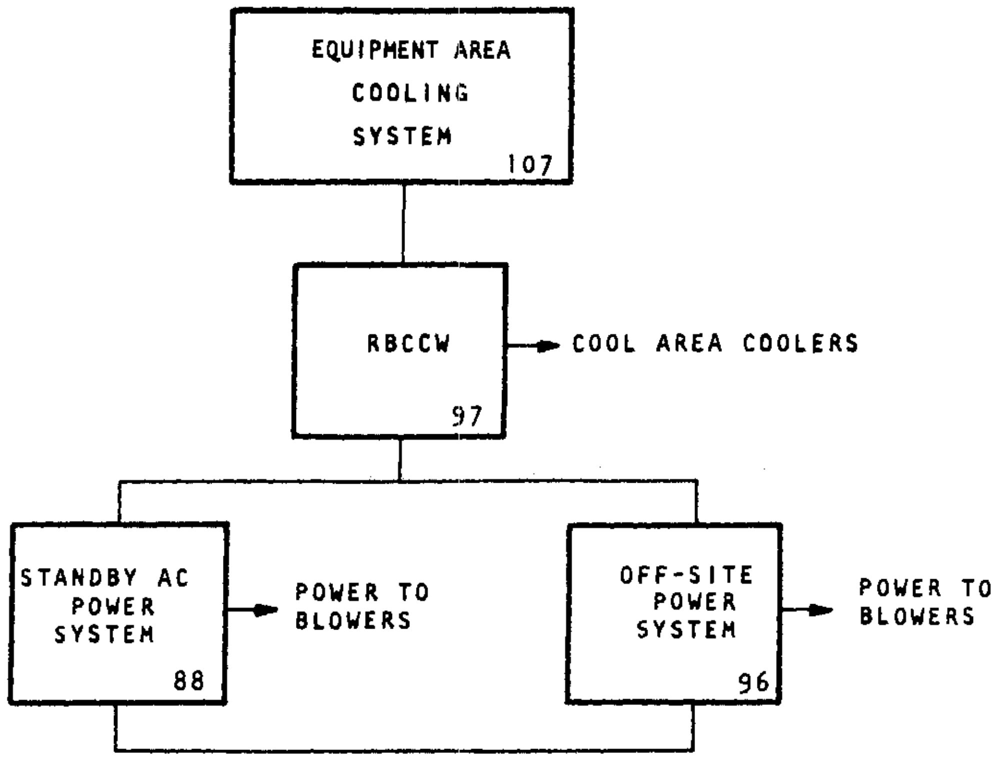


FIGURE G.5-18
EQUIPMENT AREA
COOLING SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

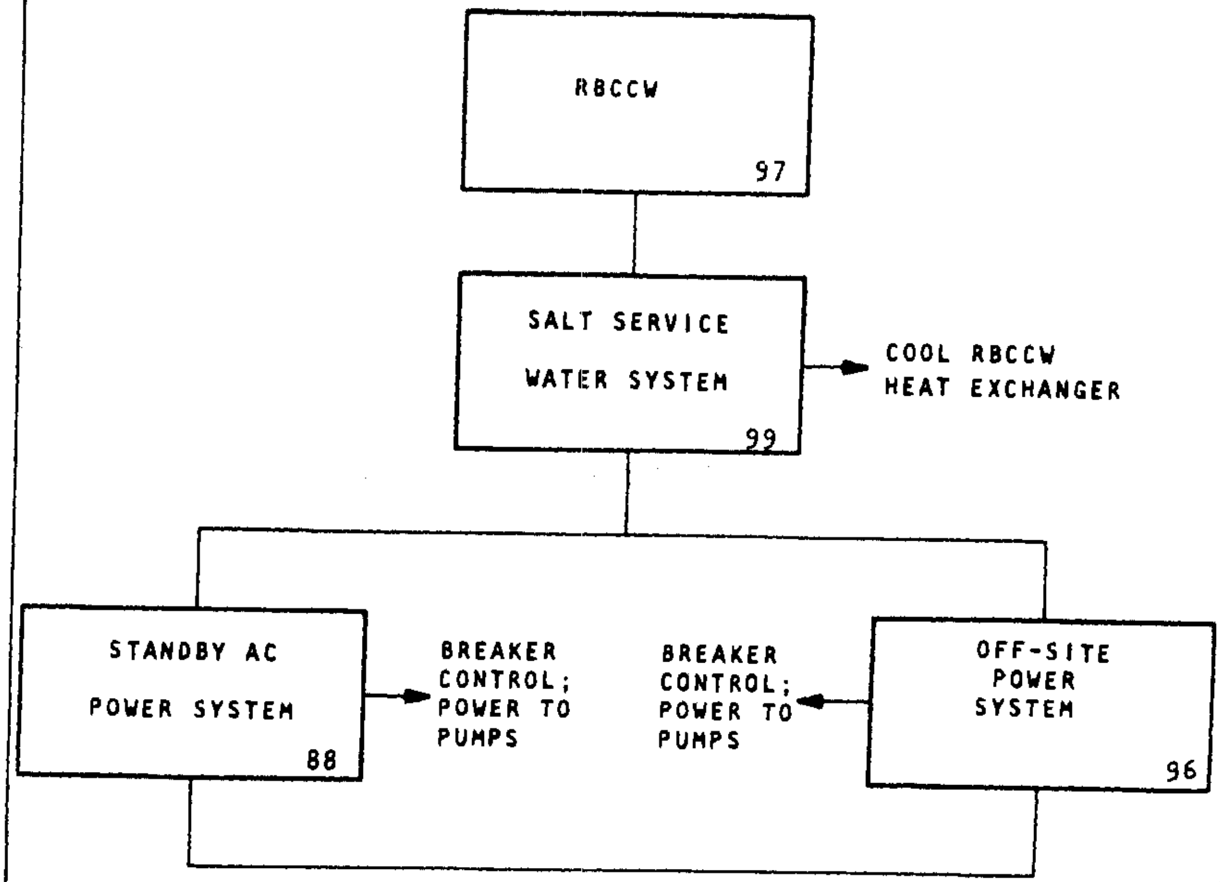


FIGURE G.5-19
REACTOR BUILDING CLOSED COOLING WATER SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

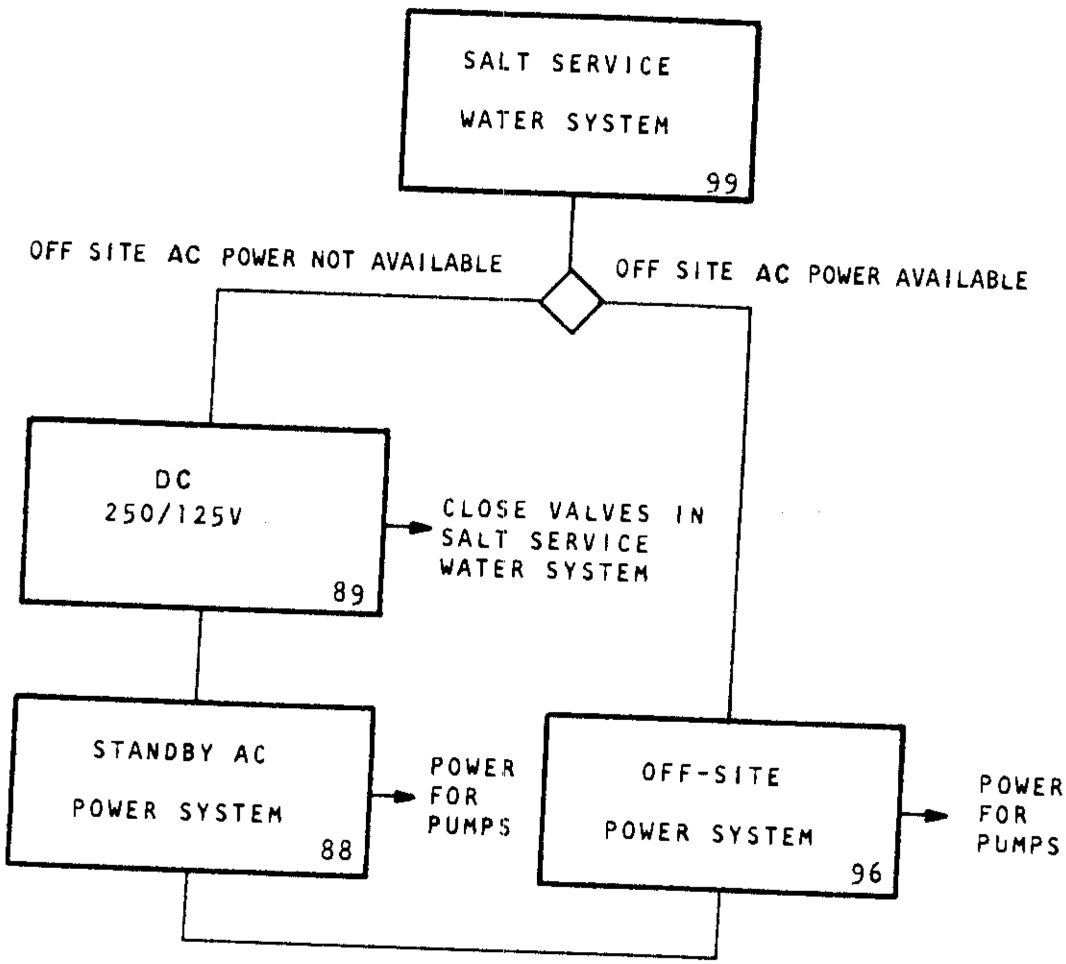


FIGURE G.5-20
SALT SERVICE WATER
SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

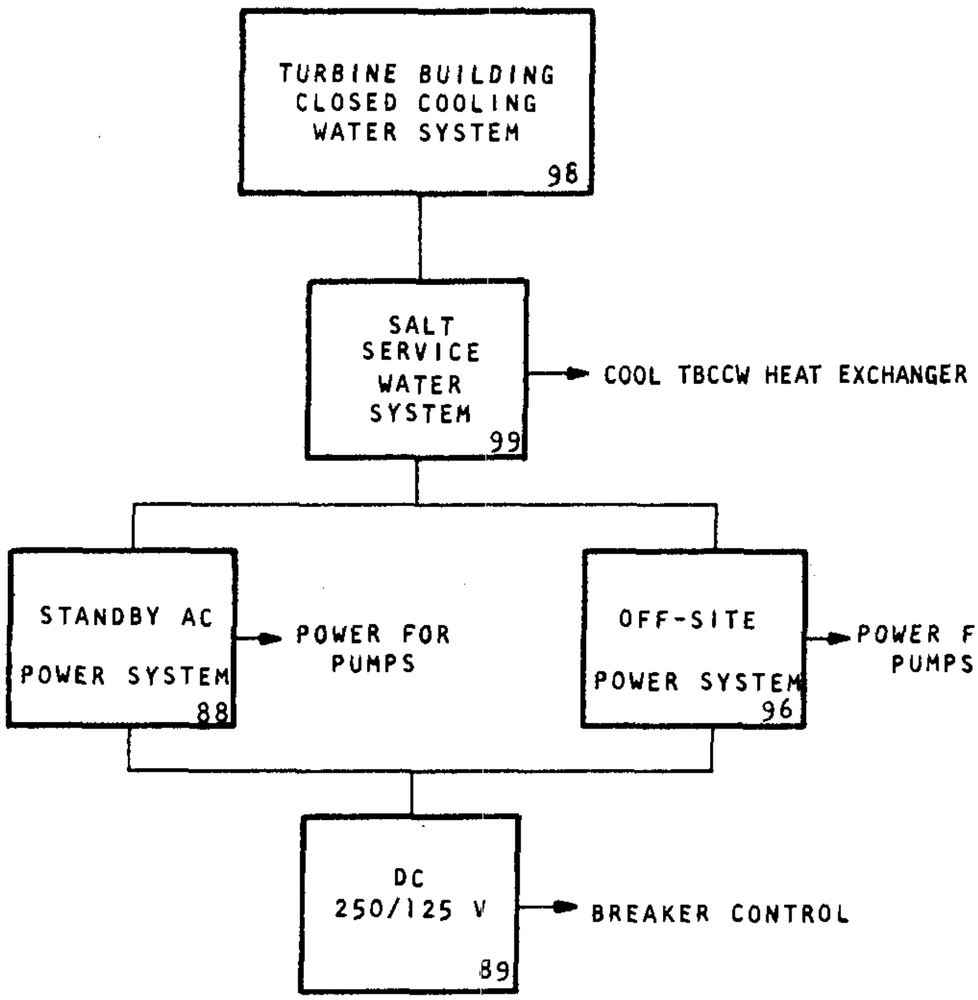


FIGURE G.5-21
TURBINE BUILDING
CLOSED COOLING WATER
SYSTEM AUXILIARIES
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

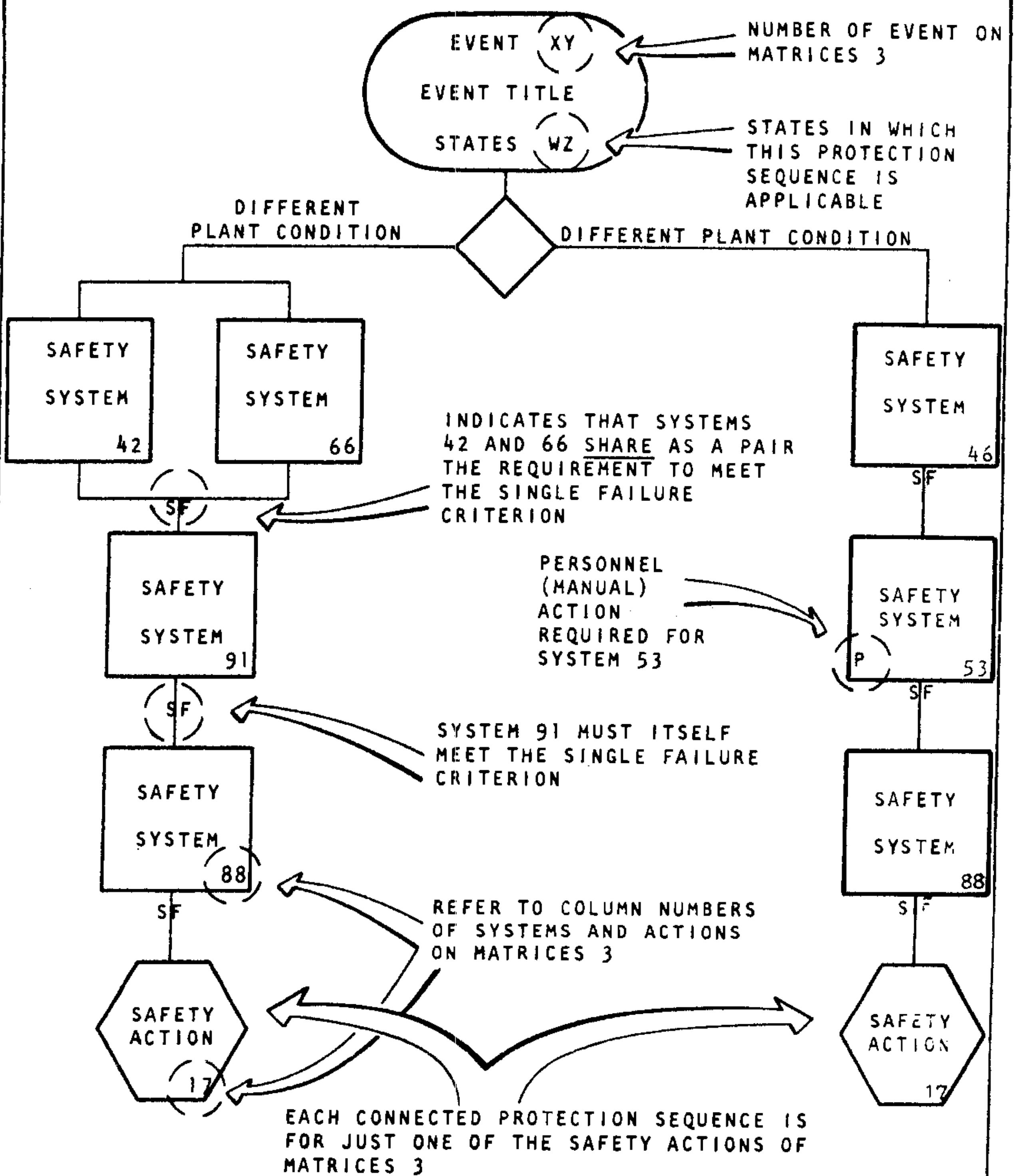


FIGURE G. 5-22
PROTECTION SEQUENCE
DIAGRAM EXAMPLE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

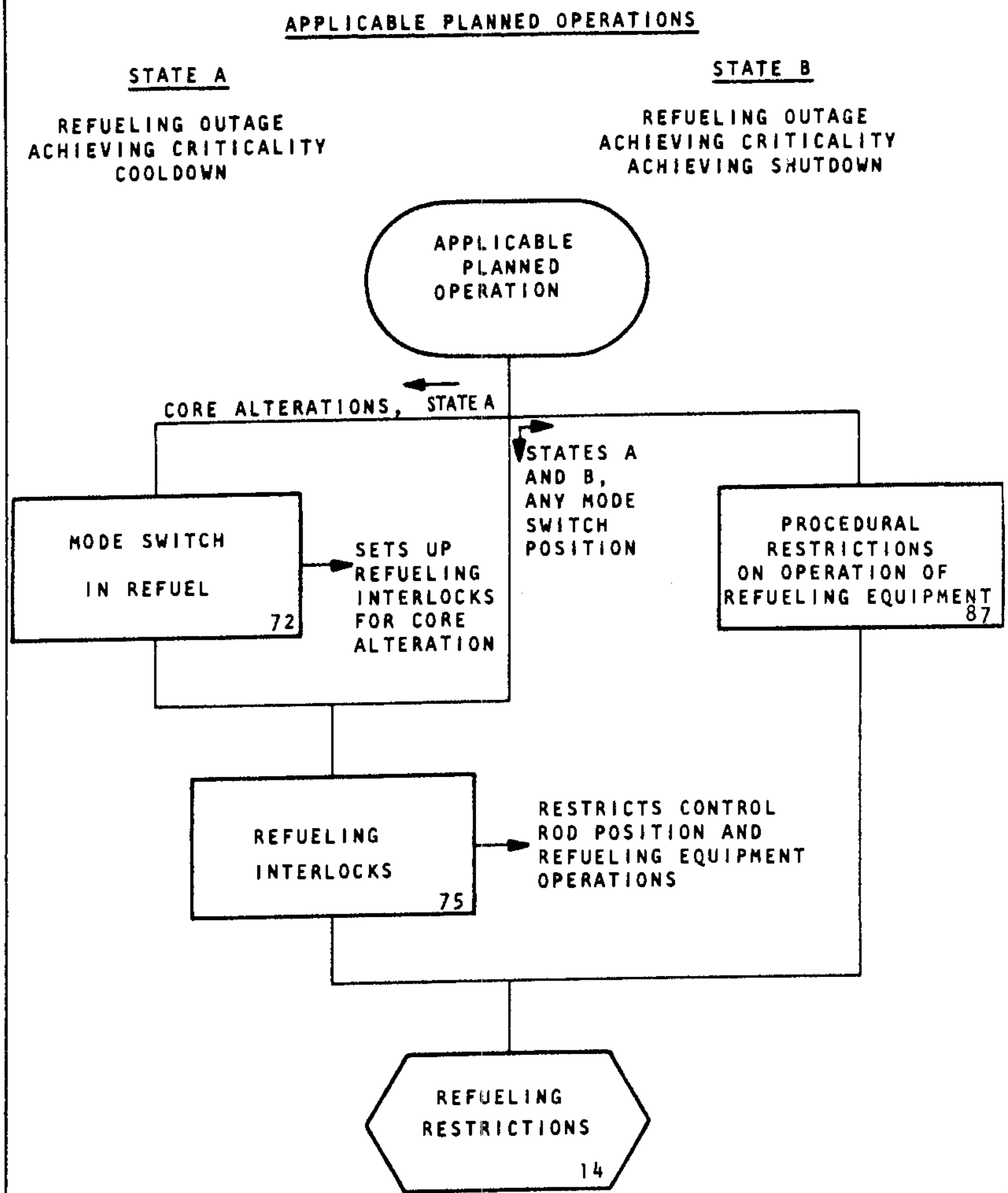


FIGURE G. 5-23
SEQUENCE FOR
REFUELING RESTRICTIONS
PILGRIM NUCLEAR POWER STATION
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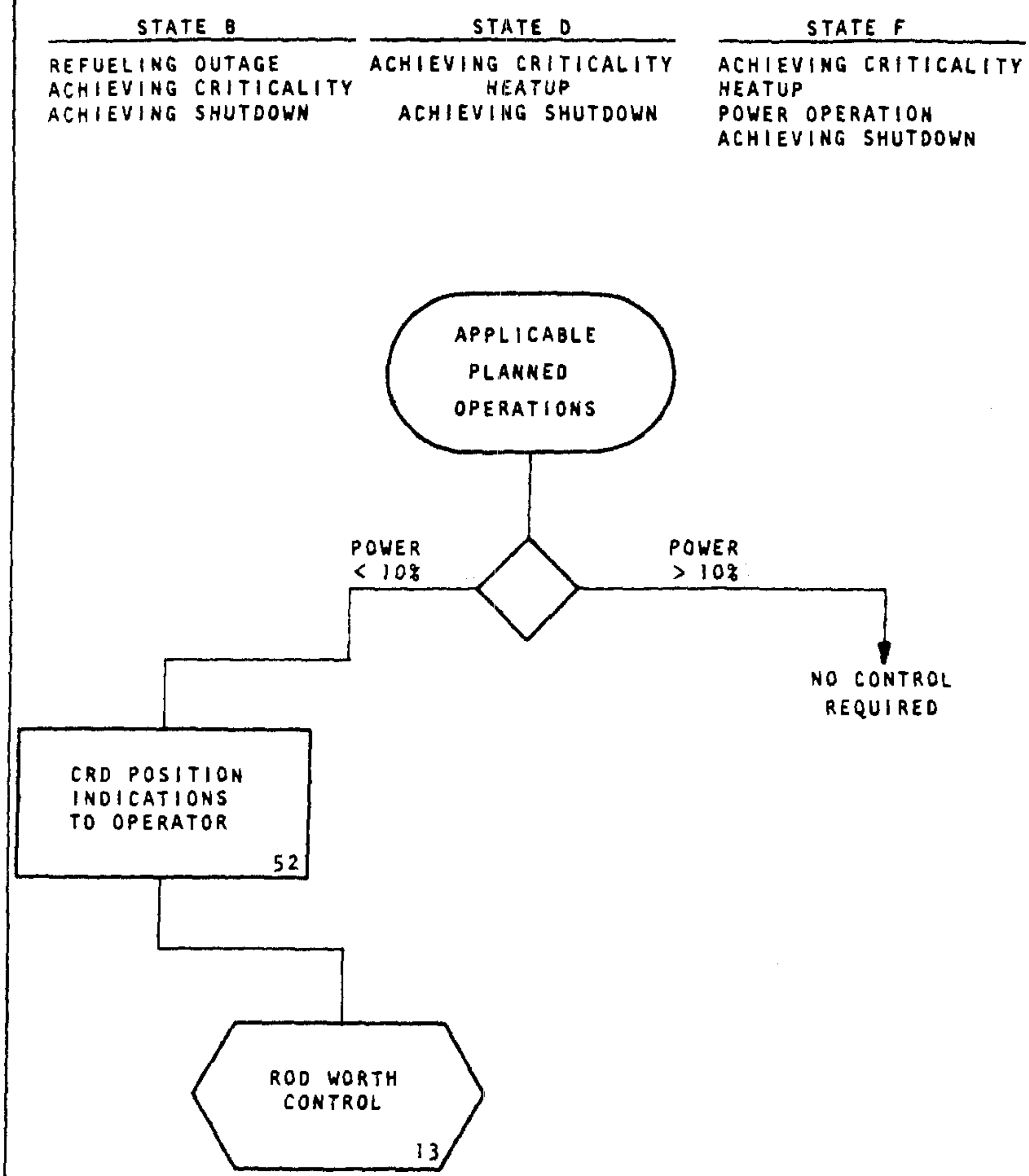

FIGURE G.5-24
SEQUENCE FOR ROD WORTH CONTROL
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OPERATIONAL NUCLEAR SAFETY ANALYSIS MATRIX
(EXAMPLE)

BWR Operating State	SAFETY ACTIONS											PLANT SYSTEMS							
				Power Level Control			Scram	Pressure Relief		Core Cooling		Core Spray System	Fuel	Reactor Protection System	Safety and Relief Valves	Control Rod System		LPCI	Standby a-c Power System
	1	2	3	4	5		17	18		22		45	46	48	52	53		76	88
Types of Operation and Events 1. 2.																			
3.																			
6. Power Operation				1-2 1-4									4L						
14. Turbine Trip							2-2 2-3	2-3						17 SF	18 SF	17 SF			
40. Loss of Coolant Accident							3-2			3-1 3-2		22 SF (S76)		17 SF		17 SF		22 SF (S45)	45-76SF
43. 44.																			

See explanatory notes on following pages.

FIGURE G.5-45 (SH. 1)
EXAMPLE OF MATRIX 3 INFORMATION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

FIGURE G.5-45

EXPLANATION OF SYMBOLS USED

SF (single failure)	System is required so that an essential safety action will meet the single failure criterion as stated by the nuclear safety operational criteria.
S (shared)	Used with other symbols to indicate that the system shares with another system the obligation to perform an action or to meet the single failure criterion. The column number of the system with which the obligation is shared is written inside parentheses with the S.
R (restricted)	One or more of the system's functions either is not acting or in not capable of acting in order to satisfy operational nuclear safety criteria while the reactor is in the designated operating state.
L (limit)	One or more of the key process parameters must be limited to satisfy nuclear safety operational criteria while the reactor is in the designated operating state.
P (personnel action)	Credit is taken for personnel action (manual control) of the corresponding system.
Blank	None of the system's functions are required or need to be restricted to satisfy nuclear safety operational criteria while the reactor is in the designated operating state.
Dark Frame Around Block	Represents the most significant or demanding condition from which an operational nuclear safety requirement for the system is derived.

INTERPRETATION OF SYMBOLS USED IN EXAMPLE MATRIX

	Event	System or Safety Action (Column Number)	Symbol (Column Entry	Meaning
6.	Power operation	Power level control (4)	1-2	Power level control is essential to avoiding unacceptable results 1-2.
6.	Power operation	Power level control (4)	1-4	Power level control is essential to avoiding unacceptable results 1-4.

FIGURE G.5-45 (cont)

	Event	System or Safety Action (Column Number)	Symbol (Column Entry	Meaning
6.	Power operation	Fuel (46)	4L	There is a limit on power level based on the fuel design that must be observed to satisfy the need for safety action 4 (power level control).
14.	Turbine	Scram (17)	2-2	Scram is essential to avoiding unacceptable results 2-2 and 2-3.
14.	Turbine trip	Pressure Relief (18)	2-3	Pressure relief is essential to avoiding unacceptable result 2-3.
14.	Turbine trip	Reactor protection	17	It is essential that the reactor protection system be capable of operating to achieve safety action 17 (scram).
14.	Turbine trip	Reactor protection system (48)	SF	It is essential that the reactor protection system be in a condition to meet the single failure criterion.
14.	Turbine	Reactor		The dark frame around the matrix block indicates that this block represents the most significant or demanding condition from which at least one operational nuclear safety requirement for the system is derived. This matrix block would be referenced (using the block coordinates X 14-48) in the operational nuclear safety requirements portion of the FSAR subsection describing the system. The block coordinates correspond to the BWR operating state letter, the row number, and the column number in that order.

FIGURE G.5-45 (cont)

	Event	System or Safety Action (Column Number)	Symbol (Column Entry)	Meaning
14.	Turbine trip	Safety and relief valves (52)	18SF	It is essential that the valves be capable of operating to achieve safety action 18 (pressure relief), and be in a condition to meet the single failure criterion.
14.	Turbine trip	Control rod drive system (53)	17SF	It is essential that the control rod drive system be capable of operating to achieve safety action 17 (scram), and be in a condition to meet the single failure criterion.
40.	Loss of coolant accident	Scram (17)	3-2	Scram is essential to avoiding unacceptable result 3-2.
40.	Loss of coolant accident	Core cooling (22)	3-1 3-2	Core cooling is essential to avoiding unacceptable results 3-1 and 3-2.
40.	Loss of coolant accident	Core spray system (45)	22	It is essential that the core spray system be capable of operating to achieve safety action 22 (core cooling).
40.	Loss of coolant accident	Core spray system (45)	SF(S76)	It is essential that the core spray system be in a condition to meet the single failure criterion. The (S76) indicates that the core spray system shares with system 76 (LPCI) the obligation to satisfy the single failure criterion.

FIGURE G.5-45 (cont)

Event	System or Safety Action (Column Number)	Symbol (Column Entry)	Meaning
			least one operational nuclear safety requirement for the system is derived. This matrix block would be referenced (using the block coordinates X 40-76) in the operational nuclear safety requirements portion of the FSAR section describing the system.
40. Loss of coolant accident	Standby ac power system (88)	45-76SF	It is essential that the standby ac power system be single failure proof relative to the system pair consisting of the core spray system (45) and LPCI (76).

G.6 CONCLUSION

It is concluded that the proposed operational nuclear safety criteria for the intitial station operating license are satisfied when the station is operated in accordance with the proposed nuclear safety requirements for initial plant operation determined by the method presented in this Appendix.

APPENDIX H

TORNADO CRITERIA FOR NUCLEAR POWER PLANTS

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TORNADO CRITERIA FOR NUCLEAR POWER PLANTS

FOREWORD

This report establishes a set of tornado criteria for nuclear power plants. The criteria are identified, developed, justified, and the application of each explained. This report[1] summarizes the tornado studies performed by Bechtel Corporation as a result of a commitment made during the review process for a Construction Permit for Pilgrim Nuclear Power Station.

When the criteria developed in this report are considered simultaneously, the resulting design parameters prescribe a tornado-resistant structure. The application of the criteria results in a building with energy absorption characteristics much higher than those of buildings that have withstood tornadoes.

APPENDIX H

TORNADO CRITERIA FOR NUCLEAR POWER PLANTS

H.1 INTRODUCTION

In the General Design Criteria for Nuclear Power Plants, the United States Atomic Energy Commission (AEC) relates tornadoes to nuclear power plant design:

> "Criterion 2 - Performance Standards (Category A)
>
> Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences, shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and surrounding area, and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitablilty as a basis for design."(2)

In 1969, the AEC staff expressed concern about four facets of tornado design: wind loading, pressure differential, protection of the spent-fuel storage pool, and missiles.

This report intends to identify the tornado design criteria, present background information, and show the methods by which the criteria are applied.

H.2 CHARACTERISTICS OF TORNADOES

Tornadoes are the most violent of all storms. Although they are local, of short duration, and have low probability of occurrence, their damage potential to nuclear power plants warrants intensive evaluation. Adequate protection must be provided for those components whose failure could cause a significant release of radioactivity or which are required for safe shutdown of the plant and removal of decay heat.

H.2.1 Sources of Information

In order to develop the tornado design criteria, five sources of information were considered: eyewitness accounts from persons making observations during tornadoes, analyses of photographs taken during tornadoes, analyses of tornado-caused damage or lack of tornado-caused damage, analyses of the physical aspects of tornado phenomena, and recommendations of persons familiar with the effects of tornadoes.

There are numerous recorded eyewitness accounts of tornadoes; however, the credibility of many accounts is questionable since the observers were often close to the tornado and under great emotional stress. Although individual reports are sometimes questioned, there are certain similarities in reports of like phenomena. These reports provide a greater understanding of tornado characteristics. Probably the largest single collection of eyewitness accounts is found in S.D. Flora's book, Tornadoes of the United States;[3] however, many others, such as Battaan,[4] have compiled interesting and informative reports. Technical journals, such as the Bulletin of the American Meteorological Society and the Monthly Weather Review, commonly make comprehensive reports of extraordinary features of tornadoes. In addition, Bechtel Corporation sent structural engineers to examine the damage caused by the June 8, 1966, tornado which passed through Topeka, Kansas; the May 15, 1968, tornado which passed through Jonesboro, Arkansas; and the January 23, 1969, tornado which passed through Jackson, Mississippi. Their findings are presented here as a supplement to the research work done and the information published in technical literature. Additionally, Bechtel Corporation has retained Mr. C.F. Van Thullenar, former Regional Director of the U.S. Weather Bureau, as a consultant on tornadoes.

Although many photographs have been taken of tornadoes, there are too few on file to support a general quantitative analysis. On two occasions-the Dallas tornadoes of April 3, 1957, and the Fargo tornadoes of June 20, 1957-enough movies of tornadoes were taken to permit detailed analyses. The analyses performed by Hoecker[5] on the Dallas tornadoes and Fujita[6] on the Fargo tornadoes give a great deal of insight into the forces that a tornado could impose on a structure.

Based on analyses of the existence of or the lack of structural damage, several reports have been prepared to determine lower and

upper bounds of various tornado forces. The analysis by Segner[7] for the Dallas tornadoes is a good example of this method.

H.2.2 Tornado Design Criteria

A tornado can be characterized as a vortex possessing tangential, radial, and translational velocity. By establishing a reasonable tornado model, the magnitude of the loading parameters can be determined. When the above loading conditions are compared to the criterion, the conservatism of the criterion becomes evident.

Numerous methods have been suggested to increase the tornado resistance of a structure; for example, recommendations of Bates[8] and Reynolds[9] illustrate simple precautions that can be taken.

From these sources of information, the following basic criteria have been adopted for nuclear power plant design:

1. The velocity components are applied as a 300 mph horizontal wind applied over the full height of the structure

2. The pressure differential is applied as a 3 psi positive (bursting) pressure occurring in 3 sec

3. The missiles are applied as a 4,000 lb automobile flying through the air at 50 mph but not more than 25 ft above ground, as a 4 in by 12 in by 12 ft plank (108 lb) traveling end on at 300 mph over the full height of the structure, or as a 3 in dia Schedule 40 pipe 10 ft long traveling end-on at 100 mph over the full height of the structure

All three loading conditions are applied simultaneously, and , except for local crushing at the missile impact area, the allowable stresses on a structure, as calculated by the working stress design method of analysis, are limited to 90 percent of the yield stress of the reinforcing steel and 75 percent of the 28-day specified strength of the concrete and 150 percent of AISC Code allowable stress for structural steel provided that the primary stress is smaller than the yield stress. If ultimate strength-design methods of concrete members are used in accordance with ACI 318-63, approximately the same margin of safety is provided through appropriately assigned load factors.

H.3 TORNADO PROBABILITY

At the present time, probability does not affect the magnitude of the tornado criteria. The tornado design criteria for nuclear power plants is unchanged for sites which have a tornado probability of occurrence greater than one in ten thousand. Sites which have a probability of occurrence of less than one in ten thousand have generally not adopted tornado design criteria.

Tornado probability is a function of damage area and frequency. The general formula used in probability studies is:

$$\text{Probability} = \frac{\text{Average damage path area x frequency}}{\text{Area of observation}} \qquad (H.3\text{-}1)$$

This probability study is based on Thom's work.[12] He studied tornado path records in Iowa and Kansas and found a mean damage path length of 3.935 mi and a mean damage path width of 466 ft. Thom used a log-normal distribution and found the average damage path area to be 2.8209 mi^2.

Thom's paper includes a map of the United States with the mean annual frequency of tornadoes in 1 deg areas. The map is based on a 10 yr period (1953-1962), since earlier data were inconsistent. This map is more reliable; as population density increases and weather recording instruments become more sophisticated, more widespread reporting increases the figure for annual frequency. Figure H.3-1 shows probability curves based on:

$$P = \frac{\bar{a}t}{A} \qquad (H.3\text{-}2)$$

where

P = probability (chances in 10,000)

$\bar{a}$ = average area of damage path (2.8209 mi^2)
t = mean annual frequency for 1 deg square
A = area of 1 deg square (varies with latitude)

These probability curves do not consider the severity of the tornadoes or any historical data on the damage path area observed regionally.

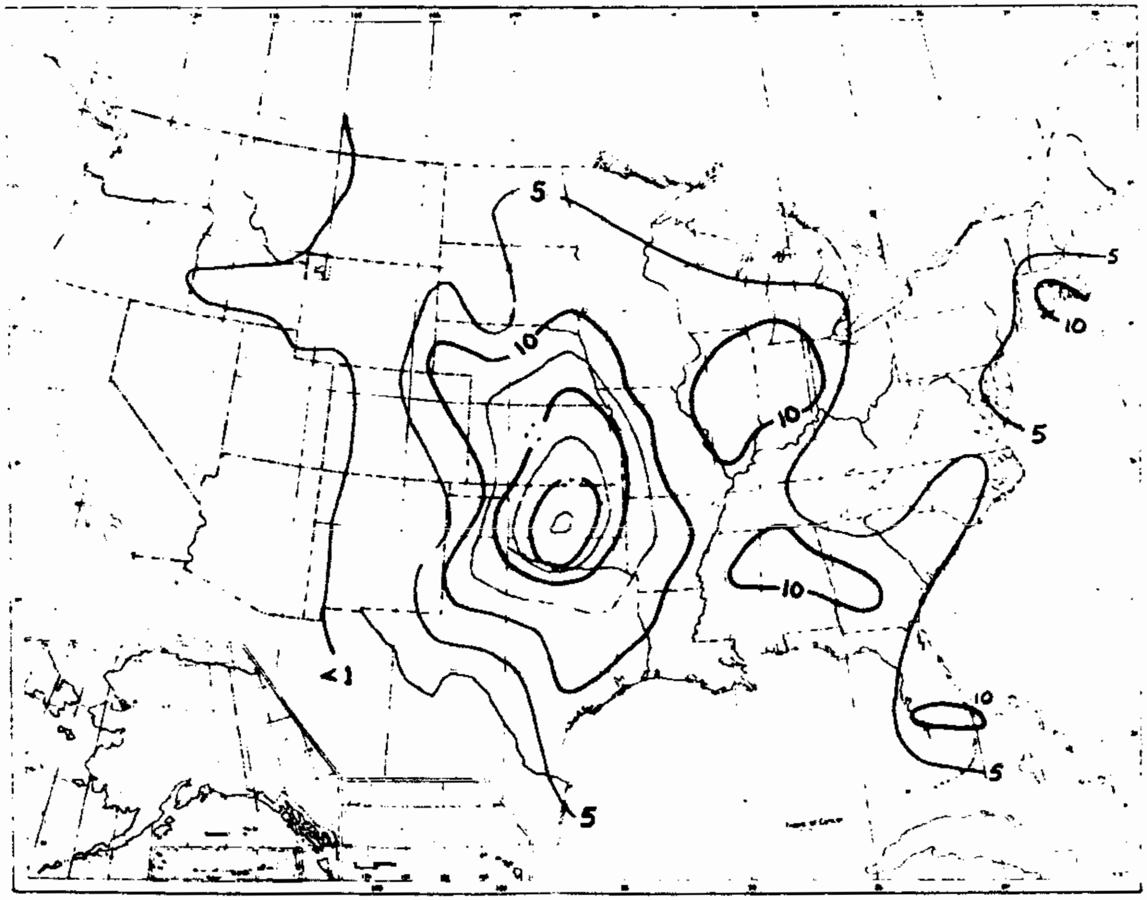

FIGURE H.3-1
ANNUAL PROBABILITY OF TORNADOES
(CHANCES IN 10,000)
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

H.4 WIND LOADING

H.4.1 Design Criterion

The velocity components are applied as a 300 mph horizontal wind over the full height of the structure.

H.4.2 Discussion

The design basis tornado model consists of a maximum tangential velocity component of 300 mph and a maximum translational velocity of 60 mph. The complexity of the tornado phenomenon requires considerable engineering judgement in establishing a wind loading criterion. The velocity components are applied as a 300 mph wind over the entire surface of the structure. Portions of ASCE paper 3269[10] are used to establish appropriate shape factors. Provisions for gust factors and variation of velocity with height are not applicable. The radial velocity component is neglected since it precedes the maximum tangential component, acts perpendicularly, and is of a magnitude 1/4 to 1/2 the maximum tangential velocity component.[5] The vertical component is also neglected since it acts parallel to the structural walls.

The tornado design criteria for various critical structures have been checked against extrapolations of Hoecker's[5] velocity profiles developed for movies of the Dallas tornadoes of April 2, 1957. Hoecker's studies were selected because they are the most detailed tornado analyses available. Figure H.4-1 shows the tangential velocity distribution of the design basis tornado as extrapolated from Hoecker's data. Figure H.4-2 shows a section of the design basis tornado at the height of maximum low-level tangential velocity for both 0 mph and 60 mph transverse wind velocities. These curves may be compared with wind loading criterion shown on Figure H.4-3.

The average wind loading on a structure is obtained by integrating the wind created by the design basis tornado over the surface of the structure. The design basis tornado with a translation velocity of 60 mph imposes on a typical BWR building an average wind loading of 220 mph velocity. This is 80 mph below the design wind loading and denotes the conservatism of the established wind loading criterion. Figure H.4-4 shows the design basis tornado wind velocity envelope on a typical Reactor Building wall, and, for comparison, Figure H.4-5 shows the design criterion wind velocity envelope.

Further confirmation of the conservatism of the design wind loading criterion can be obtained from analysis of photographs of tornadoes or analysis of the presence, or absence, of tornado-induced structural damage. No actual wind velocity measurements have been made for reasons Battaan[4] has stated:

> "The maximum speeds have never been recorded because the anemometers in a position to make the measurements have never survived. Speeds as high as 120 mph have been measured, but, to

cause the types of damage observed, much higher velocities would be needed."

On at least two occasions - the Dallas tornadoes of April 2, 1957, and the Fargo tornadoes of June 20, 1957 - sufficient photographs and movies were taken to permit detailed analyses. Hoecker[5] calculated a relative maximum low-level (below the storm cloud) tangential velocity of approximately 170 mph and a translational velocity of approximately 30 mph for the Dallas tornadoes; Fujita[6] calculated a relative maximum low-level tangential velocity of approximately 230 mph and a translational velocity of approximately 30 mph for the Fargo tornadoes.

Many reports attempt to place an upper and lower bound on the wind loadings caused by a particular tornado. One of the best examples of this is Segner's[7] analysis of the Dallas tornado. He was able to determine a lower bound of the wind loading from overturned freight cars by noting their positions and loads, yet there were enough loaded freight cars left standing in the direct path of tornado to establish an upper bound in wind loading. His calculations show that wind velocities on the order of 200 mph would be necessary to perform some of the observed incidents. Reynolds'[11] studies of nine different tornadoes — including the major ones that struck Vicksburg, Mississippi; Waco, Texas; and Wood River, Illinois — state that the damage patterns suggest maximum wind velocities not exceeding 200 or 250 mph. Bates[8] has summarized many studies of tornado-induced wind damage in this statement:

"Maximum wind speeds producing damage (threshold speeds to produce damage observed) in tornadoes have been estimated to be in the 250 mph range in several studies. With greatest frequencies, the wind speeds appear to be in the range of 100 to 150 mph. Unfortunately, estimates of maximum wind speeds (speeds which could not have been exceeded) are more difficult to determine from observed damage. An outstanding case was found in a watertower directly on the track of the Ruskin Heights, Missouri, tornado of 1957. Engineering estimates indicate that this tower, which was essentially undamaged, could have with-stood wind speeds up to 500 mph. It appears from these several studies that a conservative maximum wind speed at anemometer height is 300 mph."

H.4.3 Additional Analysis

If a noncritical portion of a critical structure contains a light gage metal siding, or if a portion of a critical structure could be exposed to wind velocities in excess of the 300 mph design wind loading criterion, additional analyses must be performed.

In the superstructure there are often portions of the structure housing no unprotected critical equipment and consisting of a steel frame clad with light gage metal siding. Under tornado wind loading conditions, it is assumed that one third of the siding remains in place. This assumption is quite reasonable because the applied force

is approximately 185 psf on the windward side and 110 psf on the leeward side; the normal allowable design pressure is on the order of 25 psf. If the structure supports a bridge crane, the steel frame is designed so that it will not collapse or distort and allow the crane to fall. Steel frames supporting the roof and crane are checked to assure that stresses do not exceed 90 percent of the minimum yield stress if one third of the light gage metal siding remains in place.

Tall, tornado-resistant structures could be subjected to wind velocities in excess of 300 mph. As an example, Figures H.4-4 and H.4-5 indicate a typical BWR Reactor Building on which the design basis tornado wind loading and the design wind loading criterion are imposed. For the portion of those structures that could experience wind velocities in excess of the 300 mph design wind loading, analysis using the wind loadings from the design basis tornado is performed to assure that no unacceptable structural failure will occur. For this type of analysis, limiting values are established to control the maximum structural deformations to within defined limits and to provide strength equal to or greater than that required to sustain the loads and limit the deformations.

Since the probability of the occurrence of a design basis loss of coolant accident or a design basis tornado during the life of a plant is small, the probability of the simultaneous occurrence of these two independent events is vanishingly small. Therefore, the need to control leakage is not considered for tornado loadings.

H.4.4 Conclusions

Available information on tornado effects supports the conservatism of a 300 mph wind loading as the design wind loading criterion. Analysis of the postulated design basis tornado (300 mph maximum tangential velocity with a 60 mph translation velocity) also indicates the conservatism of the design criterion. Therefore, it is reasonable to assume that structures designed to meet this criterion will provide protection from tornado-induced wind.

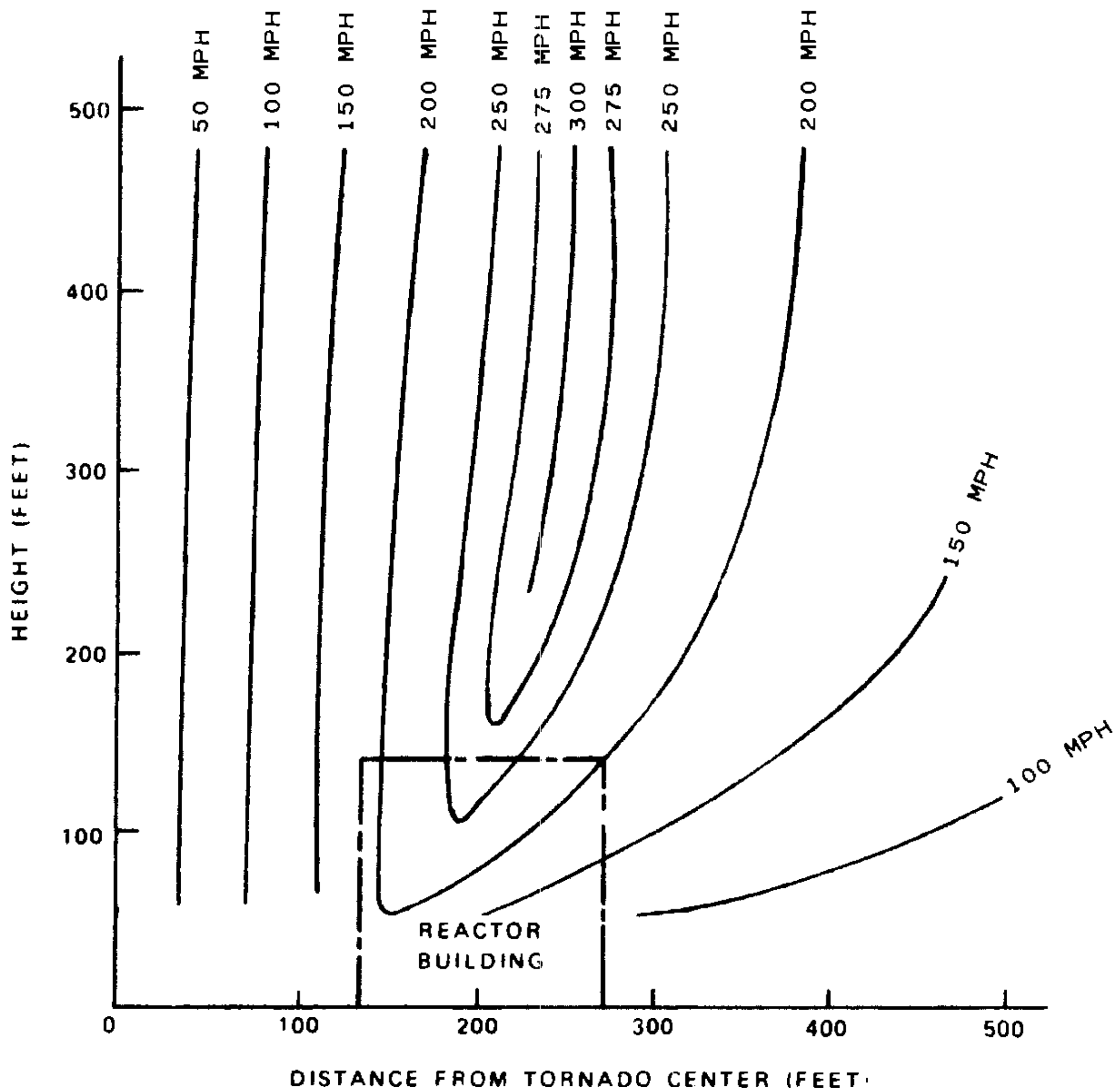

FIGURE H.4-1
DESIGN TORNADO TANGENTIAL VELOCITY DISTRIBUTION
PILGRIM NUCLEAR POWER STATION
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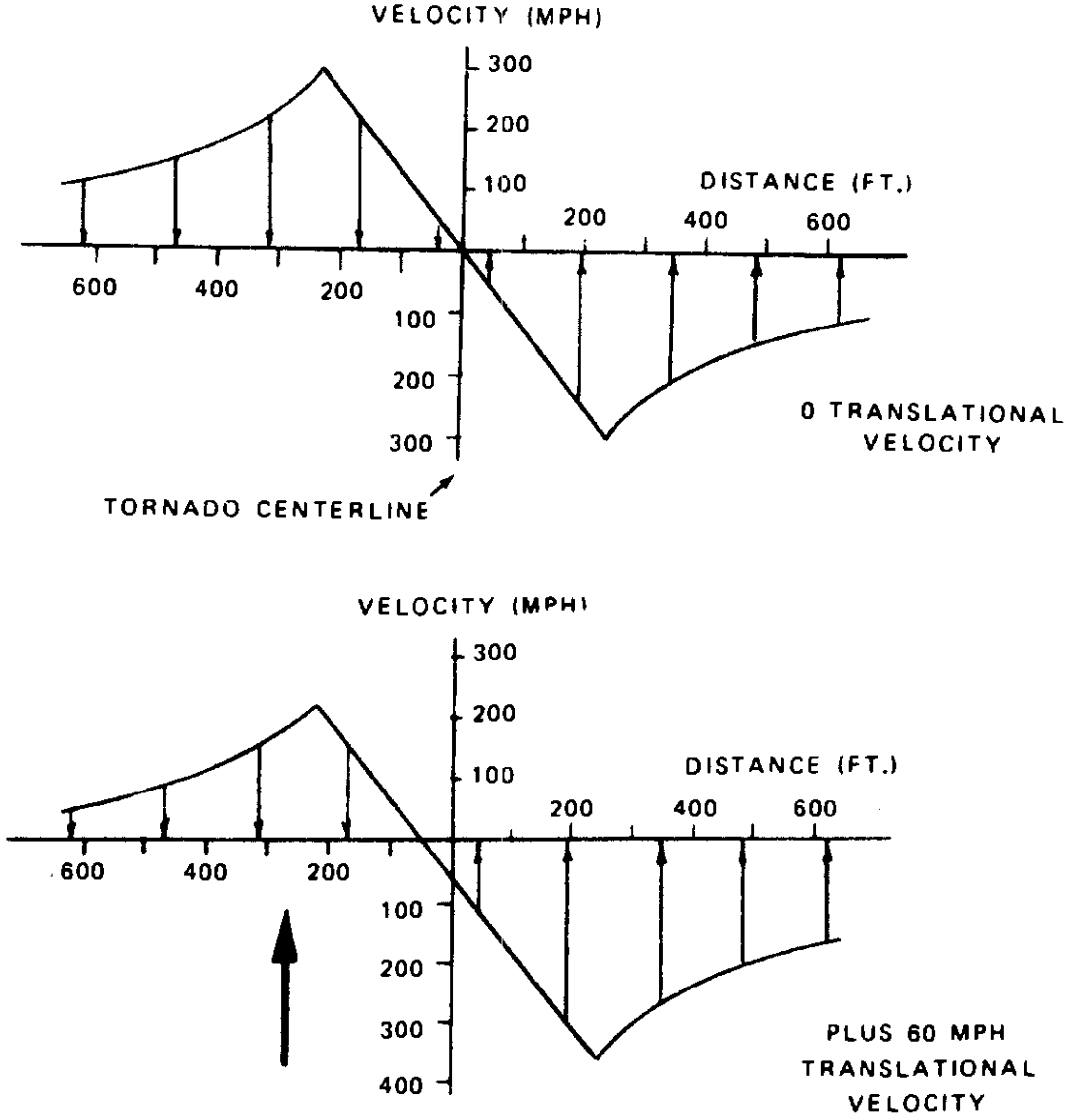

FIGURE H. 4-2
DESIGN TORNADO
TANGENTIAL VELOCITY
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

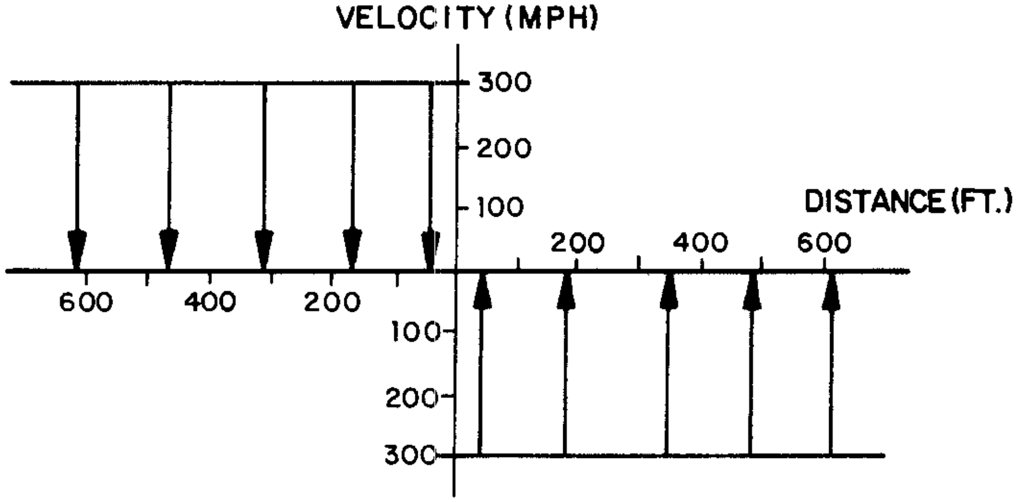

FIGURE H.4-3
TORNADO WIND VELOCITY
DESIGN CRITERION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

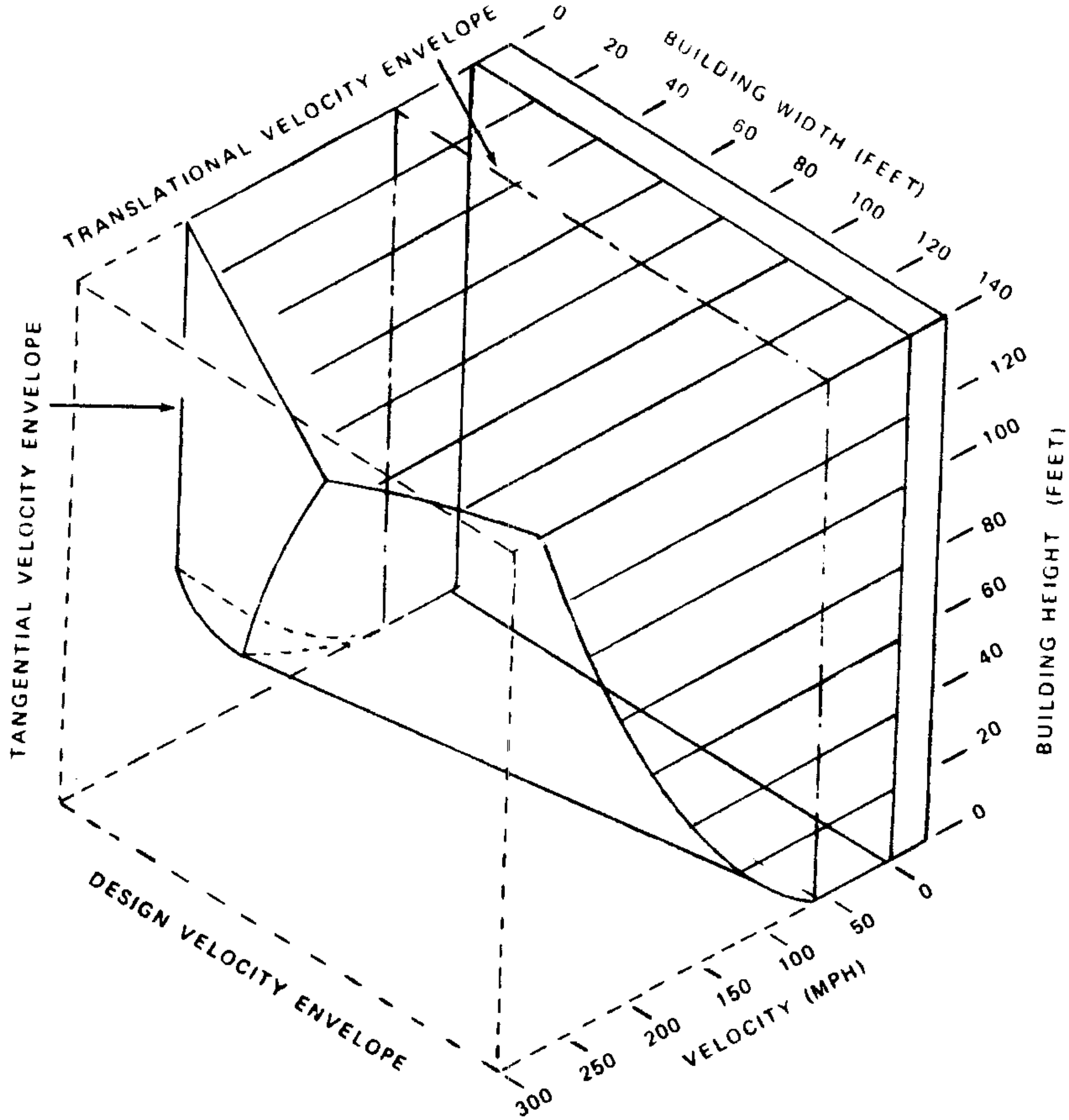

FIGURE H.4-4

WIND VELOCITY ENVELOPE FOR A TORNADO WITH A MAXIMUM TANGENTIAL OF 300 MPH AND A TRANSLATIONAL VELOCITY OF 60 MPH

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

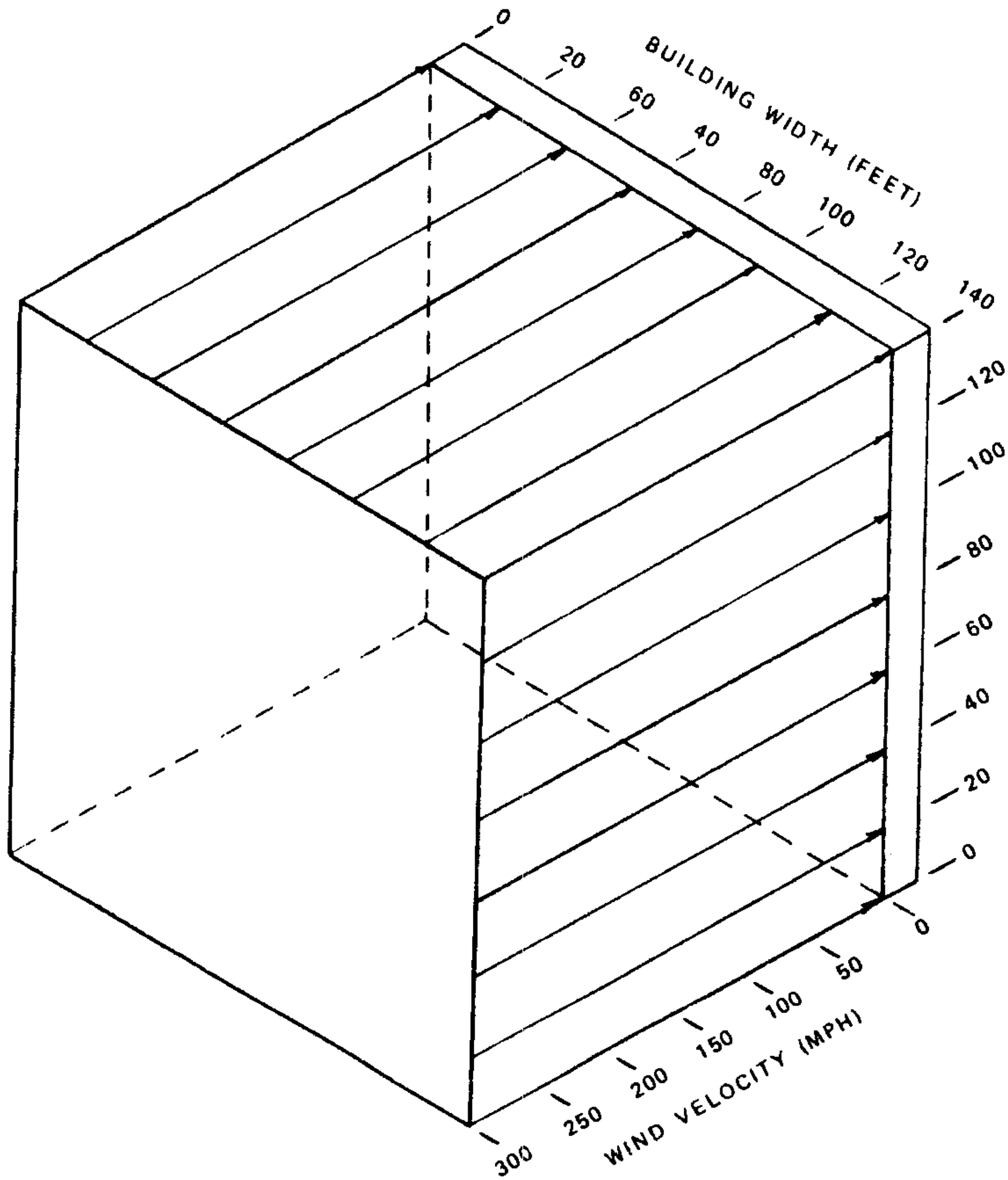

FIGURE H. 4-5
DESIGN WIND-VELOCITY ENVELOPE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

H.5 PRESSURE DIFFERENTIAL

H.5.1 Design Criterion

The pressure differential is applied as a 3 psi positive (bursting) pressure occurring in 3 sec.

H.5.2 Discussion

All critical structures are designed to withstand a tornado-induced depressurization rate of 1 psi/sec for 3 sec, a calm for 2 sec, and a repressurization to ambient pressure in the following 3 sec. Calculational confirmation of the design criteria may be obtained from the cyclostrophic wind equation

$$\delta P/\delta r = \rho v^2/r \tag{H.5-1}$$

where

P = pressure
r = radial distance
ρ = density
v = velocity

once the tornado tangential velocity radius relationship is established. From the design basis tornado tangential velocity distribution shown in Figure H.4-2 and the cyclostrophic wind equation, the pressure-versus-distance relationship has been calculated. This calculation shows that the design basis tornado has a maximum pressure drop of 3 psi at the center of its vortex or that the atmospheric pressure drops to 11.7 psia from an assumed ambient pressure of 14.7 psia. The pressure time history of the design basis tornado has been calculated by assuming that the tornado has a translational velocity of 60 mph as it passes a point on its path centerline. This relationship is shown on Figure H.5-1. For comparison, the design criteria of a 1 psi/sec depressurization rate for 3 sec, followed by a constant pressure for 2 sec, and a repressurization rate of 1 psi/sec for 3 sec are superimposed.

Further confirmation of the conservatism of the pressure differential design criterion is gained from surveying the (officially recorded) measurements and eyewitness accounts of extreme pressure drop. Reynolds[9] has summarized the officially recorded measurements of significant pressure drops.

The following table lists these pressure drops:

	Location	"hg	psi
1.	Dyersburg, Tennessee	0.65	0.32
2.	Sidney, Nebraska	0.48	0.24
3.	Minneapolis, Minnesota	0.42	0.21
4.	Little Rock, Arkansas	0.38	0.19
5.	Cleveland, Ohio	0.25	0.12

Although these are the greatest officially recorded pressure drops, some of the eyewitness accounts are far more spectacular. Flora[3] has recorded the following two incidents:

"One of the few actual measurements of this partial vacuum was due to a fortuitous combination of circumstances during the very destructive tornado that struck St. Louis on May 27, 1896. Near Lafayette Park, where destruction was especially severe, the vortex passed directly over the residence of the widow of former Commissioner Richard Klemm. Mrs. Klemm's son happened to notice that the hand of an aneroid barometer in the house pointed straight down when the storm was at its height. This position was so unusual that he remembered it. The reading was reported to H. C. Frankenfield, official in charge of the local Weather Bureau office, who had the barometer carefully tested for accuracy at this reading. After the necessary corrections had been made, it was determined that the correct low reading, reduced to sea level, was 26.94 in, a drop of approximately 10 percent from normal.

"Another record of even greater decrease in pressure exists in connection with the destructive tornado that struck Minneapolis on August 20, 1904. Here is a report of it:

"Two reliable gentlemen living near the residence of Mr. W. D. Washburn, which was near the center of the wide path of greatest damage, were watching an aneroid barometer at the time of the storm. They stated that the needle (the indicating hand) of the barometer went down to 23 in and returned immediately to near its former reading. This barometer had been compared with the official barometer at the Weather Bureau office not very long before the storm and found correct. Even allowing for error because of a possible momentum gained by the needle in its rapid drop, the reading was a remarkably low one."

Reynolds[9] states that he is skeptical of both these reports since the observations were made under conditions of stress, hardly the circumstances for taking a careful barometric reading. In addition, he states that the majority of the damage done by the Minnesota tornado showed the "effects of a straight blow of hurricane force," not the explosive type damage that would be associated with the pressure drop of 5 to 6 in (2.5 to 3 psi).

Although there is a great deal of skepticism about such eyewitness accounts, the design criterion of 3 psi is greater than the highest observed pressure drops described above. In addition to the officially recorded measurements and eyewitness accounts, Hoecker[12] has calculated that there was a maximum pressure drop of 0.86 psi, 75 percent of which occurred in 5 sec during the Dallas tornado of April 2, 1957. This pressure drop rate is much less than the tornado criterion of 100 percent of the total pressure drop occurring in 3 sec.

H.5.3 Additional Analysis

In addition to checking all nonvented compartments to verify that they are capable of withstanding a tornado-induced pressure differential of 3 psi, all structural components in vented compartments are checked to confirm that they will withstand the maximum calculated transient pressure differential. The transient pressure for vented compartments is calculated by constructing a flow diagram of all air volumes and interconnecting vent areas. An instantaneous pressure drop, caused by opening of the relief panel, is followed by a further depressurization to 3 psi at a rate of 1 psi per sec, a steady pressure for 2 sec, and repressurization to atmospheric pressure at a rate of 1 psi/sec. The maximum calculated transient pressure differential for a structural component is the maximum difference in pressure between the air volumes it separates.

H.5.4 Conclusions

Available information on tornado effects supports the conservatism of the tornado pressure differential design criterion of a 3 psi pressure drop occurring in 3 sec. Analysis of the postulated design basis tornado (maximum tangential velocity of 300 mph with a translational velocity of 60 mph) also indicates the conservatism of the design criterion. Therefore, it is reasonable to assume that structures designed to meet this criterion will provide protection from tornado-induced differential pressure.

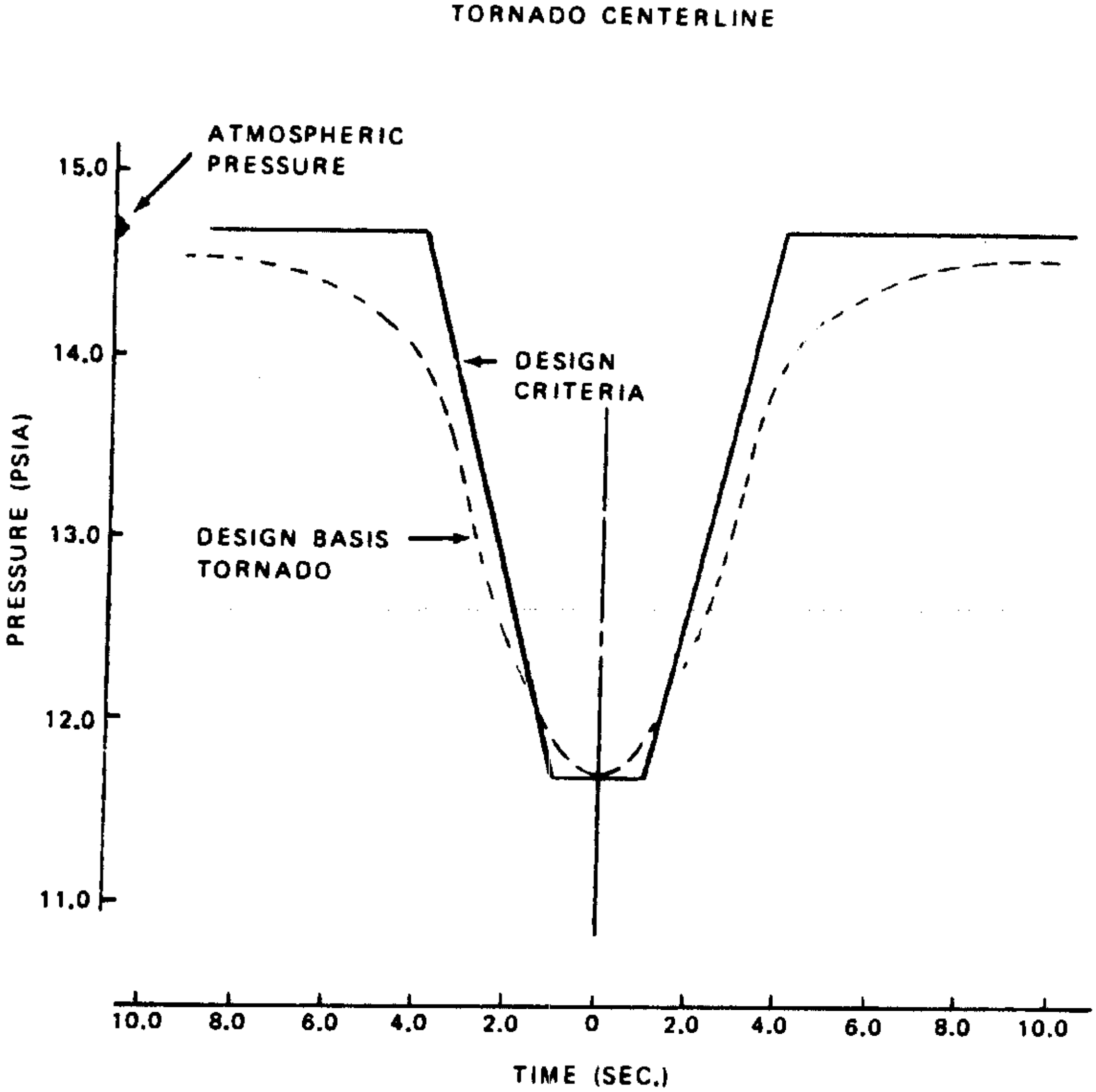

FIGURE H. 5-1
TIME-PRESSURE HISTORY OF THE DESIGN BASIS TORNADO WITH A TRANSLATIONAL VELOCITY OF 60 MPH
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

H.6 WATER LOSS

H.6.1 Design Criterion

A significant amount of water will not be removed from the fuel pool during a tornado.

H.6.2 Discussion

The primary damage a tornado can inflict on the fuel pool is the loss of water as a cooling medium. The potential means of water removal are direct removal by the tornado or the breaching of the fuel pool structure by tornado missiles. The direct removal of water by the tornado requires a combination of two mechanisms: The first mechanism is pressure-induced displacement, and the second mechanism is water entrainment.

Pressure-induced displacement is the lifting of water by differential pressure. This is not likely over a pool because of the relative size of the pool compared with the tornado. Figure H.6-1 indicates the position of the pool at the maximum pressure drop rate. The pressure differential is enough to tilt the water of a 40 ft long pool approximately 11 in. as indicated on Figure H.6-2. This distortion, acting alone, cannot remove water from the pool.

The second means of direct water removal by the tornado is water particle entrainment. The strong convergent forces of a tornado produce frictional drag on the water surface, causing waves which are immediately entrained by the wind. This effect is observed in the spray bush of waterspouts (tornadoes over water). Since the tornado is large, compared with the pool, the wind can be approximated as a two dimensional rather than a three dimensional vortex. As the wind passes over the pool, an eddy forms at each step. Figure H.6-3 shows the effectiveness of the freeboard in reducing the wind contact area. The areas under the eddies are relatively stagnant and, therefore, not capable of creating the surface disturbance necessary for water entrainment. Experiments by Liu, Kline, and Johnston[13] have shown the stagnant area to be a function of freeboard height. The numerical values of the eddy effect are shown on Figure H.6-3. Water removal occurs only over the attached area "La." Based on these experiments, when freeboard height equals pool length/7, ($h = L/7$), the attached area is zero and there can be no direct water removal. The experiment indicates that the maximum depth of water removed is 1/7 times the exposed pool length, as shown on Figure H.6-4 (less than 6 ft in a 40 ft long pool). The limits are conservative since no credit is taken for building protection or time. If building protection is available, no direct water removal will occur because of the great effective freeboard.

The mechanisms discussed above are theoretical and appear to be idealistic when compared with documented occurrences. One of the first reports to cause alarm concerned a 39 ft well that was "sucked dry" by a tornado at Marshall, Missouri, April 18, 1880.[3] This occurrence is, of course, impossible theoretically and was refuted by

a local newspaper the following week. An excerpt from the "Springfield Patriot Advertiser," dated April 29, 1880, said:

> "and right here we may remark that many people have drawn largely upon their imaginations -- the story of the child blown away and found hanging next morning in the tree; of cattle taken up and carried a long distance and deposited without serious injury; of wells sucked dry -- are pure fiction."

Other investigations of tornado effects on bodies of water do not indicate major water losses. Two bodies of water in the path of the Jonesboro, Arkansas, tornado (May 15, 1968) experienced no detectable water loss.[14] A pond directly in the tornado's path had overflowed its dike because of heavy rains following the storm. This did not indicate water loss from the tornado. Also, no water loss was noted by workmen at the pond the following morning. The second body of water affected was a swimming pool that underwent high winds on the periphery of the tornado. Figure H.6-5 shows about 2 ft of freeboard. It is doubtful that any water was removed from the pool because the stains below the water surface and the sun-bleached steps above indicate the same water level existing before the tornado. A third report of water loss occurred in Jackson, Mississippi, on January 23, 1969. Reports of 40 ft of water removed by the tornado have been investigated. The pond was less than 20 ft deep when full, and following the tornado no water loss was noticed by the sheriff or other reliable sources. Spot interviews by a ground survey team reported no unusual distrubances to any open water surfaces along the path of the tornado. The above occurrences have been confirmed in a discussion with the state climatologist at Jackson. A complete survey of the Topeka, Kansas, tornado of June 8, 1966, did not find any unusual disturbances of bodies of water. The above discussion is not intended to imply that no water is removed by tornadoes; that water removing forces are present in tornadoes is indicated by the spray bush associated with waterspouts. The implication of the above phenonemon is that the idealistic mechanism of water removal is, in actuality, very conservative because it neglects the transient dependence upon time of contact between the tornado and the water surface.

The only other potential for gross water loss from the fuel pool is the breaching of the pool structure by a tornado missile. The pool structure generally consists of 6 ft thick reinforced concrete walls and base. The missile required to penetrate a 6 ft slab is given in the following table.

Weight of missile per ft^2 (lbs)	1,000	500	100
Required impact velocity for 6 ft penetration (fps)	815	6,030	532,000

NOTE: 300 mph = 440 fps

Required impact energy for 6 ft penetration (ft lbs)10 X 10	10 X 10^6	282 X 10^6	47 X 10^{10}

The above values are based on the modified Petry penetration formula and are felt to be conservative as explained in the discussion of missile criteria. The velocities indicated in the table are obviously not available for tornado missiles. Further evidence of the pool integrity is obtained by noting that the above values are for missiles striding the slab normally. Any induced angle of impact will decrease the penetration of the missile. The missile will ricochet off if the impact angle is less than 80 deg, thus eliminating the possibility of a penetrating impact on the lower portion of the fuel pool walls.

The water in the pool is an effective deterrent of missiles. The 40 ft of water above the base slab would absorb a large portion of the missile energy, thus decreasing the missile's penetrating capability. Figure H.6-6 indicates the initial and final energies of missiles going through 40 ft of water. The higher the initial energy, the better the absorbing characteristics of the water. Calculations have shown that a base slab thickness of 1 ft would provide adequate protection from tornado-hurled missiles. Therefore, other structural and radiation requirements govern the base slab design. The 6 ft slab, common to most fuel pools, is more than adequate to provide structural integrity during a tornado.

H.6.3 Conclusions

The two potential means of water removal from a fuel pool are direct removal by the tornado and water leakage caused by missile impact.

The direct removal of water by the tornado is limited by an eddy formation over the pool. Water removed in this way still leaves adequate coverage of the fuel elements.

The second possibility of gross water loss would be from a rupture of the pool by tornado missiles. The pool structure usually consists of 6 ft thick walls and base which will withstand any tornado-induced missiles. The possible angles of impact and energy absorption characteristics of the water add to the integrity of the pool structure. The pool, therefore, will not lose a critical amount of water during a tornado.

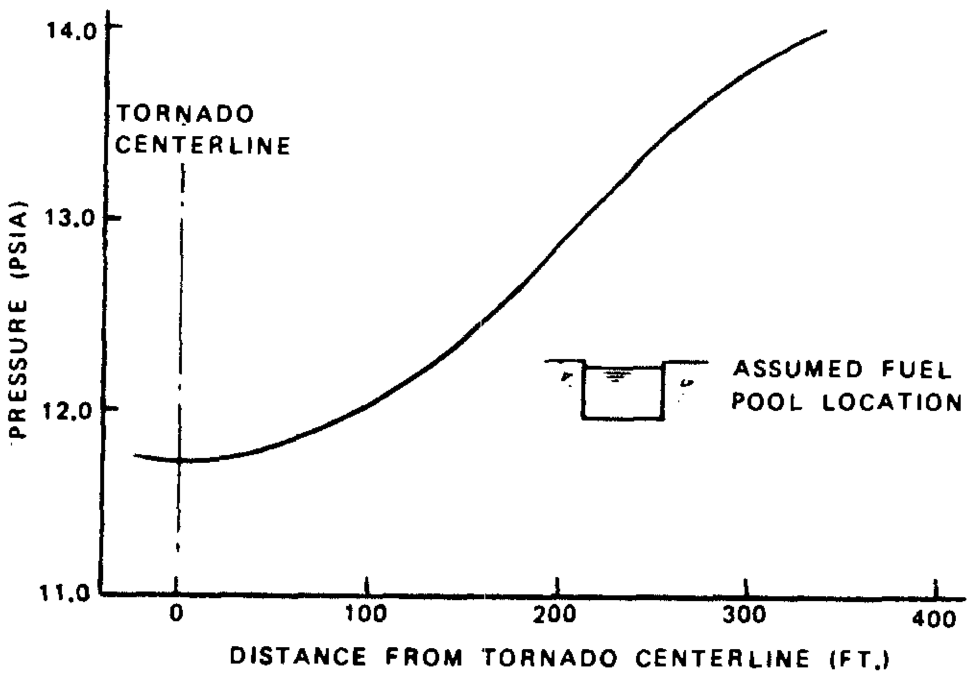

FIGURE H.6-1
FUEL POOL SIZE
RELATIVE TO TORNADO
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

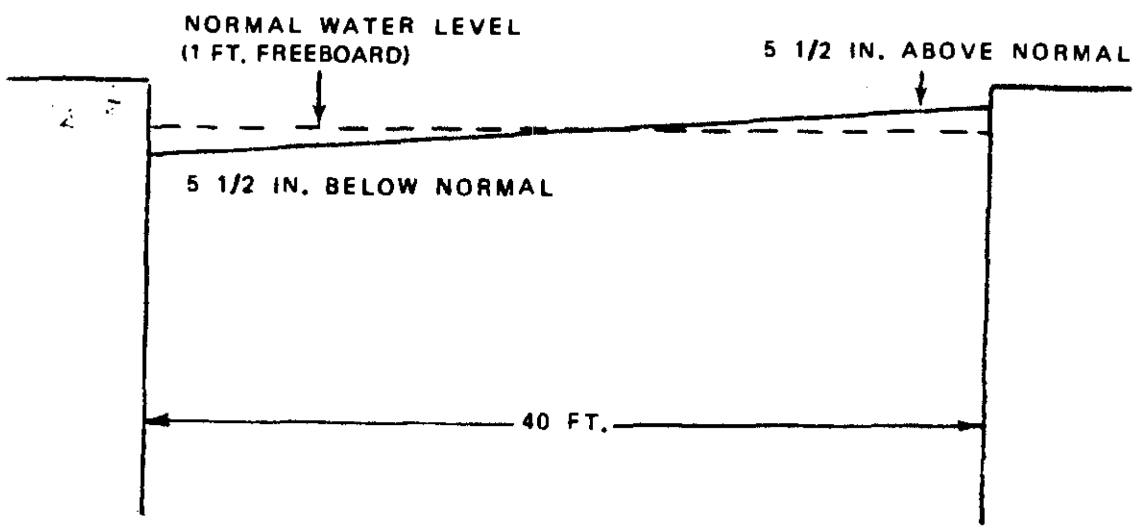

FIGURE H.6-2
MAXIMUM WATER SURFACE DISTORTION DUE ONLY TO PRESSURE REDUCTION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

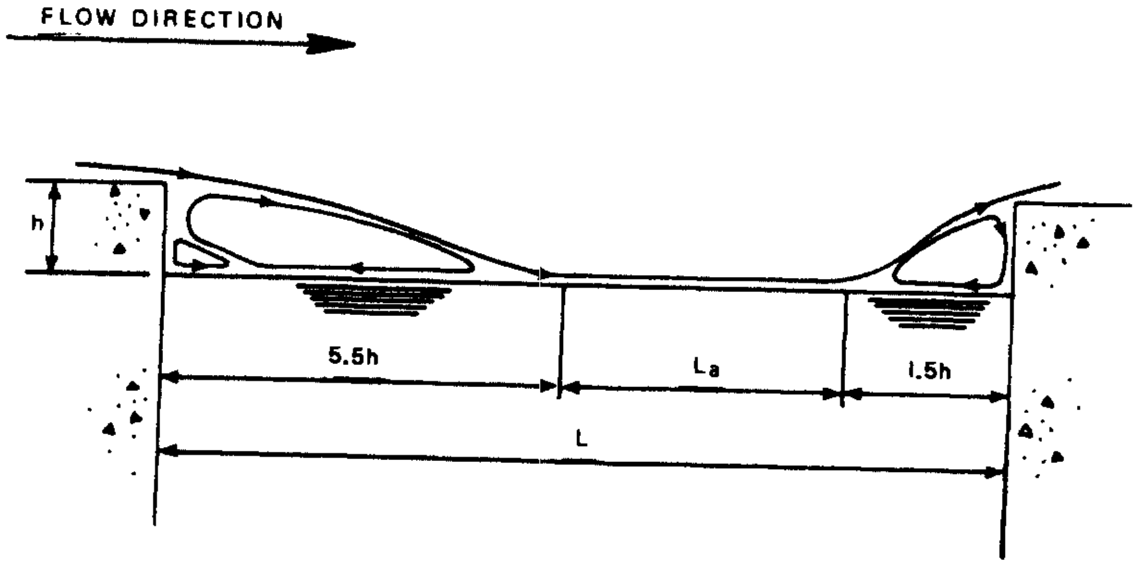

FIGURE H.6-3
AIR FLOW OVER POOL
DURING WATER REMOVAL
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

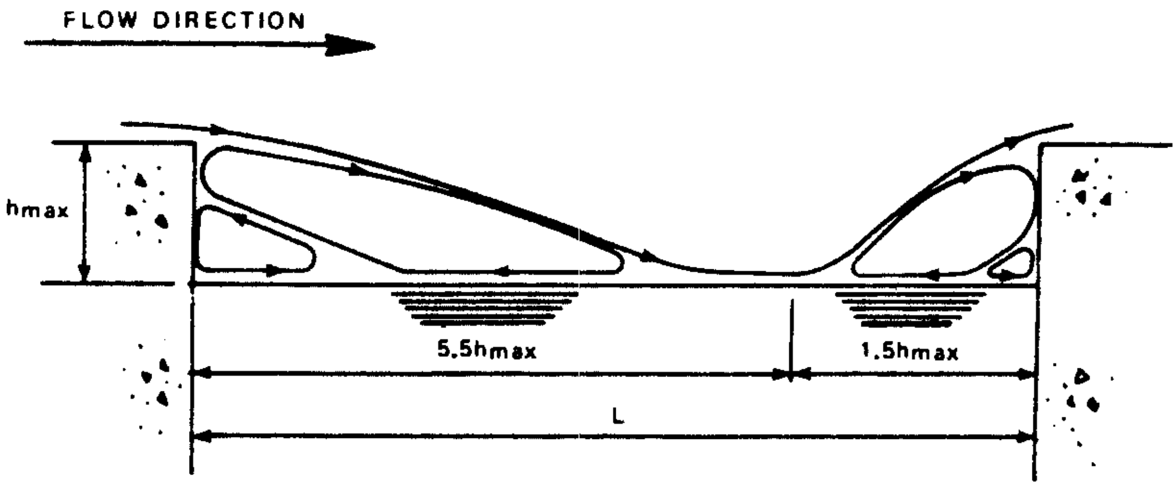

FIGURE H.6-4
AIR FLOW OVER POOL AFTER MAXIMUM WATER REMOVAL
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

NOTE: LITTLE OR NO WATER LOSS APPARENT
DUE TO SUN-BLEACHED STAIRS AND
LACK OF STAINS ON WALLS

FIGURE H.6-5
SWIMMING POOL IN JONESBORO, ARKANSAS, TORNADO PATH
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

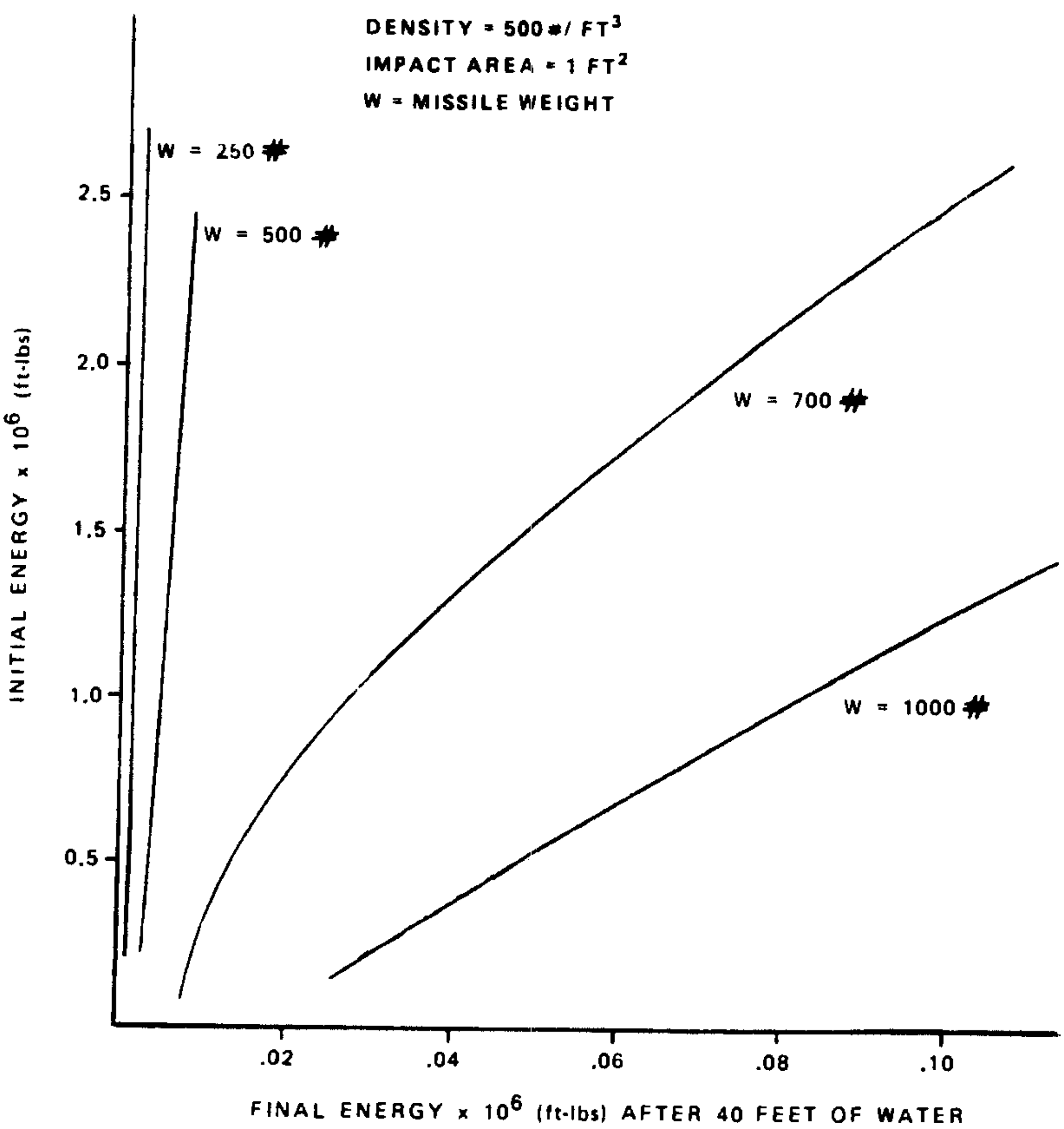

FIGURE H. 6-6
ENERGY DISSIPATION
OF MISSILES IN WATER
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

H.7. TORNADO MISSILES

H.7.1 Design Criteria

The effects of the following tornado-induced missiles shall be evaluated:

1. A 4 in x 12 in x 12 ft long wood plank (108 lb) traveling end-on at 300 mph over the full height of the structure

2. A 3 in id pipe (ASA Schedule No. 40) 10 ft long traveling end-on at 100 mph over the full height of the structure

3. A passenger auto (4,000 lb) traveling end-on at 50 mph with a contact area of 20 ft^2 and at a height not greater than 25 ft above ground

H.7.2 Discussion

These criteria represent a conservative spectrum of tornado missiles. The three missiles cited do not necessarily represent realistic occurrences but are as upper design parameters. The criteria represent both light missiles subject to high velocities and heavy objects that stay relatively close to the ground.

The wood plank represents a conservative upper limit of the spectrum of missiles that might obtain sustained flight in a tornado. Injection and acceleration of specific missiles are impossible to determine since drag and lift coefficients, missile orientation, and the effect wind components are not accurately determinable for nonaerodynamic objects. Because of the complexity of the problem, the assumption is made that objects with low weight-to-area ratios will reach velocities approaching the maximum horizontal wind velocity of 300 mph. This, of course, is conservative, since drag force, tumbling, and erratic wind conditions will not allow the missile to approach this velocity. The light missiles are assumed to be injected at any height by the vertical velocity component; thus, they may act at any height. Missiles with a low weight-to-area ratio usually possess crushing characteristics upon impact. Calculations made on the plank, assuming a tumbling sequence, velocity forces, shape factors, and injection height, indicate a more realistic impact velocity of about 150 mph. These calculations, plus observed phenomena, confirm the conservatism implied.

The pipe, such as a portion of maintenance scaffolding, is typical of a missile with a high weight-to-impact-area ratio that might conceivably be temporarily suspended in the tornado. Information on this type of missile is limited, and thus deductive limitations must be incorporated. The injection and

suspension of any missile is dependent upon the object's shape. A round object will be rolled along the ground rather than injected. If the object is injected, its nonaerodynamic properties will cause it to fall. Preliminary calculations on the pipe with assumed parameters similar to the plank's indicate an impact velocity close to 50 mph. The 100 mph velocity of the pipe more than accounts for the uncertainties involved in pipe acceleration.

The third category of missiles is heavy masses that can be moved by the tornado, such as automobiles. The problems of theoretical calculations mentioned in the two previous discussions of missiles are also applicable here, with some deductive limitations applied. The most logical hypothesis of automobile movement is that of tumbling. As the object is injected, the wind level causes a rotation of the car, its rotation cancels the lift force and causes the object to fall. When this is repeated, tumbling occurs. The height of injection cannot be calculated since it is a function of lift coefficient, time, and erratic wind forces, but empirical evidence indicates that injection heights are low (possibility less than the height of the object). Based on observed incidents, a 50 mph horizontal velocity and a 25 ft injection height are believed to be conservative parameters.

Observations made of the Jonesboro, Arkansas, tornado of May 18, 1968, indicate that small debris was caught by fences and trees. This does not imply high altitudes or high velocities. Further observation concluded that a majority of the objects injected as the tornado passed through the town of Oil Trough, Arkansas, did not clear the adjacent river (100 yd).[14] Figure H.7-1, taken at the river, supports this conclusion. Automobiles observed along the same tornado path are damaged on all sides, as shown on Figure H.7-2. This type of damage implies tumbling. The low injection height hypothesis is supported by Figure H.7-3; the car shown did not clear the railroad embankment but did shear off a telephone pole 1 ft above the ground. Similar observations, such as cars shearing trees at grade, support the hypothesis of tumbling.

H.7.3 Analysis

Once the missile parameters have been established, it is necessary to determine how the criteria will be incorporated. To prevent a loss of function due to tornado-generated missiles, both structural stability and penetration must be investigated.

Structural stability is a function of momentum transfer. When a tornado-induced missile strikes a surface of a structure, the energy dissipated by the structure is usually very small because of the large difference between the masses of the missile and the wall, thus the missile effect on the structural stability is

neglected for the two lighter missiles. The car missile effects on the stability of the structure must be considered.

The study of missile penetration is a very complex problem. In general, the penetration of a wall involves the simultaneous action of elastic and plastic wave propagation, crack formation, friction, heating, spalling, and in some cases shattering of the missile. Obviously, any analytical solution regarding all the above phenomena would be a very tedious, if not impossible, problem. Therefore, an empirical solution which is applicable and reasonably conservative is desirable. A general empirical expression for missile penetration has been proposed by different authors expressing the resisting force of a wall as:

$$F = -m\frac{dv}{dt} = B_1V^2 + B_2{}^2V + B_3 + B_4f(v) \qquad \text{(H.7-1)}$$

where:

v = missile velocity

m = mass of missile

B_1 = empirical constant to account for acceleration of the wall material adjacent to the projectile

B_2 = empirical constant to account for frictional forces

B_3 = empirical constant for cohesive strength of the wall

B_4 = empirical constant for hypervelocity effect

The evaluation of these constants is dependent on the conditions of the experiment, and thus under difficult circumstances one or two of the constants are usually assumed to be zero. Obviously then, the circumstances of formula development must be studied before the selection of any empirical formula.

At tornado velocities, the hypervelocity and frictional effects are neglected for most missiles. From the equation of motion, the general penetration formula becomes:

$$\overline{X} = 1/2\,\frac{m}{B_1}\ln\left[\frac{B_1}{B_3}Vo^3 + 1\right] \qquad \text{(H.7-2)}$$

For velocities above ordnance range, the penetration value is a function of nose shape and kinetic energy of the missile.

The general penetration formula becomes:

$$X = B_4 m^{1/3} \quad Vo^{2/3} \qquad \text{(H.7-3)}$$

The formula applicable to tornado velocity missiles is very similar to those discussed by J. S. Rinehart and A. Amirikian,[21,18] while the general high velocity penetration formula is similar to those of the Ballistic Research Laboratories, Army Corps of Engineers, National Research Committee[17], and Ammann and Whitney's formula,[19] a careful evaluation of applicable constants (B_1, B_2, B_3, B_4) should be made to obtain a proper penetration formula. Formulas such as the modified Petry (Amirikian) do not consider the effects of nose shape or crater formation and are thus invalid for hypervelocity missiles. It should be noted that most other missile penetration formulas developed for high velocity missiles are not applicable for lower velocities. Many of these formulas have limits of applicablility; these should be adhered to. The normal lower limit for high velocity formulas is 500 ft/sec.

For tornado-generated missiles the most acceptable empirical penetration formulas are those of Rinehart and Amirikian. The numerical values of the constants in Rinehart's formula are not available at this time; therefore, Amirikian's modified Petry formula is suggested. It is:

$$D = K_1 \frac{W}{A} \log_{10} \left[1 + \frac{V^2}{215{,}000} \right] \qquad \text{(H.7-4)}$$

where:

D = depth of penetration into infinite thickness (ft)

W = missile weight (lb)

A = maximum effective cross sectional area of the missile (ft^2)

V = impact velocity (ft/sec)

K_1 = experimentally obtained material coefficient for penetration ($K_1 = 4.76 \times 10^{-3}$ for normal reinforced concrete)

Penetration into a finite slab is a multiple of D.

$$D_1 = K_2 D \qquad \text{(H.7-5)}$$

where:

D_1 = penetration into finite slab

D = penetration into infinite slab

$K_2 = 1 + e^{-4x}$ where $x = [(T/D)-2]$

T = slab thickness

When $T = 2D$ the missile will just perforate the slab, thus a limiting value of penetration in finite slabs is:

$$D_1 = 2D \qquad \text{(H.7-6)}$$

A preliminary analytical analysis utilizing a finite element technique gives good agreement to the penetration values of the modified Petry formula.

Studies of the effects of missile impact on spalling have been made for high velocity missiles.[19,20,22]. Results indicate that the spalled portion obtains a velocity less than 20 ft/sec. The reinforcing steel limits the spalled thickness. Thus, for the relatively low impact velocities involved, the effect of spalling is neglected.

All of the penetration formulae are for missiles of steel; therefore, the effect of missile crushing is neglected. The obvious result is that the crushing upon impact of wood missiles will eliminate much of the penetration capacity. The effect of crushing obviously should be considered for critical wood missiles. No formulae are known to account for the effect of missile crushing, but a reasonable solution is to subtract the crushing energy of the missile from the impact energy and assume the remainder is dissipated in penetration by an equivalent steel missile.

Penetration values of the pipe missile should consider the effect of friction due to an inside and outside shearing surface. As previously discussed, friction is an important parameter in deterring penetration. The modified Petry formula does not emphasize friction and therefore, adjustments must be incorporated to obtain penetration values. The modified Petry formula suggests an area parameter based upon the net area of the missile.

The empirically derived constant (K_1) accounts for normal frictional effects. The additional frictional effects due to the geometry of a pipe are accounted for by modifying the net area to an "effective area." This effective area is a function of perimeters and is assumed to have the following relationship:

$$A = A_{net} \frac{P_O + P_I}{P_N} \qquad \text{(H.7-7)}$$

where:

A = Effective cross sectional area

A_{net} = Actual cross sectional area of the missile

P_O = Outside perimeter of object
P_I = Inside perimeter of object
P_N = The perimeter of a solid rod with an area equivalent to A

The effective area modification is consistent with solid missiles as P_I becomes zero and $A = A_{net}$.

H 7.4 Missile Spectrum Analysis

Further examination of the objects in and around the power plant that can become tornado missiles reveal three other categories.

Typical missiles that might be generated at grade level around the plant are shown on Table H.7-1.

Missiles with a large weight-to-projected-area ratio cannot be propelled higher than the walls around the refueling floor which are 145 ft above grade. Lighter missiles, such as items 6 and 7 on Table H.7-1, may be carried high enough to clear the walls around the refueling floor. As will be discussed later, none of these missiles have sufficient energy to rupture a fuel rod.

Missiles that might be generated at a high level are shown on Table H.7-2.

These missiles cannot enter the refueling area by penetrating the precast concrete walls as discussed in Section 3.1.3 in Amendment 4 to DAR. There could, however, be objects falling vertically after they have lost their horizontal velocity. In the latter case, they have been conservatively assumed to fall freely to the water surface neglecting aerodynamic braking effects and assumed to penetrate the water through the path of least resistance. These falling objects and debris which may fall into the fuel pool will then be decelerated as they travel through the 25 ft of water cushion above the spent fuel assemblies and racks.

The third classification of missiles are those which might be generated by a tornado which has entered the refueling floor. The potential missiles in this category are shown on Table H.7-3.

The energy required to rupture a single fuel rod in compression is discussed in Section XIV-3.2 of the DAR and is 250 ft lb. Because of the size of the reactor head insulation and the ventilation ducts, the impact energy would most probably be transmitted to several fuel assemblies, and no fuel rods would fail.

The examination of objects in and around the power plant included many potential missiles which can be eliminated for the reasons given below:

1. Transformers - small transformers will be bolted to their foundations and main transformers will be too heavy to become missiles

2. The main off-gas stack - see Amendment 4, Part I, Section 3.1.2 which concludes that this will not become a missile

3. Painter's trolley on stack - will be designed so that it will not separate from the stack monorail in tornado winds

4. Stack ladder, 10 ft long - a permanent structure and welded or bolted connections will be adequate to prevent its separation from the stack in a 300 mph wind

5. Pipe, 4 ft dia C.I. x 12 ft 0 in - observations of many tornados confirm that this type object cannot be lifted by 300 mph tornado winds but may be tumbled along the ground

6. The refueling crane - must not be allowed to fall into the fuel pool; therefore, dead-man type automatically operated rail clamps will be provided which are actuated when all controls are released. The entire crane structure will be designed to remain intact when subjected to 300 mph winds. The clamps, rails, and rail attachments will be designed to sustain the stresses generated by these forces

7. The reactor stud tensioner - too heavy to be lifted by tornado winds which can enter the refueling floor with the walls intact. To prevent sliding this equipment into the fuel pool, concrete curbs will be provided

8. The reactor service platform - provision will be made for bolting this item to the floor during tornado alerts

9. Reactor head strongback - provision will be made to bolt this item to the floor during tornado alerts

10. Stud tensioner power unit - a heavy compact item which cannot be lifted by the tornado winds. Curbs will be provided to prevent its sliding into the fuel pool

11. Equipment hatch covers on the refueling floor - bolted down or stored on other floors during tornado alerts except when they are small and can be shown to have acceptably low energy of impact on the fuel element racks

12. Jib cranes - designed so they will not be released from their sockets in tornado winds

13. The portable ladder for the reactor cavity - stored below the refueling floor when not in use

14. Metal grating at the reactor head cleaning station - not removable; the grating and framework will therefore be designed to remain intact during tornado winds

15. Impact wrenches and other tools - stored where they cannot be subjected to tornado winds; operating procedure will require that they be stored during tornado alerts

The spectrum of potential missiles for which impact energies are tabulated was analyzed by calculating the free-fall velocity in air at the fuel pool surface, neglecting air resistance, then calculating the deceleration effect of a 25 ft depth of water to arrive at the impact velocity and impact energy where the missiles make contact with the fuel racks. In all cases investigated, the energy of impact was less than 840 ft lb.

In summary, the potentially damaging missiles have been eliminated by adequate permanent attachments, automatic clamps, or operational procedures for securing during tornade alerts. The remaining spectrum of potential missiles can be shown to impact against the fuel element assemblies with a kinetic energy low enough to prevent rupture.

H 7.5 Conclusions

The missile criteria are intended to give useful design parameters. The missiles cited are representative of three particular categories: those that become suspended, those that are temporarily suspended, and those that are tumbled along the ground. Since the problem is highly complex, reliable theoretical computations cannot be made; conservative assumptions must be incorporated into the criteria.

Once the design parameters are established, the analysis of the missile impact involves energy transfer and penetration. For the light tornado missiles, the mass of the missile is relatively small, therefore the energy to be dissipated is negligible. Thus, the effect of the missile impact on the structural stability can be neglected except for the automobile.

The designer must use caution when applying available penetration formulae. The general equation for missile penetration contains parameters which are not all valid simultaneously. The penetration analysis of tornado-induced missiles can neglect the effect of hypervelocity. The resulting general penetration equation is very similar to formulae suggested by Rinehart and Amirikian. The modified Petry formula, suggested by Amirikian, is used due to accessibility of the penetration constants.

Two modifications are made to account for the particular missile characteristics: crushing is accounted for in the case of the wood missile, and the large shearing surface is accounted for in the case of the pipe.

By using the suggested missile criteria and the appropriate penetration formula, the structure will be capable of withstanding tornado-induced missiles.

Lack of debris on opposite bank indicates a majority of the debris was not carried over the river (100 yards).

FIGURE H. 7-1
DAMAGE FROM JONESBORO, ARKANSAS, TORNADO
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

Car appears to have been bounced along rather than injected and thrown about.

FIGURE H.7-2
DAMAGE FROM JONESBORO, ARKANSAS, TORNADO
PILGRIM NUCLEAR POWER STATION
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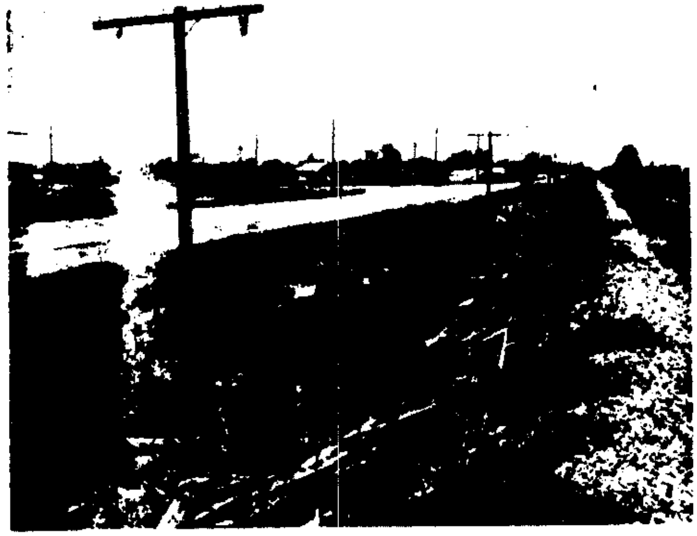

Car did not clear 6-ft railroad embankment. It also appears to have been tumbled. Note debris caught by embankment.

FIGURE H.7-3
DAMAGE FROM JONESBORO, ARKANSAS, TORNADO
PILGRIM NUCLEAR POWER STATION
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TABLE H.7-1

TYPICAL MISSILES AT GRADE LEVEL

	Weight (lb)	Minimum Impact Area (ft^2)	Original Elevation Above Grade (ft)	Maximum Elevation Above Grade (ft)	Impact Energy* (ft lb)
1. Passenger Auto	4,000	30	0	3	-
2. Steel Plate 8 ft x 8 ft x 3/8 in	1,000	1.0	0	Tumbles	-
3. Crated Motor	1,000	16	0	Skids	-
4. Pipe 4 in CI x 12 ft 0 in	240	0.14	0	Tumbles	-
5. Wood Plank 4 in x 12 in x 12 ft	108	0.33	0	50	-
6. Street Light Fixture 1 ft x 4 ft x 6 in	25	0.5	20	300	2
7. Crushed Rock 1 1/2 in	.25	0.01	0	500	1

* Impact energy is listed for missles which can enter the fuel pool, and is the total kinetic energy, after deceleration by falling through 25 ft of water above the spent fuel racks.

TABLE H.7-2

TYPICAL MISSILES AT HIGH LEVEL

	Weight (lb)	Minimum Impact Area (ft^2)	Original Elevation Above Grade (ft)	Maximum Elevation Above Grade (ft)	Impact Energy (ft lb)
1. Microwave Antenna, 10 ft x 15 ft	750	1.0	300	300	840
2. Piece Roof Deck 4 ft x 16 ft	130	1.0	150	450	257
3. Warning Light	10	0.2	300	300	2

TABLE H.7-3

POTENTIAL MISSILES GENERATED BY A TORNADO

	Weight (lb)	Minimum Impact Area (ft^2)	Original Elevation Above Grade (ft)	Maximum Elevation Above Grade (ft)	Impact Energy* (ft lb)
1. Reactor Head Insulation*	4,000	1.0	100	100	736
2. Ventilation Ducts	160	0.333	140	140	300
3. 8 ft Handrail Section	50	0.05	100	100	26

* Insulation will be buoyant temporarily but will lose its buoyancy at some depth below water.

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APPENDIX I

SITE INVESTIGATION OF THE SEABREEZES

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SITE INVESTIGATION OF THE SEABREEZES

I.1 INTRODUCTION

During the summer of 1969 a series of smoke releases were made from the top of the 300 ft msl meteorological tower at the site in conjunction with an intensive sea and air monitoring program. Purpose of the tests was to determine the behavior of an elevated plume discharged into the transition zone from over water to overland air flow and, in particular, to investigate the occurrence of "fumigation" as referred to by Hewson[1] and Van der Hoven.[2]

Specific objectives were to determine whether the point inland at which fumigation occurs can be predicted by Van der Hoven's model by a combination of sea surface temperature measurements and onshore tower data and whether his nomogram (see Figure I.1-1) could be applied to hilly terrain. If this proved to be the case, then a "fumigation climatology" could be compiled with relatively little effort in addition to the routine meteorological observations already being maintained at the site.

This report summarizes the findings of the summer seabreeze program. Conclusions are based on the results of 10 smoke releases over the period July 9 through August 26, 1969. Results of three tests (1,3 & 4) were disregarded because of aircraft equipment malfunction.

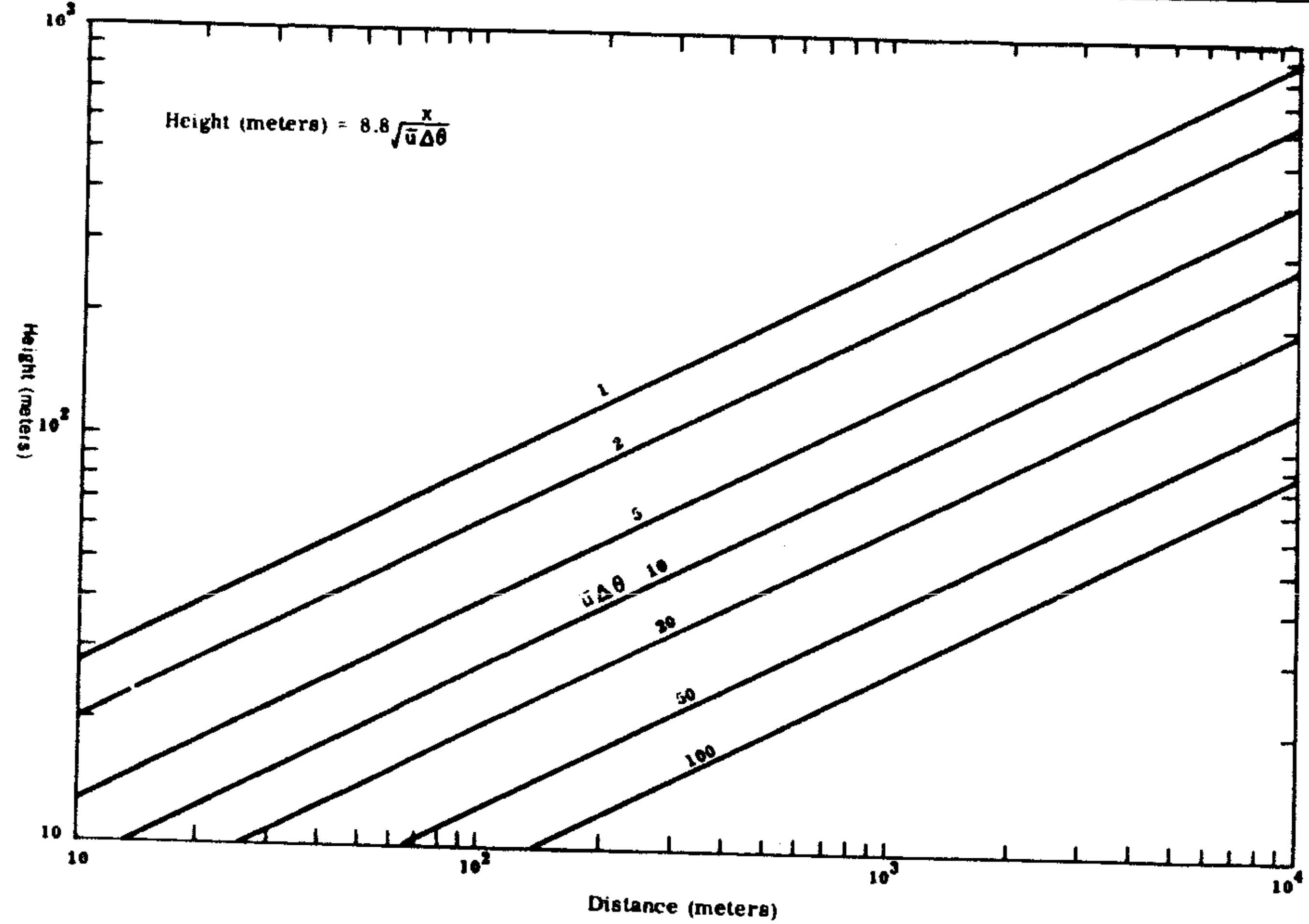

FIGURE I. 1-1
MIXING DEPTH AS FUNCTION OF STABILITY, WINDSPEED, AND INLAND TRAVEL DISTANCE
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

I.2 RESULTS AND CONCLUSIONS

1. A seabreeze occurs at the site only when a weak overall pressure gradient exists - usually when a high pressure area is situated over New England.

2. The seabreeze is most pronounced in spring and early summer when the contrast between sea and land temperatures is the greatest. (i.e., when water temperatures are low and the land surface is warming rapidly.)

3. The onset of a seabreeze is marked by a light northerly wind which gradually veers to east and southeast.

4. Fumigation frequently occurs from a 300 ft release, but the point at which it occurs varies considerably over a short span of time.

5. Calculated locations at which fumigation was expected to occur were in good agreement with Van der Hoven's prediction technique on six occasions and fair on one occasion. Tower data, coupled with sea water temperatures, would have correctly predicted occurence or non-occurrence of fumigation, but would have placed it in the wrong location on one occasion.

6. Predictions made from aircraft soundings are more reliable than tower soundings when the mixing layer has already reached the tower. Tower soundings can be used to predict fumigation inland from that point only if the upper half shows a stable condition. Fumigation occurring between the tower and the shore will not be detected by the tower if its measurements show nonstable conditions.

7. Plumes blowing towards Plymouth (E to ESE wind) frequently touch the ground over Rocky Point, or come down to the surface over Plymouth Harbor. Four factors seem to be involved, sometimes working together, sometimes independently. These are:

    a. Fumigation

    b. "Looping" due to superadiabatic lapse rates in the lower levels

    c. Normal vertical growth downward of the plume

    d. Aerodynamic downwash due to terrain features - a possibility suggested by the photographs

8. During normal seabreeze days, a wind from the NNE towards Manomet Hill was usually transient in nature and persisted hardly long enough for smoke photographs to be obtained. A persistent wind from this direction is usually associated with a northeaster, and "dry" northeasters are a rarity. Run 8 was during such an event and fumigation occurred very close to the tower. In fact, a

visibility restricting plume descended on Rocky Hill Road, less than 1000 ft distant.

9. The general conclusions are:

   a. Van der Hoven's model, modified to fit irregular terrain, appears to provide a reliable tool for predicting the occurrence of fumigation and the point at which the plume will touch the ground

   b. The most frequently "fumigated" area at the site is the section of Rocky Point between the site and Plymouth Bay

   c. The area in which fumigation occurs the closest to the tower is to the SSW towards Manomet Hill

From the above results and conclusions it is our recommendation that short term dosage calculations for the accident analysis should consider the effects of fumigation occurring at or beyond the site boundary. A two meter per second minimum wind speed would appear to be justifiable during periods when a sea breeze prevails. It also seems reasonable to assume a moderately stable lapse (+1.5 to +3.0°C/100 m) as the initial over water condition for the layer of air extending from the surface to the top of the stack.

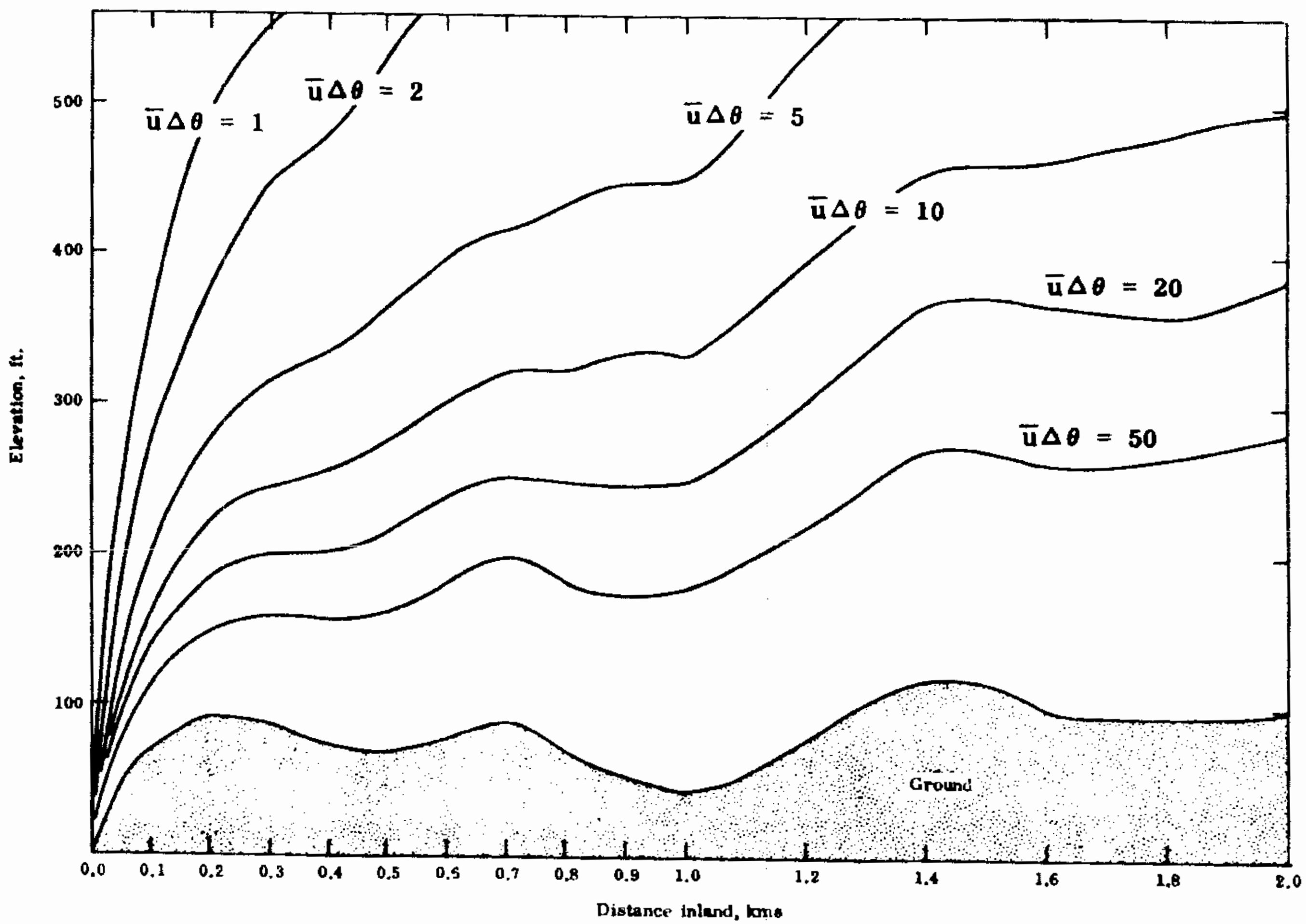

FIGURE I. 2-1
VERTICAL MIXING DEPTHS
NNW WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

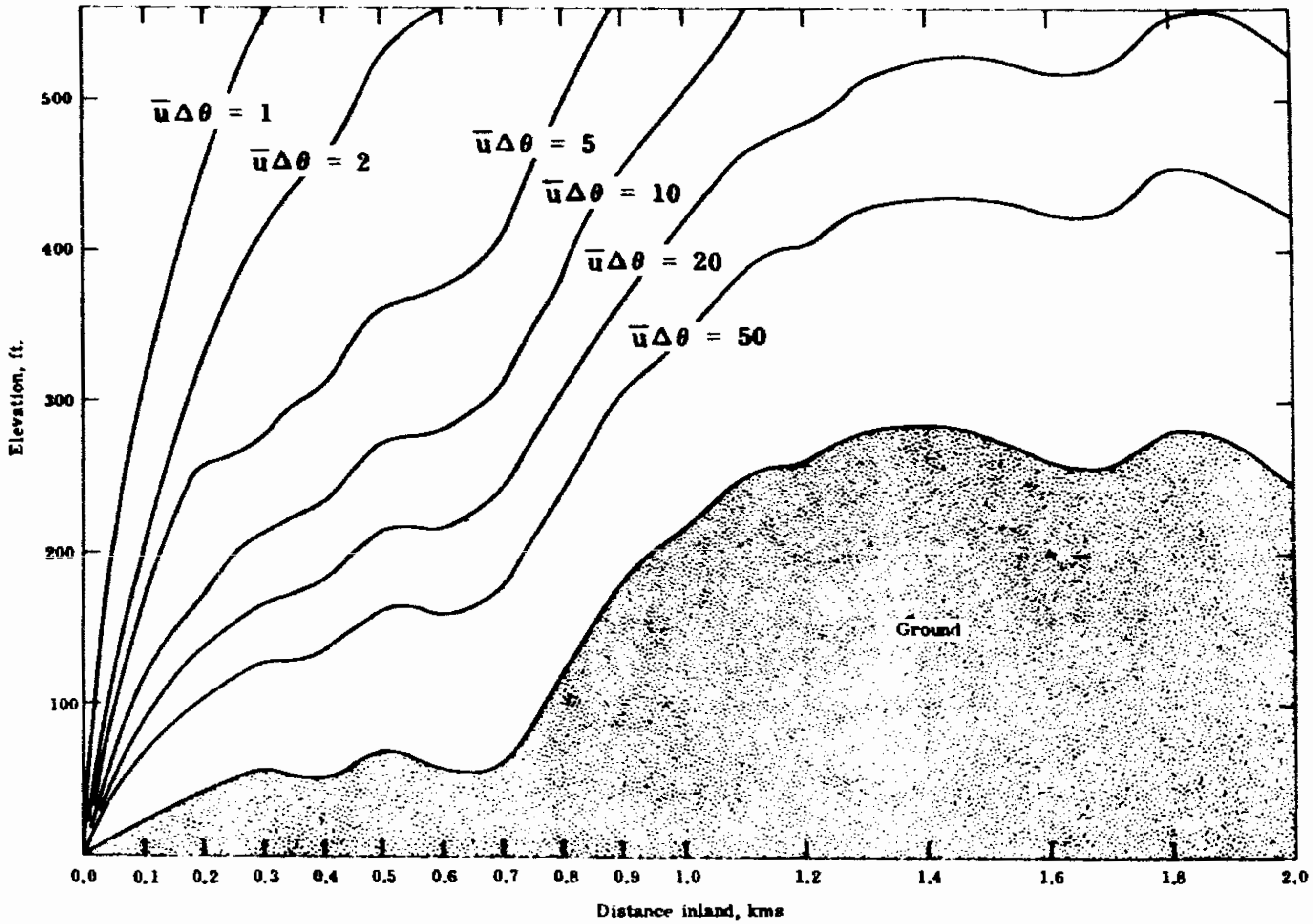

FIGURE I. 2-2
VERTICAL MIXING DEPTHS N WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

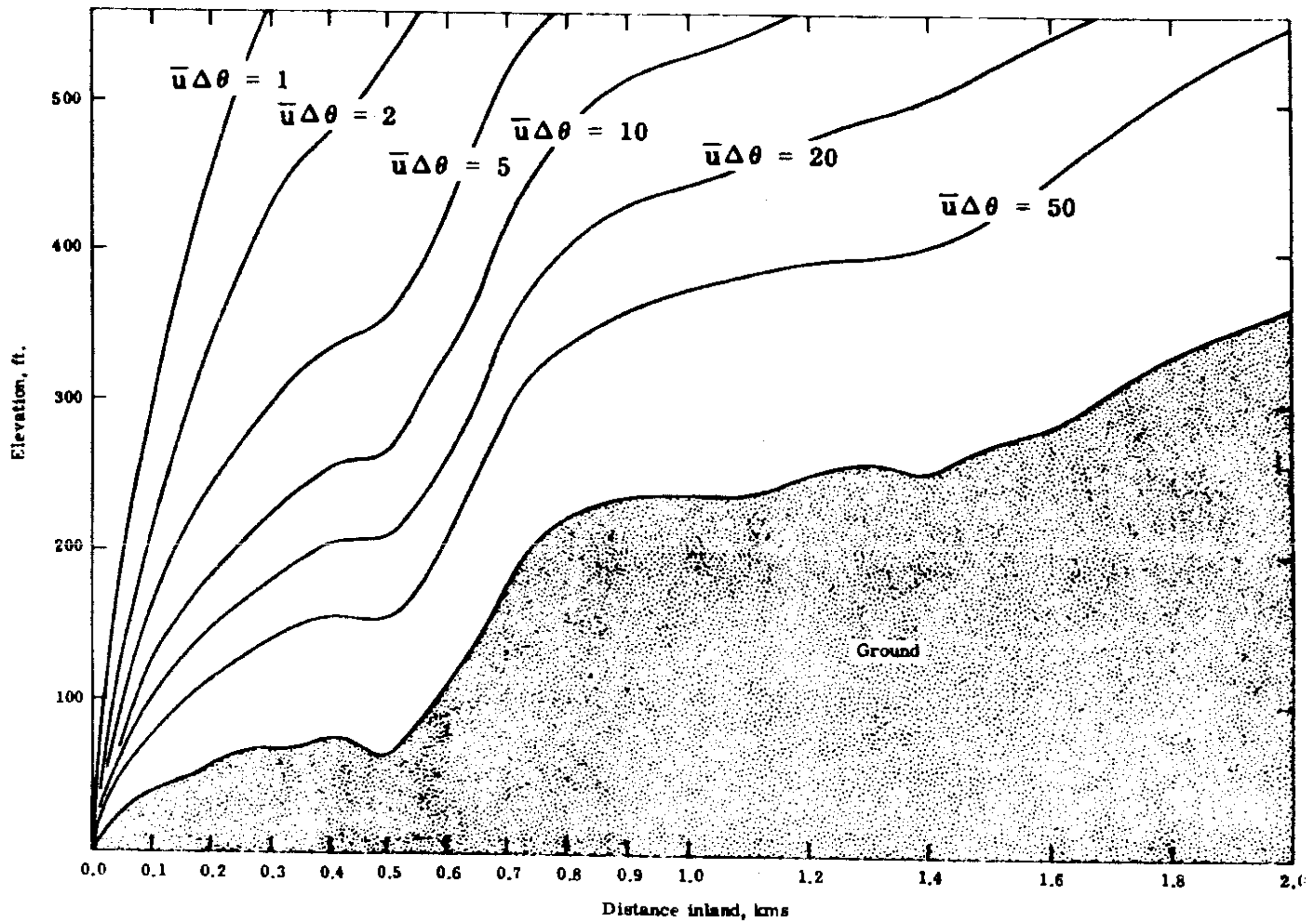

FIGURE I. 2-3
VERTICAL MIXING DEPTHS NNE WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

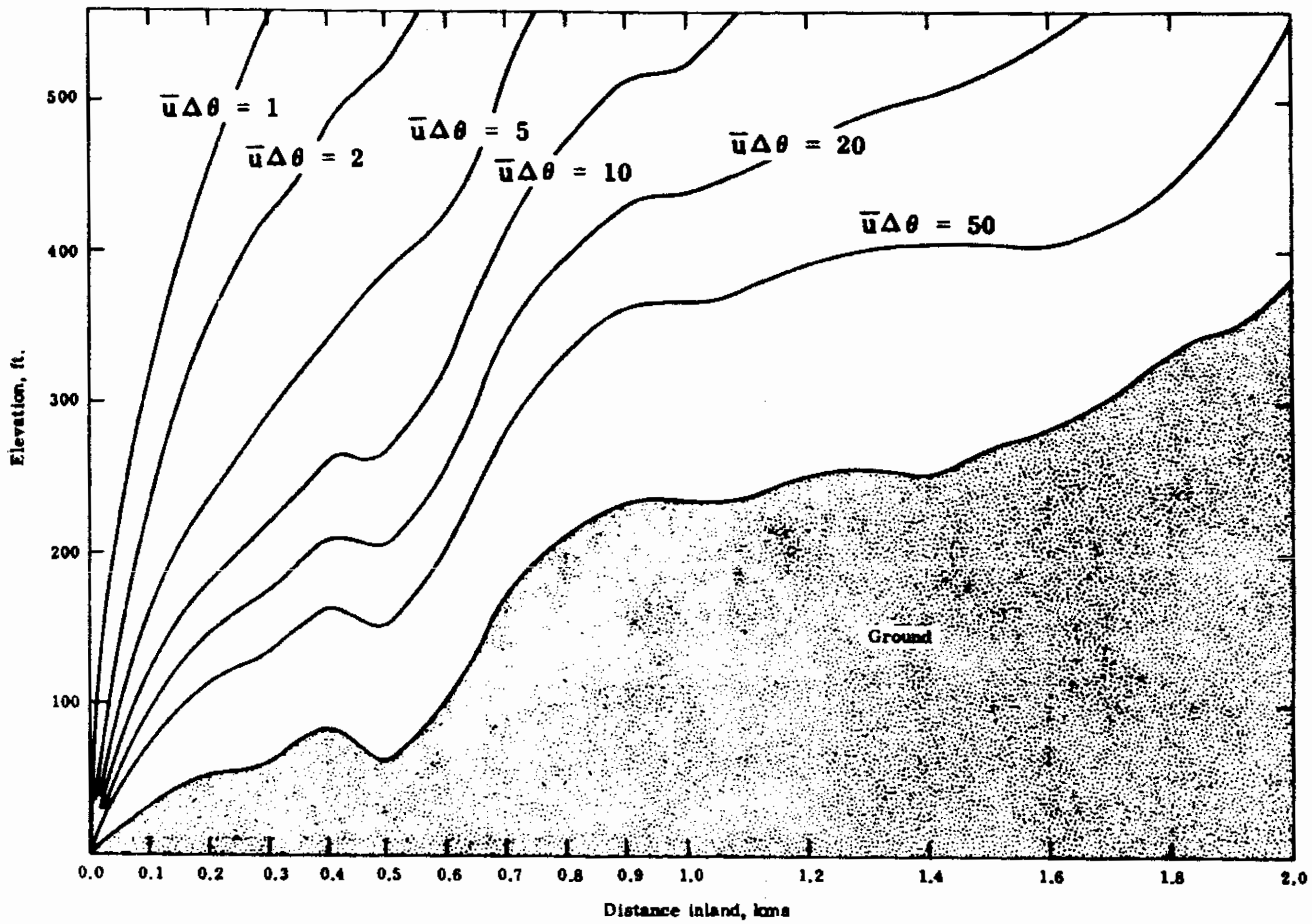

FIGURE I.2-4
VERTICAL MIXING DEPTHS NE WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

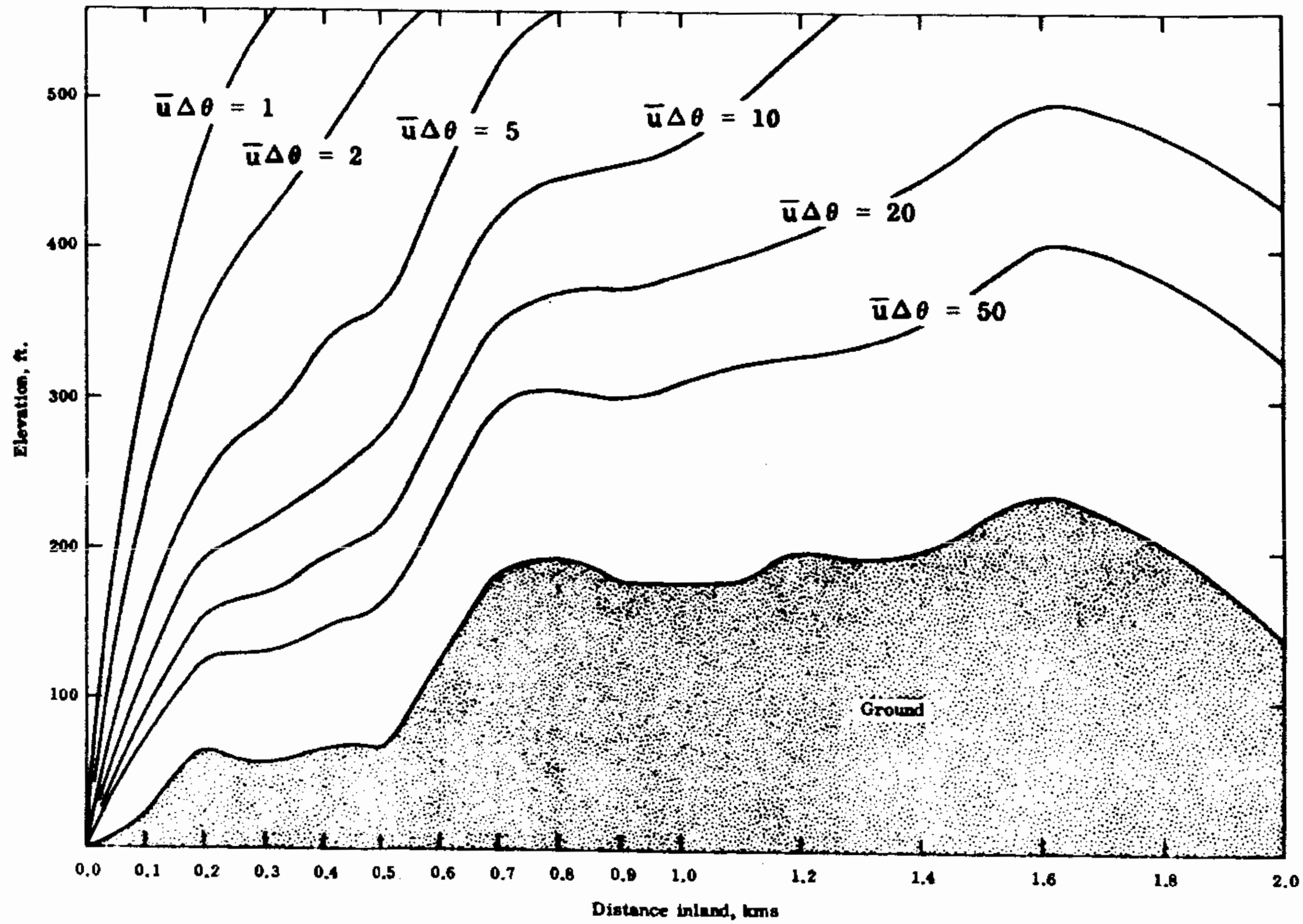

FIGURE I.2-5
VERTICAL MIXING DEPTHS ENE WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

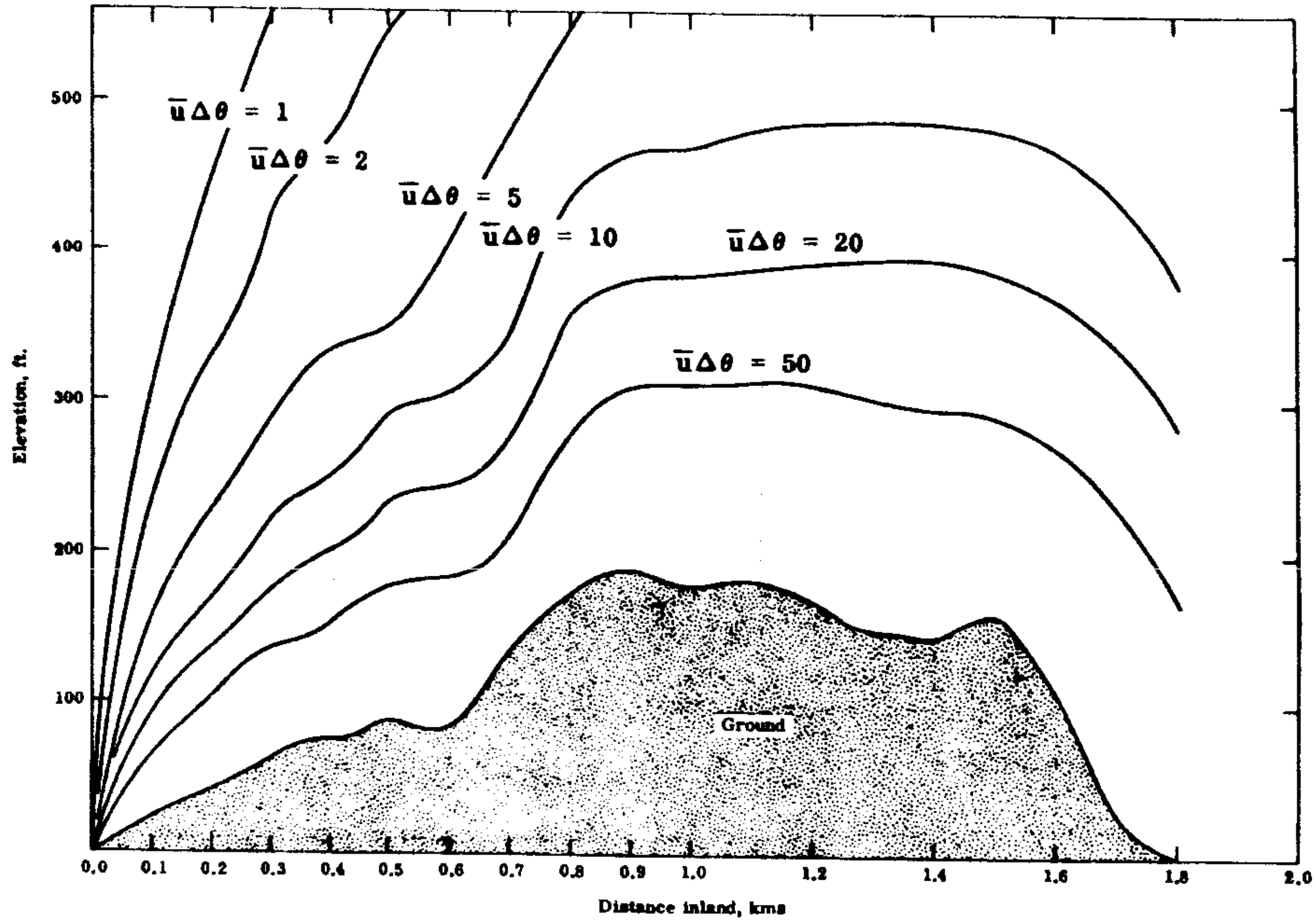

FIGURE I.2-6
VERTICAL MIXING DEPTHS E WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

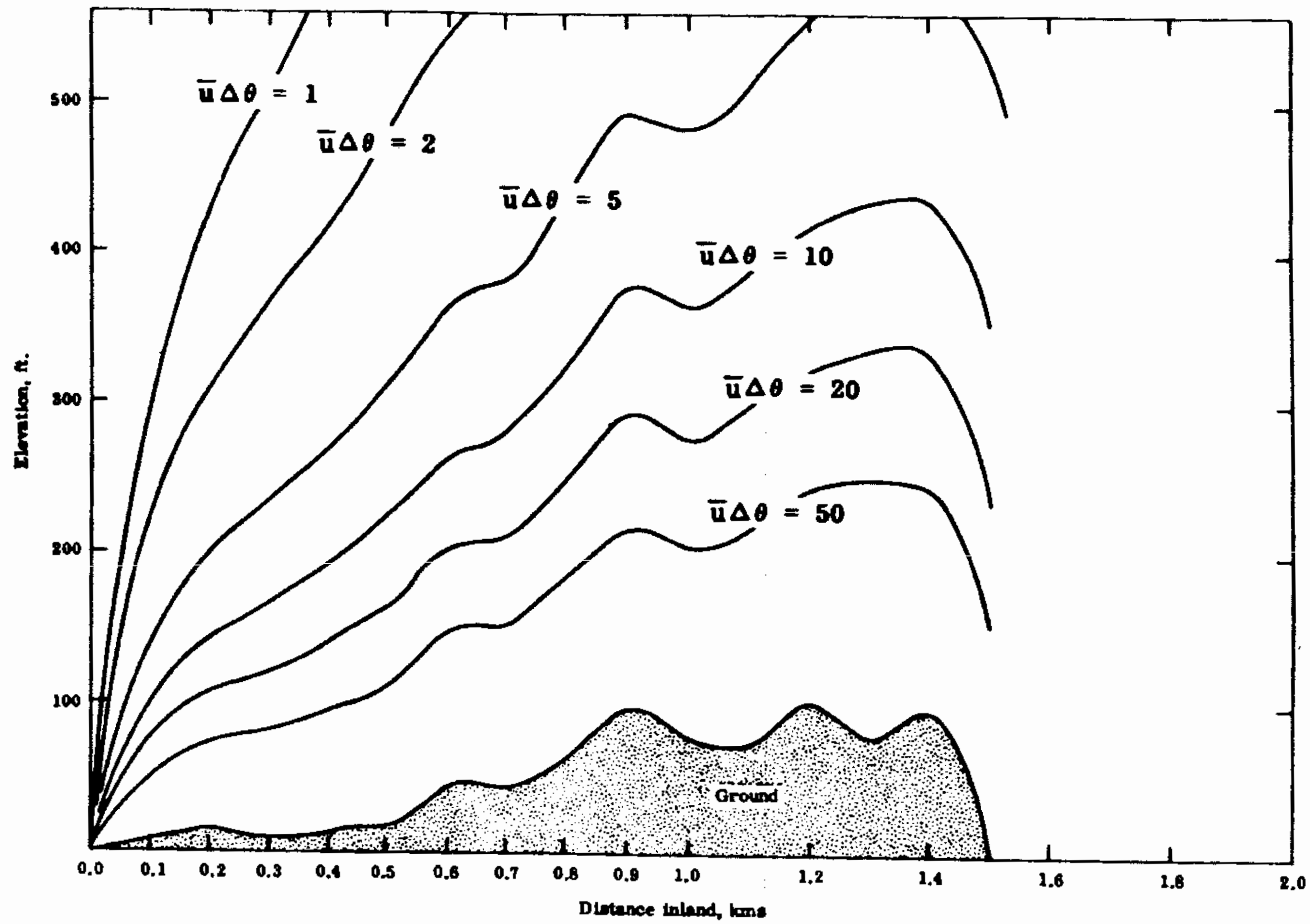

FIGURE I. 2-7
VERTICAL MIXING DEPTHS ESE WIND
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

I.3 DISCUSSION

I.3.1 Theoretical Considerations

When air follows a trajectory over a relatively cold surface (in this case the Atlantic Ocean and Cape Cod Bay), the lower layers of the atmosphere become progressively cooler such that a stable condition develops, often to a depth of several hundred to over a thousand ft. During an onshore wind, such as occurs with a seabreeze, this cool, stable, marine air becomes heated from below and assumes a neutral or superadiabatic lapse rate in the lower levels while retaining stable conditions at higher levels. With increased time and distance from the shoreline, the heated zone, or "mixing level" grows vertically until the last remnant of the stable layer has been "burned off".

If a tall stack located near the shore discharges into the stable layer, the effluent plume will disperse very little and hold together in a steady cone as it moves downwind. At some point downwind the mixing layer will extend upward to the plume level. At this point material in the plume mixes rapidly downward to cause "fumigation". As the mixing layer continues to grow upward, then vertical mixing also takes place above the plume, and gradually "normal" diffusion conditions take over.

In a paper entitled Atmospheric Transport and Diffusion at Coastal Sites by Dr. Isaac Van der Hoven appearing in Nuclear Safety, Volume 8, No. 5, 1967, a nomogram is presented for determining the depth of the mixing layer as a function of initial over water stability, and overland travel distance. With a fixed source elevation it is then possible to determine at what point the plume will intercept the slope of the mixing zone and fumigation will occur.

Van der Hoven expresses the product of wind speed and initial stability as $\bar{u}\Delta\theta$, where $\bar{u}$ = average wind speed over depth of interest and $\Delta\theta$ is the difference in potential temperature between the top and bottom of the initial inversion layer.

Because of the difficulty in obtaining overwater vertical air-temperature profiles, a program was initiated at the site to measure the stability indirectly from the 220 ft meteorological tower (300 ft msl) temperatures and sea surface temperatures. The assumption was made that if the tower was close enough to the shore and had sufficient elevation to remain in the stable layer during the invasion of a deep layer of marine air, initial overwater stability could be determined. This is done by taking the air temperature at the top of the tower, and increasing it adiabatically by lowering the air parcel to sea level. $\Delta\Theta$ then becomes the 300 ft tower temperature plus 0.91°C minus the sea surface or sea air interface temperature. $\bar{u}$ is taken as the mean wind speed as observed at 150 ft msl on the tower.

Van der Hoven's nomogram (Figure I.1-1) applies to level terrain. The technique was modified for irregular terrain by assuming

"streamlines" followed the topography. Elevation profiles for various onshore directions were thus added to provide the working nomograms as shown on Figures I.2-1 through I.2-7. In reality, the wind streamlines would tend to approach closer to the ground as air passed over a ridge. At the same time the rate of heating on a slope would very likely be greater than on a level plain. Hence, the two factors would tend to nullify each other, and making the mixing depth isopleths follow the terrain contours seems to work in practice.

As the results show, Van der Hoven's model provides acceptable predictions provided the initial overwater stability is accurate. Tower data can be used with confidence if it is certain the top temperature measurement is representative of initial overwater conditions. Unfortunately, tower data could not always be used because of the rapid upward growth of the mixing level once the air passed over the warm land surface. Minimum distance between the shore and the tower is 0.17 kms. If the fumigation distance is equal to or less than 0.17 kms the tower would not detect it. This is not serious, however, if the stack is at least 0.17 kms inland, because beyond that point good vertical mixing takes place, both below and above the stack.

I.3.2 Field Program

Routine meteorological observations of wind at 150 and 300 ft msl and temperatures at 85, 295, and 300 ft msl have been maintained from a tower at the site for over a yr. See Section 2.3. During the summer months of 1968 and 1969 sea surface temperature measurements were taken on a semi-routine basis by a variety of methods; hand held thermistor probes, moored buoys - both automated and manually serviced, and by airborne infra-red radiometers.

Prior to the smoke-release program of 1969 an IR survey of the entire Cape Cod Bay was conducted May 1, 1969 to determine the optimum location for a surface water temperature sensing buoy so as to avoid anomalies that might be created by local currents. A spot about 0.25 mi offshore was selected as being satisfactory. The buoy holds two self contained thermographs near the water surface. As an interim measure, once-daily observations were made from a boat at approximately 1400 hr local time by means of a thermistor probe and sounding line.

On days smoke was released, additional water surface temperature measurements were made from an aircraft with an IR radiometer to get average values over Cape Cod Bay. IR measurements and water temperatures measured from boat or buoy agreed within 0.5°C.

Vertical temperature soundings from 30 ft above the water up to 1500 ft were made with a wing mounted bead thermistor. Pressure-altitude was determined from a strain gauge pressure sensor. Readings from all of these sensors were recorded by an airborne multichannel recorder.

Smoke was generated from a smoke generator mounted on a small platform near the top of the meteorological tower. Oil and water were pumped to the generator from tanks at the bottom of the tower. Oil used was U.S. Navy specification SGF1.

The aircraft also served as the photographic base. Plume behavior was photographically documented when the aircraft was flying parallel to, and at essentially the same altitude as the plume.

I.4 DATA AND CALCULATIONS

To determine the point at which the mixing layer theoretically intercepted the elevated plume, Van der Hoven's corrected nomogram, Figure I.1-1, (the one in Nuclear Safety was drawn incorrectly) was adjusted by preparing terrain profiles from the USGS Manomet quadrangle and adding the ground elevation to the depth of the mixing level every 100 m downwind. These terrain-adjusted profiles are shown for various plume trajectories on Figures I.2-1 through I.2-7.

Presented here are the significant data, $u\Delta\theta$, point of interception of the 300 ft plume with the mixing level, etc., for each smoke release. Actual temperature profiles for each test are shown on Figures I.4-1 through I.4-7. Dashed lines are adiabatic lapse rates. A summary of the smoke tests is shown on Table I.1-1.

TEST RESULTS

Test No.	1	2
Date	July 9, 1969	July 15, 1969
Wind Direction	SE	ESE
Wind Speed ($\bar{u}$)	3.6 $msec^{-1}$	3.2 $msec^{-1}$
Water Temperature	17.1°C	18.0°C
$\Delta\theta$ (Aircraft)	Not Available	5.9
$\Delta\theta$ (Tower)	Not Applicable	4.6
$\bar{u}\Delta\theta$	Test Unsatisfactory	16.8
How Obtained		Average Aircraft and Tower
*Predicted Point of Interception		0.74 km
Observed Point of Interception	"Looping" Plume	0.71 km
How Obtained	Aerial Photo	Aerial Photo

REMARKS:

Test 1: Superadiabatic lapse rate was observed on the tower, which meant that plume was already in the mixing layer. This was verified by photographs. "Looping" plume descended to surface on Plymouth Bay about 1.4 kms from source. Tower could not be used as a predictor and aircraft soundings were not available.

*Distance from shore, not point of smoke release

Test 2: Excellent agreement between observed and predicted. Aircraft sounding showed stable over water sounding. Upper half of tower showed neutral condition but essentially same temperature at 300 ft.

TEST RESULTS

Test No.	3	4
Date	July 23, 1969	July 23, 1969
Wind Direction	E	SE
Wind Speed	3.0 $msec^{-1}$	4.0 $msec^{-1}$

REMARKS:

Aircraft out of servive. Instrumented aircraft not available. Both runs unsatisfactory.

TEST RESULTS

Test No.	5	6
Date	July 24, 1969	July 31, 1969
Wind Direction	NE	ESE
Wind Speed	9.0 $msec^{-1}$	3.1 $msec^{-1}$
Water Temperature	18.3°C	17.6°C
$\Delta\theta$ (Aircraft)	0.3	6.9
$\underline{\Delta}\theta$ (Tower)	0	4.5
$\bar{u}\Delta\theta$	2.7	12.2-18.6
How Obtained	Aircraft	Tower - Aircraft
Predicted Point of Interception	None (Adiabatic)	0.88 - 1.2 km
Observed Point of Interception	See Remarks	Vbl 0.6 - 1.0 km
How Obtained	Aerial Observer	Aerial Photos and Observer

REMARKS:

Test 5: Aircraft sounding showed nearly neutral condition, tower showed superadiabatic condition. Tower is 0.26 km from shore, hence plume was enveloped in mixing layer immediately and vertical mixing upward and downward occurred. Lower edge of cloud hit hill beyond road to SW, but this was not "fumigation."

Test 6: During stable portion of plume it tended to follow contours of terrain. Breakup thereafter seemed to be as much a function of aerodynamic as thermal turbulence. Photography limited by high clouds. Tower again showed superadiabatic conditions.

TEST RESULTS

Test No.	7	8
Date	August 7, 1969	August 12, 1969
Wind Direction	ESE	NNE
Wind Speed	2.5 $msec^{-1}$	1.8 $msec^{-1}$
Water Temperature	19.3°C	19.1°C
$\Delta\theta$ (Aircraft)	4.0	1.8
$\Delta\theta$ (Tower)	6.0	5.3
$\bar{u}\Delta\theta$	12.5	3.2
How Obtained	Average	Aircraft
Predicted Point of Interception	0.79 km	0.31km
Observed Point of Interception	0.87 km (Est)	Vbl 0.32 - 0.37 km
How Obtained	Aerial Photos	Aerial and Ground Observers

REMARKS:

Test 7: Plume followed contours of hill. Exact point of interception with mixing layer hard to determine from photograph. Observer noted smoke on ground at edge of Plymouth Bay, occasionally on shore.

Test 8: Very persistent wind from NNE after initially veering from NNW. Plume descended to ground close to tower and restricted visibility on Rocky Hill Road. Aircraft showed stable conditions to at least 800 ft. Tower showed neutral in upper half, unstable below.

TEST RESULTS

Test No.	9	10
Date	August 13, 1969	August 26, 1969
Wind Direction	ESE	NNE
Wind Speed	3.3 $msec^{-1}$	4.9 $msec^{-1}$
Water Temperature	19.5°C	19.5°C
$\Delta\theta$ (Aircraft)	4.1	0
$\Delta\theta$ (Tower)	5.4	0
$\bar{u}\Delta\theta$	15.7	0
How Obtained	Average	Average
Predicted Point of Interception	0.90	See Remarks
Observed Point of Interception	1.0 km	Not Applicable
How Obtained	Aerial Photo and Observer	

REMARKS:

Test 9: Stable plume was again observed to follow contours of terrain until it intercepted mixing level. Considerable vertical mixing over, and to leeward of, small ridge on Rocky Point.

Test 10: Although an onshore wind prevailed during this test the sounding was essentially adiabatic from the surface to over 1000 ft elevation. Mixing occurred both vertically upward and vertically downward at the same rate and fumigation did not occur.

TABLE I.4-1

SUMMARY OF SMOKE TESTS

Test No	Wind	Fumigation Observed	Pt. of Interception with Mixing Layer: Observed	Pt. of Interception with Mixing Layer: Predicted	Prediction Rating	Tower Data Reliable As Predictors
1	SE	No	Insufficient data because of equipment malfunction			
2	ESE	Yes	0.71 km	0.74	Good	Yes
3	E	Yes	Insufficient data because of equipment malfunction			
4	SE		Insufficient data because of equipment malfunction			
5	NE	No	Would have predicted no fumigation		Good	Yes
6	ESE	Yes	0.6-1.0	0.9-1.2	Fair	Yes
7	ESE	Yes	0.87	0.79	Good	Yes
8	NNE	Yes	0.32	0.31	Good	No*
9	ESE	Yes	0.90	0.85	Good	Yes
10	NE	No	Would have predicted no fumigation		Good	Yes

Prediction Rating: Good - Within 10% of Observed
Fair - Within 20% of Observed

* Tower data would have predicted fumigation, but too far from the source.

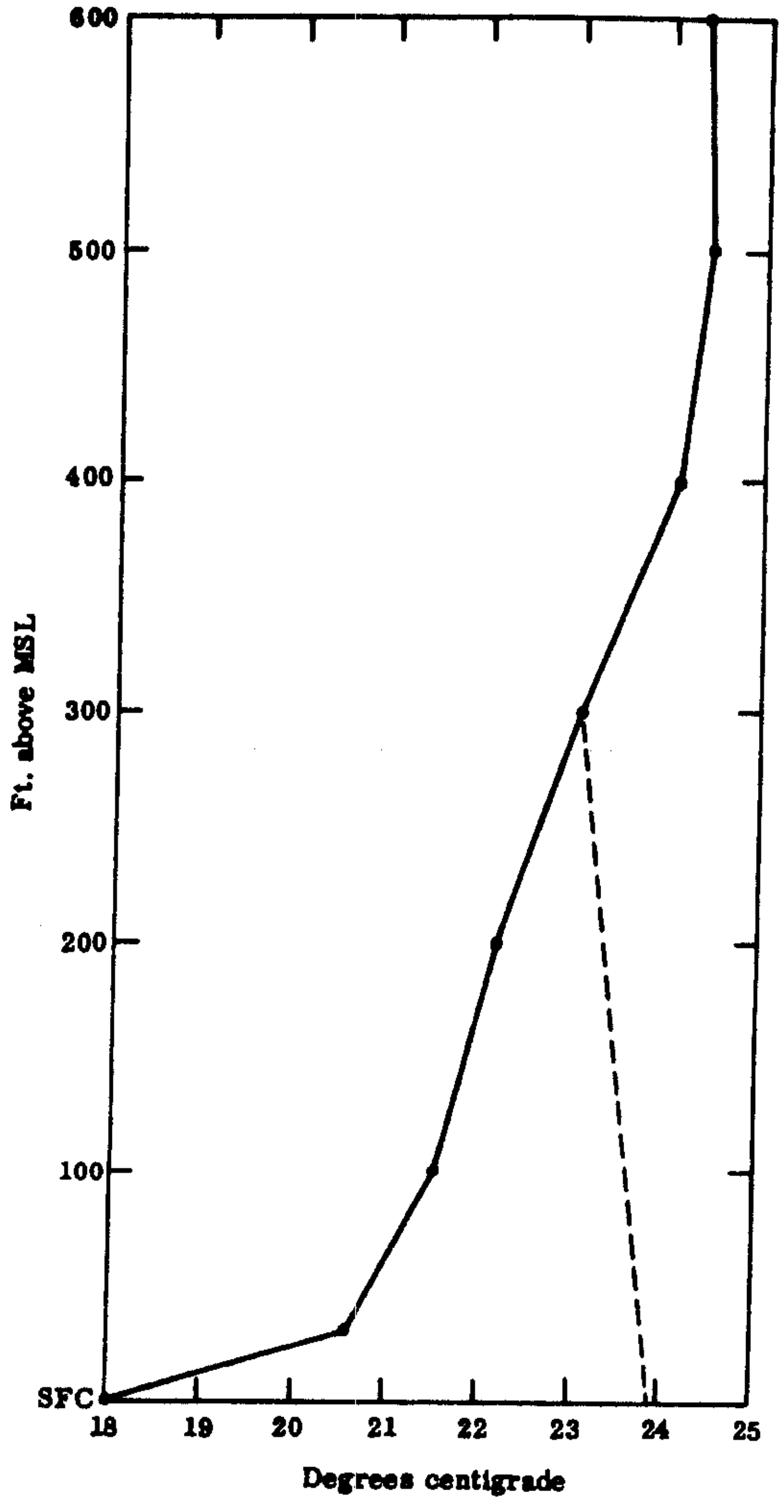

FIGURE I 4-1
TEMPERATURE PROFILE TEST NO. 2
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

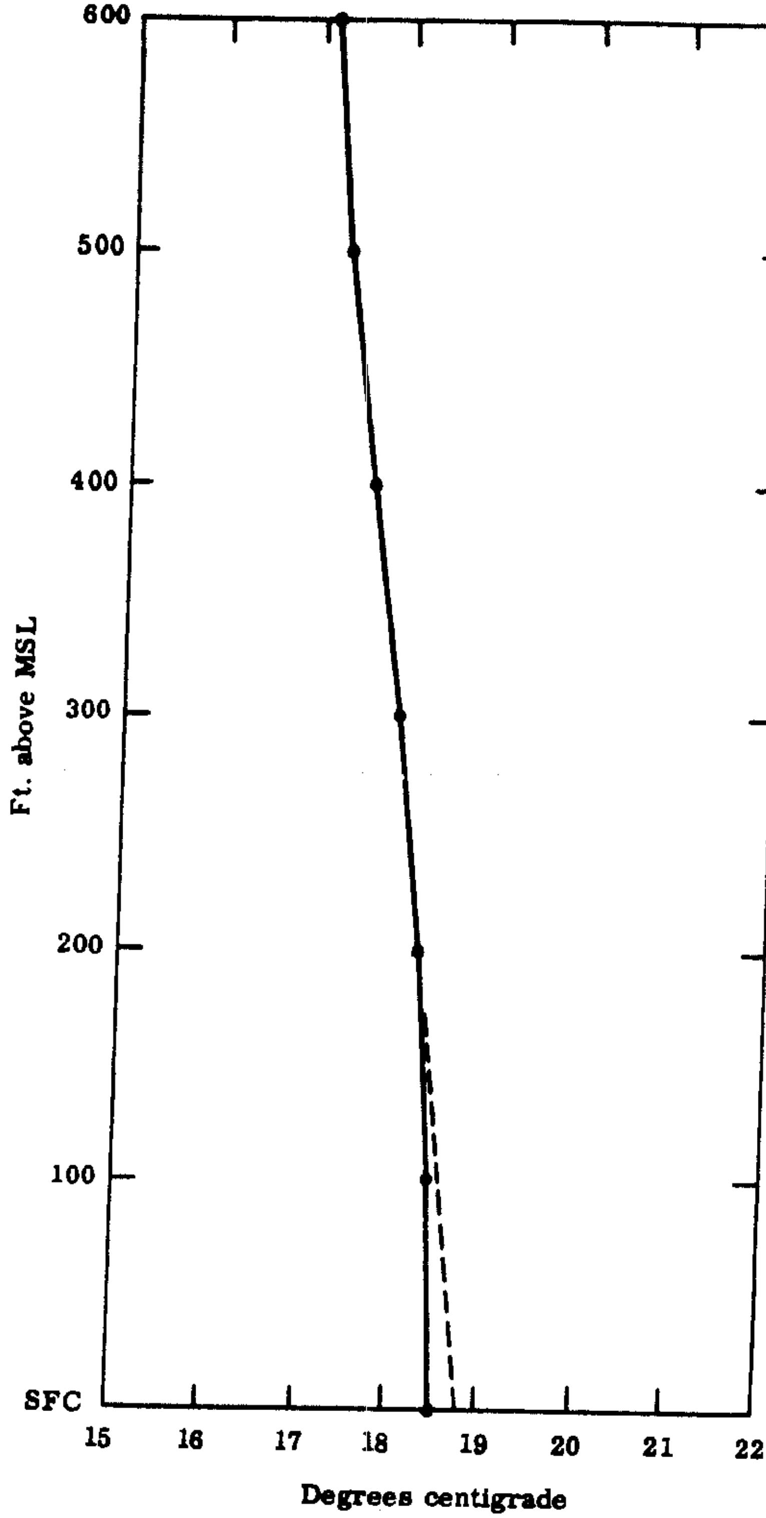

FIGURE I 4-2
TEMPERATURE PROFILE TEST NO.5
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

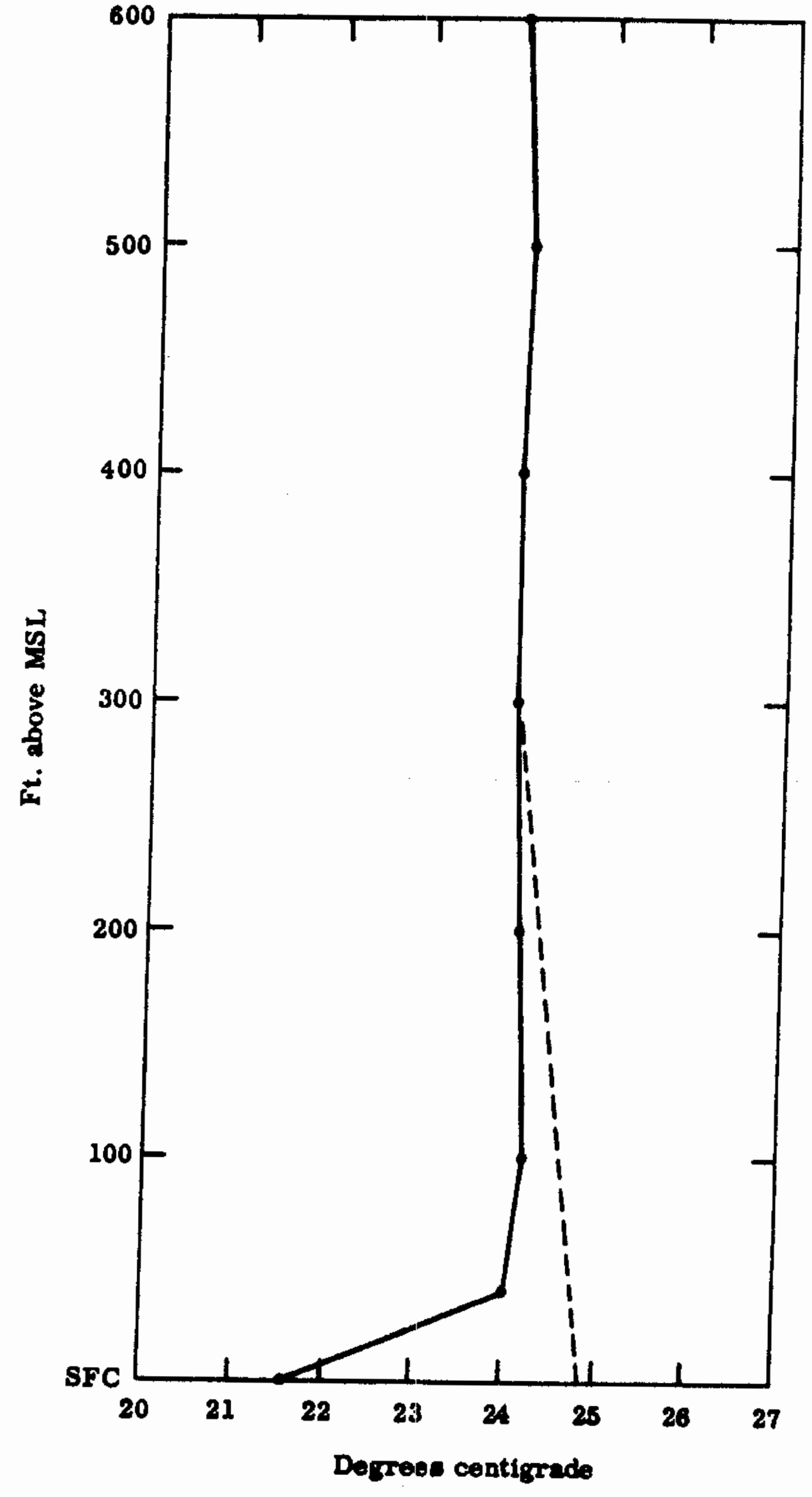

FIGURE I. 4-3
TEMPERATURE PROFILE TEST NO. 6
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

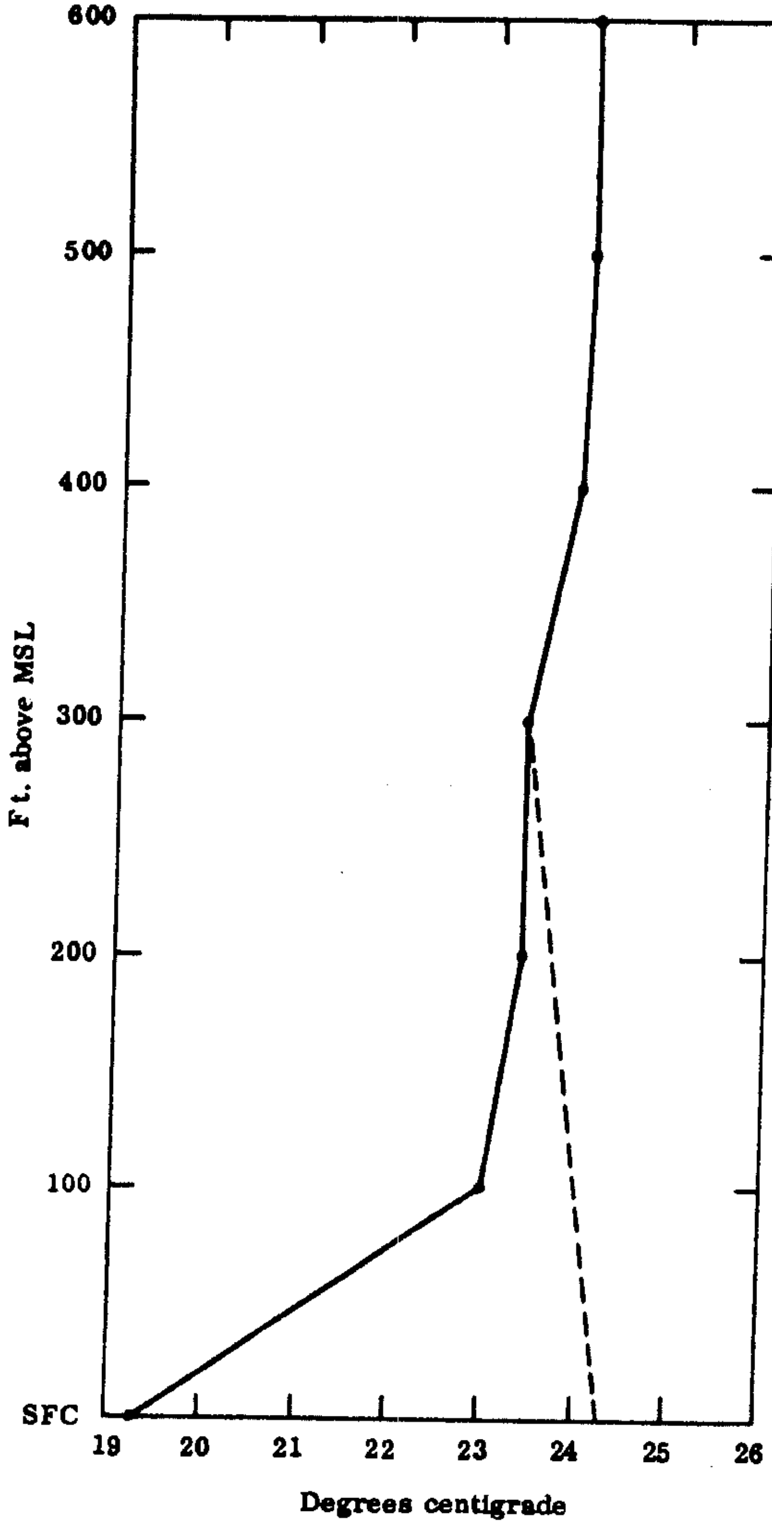

FIGURE I.4-4
TEMPERATURE PROFILE TEST NO. 7
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

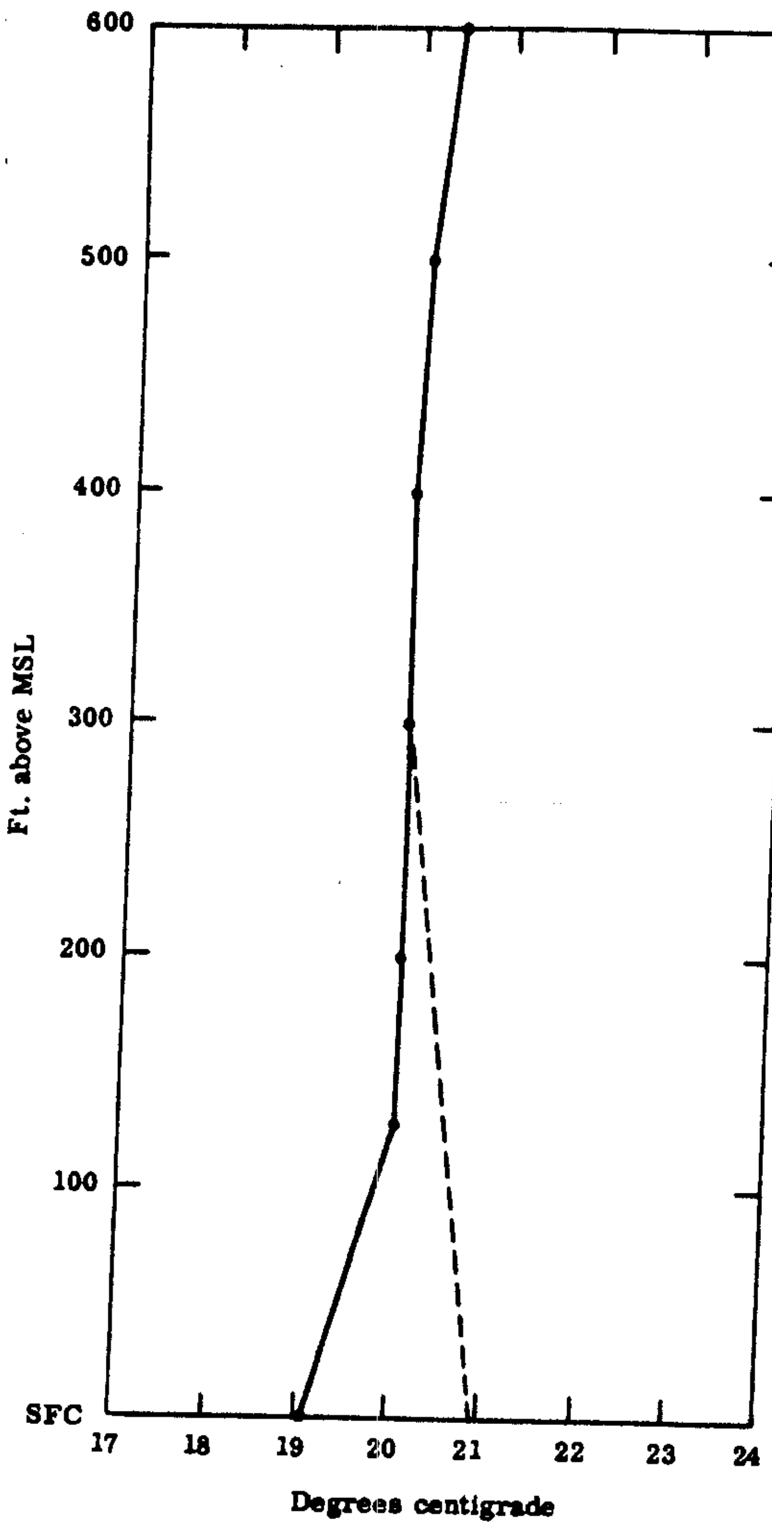

FIGURE I.4-5
TEMPERATURE PROFILE TEST NO. 8
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

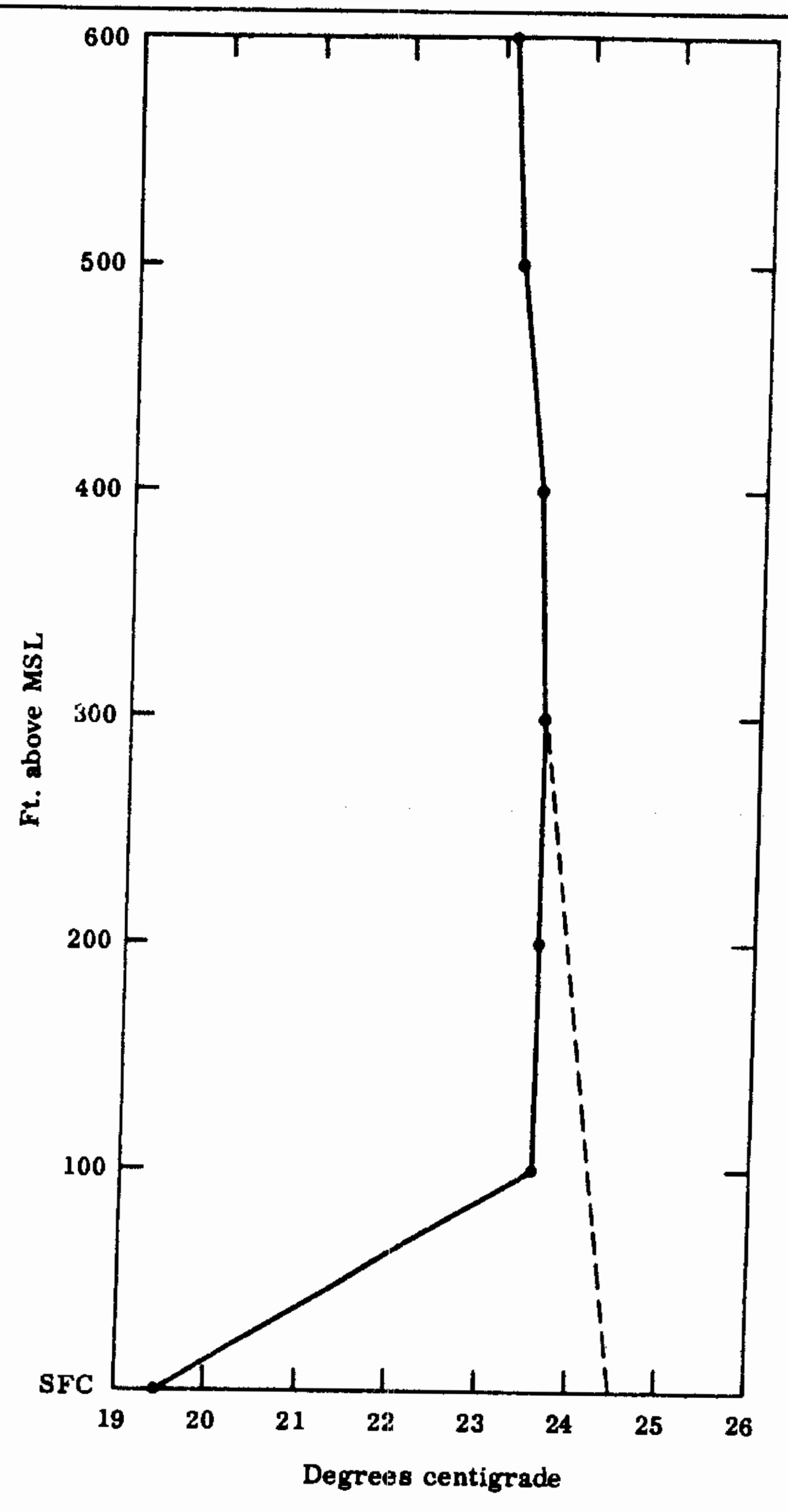

FIGURE I.4-6
TEMPERATURE PROFILE TEST NO. 9
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

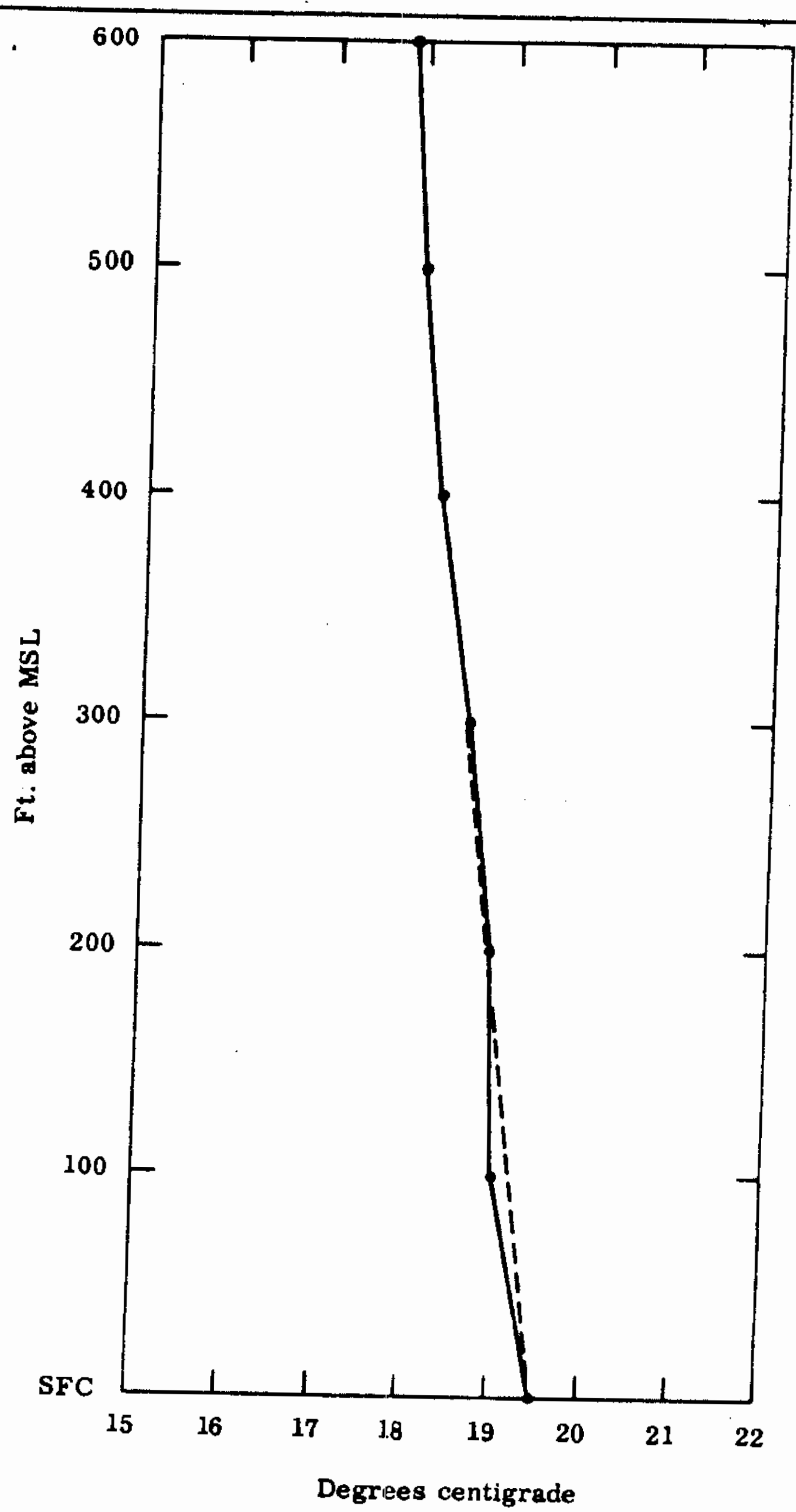

FIGURE I. 4-7
TEMPERATURE PROFILE TEST NO. 10
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

I.5 REFERENCES

1. Hewson, Meteorology & Atomic Energy. P. 105-107, 1968.

2. Van derHoven, I. Nuclear Safety, Vol. 8, No. 5, p. 490-499, Sept-Oct 1967.

APPENDIX J

STATION RESEARCH, DEVELOPMENT, AND FURTHER INFORMATION REQUIREMENTS AND RESOLUTIONS

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APPENDIX J

STATION RESEARCH, DEVELOPMENT, AND FURTHER INFORMATION REQUIREMENTS AND RESOLUTIONS

J.1 RESOLUTION OF ACRS CONCERNS

The areas specified in the ACRS Construction Permit Letter, Construction Permit Safety Evaluation Report and other recent, related ACRS Construction Operating Permit letters (up to October 1969) have been formally answered by the resolutions set forth in Appendix J.

Appendix J remains intact in the updated FSAR in order that NRC may locate previously submitted information in one document. This requirement to maintain a historical record meets the intent of the FSAR Update Rule (10CFR50.71(e)) in NRC letter of December 15, 1980.

J.1.1 Summary Description

The design of the General Electric boiling water reactor (BWR) for this station is based upon proven technological concepts developed during the development, design, and operation of numerous similar reactors. The AEC Staff and the Advisory Committee for Reactor Safeguards (ACRS) have, in their review of reactor projects, identified several technical areas for which further detailed support information should be obtained. All of these development efforts are of three general types: (a) those which pertain to the broad category of water cooled reactors, (b) those which pertain specifically to BWRs, and (c) those which have been noted particularly for this facility during the construction permit licensing activities by the AEC Staff and ACRS reviews.

The following discussion is a complete, comprehensive examination of these concern areas and the Pilgrim Nuclear Power Station Construction Permit concerns, indicating the planned or accomplished resolution:

1. Areas Specified in the Pilgrim ACRS Construction Permit Letters.

2. Areas Specified in the Pilgrim AEC Staff Construction Permit Safety Evaluation Report.

3. Areas Specified in Other Recent Related ACRS Construction and Operating Permit Letters.

The scope of many of the areas of technology for items in 1, 2, and 3 above is discussed in detail as part of an official response[1] by the General Electric Company to the various ACRS concern subjects. General Electric has submitted many Topical Reports to the AEC in support of this application and those of other GE-BWR facilities. Refer to Tables 1.11-2 to 1.11-4.

The following significant design changes in Class I components were incorporated in the PNPS to reflect design changes incorporated on other, more recent, BWR plants:

1. Incorporated an auto-blowdown interlock based on CSCS pump discharge pressure rather than availability and simplified the ADS logic. See Section 7.4

2. Incorporated low drywell pressure permissive for containment spray. See Sections 5.2 and 7.3

3. Improved testing capability of CSCS circuits. See Section 7.4

4. Designed Reactor Protection System (RPS), Primary Containment Isolation System (PCIS), and initiation of CSCS to conform with the intent of IEEE-279 Standard. See Sections 7.2, 7.3, and 7.4

5. Deleted baffles in suppression pool

The following significant design changes in Class I components were incorporated as a result of normal design development and engineering refinement:

1. Incorporated line filling connections on Core Spray and Residual Heat Removal Systems. See Figures 7.4-8 and 7.4-10

2. Incorporated additional refueling ventilation exhaust radiation monitors for improved testing capability. See Section 7.12

3. Incorporated low RPV pressure permissive in series with low RPV level for CSCS initiation to permit continued operation of drywell coolers in the event of loss of offsite ac power without loss of coolant accident (LOCA). See Figures 7.4-9 and 7.4-11 through 7.4-13

4. Incorporated four constituent enrichments in fuel assembly design to reduce power peaking and increase ultimate core output capability. See Section 3.7

5. Incorporated manual isolation of nonessential cooling loads on the RBCCW System to improve reliability of system during normal operation. See Figure 10.5-1

6. Incorporated small bypasses around testable check valves in CSCS to improve testing capability. See Figures 4.7-1, 4.8-3, 7.4-1, and 7.4-8

7. Incorporated capability to individually synchronize the onsite diesel generators to the unit auxiliary transformer to improve testing capability of the diesel generator. See Section 8.5

8. Increased the height of the main stack to increase allowable stack release limits. See Section 12.2

The operating license was requested at 1,998 MWt corresponding to the highest reactor core power level considered in the evaluation of CSCS performance.

J.2 AREAS SPECIFIED IN THE ACRS CONSTRUCTION PERMIT LETTER FOR PILGRIM NUCLEAR POWER STATION

J.2.1 Introduction

"At its ninety-sixth meeting, on April 4-6, 1968, the Advisory Committee on Reactor Safeguards reviewed the application by the Boston Edison Company for authorization to construct its Pilgrim Nuclear Power Station. An ACRS Subcommittee had previously reviewed the project and visited the site during a meeting with the applicant in Boston, Massachusetts, on March 26-27, 1968. During a review, the Committee had the benefit of discussions with representatives and consultants of Boston Edison Company, General Electric Company, Bechtel Corporation, and the AEC Regulatory Staff."[2]

The ACRS areas of concern as reported in their letter of April 12, 1968 and the Applicant's resolution of these areas are described in the following sections.

J.2.2 Station Wave Runup and Water Rise Studies

Concern

"The applicant is continuing his studies of water rise and runup during severe coastal storms. The design of the structures is stated to be sufficiently flexible to perform adjustment for an unexpectedly high calculated flood level."[2]

Resolution

Maximum design still water levels for Pilgrim Station of 17.6 ft mlw has been established from analysis based on ESSA Hurricane Report 7-97.

A series of hydraulic model studies of wave action have been performed to demonstrate the adequacy of the site protection provided by the offshore breakwaters and the revetment. It was concluded from the test results that no reactor building flooding would occur. Refer to Section 2 for further details.

J.2.3 Problem Areas Pertaining to Large, Water Cooled, Power Reactors

Concern

"The Committee has, in the past, called attention to several problem areas pertaining to large, water cooled, power reactors, - these apply also to the Pilgrim plant."[2]

Resolution

Areas of concern, which the Applicant believes may be applicable to Pilgrim Station, that were reported in the ACRS construction permit

letters for the Vermont Yankee, Browns Ferry, Peach Bottom, and Diablo Canyon facilities are discussed in the following Sections.

J.2.3.1 Effects of Fuel Failure on Core Standby Cooling System Performance

Concern

"Analysis indicates that a large fraction of the reactor fuel elements may be expected to fail in certain loss-of-coolant accidents. The applicant states that the principal mode of failure is expected to be by localized perforation of the clad, and that damage within the fuel assembly of such nature or extent as to interfere with heat removal sufficiently to cause clad melting would not occur. The Committee believes that additional evidence, both analytical and experimental, is needed and should be obtained to demonstrate that this model is adequately conservative for this power density and fuel burnup proposed."[3]

Resolution

General Electric Company has conducted both analytical studies and experimental tests to develop information on the effect of fuel rod failure caused by deformation of the Zircaloy cladding following a postulated Loss of Coolant Accident (LOCA).[4] Other tests conducted by the nuclear industry and the national laboratories confirm these findings.

The overall results indicate that the maximum expected distortion as a result of bowing, ballooning, and perforation of the cladding would not significantly change the ability of the BWR Emergency Core Cooling System to accomplish its design objective. Furthermore, it is concluded that the BWR CSCS can withstand significant degradation before any gross core damage could result.

J.2.3.2 Effects of Fuel Bundle Flow Blockage

Concern

"The applicant considers the possibility of melting and subsequent disintegration of a portion of a fuel assembly, by inlet coolant orifice blockage or by other means, to be remote. However, the resulting effects in terms of fission product release, local high pressure production, and possible initiation of failure in adjacent fuel elements are not well known. Information should be developed to show that such an incident will not lead to unacceptable conditions."[3]

Resolution

The effects of a fuel bundle flow blockage in a BWR have been investigated.[5] The consequences in terms of fuel damage, fission product release, local high pressure production, and possible propagation to adjacent assemblies have been evaluated. The

conclusions reached are that a flow blockage incident will not result in local high pressure production or propagate to adjacent assemblies even at full reactor power conditions. Fuel cladding failure will only occur for blockages greater than 90 percent even for the most severe combination of events, namely a complete flow blockage at full power. The Reactor Protection System (RPS) offers adequate protection in preventing cladding or fuel melting.

J.2.3.3 Verification of Fuel Damage Limit Criterion

Concern

"A linear heat generation rate of 28 kW/ft is used by the applicant as a fuel element damage limit. Experimental verification of this criterion is incomplete, and the applicant plans to conduct additional tests. The Committee recommends that such tests include heat generation rates in excess of those calculated for the worst anticipated transient and fuel burnups comparable to the maximum expected in the reactor."(3)

Resolution

A report(6) has been published which presents a review of General Electric and non-GE experience and development data. It is shown that the area of technology demonstrated by irradiations of Zircaloy clad UO_2 fuel rods and pins in reactors, loops, and capsules goes beyond the combination of fuel rod power and exposure that will be experienced in modern BWRs not only for normal continuous operation, but also for anticipated power transients. A larger quantity of data at these combinations of fuel rod linear power and burnup will become available in the future, but the recorded successful irradiations demonstrate that not only safe but reliable fuel can be designed for modern BWR conditions. A value of one percent plastic strain of Zircaloy cladding is conservatively defined as the limit below which fuel damage due to overstraining is not expected to occur. The linear heat generation rate required to cause this amount of cladding strain is calculated to be approximately 28 kW/ft in fresh fuel. The basis for this definition and its inherent conservatism is demonstrated.

J.2.3.4 Effects of Cladding Temperatures and Materials on Core Standby Cooling System Performance

Concern

"In a loss-of-coolant accident, the core spray and flooding systems are required to function effectively under circumstances in which some areas of fuel clad may have attained temperatures higher than those at which such cooling mechanisms have been tested to date. The applicant is conducting tests of these devices at increased temperatures and has reported preliminary results which are promising. The Committee again urges that these tests be extended to temperatures as high as practicable. The use of stainless steel in

these tests for simulation of the Zircaloy clad appears suitable, but some corroborating tests employing Zircaloy should be included."[3]

Resolution

The effects of cladding temperature and material on the BWR Emergency Core Cooling System (ECCS) performance have been investigated.[7] The experimental results of simulated full length, full size bundles with electrically heated rods subjected to cooling by spraying or flooding are presented. The results of these experimental tests indicate that the ECCS is effective at temperature levels significantly higher than those predicted, over the entire postulated LOCA break area range up to and including the design basis accident. Therefore, the ECCS can maintain basic core geometry and prevent melting of the fuel rod cladding. Furthermore, any effect of material on the performance of the ECCS is limited to temperature levels above approximately 2,300°F where the metal-water reaction of the Zircaloy becomes significant. Also the basic differences in material properties can be accounted for by analysis.

J.2.3.5 Design of Piping Systems to Withstand Earthquake Forces

Concern

"The Committee recommends that the applicant give special attention to the design of the critical elements of the plant piping, including the drywell torus connections, to ensure that these elements are not overstressed under maximum earthquake forces."[8]

Resolution

Critical elements of the station piping, including the connections of that piping to the drywell and torus of the primary containment, are designed to withstand, without overstress, the maximum forces resulting from the maximum credible earthquake (0.15g) which is approximately two times the design earthquake (0.08g) to be expected at the site. This was accomplished by the performance of an appropriate static or dynamic analysis of the important piping in systems critical to reactor safety or to safe shutdown of the station. The stresses resulting from these earthquake forces have been calculated and are within the limits for the piping materials and other associated components involved, according to appropriate USAS and ASME Codes. Refer to Section 12 and Appendix C for further information.

J.2.3.6 Reevaluation of Main Steam Line Break Accident

Concern

"Fuel clad temperatures following a steam line break should be further evaluated during detailed design, with due attention to using conservative assumptions and methods in calculating these temperatures. Steam line isolation valve closure time as short as three seconds may be required to maintain acceptably low fuel clad temperatures in this accident. This applicant has stated that isolation valves with closure times adjustable from 3 to 10 seconds will be obtained for the plant."[8]

Resolution

The resolution plan for the above concern item was presented in a GE Topical Report[1] submitted to the AEC in April, 1968. The FSAR justifies a 10 sec closure time, both thermal-hydraulic-wise and radiologically. Refer to Section 14.

A more extensive study of this phenomena was undertaken. The program was completed and a GE Topical Report[9] was submitted to the AEC in October, 1969.

J.2.3.7 Control Rod Block Monitor Design

Concern

"The rod block monitor system for the Vermont Yankee reactor is a two-channel system, with one channel required for rod blocking action. The applicant has proposed that, if one channel is bypassed for maintenance, an appropriately shorter interval between tests will be used for the operating channel. The Committee believes that, if one channel of rod block monitor system is to be out of service for a long period of time, other measures, in addition to frequent testing of the operative channel, should be taken to ensure that improper rod withdrawal is not allowed to occur."[8]

Resolution

The Rod Block Monitor (RBM) System was incorporated for operational reasons for the purpose of backing up the reactor operator in preventing a single operator error or a single equipment malfunction from causing local fuel damage. The level of reliability provided by the RBM System is consistent with this application and the Applicant is satisfied with the level of financial protection provided by this design.

The control rod block action of the RBM System is not to be confused with the NIS-APRM rod block function. The RBM is a local power control system. The APRM rod block is a bulk power control system. Refer to Sections 7 and 14 for further details and description on this system.

An operational analysis was performed on this system.

The analysis detailed in Appendix G support the non-safety status of this system by indicating that no unacceptable safety results are encountered because of single operator or single equipment failures associated with the Control Rod System assuming that the RBM System was completely unavailable.

J.2.3.8 Main Steam Line Isolation Valve Testing Under Simulated Accident Conditions

Concern

"Steam line isolation valves are provided which constitute an important safeguard in the event of failure of a steam line external to the containment. One or more valves identical to these will be tested under simulated accident conditions prior to a request for an operating license."[10]

Resolution

General Electric Company implemented a program to test a full size main steam line isolation valve under simulated accident conditions. The detailed description of the program was presented in a GE Topical Report[1] submitted to the AEC in April, 1968. The testing programs under simulated accident conditions have been successfully completed and reported in a GE Topical Report[11] submitted to the AEC in March, 1969. Analysis of the accident event is discussed in a GE Topical Report[9] submitted to the AEC in October 1969.

J.2.3.9 Depressurization Performance of High Pressure Coolant Injection System

Concern

"The film condensation coefficient used to predict the depressurization performance of the High Pressure Coolant Injection (HPCI) System is based on extrapolation of available heat transfer data. Additional experiments or other supporting studies are needed to confirm the effectiveness of the HPCI system, and the results should be reviewed by the Regulatory Staff."[12]

Resolution

The resolution of the above concern item is presented in the GE Topical Report[1] submitted to the AEC in April, 1968.

The primary function of the HPCI System is to provide coolant makeup to the reactor vessel to keep the reactor core covered and cooled for small system breaks. The secondary function is to depressurize the reactor so that the Low Pressure Coolant Injection System (LPCI) or the RCSC System in the CSCS network can become effective for somewhat larger breaks than can be handled entirely by HPCI System inventory makeup. An analytical model based upon solution to the mass and

energy balances for the system assuming thermodynamic equilibrium is used to predict the depressurization characteristics due to HPCI System operation. Because equilibrium does not actually exist, a calculated "mixing efficiency" is used to represent how nearly the injected subcooled water is raised to the temperature of the reactor vessel fluids.

Engineering tests were conducted in which subcooled water was injected into a constant volume, High Pressure Steam-Water System designed to simulate reactor conditions and geometry. Depressurizaion rate, inlet and fluid temperature were measured. An overall mixing efficiency was evaluated. A sufficient range of variables were included in the tests to determine a mixing efficiency for each reactor primary system. Refer to Section 6 for further details of the HPCI System depressurization performance.

The results of this test program were submitted to the AEC in a GE Topical Report[13] in June 1969.

J.2.3.10 Core Standby Cooling System Thermal Effects On the Reactor Vessel and Internals

Concern

"The Regulatory Staff should review analyses of possible effects upon pressure vessel integrity, arising from thermal shock induced by ECCS operation."[14]

Resolution

A detailed reactor vessel thermal shock analysis was performed on a representative GE BWR reactor vessel. The thermal shock analysis simulating CSCS-LOCA operation was performed on similar reactor vessel design to the Pilgrim vessel and is reported in a GE Topical Report[15] submitted to the AEC in July, 1969.

The thermal shock analysis simulating CSCS-LOCA conditions was made on the reactor internals, including the core spray sparger and the reactor vessel shroud and is described in Sections 3 and 4.

J.2.3.11 Effects of Blowdown Forces on Reactor Primary System Components

Concern

"The effects of blowdown forces on core and other primary system components should be analyzed more fully as detailed design proceeds."[14]

Resolution

The reactor core structural components are designed to accomodate the loadings applied during normal operation and maneuvering transients. Deflections are limited so that the normal functioning of the

components under these conditions are not impaired. Where deflection is not the limiting factor, the ASME Boiler and Pressure Vessel Code, Section III, was used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure.

The loading conditions which occur during excursions or design basis LOCAs were examined. The reactor core shroud, shroud support, and jet pump body, which comprise the inner vessel around the core within the reactor vessel, are designed to maintain a reflooding capability following a design basis LOCA. Reflooding the reactor core to the top of the jet pump inlets provides adequate cooling of the fuel.

The design of the jet pump parts takes into account the pressure loading both in normal and accident conditions, and the reactions at the supporting brackets due to differential thermal expansion of the pump and reactor vessel.

The reactor internals were designed to preclude failure which would result in any part being discharged through the main steam line, in the event of a steam line break, which might block a main steam line isolation valve.

The structural components which guide the control rods were analyzed to determine the loadings which would occur in a design basis LOCA. The reactor core structural components are designed so that deformations produced by accident loadings do not prevent insertion of control rods.

Refer to Sections 3, 4, and Appendix C for additional details.

J.2.3.12 Instrumentation for Prompt Detection of Gross Fuel Failure

Concern

"Considerations should also be given to the development and utilization of instrumentation for prompt detection of gross failure of a fuel element."[14]

Resolution

Refer to Responses 7.5 and 9.4.2 of Supplement No. 3 and to Response 7.5 of Supplement No. 5 of the Brunswick Steam Electric Plant, Units 1 and 2 (AEC Docket Nos. 50-324 and 50-235) where it is shown that the GE BWR failed fuel element detection capability for gross failure is conservatively responsive and well within the design requirements of the concern.

The Brunswick submittal (referenced) discusses the design criteria for the instrumentation for prompt detection of gross failure of a fuel element which is also applicable for this facility. The detection system promptly detects and takes the necessary corrective action for not only gross, immediate, but also minor, long term fuel failures.

Refer to Section 7 for further details.

J.2.3.13 Diversification of the Core Standby Cooling System Initiation Signals

Concern

"Also, he will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates emergency core cooling, to provide additional assurance against delay of this vital function."[10]

Resolution

The preliminary design of sensors for the CSCS equipment consisted of a reactor vessel low water signal from either of two independent instrumentation sources to activate the pumping equipment. Further studies were conducted to ascertain whether reliability could be improved by utilizing alternate or improved sensors. As a result of these studies, instrumentation which detects high pressure in the drywell has been incorporated in addition to the reactor low water level instruments to actuate RCSC, HPCI, and LPCI, and the Standby Diesel Generator Systems.

Diversity of sensors which initiate CSCS functions has also been incorporated into the design. That is, two different types of pressure interlock sensors bellow type and bourdon tube type, are used for this function in order to circumvent any unknown phenomenological uncertainties associated with pressure parameter measurements.

J.2.3.14 Control Systems for Emergency Power

Concern

"The applicant stated that the control systems for emergency power will be designed and tested in accordance with standards for reactor protection systems."[10]

Resolution

Class IE Electrical Systems provide emergency power to Pilgrim Station. The ac and dc subsystems are each divided into two redundant systems with independent control systems. These control systems are designed and will be tested in a manner which meets the requirements of the IEEE criteria for Class IE electrical systems and the applicable sections of IEEE-279. These tests will show that the ac and dc subsystem meet the intent of the requirements of both of the above IEEE documents.

Refer to Section 8 for details.

J.2.3.15 Misorientation of Fuel Assemblies

Concern

"Operation with a fuel assembly having an improper angular orientation could result in local thermal conditions that exceed by a substantial margin the design thermal operating limits. The applicant stated that he is continuing to investigate more positive means for precluding possible misorientation of fuel assemblies."(10)

Resolution

Operation with a misoriented fuel assembly would be an economic rather than a safety concern. Analyses have shown that less than 10 fuel rods in a misoriented assembly would experience a Minimum Critical Heat Flux Ratio (MCHFR) less than 1.9. Under normal operating conditions these 10 fuel rods would, even in the peak power position, remain at a MCHFR greater than 1.0 and peak linear heat generation rate less than 28 kW/ft.

Studies into means of precluding possible fuel misorientation have been completed. It is concluded that the present method of procedural controls is the most desirable of the alternates. Fuel handling operations at operating GE BWRs have shown this to be an efficient, effective method.

Various mechanical devices to prevent inserting a misoriented fuel assembly were also studied and eventually discarded. These devices tended to provide greater potentials for fuel damage during loading and storage operations than the misorientation they were designed to prevent.

Visual identification has been successfully used in all BWRs operated to date to provide assurance of fuel location and orientation. Photos taken of the KRB (Gundremmingen) core after the initial fuel loading clearly showed four different means of identifying a misoriented fuel assembly: (1) All the assembly numbers point towards the center of the cell, (2) the spring clip assemblies all face the control rod, (3) the lugs on the handles point towards the control rods, (4) cell to cell symmetry. Experience has shown that the distinguishing features will be visible during the design lifetime of the fuel. In all cases, fueling procedures require that the fuel assembly number be verified. As a result of this study and the accumulated fuel handling experience, no further work with respect to providing an alternate means of preventing fuel assembly misorientation is planned. The Applicant is satisfied that these methods and procedures provide adequate protection for the fuel investment.

Refer to Section 3 for further details.

J.2.4 AEC General Design Criterion Number 35 - Design Intent and Conformance

Concern

"The applicant and the Staff should resolve the manner in which the intent of General Design Criterion Number 35 (10CFR50.34 proposed July 11, 1967) will be met for the Pilgrim Plant."[2]

Resolution

The piping and pressure containing parts of the reactor coolant pressure boundary will conform to the Non-Destructive Testing (NDT) requirements of Criterion 35 (10CFR50.34 , proposed July 11, 1967) as follows:

a. The fracture or notch toughness properties and the operating temperature of ferritic materials of the reactor coolant pressure boundary will be controlled to ensure adequate toughness by maintaining a material service temperature at least 60°F above the nil ductility transition temperature, when the system is pressurized to more than 20 percent of the design pressure.

b. Charpy V notch tests will be performed to demonstrate that materials and weld metal will meet brittle fracture requirements at test temperature. The welding procedures used must be qualified by impact testing of weld metal and heat affected zones to the same requirements as the base metal.

c. Piping and equipment pressure parts having a nominal wall thickness of 1/2 in or less need not be impact tested provided the material is normalized or has been fabricated to a fine grain melting practice.

d. Impact testing is not required on components or piping within the boundary having a minimum service temperature of 250°F or more.

Protection against the brittle fracture or other failure modes of the Reactor Coolant Pressure Boundary System components is provided for all potential service loading temperatures.

Control is exercised in the selection of materials and fabrication and design of equipment and components to meet the above criteria.

Refer to Appendix A for a more detailed discussion of conformance to the NDT requirements of Criterion 35.

J.2.5 Fuel Clad Disintegration Limitations

Concern

"In connection with postulated loss-of-coolant accidents, the applicant stated that, using conservative assumptions and allowing appropriately for fuel element distortion from the original core geometry, the emergency core cooling systems will be designed to keep fuel-clad temperatures below the point at which the clad may disintegrate upon subsequent cooling."[2]

Resolution

With respect to this overall concern of CSCS effectiveness to cool overheated fuel rods, the industry had selected a maximum allowable temperature of 3,371°F, the melting temperature of Zircaloy. This selection was based on a desire to keep the fuel bundle geometry intact. Refer to a GE Topical Report[1] submitted to the AEC.

Even though this criteria has been adopted for CSCS equipment design, experimental effort continues at the General Electric Company and elsewhere to further refine our knowledge with respect to a proper tolerable maximum fuel temperature during LOCAs.

Some preliminary data from Argonne which has not been fully evaluated at this time tends to indicate that possible clad shattering rather than clad melting would be a more conservative criteria for maintaining fuel bundle geometry. This clad shattering might occur at temperatures as much as 400°F below the Zircaloy melt temperature. Current testing programs at GE for our specific fuel bundle designs are fully investigating this possibility. GE is also fully evaluating this new Argonne National Laboratory data.

The current conservative core cooling evaluation techniques used on this station indicate that the maximum predicted fuel temperatures following postulated design basis LOCAs (less than 2,200°F) are sufficiently below the temperatures of clad shattering that there is no concern regarding loss of fuel geometry. Refer to Section 6 for details. See Section J.2.3.4.

J.2.6 Automatic Depressurization System - Initiation Interlock

Concern

"The applicant stated that he would give further consideration to a suitable interlock to ensure that low-pressure cooling capability would be available before the auto-relief depressurization could be initiated."[2]

Resolution

An auto-relief interlock is included in the design to ensure that low pressure core cooling capability will be available before the auto-relief depressurization can be initiated. A system has been

installed to sense pressure downstream from the core spray and the residual heat removal (RHR) pumps which will prevent auto depressurization unless sufficient pumps are operating to assure the required capability of low pressure core cooling.

Pressure sensors have been installed downstream from each of the low pressure CSCS pumps. The signals from these pressure sensors feed a logic matrix unit. If the logic matrix unit determines from the sensed pressure that adequate core cooling capability is available, auto blowdown will be permitted to proceed upon receipt of its own initiation signals.

J.2.7 Applicants' Role - Quality Assurance Program

Concern

"The Committee recommends that the Boston Edison Company assume an active role in quality assurance at all stages of fabrication and construction."[2]

Resolution

The Applicant has established a comprehensive quality assurance program for the Pilgrim Station.

Refer to Appendix D for program details.

J.2.8 Offsite Emergency Plans

Concern

"The Committee was informed that the Commonwealth of Massachusetts is responsible for preparing off-site emergency plans, with inputs provided by the applicant. The Committee believes that the applicant should assure himself of the adequacy of all emergency plans."[2]

Resolution

The Applicant assumes complete responsibility for the preparation and adequacy of all emergency plans for the Pilgrim Nuclear Power Station which include both onsite and offsite procedures. The offsite emergency plans and procedures are developed in cooperation with responsible agencies of the Commonwealth of Massachusetts and of the Town of Plymouth to assure their familiarity with such plans and to assure coordination of the Applicant's emergency plans with those of other authorities that will assist the Applicant in the event of an offsite emergency. Refer to Section 13 and Appendix N for additional details.

J.3 AREAS SPECIFIED IN THE AEC STAFF CONSTRUCTION PERMIT - SAFETY EVALUATION REPORT FOR PILGRIM NUCLEAR POWER STATION

J.3.1 General

The AEC Staff Construction Permit - Safety Evaluation Report (SER) of May 20, 1968, identified three general areas of specific concerns:

1. SER Section 5.0, Requirements Further Technical Information

2. SER Section 7.0, Report of the Advisory Committee on Reactor Safeguards

3. Miscellaneous Selective SER Section Concerns

These areas are discussed in the following sub sections. Also refer to Table 1-11-3 of the AEC Staff Construction Permit-SER of May 20, 1968.

J.3.2 Requirements for Further Technical Information

J.3.2.1 Introduction

"5.0....A number of areas have been identified that require further technical information to support the design and safety features of boiling water reactors (BWR's) similar to Pilgrim Station. Some of these areas are pertinent not only to BWR's, but also to all large water-cooled power reactors and have been noted in recent ACRS reports."(16)

J.3.2.2 Fuel Damage Limits

Statement

"5.1....(1) Fuel Damage Limits - The objective of the program is to establish the fuel damage threshold for successively larger energy deposition rates as a function of fuel burnup. The applicant established a damage limit corresponding to a linear power generation of 28 kilowatts per foot. Experimental data to date support its model. Additional work is planned and the results of this program are expected to be available to confirm the validity of this limit prior to Pilgrim Station startup operation. The program scope also includes work sponsored by the Commission at Argonne National Laboratory and at the SPERT facilities. We conclude that there is reasonable assurance that this program will provide the necessary information prior to initial operation of the Pilgrim Station."(16)

Resolution

Refer to Appendix J, Section J.2.3.3.

J.3.2.3 Flow Channel Blockage

Statement

"5.1....(2) Flow Channel Blockage - The Applicant indicated that flow blockage during normal operation is local in nature, and cannot propagate to affect the remainder of the core. Nevertheless, the applicant has stated that additional analytical and experimental work will be conducted to confirm the results of previous studies. At this stage of the Pilgrim review, we are satisfied with the applicant's plans to conduct additional analytical and experimental work. The results from this program will be available for review prior to the proposed initial operating date for Pilgrim Station."(16)

Resolution

See Appendix J, Section J.2.3.2.

J.3.2.4 Effect of Fuel Clad Failure on Emergency Core Cooling

Statement

"5.1....(3) Effect of Fuel Clad Failure on Emergency Core Cooling - Based upon analytical and experimental work done to date, clad perforation occurs at a localized area of a fuel rod. Perforation is caused by high internal fuel rod pressure and the region at which perforation occurs has been observed to be random and localized over the length of the fuel rod. Further experimental and analytical work will be continued in order to confirm and further refine the understanding of this fuel damage model. This work will include further perforation tests of fuel cladding under various conditions of temperature, pressure and metal ductility, further heat transfer analysis of fuel bundles under accident conditions, and other tests as appropriate. A report of the results of these tests is scheduled for the end of 1968. Based upon the work done to date and the scope and schedule for the test programs, we believe there is reasonable assurance that this area will be satisfactorily resolved prior to the date proposed for initial operation of Pilgrim Station."(16)

Resolution

See Appendix J, Sections J.2.3.1 and J.2.3.4.

J.3.2.5 Control Rod Worth Minimizer

Statement

"5.2....(1) Control Rod Worth Minimizer - The Control Rod Worth Minimizer is designed to back up operating procedures in assuring that control rod patterns do not result in excessive control rod worths. Design of the RWM is essentially complete and has been evaluated on a preliminary basis in connection with the review of the Oyster Creek, and Nine Mile Point plants for a provisional operating

license. We expect that the operating data that will be forthcoming from these reactor plants will be sufficient to determine the adequacy of the RWM for the Pilgrim Station."[16]

Resolution

The design of the Control Rod Worth Minimizer (CRWM) is complete as reported in a GE Topical Report[17] submitted to the AEC in March 1967. See Section 7 for further details.

J.3.2.6 Control Rod Velocity Limiter

Statement

"5.2....(2) Control Rod Velocity Limiter - The Control Rod Velocity Limiter is designed to limit the freefall velocity of a control rod to mitigate the consequences of a reactivity insertion transient. The design is complete and the test program has been reported by General Electric to have successfully achieved the design objectives. As in item (1) above, the design aspects have been reviewed for the Oyster Creek and Nine Mile Point plants and the device will have been installed in other boiling water reactors prior to its application in Pilgrim Station."[16]

Resolution

The design and final test program of the Control Rod Velocity Limiter (CRVL) is complete as reported in a GE Topical Report[18] submitted to the AEC in March, 1967. Refer to Section 3 for further details.

J.3.2.7 In-core Nuclear Instrumentation

Statement

"5.2....(3) In-Core Nuclear Instrumentation - The In-Core Nuclear Instrumentation consists of in-core chambers to monitor local power density. General Electric has reported on the performance of in-core monitors from operating experience at the Big Rock Point and KRB reactors. The applicant states that sufficient data is available to demonstrate the adequacy of this component performance.

We have not reviewed such data in detail at this time, but we expect to closely follow the operation of Oyster Creek and Nine Mile Point plants in this regard. As with the previous items, we believe that satisfactory in-core testing will have been conducted prior to operation of Pilgrim Station."[16]

Resolution

The design and adequate performance demonstration of the Incore Nuclear Instrumentation System is complete and is reported in a set of GE Topical Reports[19,20] submitted to the AEC in August, 1968 and November, 1968, respectively. See Section 7 for further details.

J.3.2.8 Jet Pump Development

Statement

"5.2....(4) Jet Pump Development - Considerable analytical and test work has been completed on the jet pump system for reactor coolant recirculation to establish basic design characteristics. Continued development in progress and planned is summarized in Amendment No. 2. The development program, and the fact that this component will have been operated in similar reactor plants prior to its operating application in Pilgrim, will be adequate to establish its capability at the operating license review stage."[16]

Resolution

The design and test program of the Jet Pump Assemblies is complete and is reported in a GE Topical Report[21] submitted to the AEC in September 1968.

J.3.2.9 Load Control Using Variable Speed Recirculation Pumps

Statement

"5.2....(5) Load Control by Variable Speed Recirculation Pumps - This is principally an analytical program to accurately model the performance of the recirculation pumps and associated reactor response. The adequacy of the model will be demonstrated by comparison of prediction with the results of Oyster Creek and Nine Mile Point reactor startup tests and substantiated in other BWR plant operations prior to the Pilgrim Station startup."[16]

Resolution

The objective of this program is to accurately model the performance of the Reactor Coolant Recirculation System (RCRS) pumps, and the reactor response for this system. The modeling program is complete, with appropriate parameters modified as particular equipment is designed or purchased. The adequacy of the model is routinely verified by comparison of the prediction with the results of startup tests. See Sections 7 and 14 for further details.

J.3.2.10 Core Spray Cooling Effectiveness

Statement

"5.2....(6) Core Spray Cooling - The objectives of this program are to confirm the core spray cooling and flooding effectiveness of the optimized core cooling systems applicable to the current design of GE-BWR's. An objective of the program will be to determine core cooling effectiveness of a wide range of clad temperatures which will include predicted temperatures based on the expected performance of the proposed Pilgrim ECCS. The tests will also include studies with cladding material similar to that which will be used in the Pilgrim reactor.

The results of analytical and test work to date are provided in documentation for previously approved BWRs. These results, as applicable to Pilgrim, are summarized in Appendix A of the DAR and in Amendment Nos. 2 and 4. Tests have been done and will continue on core spray distribution over a simulated reactor core. In view of the effort expended on this matter to date and the plans for continued work scheduled through 1968, we believe that this matter will be resolved satisfactorily prior to initial operation of Pilgrim Station."[16]

Resolution

The design and final test program of the effectiveness of the Core Spray Cooling System (CSCS) has been completed and reported in a GE Topical Report[22] submitted to the AEC in March, 1968. See Appendix J, Section J.2.3.4.

J.3.2.11 Boiling Water Reactor System Stability Analysis

Statement

"5.2....(7) System Stability. The objective of this program is to develop an analytical model which would predict the onset of instabilities in the reactor core. Tests have been conducted at other GE-BWR's, notably the SENN and KRB reactors, and experimental data were found to agree well with model predictions which show no significant tendencies for system instabilities. The General Electric Company has indicated that it is continuing its studies on this matter and will keep us informed of the findings as they become available. We will continue our review of this matter. Additional analytical results and reactor operating data will become available prior to anticipated initial operation of Pilgrim Station."[16]

Resolution

The development of a BWR Stability Model which would predict the onset of instabilities in the reactor core in this station has been completed, and the excellent agreement between model predictions and experimental data has been reported in the following GE Topical Reports[23,24] submitted to the AEC in April 1969 and June 1968 and in GE memorandum[25], submitted on Peach Bottom Atomic Power Station, Units 2 and 3, AEC Docket Nos. 50-277 and 50-278. See Section 7.16 for further details.

J.3.2.12 Provisions for Inservice Inspection

Statement

"5.2....(8) Provisions for In-Service Inspection - The Applicant has stated that provisions are being incorporated in the design by the vendor of the nuclear steam supply components to facilitate inspection of selected areas of the interior of the reactor vessel and its components, as recommended in the report, APED-5450. This report, titled "Design Provisions for In-Service Inspection," is a

topical report submitted by the General Electric Company. At this stage of the review, we are satisfied with the applicant's awareness of the requirements to provide a detailed in-service inspection program at the operating license stage, and of its stated intentions to do so."[16]

Resolution

The Applicant retained the Southwest Research Institute to provide independent advice regarding inservice inspection techniques and capabilities. The Southwest Research Institute provided recommendations during the design effort regarding access provisions for inspection and assisted the Applicant in the development of an inservice inspection program. This inspection program complies with the intent of the ASME Inservice Inspection Code to the maximum extent practicable. The Applicant also retained the Southwest Research Institute to perform the initial testing required during station startup to obtain base or reference data. See Appendix K for a detailed description of the Boston Edison Inservice Inspection Program.

J.3.2.13 Analysis of Thermal Shock Effects from Core Standby Cooling Systems

Statement

"5.2....(9) Analysis of Thermal Shock Effects from ECCS. The effect of thermal shock on the reactor vessel and its appurtenances induced by injection of emergency core cooling water into the higher temperature reactor system has not yet been fully analyzed. The Applicant has responded in Amendment No. 4 that General Electric will perform a detailed stress analysis in connection with the Millstone Point provisional operating license review. At this stage of the Pilgrim review, we are satisfied with the applicant's awareness of the requirement for a detailed thermal shock analysis at the operating license stage and of its stated intentions to provide one."[16]

Resolution

See Appendix J, Section J.2.3.10.

J.3.2.14 High Pressure Coolant Injection System- Depressurization Model (Peak Clad Temperatures)

Statement

"5.2....(10) HPCI System Depressurization Model (Peak Clad Temperatures). The principal function of the HPCI system is to maintain water inventories sufficient to assure core cooling for small breaks. For intermediate breaks, it also serves to depressurize the pressure vessel to a level such that the core spray system or low pressure injection system can reach rated flow. The HPCI system is designed to pump 4250 gpm into the reactor pressure

vessel within a reactor pressure range of about 1100 psig to 150 psig. Our concern continues to be with the accuracy of the analytical techniques in predicting vessel depressurization using the HPCI system. In the Cooper Station application, it was stated that the required mixing efficiency for the HPCI system would be obtained by optimizing the feedwater sparger design; i.e., by using a large number of small sparger holes to maximize the HPCI exit velocity and jet surface area, and by aiming the sparger holes to maximize wetted film area. This proposed optimization of the feedwater sparger design would be based on an analytical model which has not been verified experimentally. As a result of the continuing interest in the area of depressurization model and peak clad temperatures, the General Electric Company has formulated an experimental test program to determine HPCI mixing efficiencies. It is planned that the proposed tests will be completed in 1968....

...After the depressurization model is evaluated by comparison with experiments, the analytical model can be refined to establish the calculated peak clad temperatures. We believe that the HPCI depressurization principle is feasible and that the experimental program can be expected to test its applicability. The results from this program will be available for review prior to the proposed operating date for Pilgrim Station."[16]

Resolution

See Appendix J, Section J.2.3.9.

J.3.2.15 Main Steam Line Isolation Valve Operability

Statement

"5.2....(11) Main Steam Line Isolation Valve Operability - General Electric Company is currently undertaking a program to test main steam line isolation valves under simulated accident conditions. This testing program will include testing of valves on a scale to permit evaluation of hydrodynamics of the blowdown under prototypical conditions, testing of valves designed for other boiling water reactors under simulated accident conditions, and testing the steam line isolation valves during the preoperational test phase to verify that the valves as installed will meet all functional requirements. The results of these tests are expected to be available prior to the Pilgrim Station operating license review. We are presently reviewing the scope and schedule for these tests and conclude that there is reasonable assurance that this matter will be satisfactorily resolved prior to the date proposed for initial operation of Pilgrim Station."[16]

Resolution

See Appendix J, Section J.2.3.8.

J.3.2.16 Electrical Equipment Inside Containment Test Program

Statement

"5.2....(12) Electrical Equipment Inside Containment - Electrical equipment which must operate inside primary containment in an accident environment is limited to cables and operators for isolation valves. Where practical, the valves are designed to fail "as is" or closed (safe failure). A circuit failure after the valve has closed will be a safe failure. In addition to designing the equipment to withstand the accident environment long enough to operate the valves, the applicant has agreed to test the performance of this equipment. In Amendment No. 4, the applicant outlines the components and subsequent tests of the materials to be installed in the Pilgrim Station primary containment. The tests will demonstrate that the material and equipment will survive the accident conditions of simultaneous pressure, temperature, and humidity for a period of time essential for their operation. Successful demonstration by these tests will satisfy our requirements."[26]

Resolution

Tests have been completed on cables and operators for isolation valves, recirculation line valves and relief valves. The test results indicate the capability of the equipment to meet or exceed the design requirements and to function while under postulated accident conditions.

J.3.2.17 Primary System Leak Detection

Statement

"5.2....(13) Primary System Leak Detection - Detection of leaks in the primary system will be accomplished by monitoring the sump level in the containment vessel, by monitoring the T of the cooling water of the containment vessel heat exchangers and by monitoring the containment vessel pressure. Leaks of from less than 1 gpm to up to 40 gpm can be detected by these methods. Selected areas in the reactor building in the vicinity of the RCIC and RHR equipment will also be monitored. The above represents the status of leak detection techniques which are presently proposed; however, we intend to continue our consideration of leak detection techniques to ascertain that the proposed methods are sufficiently sensitive. We conclude that there is reasonable assurance that this matter will be satisfactorily resolved prior to the date proposed for initial operation of the Pilgrim Station."[16]

Resolution

See Appendix J, Section J.4.9, J.4.11, and Section 4 for further details.

J.3.3 Report of the Advisory Committee on Reactor Safeguards

Statement

"7.0....Report of the Advisory Committee on Reactor Safeguards - The Advisory Committee on Reactor Safeguards (ACRS), by letter dated April 12, 1968 to Chairman Seaborg, reported on the proposed Pilgrim Nuclear Power Station. A copy of the letter is attached as Appendix B. In its letter, the ACRS commented or made recommendations on particular items of fact or concern, all of which have been addressed in this Safety Evaluation. An index to the discussion of these comments in this Safety Evaluation is tabulated below. (Table 7.1.)"[16]

Resolution

See Appendix J, Section J.2.

J.3.4 Miscellaneous Selective Safety Evaluation Report Section Concerns

J.3.4.1 Criterion 35 Intent

Statement

"6.0....Station Design With Respect to the General Design Criterion....The capability of satisfying Criterion 35, which relates to the prevention of brittle fracture in the pressure vessel and the other parts of the primary coolant pressure boundary, was discussed at length with the applicant. In Amendment No. 9 (Comment 4) and Amendment No. 10 (Comment 1) the applicant defines and discusses the problem. The Pilgrim pressure vessel meets the requirements of Criterion 35 as presently phrased; however, the remainder of the primary system does not meet a literal interpretation of the present phrasing of the criterion. However, the applicant states that the Pilgrim Station will meet the requirement of brittle fracture prevention in all parts of the reactor coolant pressure boundary. We believe that this approach meets the intent of Criterion 35. Details of the design and analytical techniques by which the applicant will assure prevention of brittle fracture in the primary system will be resolved between the applicant and staff."[16]

Resolution

See Appendix J, Section J.2.4

J.3.4.2 Emergency Planning

Statement

"8.3....Emergency Planning....We consider the applicant's basic planning acceptable for the construction permit review stage. A detailed evaluation of emergency planning bases and objectives, and

of instrumentation provisions within the plant for emergency off-site evaluation, will be conducted at the operating license stage."[26]

Resolution

See Appendix J, Section J.2.8.

J.3.4.3 Meteorological Program

Statement

"....2.2 Meteorology ....The applicant has agreed that a final determination of fumigation will be based on the results of the on-site meteorological program. The results of these findings will be used to establish gaseous release limits at the provisional operating license stage of review."[16]

Resolution

The results of the onsite meteorological program are given in Section 2. Appendix E contains stack release calculations and Appendix I details the results of the onsite (seabreeze) studies.

J.3.4.4 Hydrology and Oceanography Program

Statement

"....2.4 Hydrology and Oceanography....Our consultants on flooding from the Coastal Engineering Research Center indicate that the applicant's analysis does not account for the meteorological parameters associated with the Probable Maximum Northeaster. The applicant has agreed to reanalyze the maximum probable flood incorporating these influences based upon the publication of a U.S. Weather Bureau study on meterological conditions associated with Northeaster storm surge to be available by about October 1968. We conclude that this does not represent any engineering problem since the applicant will provide protection to the agreed-upon maximum probable flood level. The reports of our consultants on hydrology and oceanography are attached as Appendix F and Appendix G from U.S. Geological Survey and U.S. Army Coastal Engineering Research Center, respectively."[16]

Resolution

See to Appendix J, Section J.2.2.

J.3.4.5 Reactor Pressure Vessel-Stub Tube Design

"....3.1 Reactor ....Recently, defects have been found in the control rod drive stub tubes of the Oyster Creek reactor pressure vessel. This problem is presently under review by the regulatory staff. It is not yet entirely clear whether the....

....In order to accommodate our review of this potential problem with the Pilgrim Station vessel at the earliest time, the applicant informed us that it will provide us with the design of the stub tubes and an evaluation of the potential stub tube problems on its vessel. As indicated in the applicant's Summary of Application, it will incorporate any necessary corrective action resulting from the stub tube problem evaluation, in its vessel prior to the completion of its fabrication. The applicant has indicated that the design and evaluation will be submitted to the staff as soon as information is available. The proposed plans with respect to the reactor pressure vessel are acceptable to us."[16]

Resolution

A GE Topical Report [27] was submitted to the AEC in November, 1968 on the Stub Tube design. The report describes the design, analysis, fabrication, and test of the control rod drive penetration typically used in current General Electric Company reactor vessels. The penetration described consists of an Inconel internal stub nozzle welded inside the reactor vessel bottom head, and an austenitic stainless steel control rod drive housing penetrating the reactor vessel head and welded to the top of the Inconel stub nozzle. This penetration is typical of the Dresden II and III, Millstone, Monticello, Browns Ferry, Vermont Yankee, Peach Bottom, and Pilgrim nuclear power station plants now well along in construction and on other plants to follow in the immediate future. Although details of design and fabrication vary slightly in this series of plants, principally to accommodate the fabricators' manufacturing preferences and methods, these differences are not significant and the resulting penetrations are equivalent. See Section 4 for further details.

J.3.4.6 Flow Reference Scram Design

Statement

"....3.4 Instrumentation....Reactor scram is actuated by a fixed high-flux scram setting (120% power). We have evaluated the ability of the proposed system to prevent fuel damage at steady state reduced recirculation flow and conclude it is adequate. However, we have asked the applicant to continue reviewing the adequacy of the systems for anticipated transients at reduced recirculation flow. Results of this evaluation will determine the required APRM functions. This area will be resolved during the operating license review."[16]

Resolution

See Appendix J, Section J.4.8

J.3.4.7 Reactor Protection System-Power/Flow Instrumentation Design - (IEEE-279)

Statement

"....3.4 Instrumentation....(1) Reactor Protection System Power/Flow Instrumentation....The applicant has stated that the design of the reactor protection system power/flow instrumentation will comply with the Proposed IEEE Standard for Nuclear Power Plant Protection Systems. Our review of the preliminary design indicates that the proposed system can be built to meet the proposed standard. The adequacy of the system depends upon the manner in which channel independence and isolation are implemented in the final design. We will review this aspect of the final design in detail at the operating license evaluation."[16]

Resolution

The design basis and description of the RPS and the power/flow instrumentation are not combined, nor are the other systems mentioned in the AEC-Staff SER Section 3.4 to be associated with, nor combined with the RPS.

The RPS for Pilgrim Station is designed to meet the intent of the requirements of IEEE-279. See Appendix J, Section J.3.4.12 for further details.

J.3.4.8 Low Pressure Coolant Injection - Logic Control System Design (IEEE-279)

Statement

"....3.4 Instrumentation....(2)Emergency Core Cooling System Actuation....The above description applies to the normal two recirculation loop mode of operation with the equalizer line valves closed. Two differential pressure gages are connected across the equalizer line to identify a damaged recirculation loop during a one loop mode of operation. These gages are connected in a one out of two logic scheme. The applicant has stated that this system is being designed to the proposed IEEE Standard. We will evaluate the final design of this vital system during the operating license review to ensure that the requirements of the IEEE Standard are met."[16]*
*The equalizer line and valves were removed during the 1984 pipe replacement program.

Resolution

The design basis and description of the LPCIS Injection Control System is not in agreement with the interpretation. The Applicant's documented design is as described in Appendix J, Section J.3.4.12 and Sections 6 and 7.

J.3.4.9 Reactor Protection System - Instrumentation - Test Capability During Operation

Statement

"....3.4 Instrumentation....(3) Conclusions....All instrument channels required for reactor protection and the actuation of the engineered safety features are testable during reactor operation. Also, there is complete separation of control and safety functions."(16)

Resolution

CSCS Sensor Testability - Test jacks have been added in the CSCS initiation circuitry in order to permit testing of these circuits using permanently installed equipment. Testing of the RHR, Core Spray, HPCI, and the Automatic Depressurization Relief Systems is facilitated by this design. Additional alarms and indicating lights, provided to indicate the operational state of the sensors and relays, will aid in testing as well as help assure that circuitry is returned to normal after testing is completed. This design is similar to the design provided recently on the Dresden Nuclear Power Station, Unit 2- AEC Docket No. 50-293-Amendment 17 and the Monticello Nuclear Generating Plant, Unit 1-AEC Docket No. 50-263 - Amendment 19.

CSCS System Testability - Refer to Section 6 for the system design and test capabilities.

RPS Testability - See Section 7 for the system testability.

Primary Containment Isolation System (PCIS) Testability - See Section 7 for the system testability.

J.3.4.10 Reactor Protection System and Core Standby Cooling Instrumentation-Cable Markings and Identification

Statement

"....3.4 Instrumentation....(3) Conclusions....The applicant has stated that a means will be developed to easily distinguish protection system cables, wires and components from similar components which are not related to protection. We conclude that the stated intention is sufficient at this stage of review."(26)

Resolution

All cables in the RPS and in the CSCS are uniquely marked at each termination to distinguish them from other cables. Electrical panels and components are prominently identified by name plate.

See Section 7 for further details.

J.3.4.11 Drywell Accident Conditions - Electrical Component Testing

Statement

"....3.4 Instrumentation....(3)Conclusions....The applicant will provide test data to confirm that those electrical components, instruments and cable located in the primary containment will be capable of performing their function during and subsequent to a design basis accident for the length of time required. We consider this commitment to be adequate for the construction permit review. The data will be evaluated during the operating license review."(16)

Resolution

See Appendix J, Section J.3.2.16.

J.3.4.12 Reactor Protection System and Core Standby Cooling Instrumentation Design Criteria (IEEE-279)

Statement

"....3.4 Instrumentation....(3) Conclusions....The applicant has stated that the reactor protection system and the instrumentation which actuates the engineered safety features are being designed to the proposed IEEE Standard. Our analysis of the preliminary design indicates that these systems can be built to satisfy the requirements of the proposed IEEE Standard and we conclude that this is adequate at this stage of review."(16)

Resolution

Instrumentation in the Pilgrim Nuclear Power Station is classified as either control instrumentation or protection instrumentation. The former performs absolutely no safety or protective function, while the latter performs only protective functions. The former (control instrumentation) are not designed to meet the requirements of IEEE-279 since no safety or protective function is provided by these instruments. The latter group of instrumentation includes not only that system called the RPS, but also the initiation circuitry for the CSCS, the Reactor Vessel and PCIS, and the Reactor Building Isolation Control System (RBICS). This latter group of instruments is designed to meet the intent of the requirements of IEEE-279. Systems which are designed to meet the intent of the requirements of IEEE-279 are the following:

The Reactor Protection Systems (RPS)

Initiation logic for the Core Standby Cooling Systems (CSCS)

Initiation logic for the Primary Containment Isolation System (PCIS)

Initiation logic for the Reactor Building Isolation Control System (RBICS)

Reactor Protection System

The RPS provides protective reactor trip or shutdown to terminate any potentially unsafe trend. This system has no control function. The RPS is designed to meet the intent of the requirements of IEEE-279. This system includes the Average Power Range Monitor (APRM) and the Intermediate Range Monitor (IRM) subsystems.

RCRS flow measurement reference signals to the APRM scram circuits are designed and installed in a manner identical to Dresden 2 and Oyster Creek 1 facilities equipment arrangements.

The Rod Block Monitor (RBM) System is not a safety system (see Section 1.4) and is not designed to IEEE-279 requirements. See also Dresden 2/3-Amendments 17 through 20.

Core Standby Cooling Systems

The CSCS are made up of several subsystems. These subsystems are intended to provide two protective functions. One protective function is for large primary system breaks, where core spraying or core flooding is to be accomplished to adequately cool the core. The Core Spray Cooling subsystems and the LPCI subsystem each independently provide this protective function. This is referred to as the "low pressure core cooling" protective function. The other protective function is for small primary system breaks. In this case, the protective function occurs in two steps; the first is the depressurization of the primary system followed by the second which is spraying or flooding, as in the large break case. The depressurization can be performed rapidly by use of the Auto Depressurization Subsystem (ADS), or slowly by the HPCI subsystem which also provides makeup to the coolant inventory. The ADS and HPCIS are each, independently, capable of providing the first step in the small break protective function. This is known and referred to as the "high pressure core cooling" protective function.

Each of the two protective functions (low and high pressure core cooling) described above are accomplished by the use of one of two subsystems. Either the LPCIS or one of the two CSCS loops perform a "low pressure core cooling" function and either the HPCIS or the ADS perform a "high pressure core cooling" function. These subsystems are not individually redundant and independent, but are collectively designed so that each protective function (high and low pressure cooling) is achieved with a combined systems design which meets the requirements of IEEE-279 in both initiation and control. A discussion of each subsystem is given below in order to clarify the applicability of IEEE-279 to each protective function, and the capability of each subsystem making up the protective function to itself, meet the IEEE-279 requirements.

1. Low Pressure Core Cooling Protective Function

There are two completely independent, redundant, physically separated core spray subsystem loops. The initiation logic for these two

subsystems meets the intent of the requirements of IEEE-279. There is no control function served by this subsystem nor is there any automatic control circuitry on the subsystem. Upon initiation, the subsystem operates continuously at design conditions. The two subsystems together meet the IEEE-279 requirements.

The initiation logic for the Low Pressure Coolant Injection (LPCI) subsystem and the loop selection logic meet the single active component failure criterion. The LPCI subsystem does not meet the requirements of IEEE-279 because the protective function performed by the LPCI subsystem is redundant to and can be performed alternately by the core spray subsystems described above. These three subsystems collectively meet the intent of the requirements of IEEE-279.

The LPCIS mode of the RHRS has no automatic control circuitry associated with it. Like the core spray subsystems, the LPCIS upon initiation operates continuously at design conditions. Subsequent to reflooding the core after an accident, the LPCI subsystem can be switched to manual control and system flow reduced to that required for other system functions such as containment cooling modes of operation. The manual control circuitry required for containment cooling meets the requirement of IEEE-279.

The flow path used by the LPCIS for injecting water into the reactor vessel utilizes a single injection valve and flow path. The circuitry which operates this valve does not meet the requirements of IEEE-279 for the reasons discussed above. The shutdown cooling function of the RHRS is normally isolated during reactor operation because of two closed valves. This portion of the RHRS does not provide any safety or protective function and therefore is not designed to meet the requirements of IEEE-279.

2. High Pressure Core Cooling Protective Function

The Auto Depressurization Subsystem (ADS) initiation logic meets the intent of the requirements of IEEE-279. The automatic control of the valves meets the single active component failure criterion of IEEE-279. The protective function of the ADS is redundant to and can be performed alternately by the HPCI subsystem described below. Thus, there are two independent and fully redundant subsystems to provide the high pressure core cooling protective function. The valves, when actuated, open and remain open with no further automatic control. The valves are powered by redundant single active component failure proof power sources and power control circuitry. The manual control of the valving meets the single active component failure proof criterion.

The initiation logic of the HPCI subsystem meets the intent of the requirements of IEEE-279. This subsystem has a steam turbine that is automatically controlled to operate under a wide range of driving steam conditions from as low as 50 psig to 1,100 psig.

The control instrumentation for the turbine does not meet the requirements of IEEE-279. The protective function served by the

HPCIS is redundant to and can be performed alternately by the ADS subsystem described above. The initiation and control circuitry of these two subsystems which perform the depressurization (high pressure) protective function, when considered together, are designed to meet the intent of the requirements of IEEE-279.

3. Primary Containment Isolation Systems

The initiation logic for the automatic closure of the primary containment isolation valves meet the intent of the requirements of IEEE-279. Two valves are located on each line, one inside and one outside the containment to meet the single active component failure criteria.

4. Reactor Building Isolation Control System

The initiation logic for the RBICS is designed to meet the intent of the requirements of IEEE-279. The system meets the single active component failure criterion by including two isolation devices for each ventilation penetration.

Conclusions

A thorough treatment of the protective function design philosophy is given in Sections 1, 5, 6, 7, and Appendix G. The systems described above meet the intent of the IEEE-279 requirements as stated.

J.4 AREAS SPECIFIED IN RECENT, RELATED ACRS CONSTRUCTION AND OPERATING PERMIT LETTERS

Areas specified in related ACRS Construction and Operating Permit letters released subsequent to the Pilgrim Station Construction Permit review are discussed in this section.

J.4.1 Scram Reliability Study

Concern

"The Committee believes that, for transients having a high probability of occurrence, and for which action of a protective system or other engineered safety feature is vital to the public health and safety, an exceedingly high probability of successful action is needed. Common failure modes must be considered in ascertaining an acceptable level of protection. In the event of a turbine trip, reliance is placed on prompt control-rod scram to prevent large rises in primary system pressure. The applicant and his contractors have devoted considerable effort to providing a reliable protective system. However, systematic failures due to improper design, operation or maintenance could obviate the scram reliability. The Committee recommends that a study be made of further means of preventing common failure modes from negating scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients."[28]

Resolution

Studies have been performed by General Electric Company, a) to evaluate common mode failures which could negate scram action,[29] and b) of design features to make tolerable the consequences of failure to scram during anticipated transients.[30] Neither the Applicant nor the General Electric Company consider complete failure of the scram system (redundant RPS and the 145 individual control rods) a possibility; nevertheless the studies were performed.

The common mode failure analysis focuses primarily on the plant variables that are sensed and serve as input information to the plant protection systems. The report shows the extent to which each plant variable (i.e., flux, flow, pressure, etc.) used to initiate a protective function, is backed up by a completely different plant variable in the event that a common mode failure blocks the flow of information from the first level of protection signal. This approach to the study was selected because it cuts the broadest slice through an almost infinite common mode failure analysis problem, is the most productive of useful results for an initial effort, and narrows the scope of follow-on studies to a reasonable effort.

In the course of the study, the plant's reponse to all operational transients and accidents was tracked, given that the plant protection system was fully sensitive to information flow from the critical plant variables. Then, each information flow path was assumed to be blocked, one at a time, by an unidentified common mode failure, and the plant's response was again tracked for the same operational transients and

accident conditions. The results provide an assessment of the degree of resistance of the GE BWR RPS to unknown common mode failures resulting in loss of the primary protection initiation signals.

Although the failure to scram study identifies potential design innovations to ease the severity of the consequences of failure to scram, it is claimed that the degree of reliability, availability, and performance of the scram function does not necessitate the imposition of a new design basis or the implementation of these innovations. A description of this additional protection is presented in the report. The transients evaluated are based on a per unit generic analysis representative of all plant sizes embraced by the BWR product line series currently under construction and entering operational service at the time of Pilgrim Unit 1 startup.

J.4.2 Design Basis of Engineered Safety Features

Concern

"For purposes of design of the engineered safety features, the applicant has proposed using a fission-product source term smaller than that suggested in TID-14844, and a treatment of this source within the containment different from that recommended in the same document. The Committee believes that the assumptions of TID-14844 should be used as a design basis for the engineered safety features of the Brunswick plant, unless and until the use of a different set of assumptions has been justified to the satisfaction of the Regulatory Staff and the ACRS."[28]

Resolution

The engineered safety systems for the Pilgrim facility are designed to perform preventive and mitigative functions when called upon during the course of any credible design basis accident. These functions are related to two general objectives: (1) protect the fuel barrier (i.e., maintenance of fuel cladding integrity, prevention of clad melt, minimization of extent of fuel rod perforations, etc.), and (2) minimize potential offsite doses (i.e., mitigate the cause and consequences of accidents, containment, filter, control elevated release, etc.). The design philosophy is that these functions must be maintained under all credible design basis accident conditions.

The radiological consequences accident analysis employed in meeting the above two objectives was based upon the following criteria:

1. Liberal definitions and conservative assumptions for the initial accident event

2. Appropriate conservative proven engineering calculational methods

3. Conservative actual or predicted system/equipment performance

The design philosphy is to assure that substantial margins exist in the overall system, component, or equipment design. This is achieved from a

thorough consideration of all factors, such as the fission product source terms, plateout, partition, etc., that may be reasonably expected to occur. Margin is applied to each assumption and calculational method consistent with the confidence level one has in the data and its uncertainties to achieve an overall margin.

In the spirit of the above general design basis philosophy, the radiological consequences accident analysis methods and models employed in the design of the Pilgrim facility are those cited in the GE Topical Report submitted to the AEC in March, 1969. This document reflected the approved design described in the construction permit PDAR documentation and the "as built" station conformance to 10CFR100 limits which are described in the FSAR.

TID-14844 Capability

Although the Applicant and General Electric Company do not feel that the TID-14844 sources are credible and thus required for safety purposes, the engineered safety features have been analyzed using the TID-14844 source terms. An assessment of the capability of CSCS equipment to perform their intended functions along with the offsite radiological effects of such postulated source assumptions has been prepared and is included in Section R.6.

J.4.3 Hydrogen Generation Study

Concern

"Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. The Committee believes the applicant should evaluate all problems which may arise from hydrogen generation, including various levels of Zircaloy-water reactions which could occur if the effectiveness of the emergency core cooling system were significantly less than that predicted. The matter should be resolved between the applicant and the AEC Regulatory Staff."

Resolution

GE studies are continuing on the possible effects of radiolysis of water in the unlikely event of a LOCA. The studies will evaluate all problems which may arise from credible hydrogen generation. The study is also intended to show possible methods of handling postulated quantities of hydrogen generated by radiolysis. Details on the studies have been documented and submitted on Supplement No. 4, Brunswick Steam Electric Plant, Units 1 and 2. (AEC Docket Nos. 50-324 and 50-325).

Two GE Topical Reports, were submitted to the AEC in March, 1968 and August 1968 which established that very little hydrogen is evolved as the result of the design basis loss of coolant accident with the design minimum CSCS equipment available for operation. Even with further CSCS degradations, the calculated maximum clad temperatures (of 2,200F) would not increase to levels (2,800F) where clad shattering or one percent metal water reactions could take place.

J.4.4 Primary Containment Inerting

Concern

"The Committee believes that, with the present state of knowledge of the performance of the ECCS and the course of a postulated loss-of-coolant accident, the containment should be inerted during operation of the reactor. However, it is recognized that inerting increases problems of inspecting for and repairing leaks in the primary system. It is recommended that the requirement for inerting be periodically reviewed as operating experience and further knowledge from development work currently underway are obtained, and as other means of eliminating the hazards from accident generated hydrogen are found."

Resolution

The Applicant has incorporated a primary containment inerting capability into the facility design as agreed upon in the Pilgrim Construction Permit Application (PDAR). The Applicant does not propose, however, to operate the station with the primary containment inerted because of the danger to the health and safety of operating personnel, and because of the reduction in permissible access to the primary containment for periodic visual inspection of primary system components. The Applicant believes the overall safety of station personnel and the general public will be enhanced more by the periodic inspections and early detection of potential deterioration of the primary boundary which are permissible without inerting than by the additional containment hydrogen capability provided by inerting.

The Applicant believes that substantial protection to preclude large metal water reactions with associated gross hydrogen releases to the primary containment is provided by the redundancy and diversity of CSCS incorporated in the facility. Substantial protection exists even assuming severely degraded availability of this equipment. Two reports prepared by the General Electric Company, are on file with the AEC which contain technical evaluations showing that reasonable safety margins are provided in the primary containment design without inerting.

Comparable safety margins for operating personnel are much more difficult to maintain if inerted since significant risk of asphyxiation exists if personnel access were permitted while the containment is inerted. Even assuming no access is permitted while the containment is inerted, careful adherence to operating procedures is required during purging to assure that the inerted atmosphere is all properly vented and that all containment areas are safe for operator access. These procedures must be carried out with a minimum of errors. Unlike the CSCS design, redundancy and diversity of function are not available for these containment purging operations and fewer errors can be permitted without a total loss of the available safety margins. The Applicant believes the safety consequences of a failure to properly vent the containment after inerting are potentially more severe than any reduction of safety margin due to operation of the containment without inerting.

The Applicant has concluded therefore that operation of the containment with an inert atmosphere is not justified based on overall safety considerations considering the relative risks, design margins, and consequences of operation with and without inerting.

J.4.5 Seismic Design and Analysis Models

Concern

"The applicant is reviewing the seismic design of Class I structural and mechanical components of the plant and will complete his analysis before the reactor goes into operation. In the event that changes to the plant should be found necessary, such changes will be made on a time scale to be agreed upon between the applicant and the Regulatory Staff."[34]

Resolution

The method used to do the seismic (earthquake) analysis on the Reactor Coolant Recirculation System loop lines of the Dresden 2/3 Plants was the same as used on other GE BWR Reactor Coolant Recirculation System loop lines. A reevaluation of the results of the Dresden 2/3 recirculation lines was made by expanding the analysis to include the following alteration to the GE standard method referenced and described in a GE BWR docket [Monticello FSAR (Method I)]. In the expanded analysis the inertia forces for each mode were used to determine each mode's contribution to the total internal forces, moments, and stresses in the pipe. The total combined results were obtained by taking the square root of the sum of the squares of each parameter; i.e., forces, moments, and stresses. This new analysis is identified and called Method II.

A summary of the three highest stresses for Method I in each of the four components in the Dresden 2/3 recirculation lines and the corresponding stresses as calculated by Method II were submitted to the AEC for purposes of comparison in order to eliminate the subject concern above. The comparison of the stresses of corresponding points as calculated by Method I were substantially in agreement with the stresses calculated by Method II.

From the above comparison it may be concluded for the Reactor Coolant Recirculation System loop lines on Dresden 2/3 that either Method I or II gives results reasonably close to one another. The Pilgrim Station recirculation lines are very similar to those of the Dresden 2/3 Plants. The size of the pump suction, risers, header, and pump discharge are the same. The length and shape of components are nearly identical. Since the Pilgrim recirculation lines are similar in shape and size to the Dresden 2/3 recirculation lines, it can be concluded that applying Method II analysis to Pilgrim would result in stress differences similar to those obtained on Dresden 2/3 Plants. Therefore, continued use of the seismic results and techniques as given in the "Report on Dynamic Earthquake Analysis of the Recirculation Lines-Appendix A" of the Monticello FSAR, Volume IV (Method I) gives a conservative design of the recirculation piping

lines and their supports and is in basic agreement with methods suggested by others in reviewing the station seismic design.

The Dresden 2/3 reconfirmation of seismic design justification and conservatism was submitted to the AEC STAFF in October 1969. The same seismic techniques used on the above GE BWR's have been used on this facility.

Note: The method used to do the seismic analysis of the replacement Reactor Coolant Recirculation System loop lines and the connecting RHR and RWCU piping is described in paragraph 12.2.3.5.5.

J.4.6 Automatic Depressurization System - Single Component Failure Capability - Manual

Operation

Concern

"The automatic pressure relief subsystem should be modified so that at least the manual actuation of the subsystem would not be prevented by any single failure in the subsystem."[34]

Resolution

In order to provide an additional level of single component failure capability, the automatic pressure relief subsystem of the CSCS has been modified to provide the subsystem with the ability to sustain a dc power failure in any one of its two dc battery feeds. The subsystem has been designed and installed such that either one of the two redundant, independent 125 V dc battery system networks is available, automatically, for the required subsystem action. This modification provides the subsystem (when manually operated) with the single component failure criteria application capability.

J.4.7 Standby Gas Treatment System Design

Concern

"Several matters are still under discussion between the applicant and the Regulatory Staff. These include review of the need for separation of redundant components of the standby gas treatment system, and final revisions to the technical specifications. The ACRS believes these matters can be resolved by the applicant and the Regulatory Staff."[34]

Resolution

The Standby Gas Treatment System design provides both electrical and physical separation between the two treatment trains. This design provides the highest possible degree of independence, isolation, and redundancy between the two full capacity treatment trains. See Sections 5 and 8 for further details.

J.4.8 Flow Reference Scram

Concern

"In the area of reactor instrumentation, the Committee believes: that the flux scram point should be automatically reduced to an appropriate level as the reactor recirculation flow is reduced below the normal full-power flow."[28]

Resolution

Although the Applicant and the General Electric Company do not feel that a flow reference scram is required for safety purposes, the flow reference scram system is being designed and installed such that the flux scram point will be automatically reduced to an appropriate level as the reactor coolant recirculation flow is reduced below the normal full power flow.

The Flow Reference Scram System will sum the flow sensed in each of the reactor coolant recirculation loops and provide a flow reference signal to vary the neutron flux scram setpoint. Flow will be sensed from one flow measurement venturi in each of the two reactor coolant recirculation loops.

The station transient analyses (Section 14) demonstrated that, for all transients considered, the core is adequately protected with a fixed APRM scram trip setting at 120 percent of rated neutron flux and high pressure scram. Therefore, the Applicant proposes to ultimately replace the automatic flow referenced scram with a fixed 120 percent scram setting, providing that initial power operating confirms the nuclear behavior characteristics used in these transient analyses.

J.4.9 Development of Instrumentation - Primary Containment Leakage Detection System - Increased Sensitivity Studies

Concerns

"It is recommended that supplemental and potentially more sensitive methods of primary system leak detection be studied, evaluated, and implemented if they provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The study and evaluation should be completed within a year."[35]

Resolution

The primary containment leakage detection capability is described in Section 4.10. This system collects all the condensed leakage in the drywell floor drain sump and measures the discharge flow from the sump to the liquid radwaste system. The Applicant is confident that continuing gross leakage from the primary pressure boundary within the drywell will be detected with this method. In addition, the Applicant is studying other methods for improving the sensitivity and shortening

the time response of primary system leak rate detection through a combination of the following techniques:

1. An analytical study is in progress to define (as a function of leak rate and leak quality) the primary containment ambient pressure, temperature, and humidity.

2. The eight operating drywell unit coolers each have flow switches on their condensate drain lines that actuate at 2 gal/min. Early operating experience at the station is expected to show that these flow switches may be set to actuate at lower condensate flows to improve the sensitivity for detection of system leaks which raise the drywell humidity and increase the drywell cooler drain condensate flow rates.

3. A study of the effect of piping the normally expected valve stem leakoff to the drywell equipment drain sump as a means of identifying and quantifying a major expected source of primary system leakage is underway. If such modifications to the present design appear to promise a significant reduction in the "background" leak rate within the drywell, implementation of the modification at an early stage in the operating lifetime of the station will be considered.

4. Instrumentation is being provided within the drywell to monitor temperature and humidity as well as pressure. The Applicant intends to use this instrumentation to evaluate continuous containment atmospheric leakage rate monitoring procedures. These procedures require correction for the quantity of water vapor within the drywell and the Applicant intends to evaluate water vapor mass balances as a method for sensitive detection of primary system leakage.

J.4.10 Development of Instrumentation - Vibration and Loose Parts Detection Studies

Concern

"The applicant has stated that he plans to study possible means of instrumenting and monitoring for vibration or for the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system and, by the time of the first refueling outage, to implement such means as are found practical and appropriate."[36]

Resolution

Confirmatory vibration tests will be conducted on the Pilgrim reactor vessel internals. This program will supplement the data obtained from the Millstone 1 tests which were formulated in accordance with the acceptance criteria for testing a "first-of-a-kind" plant. See Quad-Cities Amendment 19 for a comprehensive description of the development of vibration testing of BWR reactor vessel internals. As is also indicated in Quad-Cities Amendment 19, the combined efforts of specialists at General Electric Research & Development Center,

Schenectady, and at the Atomic Power Equipment Department, San Jose, are being applied toward the development of a device to be mounted on the exterior of the reactor vessel which is capable of monitoring vibration of the reactor internal structure on a continuing and long term basis. The results of this program should be available by early 1972.

J.4.11 Core Standby Cooling System Leakage Detection, Protection, and Isolation Capability

Concern

"Engineered safety systems that are required to recirculate water after a loss-of-coolant accident should be designed so that a gross system leak will not result in critical loss of recirculation or in loss of isolation capability. The Committee believes that exception to this general rule may be made in respect to a very short run of pipe from the torus to the first valve, if extremely conservative design of the pipe (and its connection to the torus) is used and suitably remote operability of the valve is provided. The design of these systems also should provide adequate leak detection and surveillance capability."[28]

Resolution

The Pilgrim Station design provides a separate leakage detection capability in each CSCS pump compartment which alarm in the main control room in the event of water collection within the compartment. It is anticipated that adequate time would be available to close the motor operated isolation valves in the system if required to minimize continued leakage into the compartment.

The CSCS systems are provided with an automatic makeup capability to maintain the liquid inventory in the system in the event minor leakage develops when the system is in the standby mode.

The first isolation valve in each core spray and RHR pump suction line is a manually operated valve with an extension rod that permits the valve to be remotely operated from the reactor building floor above the suppression chamber.

See Sections 4, 6, and 10 for additional details.

J.4.12 Main Steam Lines - Standards for Fabrication, Quality Control, and Inspection

Concern

"The Committee has reviewed the applicant's proposal concerning standards of design, fabrication, and inspection of the steam lines downstream of the second isolation valve. The Committee concurs with the approach used in analyzing the stresses in the piping during an Operating Basis Earthquake. The Committee recommends that a program of spot radiography of the field butt welds be employed by the applicant as a quality control measure. Consideration should be given to an appropriate program of in-service inspection."[28]

Resolution

A program of 100 percent radiographic examination of welds on main steam line piping is being employed by the Applicant as a quality control measure. See Section 4.

J.5 SUMMARY CONCLUSIONS

The necessary research and development programs, additional information, or special analysis to support the application for a provisional operating permit for this station is discussed and justified in the preceding sections. Resolution of Pilgrim ACRS and AEC Staff concern items at the construction permit phase have been examined and the basis for their resolution is presented. See Section 1.11.

Thus, it is concluded that no further research and development or related activities are necessary for this facility in order to comply with the concerns and requirements identified during the review of the Pilgrim Station construction permit application.

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APPENDIX K

INSERVICE INSPECTION PROGRAM

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INSERVICE INSPECTION PROGRAM

K.1 GENERAL

K.1.1 Introduction and Summary

Construction Inservice Inspection Program

The Inservice Inspection Program set forth in Sections K.1 to K.5 and Amendment 20 represents the program implemented for the construction stages of the Pilgrim Nuclear Power Station located at Plymouth, Massachusetts. This program remains intact in the updated FSAR in order that NRC may locate previously submitted information in one document.

This requirement to maintain a historical record for the Construction Inservice Inspection Program is outlined in question/responses concerning the FSAR update rule (10CFR 50.17 (e) in NRC letter of December 15, 1980).

Operation Inservice Inspection Program

The requirement of the NRC as outlined in Sections K.1 to K.5 of Appendix K of the initial FSAR have been met with the ongoing Inservice Inspection Program. The present Inservice Inspection Program will be superseded by the next ten years interval plan.

The revised Inservice Inspection Program will have the necessary controls to comply with the requirement of the applicable portions of the Code of Federal Regulations.

Boston Edison provides a program for inservice inspection for Pilgrim Nuclear Power Station on those systems and components of the Reactor Coolant System boundary as defined in Section K.1.3..

The Pilgrim Station engineering design effort preceded the guidance provided by the ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems, and, therefore, some inaccessibility inherent in systems designs and equipment arrangements restricts the extent of inservice inspections. These inherent limitations are outlined in Section K.2.1.5, Systems Examination Exclusions.

However, an adequate inspection program has been developed by adopting, insofar as practicable, the principles and intent of the ASME Inservice Inspection Code as interpreted from the draft copy issued in October 1968.

Inspection procedures, in general, cover the portions of the pressure containing components up to and including the outermost containment isolation valve which could isolate the primary systems in the event of a Loss of Coolant Accident (LOCA). The program assumes that

inspections can be accomplished without the necessity of removing fuel elements solely for the purpose of nondestructive testing.

Certain nondestructive testing of materials, welds, and equipment were performed by qualified specialists from outside the company. Records and documentation of all information and inspection results which provide the basis for evaluation, and which facilitate comparison with results from subsequent inspections retained by Boston Edison Company for the active lifetime of the station, and are readily available for inspection purposes.

K.1.2 Program Purpose and Objectives

The Inservice Inspection Program for Pilgrim Nuclear Power Station is established to provide the following:

1. Reasonable assurance that no LOCA will occur at Pilgrim Station as a result of leakage or rupture of pressure containing components and piping of the reactor coolant system, portions of the reactor coolant associated auxiliary systems, portions of the emergency core cooling systems, and portions of the main steam and feedwater systems.

2. An initial reference base nondestructive examination of components and piping against which future examination determinations can be compared.

3. Sufficient nondestructive testing of systems and components during refueling outages to assure that the structural integrity of these systems and components remains unchanged from previous safe conditions, or that any observed changed conditions are acceptable for continued station operation.

4. Compliance with the principles and intent of the ASME Inservice Inspection Code insofar as practicable and possible by existing designs, equipment arrangements, access as available, and encountered radiation levels.

K.1.3 Definitions

K.1.3.1 Code

Code refers to the Draft Issue of ASME Code for Inservice Inspection of Nuclear Reactor Coolant Systems dated October 1968, and issued by the American Society of Mechanical Engineers, 345 East 47th Street, New York, New York.

K.1.3.2 System Boundary

System Boundary is defined as the outer limit of a particular process system wherein is contained certain portions of that system.

K.1.3.3 Reactor Coolant System

Reactor Coolant System (RCS) is that system which contains primary reactor coolant at operating pressure during normal reactor operations. The Reactor Coolant System is limited to the system boundaries given in Section K.2.1.

K.1.3.4 Class A Code Vessels

Class A Code Vessels are those pieces of equipment designed and constructed to provisions of the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels (Class A Vessels).

K.1.3.5 Inservice Inspection

Inservice Inspection denotes the program or action of examining a component or pipe by nondestructive test methods such as visual, radiographic, ultrasonic, liquid penetrant, and magnetic particle.

K.1.3.6 Ultrasonic Examination

Ultrasonic Examination - an electronic pulse echo method as described in Appendix IX - 340 Ultrasonic Examination of Welds of the ASME Boiler and Pressure Vessel Code, Section III, and in ASTM-E-114 and ASTM-A-435 Standards.

K.1.3.7 Liquid Penetrant Examination

Liquid Penetrant Examination - the surface detection of discontinuities open to the surface of ferrous and nonferrous materials which are nonporus, and performed in accordance with Appendix IX - 360 of Section III of the ASME Boiler and Pressure Vessel Code.

K.1.3.8 Radiographic Examination

Radiographic Examination - using energy sources such as x-ray, gamma ray, etc, in conjunction with radiographic film and performed in accordance with Appendix IX - 330 of Section III of the ASME Boiler and Pressure Vessel Code.

K.1.3.9 Magnetic Particle Examination

Magnetic Particle Examination - a method for the detection of rounded discontinuities, cracks, and other linear discontinuities in welds, plates, forgings, tabular products, and castings, and performed in accordance with Appendix IX - 350 of Section III of the ASME Boiler and Pressure Vessel Code.

K.1.3.10 Visual Examination

Visual Examination - that which is performed with the eye within 24 in of the examined surface and at an angle no less than 30 deg

with the examined surface, or by remote visual means such as TV cameras and monitoring systems, fiber optics, borescopes, telescopes, and periscopes where the resolution is equal to or greater than eyesight examination.

K.1.3.11 Safe End

Safe End - a metallic transition piece used to facilitate the welding of dissimilar metals in pipes, nozzles, valves, and fittings.

K.2 INSPECTION PROGRAM DEVELOPMENT

K.2.1 Systems Boundaries

The Inservice Inspection Program delineates the systems boundaries and contained piping and components which will be inspected and tested during the operating lifetime of the station. Primary consideration is given to the reactor coolant system, portions of the reactor coolant associated auxiliary systems, portions of the CSCS, and portions of the main steam and feedwater systems. In certain systems such as feedwater, RCIC, HPCI, and reactor water cleanup systems, isolation through the primary containment is accomplished by means of check valves which are included in the inspection program.

Information obtained from inservice inspections conducted during the first 5 yr of operation will be evaluated at the end of that period to take into account experience obtained as a result of the inspections performed. New inspection capabilities that may have become available during the first 5 yr period, will be reviewed to determine if a significant increase in the information available from the overall inservice inspection program would result from the use of new equipment and/or techniques.

K.2.1.1 Reactor Coolant System Boundary

The Reactor Coolant System (RCS) which contains primary reactor coolant at operating pressure during normal reactor operations is considered to extend out to and include the first containment isolation valve outside the primary containment in the main steam and reactor feedwater piping, and to include the RCS safety and relief valves.

K.2.1.2 Reactor Coolant Associated Auxiliary Systems

Associated reactor auxiliary systems in which reactor coolant is diverted from the RCS either continuously or intermittently in support of normal reactor operation are considered to extend out to and include the first containment isolation valve outside the primary containment. For piping of these reactor auxiliary systems which contain two valves, both of which are normally closed during normal operation, the boundary extends to and includes the second of the valves whether or not the system piping penetrates the primary containment.

K.2.1.3 Core Standby Cooling Systems

Core Standby Cooling Systems (CSCS) which are connected to the RCS and which penetrate the primary containment are considered to extend out to and include the first containment isolation valve outside the primary containment. For piping of these CSCSs which do not penetrate the primary containment, the boundary is considered to extend to and include the second of two valves normally closed during normal reactor operations.

K.2.1.4 Main Steam and Feedwater Systems

The main steam and feedwater systems which supply reactor feedwater and transport steam to the main turbine generator are considered to extend from and include the nearest isolation valve outside the primary containment in the case of the feedwater system, and out to and include the first containment isolation valve in the main steam lines outside the primary containment.

K.2.1.5 Systems Examination Exclusions

Those components and piping which are part of, and contained within the boundaries outlined in Sections K.2.1.1, K.2.1.2, K.2.1.3, and K.2.1.4, where postulated failure would not result in loss of reactor coolant that exceeds the capability of normal makeup systems, (as defined in Section 120 of the 1969 ISI Draft Code), for the interval of time required to permit the reactor shutdown and orderly cooldown from the respective conditions of startup, hot standby, operation, and cooldown, are excluded from Pilgrim Station's inservice inspection testing requirements as permitted by Section 120d of the Code.

The engineering, design, and construction of the station, in certain areas, do not permit compliance with that portion of Section 243 of the 1969 ISI Code, which requires that "where less than all the components are required to be inspected in the first inspection interval, a similar percentage of components not previously inspected (other than the preoperational examinations) shall be required in each successive interval."

No nondestructive inservice inspections are to be performed which require draining of the reactor vessel or removal of the core solely for the purpose of accomplishing testing. Inspections of interior surfaces and internal components of the reactor vessel, particularly in the lower regions, are in this category.

Items included in the Pilgrim program are based on the assumption that the areas involved in testing are reasonably safe for human access, and unless radiation turns out to be much higher than anticipated, testing will be accomplished as intended.

Access available inside containment penetrations, in general, preclude the testing of welded joints.

Volumetric examination by ultrasonic methods from external locations on the inner radius of reactor vessel nozzles are expected to be limited to less than 10 percent of its perimeter (those portions of the nozzle inner radius lying perpendicular to the reactor vessel centerline). However, these are believed to be the most highly stressed areas of the inner radius.

A considerable number of nozzle to pipe connections on the reactor vessel are made by means of safe ends using Inconel welding. We intend to subject these welded joints to ultrasonic examination even

though presently available methods produce results with questionable value. A more reasonable determination will be made after zero signature test results are analyzed.

Although we have made provisions for access to Category A welds, we do not except that it will be practical to examine these areas because of the high radiation exposure expected. An appropriate preservice baseline inspection of these welds is planned.

K.2.2 Systems Principal Pressure Containing Components and Piping

The principal pressure containing components and piping which are considered for inservice inspection, testing, and examination include the reactor pressure vessel and its appurtenances, pumps, primary pressure piping, and valves.

K.2.2.1 Reactor Pressure Vessel and Appurtenances

Components and appurtenances which were subjected to various nondestructive tests and examinations in and around the reactor pressure vessel included the following:

1. Reactor pressure vessel shell
2. Top vessel closure head
3. Reactor vessel nozzles and penetrations
4. Closure studs and nuts
5. Integrally welded vessel supports
6. Reactor vessel cladding
7. Closure head cladding

K.2.2.2 Pumps

The reactor recirculation pumps in the RCS, located inside the primary containment, are included in the inservice inspection program.

K.2.2.3 Primary Pressure Piping

Pressure piping up to and including the extents indicated in Section K.2.1, associated with the RCS, the reactor coolant associated auxiliary systems, the CSCS, and portions of the main steam and reactor feedwater systems were subjected to nondestructive testing and examination. Pipe and fittings within the applicable portions of the following systems were included, excepting areas through the containment penetrations:

1. Main steam lines
2. Reactor feedwater lines
3. Reactor recirculation lines
4. Control rod drive return lines
5. Residual heat removal system lines
6. Core spray system lines
7. High pressure coolant injection system steam and feed lines
8. Reactor core isolation cooling system steam and feed lines

9. Standby liquid control system lines
10. Cleanup demineralizer system lines

K.2.2.4 Valves

Containment isolation and other valves associated with the piping systems outlined in Section K.2.2.3 which are within the primary containment were subjected to inservice inspection as required.

K.3 INSPECTION PROGRAM IMPLEMENTATION

K.3.1 System Boundaries Delineation

The system boundary of the Reactor Coolant System (RCS) is that enclosing the reactor vessel and reactor vessel connected piping out to and including the first isolation valve outside the primary containment in the various systems which penetrate the primary containment.

K.3.1.1 Boundaries Diagram

Figure K.3-1 shows the components, piping and valves, within the primary containment along the portions of other systems for which inservice inspection facilities will be provided, insofar as engineering design and construction permits, and as discussed in Section K.2.

K.3.1.2 Major Test Areas Identification

Areas of the pressure containing components considered for examination and testing are those requiring examination because of their materials, geometry, stress levels, applied loads, environment, and type of fabrication.

The entire reactor vessel closure head is clad except in the area adjacent to the three low alloy steel nozzles. All carbon and low alloy steel nozzles exposed to the reactor coolant have a corrosion allowance of 1/16 in. The three nozzles in the closure head and the unclad areas adjacent to the nozzles do not experience high pressure steam flow during planned operation. No specific measurement of reactor vessel closure head or nozzle wall thickness was planned.

Itemized in Table K.3-1 are the areas to be examined and the methods employed to provide proper evaluation.

K.3.1.3 Frequency of Inspections

Reference is made to the applicable code examination categories, examination extent and frequency percentages, and nondestructive test methods to be considered.

TABLE K.3-1

INSERVICE INSPECTION PROGRAM

Section 1: REACTOR VESSEL AND CLOSURE HEAD

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
1.1	A	Circumferential shell welds in core region	Volumetric	None	5% of the length of the circumferential welds; welds; 10% of the length of the longitudinal welds	Although access has been provided through the sacrificial shield, the access is minimal. In addition, this is an early design and removal of the shielding blocks and insulation requires considerable labor. The required examinations will be made at or near the end of the 10-year inspection interval provided (1) radiation levels permit the inspections to be made without undue exposure of personnel, or (2) techniques or equipment are developed that allow the inspections to be made without undue exposure of personnel.
1.2	B	Circumferential welds in shell (other than those of Categories A and C)	Volumetric	5% of the length of the circumferential welds and 10% of the length of the longitudinal welds in the lower panel section	5% of the length of the circumferential welds; 10% of the length of the longitudinal welds	The required amount of weld lengths will be examined at or near the end of the 10-year inspection interval. Excluded are those welds in the lower head that lie within the CROM shroud assembly. Meridional and circumferential seam welds in the vessel bottom head are excluded.
1.3	C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	1/3 of the vessel-to-flange and 1/3 of the head-to-flange circumferential weld	Cumulative 100% of the vessel-to-flange weld and of the head-to-flange weld	Both of these welds are available for examination during normal refueling operations.
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused section	Volumetric	Inspection of five nozzle-to-shell welds and inner nozzle radii	Inspection of all nozzle-to-shell welds and inner radius sections	It is planned that the nozzles will be inspected from the OD. Although access to the nozzles has been provided through the sacrificial shield, the access

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
						is minimal. In addition, this is an early design and removal of the shielding blocks and insulation requires considerable labor. Inspection of the recirculation inlet and outlet nozzles will be made provided (1) radiation levels permit the inspections to be made without undue exposure of personnel, or (2) techniques or equipment are developed that allow the inspections to be made without undue exposure of personnel.
1.5	E-1	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds	(See Remarks)	(See Remarks)	(See Remarks)	The control rod drive housing to stub tube weld can be examined by removing control rod drive mechanisms, however with currently available techniques stub tube to vessel welds cannot be examined. CRDM housing supports limit mechanism ejection and limit flow such that these welds meet the exclusion criterion of ASME code, Section XI, paragraph ISI of 120(d).
1.6	E-2	Vessel penetrations, including control rod drive mechanism penetrations and control rod housing mechanism pressure boundary welds	Visual	10% of the control rod drive mechanism and instrumentation penetrations will be visually inspected for leakage	Cumulative 25% of the control rod drive mechanism and of the instrumentation penetrations will be visually inspected for leakage	None.
1.7	F	Primary nozzle-to-safe end welds	Visual and surface and volumetric	The dissimilar metal weld on five nozzles	All of the dissimilar metal welds on the vessel nozzles will be inspected	The dissimilar metal welds of each nozzle will be inspected at the same time as the nozzle-to-shell weld. Although access to the nozzles has been provided through the sacrificial shield, the access is minimal. In addition, this

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
						is an early design and removal of the shielding blocks and insulation requires considerable labor. Inspection of the recirculation inlet and outlet nozzle safe ends will be made provided (1) radiation levels permit the inspections to be made without undue exposure of personnel, or (2) techniques or equipment are developed that allow the inspections to be made without undue exposure of personnel.
1.8	G-1	Closure studs and nuts	Volumetric and visual or surface	Cumulative 50%	Cumulative 100%	100% visual each year for thread damage.
1.9	G-1	Ligaments between threaded stud holes	Volumetric	1/3 of the vessel-to-flange bolt ligaments	Cumulative 100% of the vessel flange bolt ligaments	The ligaments will be examined at the same time as the flange weld of Item 1.3.
1.10	G-1	Closure washers, bushings	Visual	Cumulative 50%	Cumulative 100%	None.
1.11	G-2	Pressure retaining bolting	(See Remarks)	(See Remarks)	(See Remarks)	There is no bolting less than 2 inches in diameter.
1.12	H	Integrally welded vessel supports	Volumetric	None	10% of the weld	None.
1.13	I-1	Closure head cladding	Visual and surface or volumetric	None	100% of selected areas areas at or near end of interval	During the 10-year period, at least 6 patches (each 36 square inches) in the vessel head would be inspected. A portion of the closure head is not clad. During the 10-year period, at least 6 points will be measured for thickness to determine the corrosion rate.
1.14	I-1	Vessel cladding	Visual	None	During the 10-year period, 6 patches (each 36 square inches)	None.

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
1.15	N	Interior surfaces and internals and integrally welded internal supports	Visual	A critical examination will be made of the interior surfaces made available by normal refueling operations at the first refueling cycle. This will be repeated at the fourth refueling cycle with the amount of inspection being dependent upon results of the first inspection and that made on other boiling water systems.	The inspections made at the fourth refueling cycle will be repeated at the seventh and tenth refueling cycle.	The examination will include internal support attachments welded to the vessel whose failure may adversely affect core integrity provided these are available for visual examination by components removed during normal refueling operations.
Section 4:	PIPING PRESSURE BOUNDARY					
4.1	F	Vessel, pump, and valve safe end-to-primary pipe welds and safe ends in branch piping welds	Visual and surface and volumetric	50% of dissimilar metal welds would be inspected	By the end of the interval, a cumulative 100% of the welds would have been inspected	See remarks of Item 1.7.
4.2	J	Circumferential and longitudinal pipe welds	Visual and volumetric	15% of the butt welds, including one foot of any longitudinal weld on either side of the butt weld	By the end of the interval, a cumulative 25% of the butt welds in the piping system would have been inspected, including one foot of any longitudinal weld on either side of the butt welds	None
4.3	G-1	Pressure retaining bolting	(See Remarks)	(See Remarks)	(See Remarks)	There is no bolting 2 inches and larger in the piping system.
4.4	G-2	Pressure retaining	Visual	50% of the bolt-	By the end of the in-	All bolting is below 2 inches

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
		bolting		ing would be inspected	terval, 100% of the bolting would be inspected	in diameter and would be visually inspected, either in place if the bolted connection is not disassembled during the inspection interval, or whenver the bolted connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of IS-120(d).
4.5	K-1	Integrally welded supports	Visual and volumetric	15% of the supports	25% of the supports	None.
4.6	K-2	Piping support and hanger	Visual	50% of the supports would be examined	By the end of the interval, a cumulative 100% of the supports would be examined	The support members and structures subject to inspection would incude those supports within the system whose structural integrity is relied upon to withstand the design loads and seismic-induced displacements. The support settings of constant and variable spring-type hangers, snubbers and shock absorbers would be inspected to verify proper distribution of design loads among the associated support components.
Section 5:		PUMP PRESSURE BOUNDARY				
5.1	L-1	Pump casing welds	(See Remarks)	(See Remarks)	(See Remarks)	There are no pumps with pressure containing welds.
5.2	L-2	Pump casings	Visual	None	By the end of the interval, a cumulative 100% of the available inner surfaces of the required pumps would be inspected if the pumps are disassembled for	The only pumps involved in this program are the recirculation pumps.

TABLE K.3-1 (Cont)

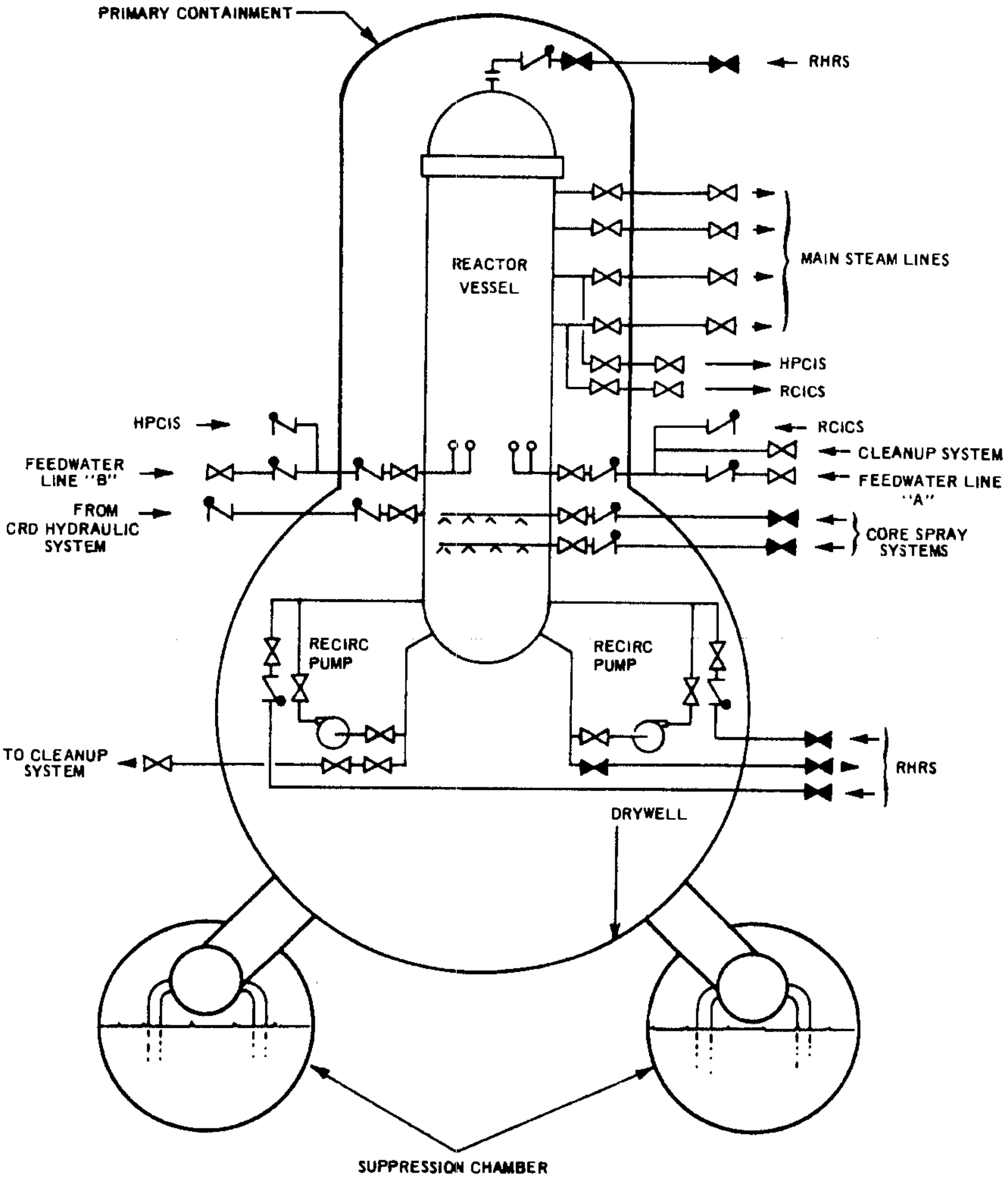
Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
					maintenance and the rotating elements are removed.	
5.3	F	Nozzle-to-safe end welds	(See Remarks)	(See Remarks)	(See Remarks)	There are no dissimilar metal welds on the pumps.
5.4	G-1	Pressure retaining bolting	Visual and volumetric	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	Bolting 2 inches and larger in diameter would be inspected either in place under tension, or when the bolting is removed or when the bolting connection is disassembled. The bolting and areas to be inspected would include the studs, nuts, bushings, threads in the base material, and the flange ligaments between threaded stud holes.
5.5	G-2	Pressure retaining bolting	Visual	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	Bolting below 2 inches in diameter would be visually inspected either in place if the bolting connection is not disassembled during the inspection interval, or whenever the bolted connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of paragraph IS-120(d).
5.6	K-1	Integrally welded supports	(See Remarks)	(See Remarks)	(See Remarks)	The pumps do not have integrally welded supports.
5.7	K-2	Supports and hangers	Visual	50% of the supports would be inspected	Cumulative 100% of the supports would be inspected	None.

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
Section 6:		VALVE PRESSURE BOUNDARY				
6.1	M-1	Valve body welds	(See Remarks)	(See Remarks)	(See Remarks)	There are no valves with pressure retaining welds in the valve bodies.
6.2	M-2	Valve bodies	Visual	50% of the valves required by the Code would be inspected	By the end of the interval, a cumulative 100% of the valves required by the Code would be inspected	Excluded from inspection are the valves in the recirculation piping. These would only be inspected if the reactor were drained and defueled for other purposes.
6.3	F	Valve-to-safe end welds	(See Remarks)	(See Remarks)	(See Remarks)	There are no valves in this system with dissimilar metal welds.
6.4	G-1	Pressure retaining bolting	(See Remarks)	(See Remarks)	(See Remarks)	There are no valves with bolting 2 inches and larger in diameter.
6.5	G-2	Pressure retaining bolting	Visual and volumetric	50% of the bolting would be inspected	By the end of the interval, a cumulative 100% of the bolting would be inspected	All bolting is below 2 inches in diameter and would be visually inspected, either in place if the bolting connection is not disassembled at the inspection interval, or whenever the bolting connection is disassembled. The bolting to be inspected would include studs and nuts. Excluded from inspection are bolting of a single connection whose failure results in conditions that satisfy the exclusion criteria of paragraph IS-120(d).
6.6	K-1	Integrally welded supports	(See Remarks)	(See Remarks)	(See Remarks)	There are no valves with integrally welded supports.

TABLE K.3-1 (Cont)

Item No.	Category	Examination Area	Examination Method	Inspection During 5-Year Interval	Tentative Inspection During 10-Year Interval	Remarks
6.7	K-2	Supports and hangers	Visual	50% of the supports and hangers would be inspected	By the end of the interval, a cumulative 100% of the supports and hangers would be inspected	The support members and structures subject to inspection would include those supports for piping, valves, and pumps within the system boundary, whose structural integrity is relied upon to withstand the design loads and seismic-induced displacements. The support settings of constant and variable spring-type hangers, snubbers, and shock absorbers would be inspected to verify proper distribution of design loads among the associated support components.



NOTE: THE PRIMARY SYSTEM BOUNDARY INCLUDES THE FIRST ISOLATION VALVE OUTSIDE THE PRIMARY CONTAINMENT

FIGURE K. 3-1
PRIMARY SYSTEM BOUNDARY
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

K.4 REFERENCE BASE EXAMINATIONS

K.4.1 Reactor Vessel Reference Base

Prior to fuel loading of the Pilgrim reactor vessel, a reference base nondestructive examination was performed to establish a base record against which future inservice inspection results could be compared to determine the integrity of the various included items throughout their lifetime.

Reference base examinations of the reactor vessel welds were by visual, dye penetrant, or magnetic particle and ultrasonic test methods because of the impracticability of radiographic methods after initial operation. Areas intended for inservice inspection testing were covered during the reference base examinations. These areas are outlined on Table K.3-1. No formal inservice inspection program was planned for systems beyond the limits of the reactor coolant pressure boundary as described on Table K.3-1. Planned surveillance testing of vital systems will normally include visual inspection of leakage when access is not precluded by high radiation condition.

K.4.2 Reactor Coolant System Reference Base

Preoperational base reference examinations of welded joints and components within the reactor coolant system were conducted prior to initial operation and were intended to be as closely representative of future examinations as is practicable.

K.4.3 Reactor Coolant Associated Auxiliary Systems Base

Portions of the Reactor Coolant Associated Auxiliary Systems, located within the primary containment and connected to the reactor vessel, were given base reference examinations.

K.4.4 Core Standby Cooling Systems Base

Piping, valves, and components associated with the Core Standby Cooling System located within the primary containment and connected to the reactor vessel were given base reference examinations.

K.4.5 Main Steam Feedwater Systems Base

Piping and valves in the Main Steam and Reactor Feedwater System were included in the inservice inspection program where these systems are located in the primary containment. Portions of both piping systems outside the primary containment were also included in the base reference examination to the limits as defined in Section K.2.

K.5 DOCUMENTATION AND RECORDS

Documentation and records of plans, schedules, and inspection reports concerned with preservice and inservice inspection are compiled and maintained by Boston Edison throughout the life of Pilgrim Station.

In general, the minimum requirements for documentation by Boston Edison are those references in the ASME-ISE Code and include full documentation of all of the preoperational base examination data and inservice inspection records of test performed. Comparative analyses reports form part of the documentary effort, in addition to corrective action reports and repair procedures where required. Originals of all inservice inspection records are maintained in a central location.

K.5.1 Preoperational Base Examination Records

Records outlining the components and piping to be examined at future dates were compiled prior to the initial operating date. These records include the components and areas subject to examination, the sampling selection, the extent of examination, the test methods employed, and the inspection frequency.

K.5.2 Inservice Inspection Records

Inservice inspection plans, procedures, methods, testing data, comparative analyses reports, and corrective actions, where required, are documented and maintained by Boston Edison.

Evaluation of recorded data from the tests is related to the original acceptance standards of the applicable design, manufacturing, and construction codes.

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CONTAINMENT REPORT

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CONTAINMENT REPORT

L.1 INTRODUCTION AND SUMMARY

The purpose of this Appendix is to provide technical information concerning the primary containment for Pilgrim Nuclear Power Station.

This appendix describes the design bases, design evaluation, loads, load combinations, and initial leak rate test of the primary containment (drywell and suppression chamber) vessels for Pilgrim Nuclear Power Station.

Section 5.6, Containment, presents the function and the design bases of the primary containment. This Appendix furnishes additional design data for the containment.

The Primary Containment System consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and water pool, isolation valves, containment cooling systems, and other service equipment. The Pressure Suppression Containment System is shown on Figure L.1-1.

The performance criteria and design information regarding the isolation valves, containment cooling systems, and other service equipment are included elsewhere in this Final Safety Analysis Report.

The Reactor Building encloses the reactor and the primary containment. This structure provides secondary containment when the primary containment is in service, and serves as the containment during periods when the primary containment is open. A detailed description of the secondary containment is included in Section 5.3, Secondary Containment.

Figure L.1-1 has been removed.

Please refer to BECo Controlled Drawing C1A 1-8.

L.2 BASIS FOR CONTAINMENT DESIGN

Sections 5.2.1 and 5.2.2 describe the safety objectives and safety design bases of the primary containment.

Section 5.2.3 gives the detailed description of the primary containment vessel. It discusses the internal and external pressures, temperatures, and other parameters to which the containment is designed. Section L.3 of this Appendix lists the loads and load combinations used for the structural design of the containment vessels. Also listed in this Section are calculated stresses of major components.

The drywell shell, pressure suppression chamber, and the penetrations together form the Containment and Suppression System.

Table L.2-1 lists the penetrations, with their functions and sizes, for the containment vessel.

Figures L.2-1 through L.2-4 show the drywell shell stretchout and penetration locations. Figures L.2-5 through L.2-24 show the details of the containment vessel and major components where stresses have been listed in Section L.3.

A complete set of detail drawings will be on file at the Boston Edison Company, Boston, Massachusetts.

Section L.4 of this Appendix describes the leak rate tests of the containment.

TABLE L.2-1

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in.)	Description
X-1	Special	1	-	10 ft id	2-1/2	Equipment Hatch
X-2	Special	1	-	8 ft 4 in id	3/8	Personnel Lock
X-4	Special	1	-	24 id	-	Head Access Hatch
X-5A	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5B	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5C	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5D	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5E	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5F	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5G	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-5H	Special	1	-	-	2-1/2 to 1/4	Vent Line
X-6	Special	1	-	24 id	1	Drive Removal Line
X-7A	1	1	20	36	2	Primary Steam
X-7B	1	1	20	36	2	Primary Steam
X-7C	1	1	20	36	2	Primary Steam
X-7D	1	1	20	36	2	Primary Steam
X-8	1	1	3	16	7/8	Steam Drain
X-9A	1	1	18	34	2	Primary Feedwater
X-9B	1	1	18	34	2	Primary Feedwater
X-12	1	1	20	36	2	RHR Suction
X-14	1	1	6	24	1-1/4 to 1/2	Cleanup Supply
X-15A	3	1	-	22	1-1/8 to 1/2	Spare
X-15B	3	1	-	22	1-1/8 to 1/2	Spare
X-15C	3	1	-	22	1-1/8 to 1/2	Spare
X-15D	3	1	-	22	1-1/8 to 1/2	Spare
X-15E	3	1	-	22	1-1/8 to 1/2	Hydrogen Analyzer
X-15F	3	1	-	22	1-1/8 to 1/2	Spare
X-15G	3	1	-	22	1-1/8 to 1/2	Drywell Pressure
X-15H	3	1	-	22	1-1/8 to 1/2	Spare
X-15I	3	1	-	22	1-1/8 to 1/2	Spare
X-15J	3	1	-	22	1-1/8 to 1/2	Spare
X-15K	3	1	-	22	1-1/8 to 1/2	Spare
X-15L	3	1	-	22	1-1/8 to 1/2	Spare
X-15M	3	1	-	22	1-1/8 to 1/2	Spare
X-16A	1	1	10	26	1-1/2 to 1/2	Core Spray
X-16B	1	1	10	26	1-1/2 to 1/2	Core Spray

TABLE L.2-1 (Cont)

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in)	Penetration Nominal Sleeve Size (Unless Noted) (in)	Sleeve Thickness (in)	Description
X-17	1	1	4	18	1	Spare
X-18	2A	1	2	6	Sch.80	Floor Drain
X-19	2A	1	2	6	Sch.80	Equipment Drain
X-20	3	1	-	6	Sch.80	Spare
X-21	2A	1	2	6	Sch.80	Spare
X-22	2B	1	3	3	Sch.80	Instrument Air
X-23	2A	1	6	12	Sch.80	RBCCW Supply
X-24	2A	1	6	12	Sch.80	RBCCW Return
X-25	2B	1	18	18	1	Vent from Drywell
X-26	2B	1	18	18	1	Vent to Drywell
X-27A	3	1	1	10	Sch.80	Core differential pressure
X-27B	3	1	1	10	Sch.80	Core differential pressure
X-27C	3	1	1	10	Sch.80	Recirc. diff. press. to RHR Loop Select
X-27D	3	1	1	10	Sch.80	Recirc. diff. press. to RHR Loop Select
X-27E	3	1	1	10	Sch.80	Recirc. diff. press. to RHR Loop Select
X-27F	3	1	1	10	Sch.80	Recirc. diff. press. to RHR Loop Select
X-28A	3	1	1	10	0.593	Core Spray Sample
X-28B	3	1	1	10	0.593	RPV Level
X-28C	3	1	1	10	0.593	RHR Interlock
X-28D	3	1	1	10	0.593	Drywell Pressure
X-28E	3	1	1	10	0.593	RPV Flange Seal Leak Detector
X-28F	3	1	1	10	0.593	RPV Wide-Range Level
X-29A	3	1	1	10	0.593	RPV Level & Pressure
X-29B	3	1	1	10	0.593	RPV Level & Pressure
X-29C	3	1	1	10	0.593	RPV Level & Pressure
X-29D	3	1	1	10	0.593	Drywell Pressure
X-29E	3	1	1	10	0.593	H_2O_2 Analyzer & PASS
X-29F	3	1	1	10	0.593	Spare
X-30A	3	1	1	10	Sch.80	Spare
X-30B	3	1	1	10	Sch.80	Spare
X-30C	3	1	1	10	Sch.80	Spare
X-30D	3	1	1	10	Sch.80	Spare
X-30E	3	1	1	10	Sch.80	Spare
X-30F	3	1	1	10	Sch.80	Spare
X-31A	3	1	1	10	Sch.80	Spare
X-31B	3	1	1	10	Sch.80	LPCI Injection
X-31C	3	1	1	10	Sch.80	Recirc. Pump B Flow
X-31D	3	1	1	10	Sch.80	Recirc. Pump B Flow
X-31E	3	1	1	10	Sch.80	Recirc. Pump B Diff. Press.
X-31F	3	1	1	10	Sch.80	LPCI Injection
X-32A	3	1	1	10	Sch.80	Spare
X-32B	3	1	1	10	Sch.80	Recirc. Pump A Flow
X-32C	3	1	1	10	Sch.80	Recirc. Pump A Diff. Press.
X-32D	3	1	1	10	Sch.80	LPCI Injection
X-32E	3	1	1	10	Sch.80	Recirc. Pump A Diff. Press.
X-32F	3	1	1	10	Sch.80	Recirc. Pump A Flow

TABLE L.2-1

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in.)	Description
X-33A	3	1	1	10	Sch.80	Recirc. Pump Seal Leak Detector
X-33B	3	1	1	10	Sch.80	Recirc. Pump Seal Leak Detector
X-33C	3	1	1	10	Sch.80	Core Spray Alarm
X-33D	3	1	1	10	Sch.80	Spare
X-33E	3	1	1	10	Sch.80	Core Spray Alarm
X-33F	3	1	1	10	Sch.80	Spare
X-34A	3	1	1	10	Sch.80	Spare
X-34B	3	1	1	10	Sch.80	Spare
X-34C	3	1	1	10	Sch.80	Main Steam Line C Flow & Isolation
X-34D	3	1	1	10	Sch.80	Main Steam Line D Flow & Isolation
X-34E	3	1	1	10	Sch.80	Main Steam Line D Flow & Isolation
X-34F	3	1	1	10	Sch.80	Main Steam Line C Flow & Isolation
X-35A	2A	1	3/8	1-1/2	Sch.80	TIP Drive
X-35B	2A	1	3/8	1-1/2	Sch.80	TIP Drive
X-35C	2A	1	3/8	1-1/2	Sch.80	TIP Drive
X-35D	2A	1	3/8	1-1/2	Sch.80	TIP Drive
X-35E	2A	1	3/8	1-1/2	Sch.80	TIP Drive
X-36	Capped	1		6	Sch.80	Spare
X-37A	3	41*	1	1	Sch.80	CRD Insert
X-37B	3	34*	1	1	Sch.80	CRD Insert
X-37C	3	34*	1	1	Sch.80	CRD Insert
X-37D	3	40*	1	1	Sch.80	CRD Insert
X-38A	3	41*	1	1	Sch.80	CRD Withdraw
X-38B	3	34*	1	1	Sch.80	CRD Withdraw
X-38C	3	34*	1	1	Sch.80	CRD Withdraw
X-38D	3	40*	1	1	Sch.80	CRD Withdraw
X-39A	3	1	10	10	Sch.80	Containment Cooling
X-39B	3	1	10	10	Sch.80	Containment Cooling
X-40A-a	3	1	1	10	Sch.80	Jet Pump Instr. and PASS
X-40A-b	3	1	1	10	Sch.80	Jet Pump Instr.
X-40A-c	3	1	1	10	Sch.80	Jet Pump Instr.
X-40A-d	3	1	1	10	Sch.80	Jet Pump Instr.
X-40A-e	3	1	1	10	Sch.80	Jet Pump Instr.
X-40A-f	3	1	1	10	Sch.80	Jet Pump Instr.
X-40B-a	3	1	1	10	Sch.80	Jet Pump Instr.
X-40B-b	3	1	1	10	Sch.80	Jet Pump Instr.
X-40B-c	3	1	1	10	Sch.80	Jet Pump Instr.

* Includes one spare penetration

TABLE L.2-1 (Cont)

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in)	Penetration Nominal Sleeve Size (Unless Noted) (in)	Sleeve Thickness (in)	Description
X-40B-d	3	1	1	10	Sch.80	Jet Pump Instr.
X-40B-e	3	1	1	10	Sch.80	Jet Pump Instr.
X-40B-f	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-a	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-b	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-c	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-d	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-e	3	1	1	10	Sch.80	Jet Pump Instr.
X-40C-f	3	1	1	10	Sch.80	Jet Pump Instr.
X-40D-a	3	1	1	10	Sch.80	Jet Pump Instr.
X-40D-b	3	1	1	10	Sch.80	Jet Pump Instr.
X-40D-c	3	1	1	10	Sch.80	Jet Pump Instr. and PASS
X-40D-d	3	1	1	10	Sch.80	Jet Pump Instr.
X-40D-e	3	1	1	10	Sch.80	Jet Pump Instr.
X-40D-f	3	1	1	10	Sch.80	Jet Pump Instr.
X-41A	3	1	3/4	6	Sch.80	Recirc. Sample
X-41B	1	1	-	6		Spare
X-41C	1	1	-	6		Spare
X-42	2A	1	1-1/2	4	Sch.80	Standby Liquid Control
X-43	3	1	-	8	Sch.80	Drywell Test Connection
X-44	2	1	-	26	1-1/2	Spare
X-46A	3	1	-	12	Sch80	Recirc. Pump Seal Purge
X-46B	3	1	-	12	Sch80	Recirc. Pump Seal Purge
X-46C	3	1	-	12	Sch80	Spare
X-46D	3	1	-	12	Sch80	Drywell Pressure
X-46E	3	1	-	12	Sch80	Backup N_2 Supply for RV-203-3B & 3C
X-46F	3	1	-	12	Sch80	H_2O_2 Anlayzer and PASS
X-47	3	1	-	6	Sch80	Drywell Test Connection
X-49A	3	1	1	10	Sch.80	Recirc.Pump Seal
X-49B	3	1	1	10	Sch.80	Recirc.Pump Seal
X-49C	3	1	1	10	Sch.80	Recirc. Diff. Press.to RHR Loop Selection
X-49D	3	1	1	10	Sch.80	RCIC Steam Line High Flow
X-49E	3	1	1	10	Sch.80	Recirc. Diff. Press.to RHR Loop Selection
X-49F	3	1	1	10	Sch.80	RCIC Steam Line High Flow
X-50A-a	3	1	1	10	Sch.80	Spare
X-50A-b	3	1	1	10	Sch.80	Spare
X-50A-c	3	1	1	10	Sch.80	RPV Pressure to RHR Vlv Closure

TABLE L.2-1

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in.)	Description
X-50A-d	3	1	1	10	Sch.80	H_2O_2 Analyzer, PASS, Leak Detection
X-50A-e	3	1	1	10	Sch.80	Spare
X-50A-f	3	1	1	10	Sch.80	Spare
X-50B-a	3	1	1	10	Sch.80	HPCI Steam Line Break
X-50B-b	3	1	1	10	Sch.80	HPCI Steam Line Break
X-50B-c	3	1	1	10	Sch.80	Recirc. Diff. Press.to RHR Loop Selection
X-50B-d	3	1	1	10	Sch.80	Recirc. Diff. Press.to RHR Loop Selection
X-50B-e	3	1	1	10	Sch.80	Spare
X-50B-f	3	1	1	10	Sch.80	Spare
X-51A	1	1	18	34	2	RHR System Return
X-51B	1	1	18	34	2	RHR System Return
X-52	1	1	10	26	1-1/2	HPCI Steam to Turbine
X-53	1	1	3	18	1	RCIC Steam to Turbine
X-54A	3	1	1	10	Sch.80	MSL High Flow Isolation
X-54B	3	1	1	10	Sch.80	MSL High Flow Isolation
X-54C	3	1	1	10	Sch.80	MSL High Flow Isolation
X-54D	3	1	1	10	Sch.80	MSL High Flow Isolation
X-54E	3	1	1	10	Sch.80	RPV Pressure Vlv Closure
X-54F	3	1	1	10	Sch.80	Spare
X-82A	Special	1	1	2	Sch.80	RPV Level & Pressure
X-82B	Special	1	1	2	Sch.80	RPV Level & Pressure
X-100A	Elec.	1	-	12	Sch.80	Neutron Monitor System
X-100B	Elec.	1	-	12	Sch.80	Neutron Monitor System
X-100C	Elec.	1	-	12	Sch.80	Neutron Monitor System
X-100D	Elec.	1	-	12	Sch.80	Neutron Monitor System
X-100E	Elec.	1	-	12	Sch.80	Thermocouples & Computer Signal
X-101A	Elec.	1	-	12	Sch.80	Power
X-101B	Elec.	1	-	12	Sch.80	Power
X-101C	Elec.	1	-	12	Sch.80	Power
X-101D	Elec.	1	-	12	Sch.80	High Range Radiation Monitor
X-102A	Elec.	1	-	12	Sch.80	Indication and Control
X-102B	Elec.	1	-	12	Sch.80	Indication and Control
X-103A	Elec.	1	-	12	Sch.80	Indication and Control
X-103B	Elec.	1	-	12	Sch.80	Thermocouples & Computer Signal
X-104A	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104B	Elec.	1	-	12	Sch.80	Reactor Manual Control

TABLE L.2-1 (Cont)

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in)	Penetration Nominal Sleeve Size (Unless Noted) (in)	Sleeve Thickness (in)	Description
X-104C	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104D	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104E	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104F	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104G	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104H	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-104J	Elec.	1	-	12	Sch.80	Reactor Manual Control
X-105A	Elec.	1	-	12	Sch.80	Power
X-105B	Elec.	1	-	12	Sch.80	Power
X-105C	Elec.	1	-	12	Sch.80	High Range Radiation Monitor
X-106A-a	3	1	1	12	Sch.80	Recirc. Line Sample
X-106A-b	3	1	1	12	Sch.80	H_2O_2 Analyzer
X-106A-c	3	1	1	12	Sch.80	Core Flow Pressure Switch
X-106A-d	3	1	1	12	Sch.80	RWCU Diff. Press.
X-106A-e	3	1	1	12	Sch.80	RWCU Diff. Press.
X-106A-f	3	1	1	12	Sch.80	Drywell Pressure
X-106B	Elec.	1	-	12	Sch.80	600V Indication and Control
X-200A	Special	1	-	48 id	1-1/2	Access Hatch
X-200B	Special	1	-	48 id	1-1/2	Access Hatch
X-201A	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201B	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201C	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201D	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201E	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201F	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201G	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-201H	Special	1	6 ft 9 in	7 ft 3-3/8 id	1-3/4	Vent Line
X-202A	Elec.	1	-	12	0.687	Indication Power-Lights, Dew Point, Temp.
X-202B	Elec.	1	-	12	0.687	Indication Power-Lights, Dew Point, Temp.
X-203A	2B	1	10	18 id	1	Vacuum Breakers
X-203B	2B	1	10	18 id	1	Vacuum Breakers
X-203C	2B	1	10	18 id	1	Vacuum Breakers
X-203D	2B	1	10	18 id	1	Vacuum Breakers
X-203E	2B	1	10	18 id	1	Vacuum Breakers
X-203F	2B	1	10	18 id	1	Vacuum Breakers
X-203G	2B	1	10	18 id	1	Vacuum Breakers
X-203H	2B	1	10	18 id	1	Vacuum Breakers
X-203I	-	-	-	-	-	Not Used

TABLE L.2-1

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in.)	Description
X-203J	2B	1	10	18 id	1	Vacuum Breakers
X-203K	2B	1	10	18 id	1	Vacuum Breakers
X-205	2B	1	20	20	1-1/4	Vacuum Relief from Bldg. & Purge Inlet
X-206A	2B	1	1	1	Sch.80	Torus Water Level Indication
X-206B	2B	1	1	1	Sch.80	Torus Water Level Indication
X-206C	2B	1	1	1	Sch.80	Torus Water Level Indication
X-206D	2B	1	1	1	Sch.80	Torus Water Level Indication
X-207A	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207B	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207C	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207D	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207E	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207F	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207G	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-207H	2B	1	1	1	Sch.80	Vent Line Drain (Capped)
X-208A	2B	1	12	12	Sch.40	Relief Valve Discharge
X-208B	2B	1	12	12	Sch.40	Relief Valve Discharge
X-208C	2B	1	12	12	Sch.40	Relief Valve Discharge
X-208D	2B	1	12	12	Sch.40	Relief Valve Discharge
X-209A	2B	1	1	1	Sch. 80	Spare
X-209B	2B	1	1	1	Sch. 80	Torus Water Temperature
X-209C	2B	1	1	1	Sch. 80	Spare
X-209D	2B	1	1	1	Sch. 80	Torus Water Temperature
X-210A	Special	1	12	16	1/2	Containment Cooling & CoreSpray Test Line
X-210B	Special	1	12	16	1/2	Containment Cooling & CoreSpray Test Line
X-211A	2B	1	6	6	Sch.80	Containment Cooling to Spray Header
X-211B	2B	1	6	6	Sch.80	Containment Cooling to Spray Header
X-212	Capped	1	-	18	1	Spare
X-213A	2B	1	8	8	Sch.80	Construction Drain
X-213B	2B	1	8	8	Sch.80	Construction Drain
X-214	Capped	1	-	4	Sch.80	Spare
X-215	Capped	1	-	4	Sch.80	Spare
X-216	Capped	1	-	2	Sch.80	Spare
X-217	Capped	1	-	2	Sch.80	Spare
X-218	Capped	1	-	10	0.594	Spare
X-219	5	1	-	10	0.594	HPCI Turbine Exhaust Vacuum and Hydrogen recombiner vent

TABLE L.2-1 (Cont)

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in)	Description
X-220	2B	1	6	6	Sch.80	RCIC Pump Suction
X-221	2B	1	16	16	7/8	HPCI Pump Suction
X-222A	2B	1	18	18	1	RHR Pump Suction
X-222B	2B	1	18	18	1	RHR Pump Suction
X-222C	2B	1	18	18	1	RHR Pump Suction
X-222D	2B	1	18	18	1	RHR Pump Suction
X-223	2B	1	24	24	1-1/4	HPCI Turbine Exhaust
X-224	2B	1	2	2	Sch.80	HPCI Turbine Exhaust Drain
X-225	2B	1	8	8	Sch.80	RCIC Turbine Exhaust
X-226	2B	1	2	2	Sch.80	RCIC Turbine Exhaust Drain
X-227	2B	1	20	20	1-1/4	Torus Purge Exhaust, Vacuum Relief, and Direct Torus Vent
X-228A	2B	1	1	1	Sch.80	Spare
X-228B	2B	1	1	1	Sch.80	Torus Pressure
X-228C	2B	1	1	1	Sch.80	H_2O_2 Analyzer and PASS Supply
X-228D	2B	1	1	1	Sch.80	Spare
X-228E	2B	1	1	1	Sch.80	Vacuum Breaker Air Supply
X-228F	2B	1	1	1	Sch.80	Spare
X-228G	2B	1	1	1	Sch.80	PASS Liquid Sample Return
X-228H	2B	1	1	1	Sch.80	PASS Liquid Sample Return
X-228J	2B	1	1	1	Sch.80	H_2O_2 Analyzer and PASS Supply
X-228K	2B	1	1	1	Sch.80	H_2O_2 Analyzer and PASS Return
X-229A	2B	1	18	18	1	Core Spray Pump Suction
X-229B	2B	1	18	18	1	Core Spray Pump Suction
X-230	3	1	-	8	Sch.80	Spare
X-240A	Special	1	1	1	Sch.80	Torus Level Transmitter
X-240B	Special	1	1	1	Sch.80	Torus Level Transmitter
X-241A	Special	1	1	1	Sch.80	Torus Level Transmitter
X-241B	Special	1	1	1	Sch.80	Torus Level Transmitter
X-242A	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242B	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242C	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242D	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242E	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242F	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242G	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242H	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242J	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242K	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242L	Special	1	7/8	15/16	.120	Torus Bulk Temperature

TABLE L.2-1

PENETRATION SCHEDULE

Penetration Number	Penetration Type	Number Required	Line Size (in.)	Penetration Nominal Sleeve Size (Unless Noted) (in.)	Sleeve Thickness (in.)	Description
X-242M	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242N	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242P	Special	1	7/8	15/16	.120	Torus Bulk Temperature
X-242Q	Special	1	7/8	15/16	.120	Abandoned in Place
X-242R	Special	1	7/8	15/16	.120	Abandoned in Place
X-243A	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243B	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243C	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243D	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243E	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243F	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243G	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243H	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243J	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243K	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243L	Special	1	7/8	15/16	.120	Torus Local Temperature
X-243M	Special	1	7/8	15/16	.120	Torus Local Temperature

The following FSAR figures have been removed from the FSAR. Please refer to the corresponding BECo controlled drawing.

FSAR FIGURE	BECO CONTROLLED DRAWING
L.2-1	C1A 2-11
L.2-2	C1A 3-5
L.2-3	C1A 4-8
L.2-4	C1A 5-11
L.2-5	C1A 6-8
L.2-6	C1A 7-6
L.2-7	C1A 8-7
L.2-8	C1A 9-7
L.2-9	C1A 10-4
L.2-10	C1A 12-5
L.2-11	C1A 13-4
L.2-12	C1A 14-4
L.2-13	C1A 15-6
L.2-14	C1A 16-4
L.2-15	C1A 17-5
L.2-16	C1A 19-10
L.2-17	C1A 20-7
L.2-18	C1A 21-5
L.2-19	C1A 50-5
L.2-20	C1A 51-7
L.2-21	C1A 52-5
L.2-22	C1A 58-4
L.2-23	C1A 62-4
L.2-24	C1A 100-3
L.2-25	C1A 185

L.3 CONTAINMENT SYSTEM DESIGN

L.3.1 General

Chicago Bridge and Iron Company (CB&I) designed, fabricated, furnished, installed, and tested the containment vessel and connecting vent piping, including bellows, jet deflectors, penetration sleeves, vessel supports, and other appurtenances. This was accomplished in accordance with Bechtel Corporation specifications.

The information in this report pertaining to the detailed design of the Pressure Containment System was taken from CB&I's Certified Stress Report, which is on file at the Boston Edison Company, Boston, Massachusetts.

The suppression pool portion of the containment has been reanalyzed by Teledyne Engineering Services using loads generated by the GE Mark I Long Term Program.

Field painting of the drywell and suppression chamber was accomplished under Bechtel specifications.

L.3.2 Design of Drywell, Pressure Suppression Chamber, and Connecting Vent System

L.3.2.1 General Description and Dimensions

The Pressure Suppression Containment System consists of a drywell, a pressure suppression chamber which stores a large volume of water, and a connecting vent system between the drywell and the water pool.

Materials, design, fabrication, inspection, and testing are in accordance with the SAME Boiler and Pressure Vessel Code, Section III, Section B, 1967 Edition, with all applicable Addenda published to June 1967, and Code Case 1177-5 and 1330-1. See Table L.3-1.

Materials used for suppression pool modifications were in accordance with the ASME Boiler and Pressure Vessel Code Section III, 1977 Edition, with Addenda up to summer 1978, and Section II.

The material for the shell of the drywell, suppression chamber, and interconnecting vent system is ASME-SA516, Grade 70 Fire Box quality fabricated to ASTM-A300. The Charpy V-notch impact tests of the material were conducted as specified in N-330, at a maximum test temperature of 0°F. This impact test temperature is based on a lowest service metal temperature of 30°F.

The drywell is a steel pressure vessel with a spherical lower portion and a cylindrical upper portion. The 34 ft 2 in dia bolted top closure is made with a double tongue and groove seal with test connection between which will permit periodic checks for tightness without pressurizing the entire vessel.

Jet deflectors are provided at the inlet of each vent pipe to prevent possible damage to the pipes or bellows assemblies from a jet force which might accompany a pipe break in the drywell, and to prevent overloading any single vent.

The free flow area around the periphery of the jet deflector plate is equal to 1.4 times the area of the 6 ft 9 in dia vent duct (1.4 x 5150 = 7210 sq in). The deflectors project approximately 2 ft 4 in into the drywell. The vent pipes are enclosed with sleeves and are provided with two-ply expansion bellows to accommodate differential motion between the drywell and suppression chamber.

During erection, the drywell vessel was supported on a steel skirt which was attached to the vessel at elevation 4 ft 1/2 in.

After the initial leak rate and overpressure testing, the drywell was embedded in concrete to elevation 5 ft 11 7/8 in thereby providing uniformity in the support by following the contour of the vessel. An embedment transition is provided for the shell from elevation 5 ft 11 7/8 in to elevation 9 ft 2 in. See Figure L.1-1.

The suppression chamber is a steel pressure vessel in the shape of a torus below and encircling the drywell. Inside the suppression chamber, also in the shape of a torus, is the vent system distribution header. Projecting downward from the header are 96 downcomer pipes which terminate below the water surface of the pool.* Columns extending from, and attached to, the bottom of the suppression chamber support the vent header and downcomers, and vent header deflector, and also resist the upward reaction from the downcomers during blowdown. The columns are pinned at the top and bottom to accommodate the differential horizontal movement between the header and the suppression chamber.

* The 48 vent header downcomer intersections are reinforced with two gusset plates and one tie plate to allow these components to withstand loads generated by a LOCA.

Vacuum breakers relieve pressure from the suppression chamber to the drywell to prevent a significant pressure differential between the drywell and suppression chamber. These vacuum breakers also prevent a backflow of water from the suppression pool into the vent system and prevent excessive water level oscillation within the downcomer pipes.

Access to the pressure suppression chamber from the Reactor Building is through two manholes with double-gasketed bolted covers, with a test connection between, which can be tested for leakage.

Access to the drywell is through the equipment hatch, personnel air lock, and through the double-gasketed drywell head, with a manhole, all of which have provisions for individual leak testing.

The pressure suppression chamber is supported on 16 pairs of equally spaced columns and 16 saddle-type supports. These supports transmit vertical loading to the reinforced concrete foundation slab of the Reactor Building. Lateral loads due to an earthquake are transmitted to the foundation by four symmetrically placed earthquake ties.

The dimensions of the Drywell and Pressure Suppression System are given on Table L.3-2.

The interior surface of the primary containment vessel is primed with an inorganic zinc coating (Carbonzinc 11) and finish-painted with a modified epoxy phenolic coating (Phenoline 368 inside the suppression chamber, and Phenoline 305 inside the drywell).

L.3.2.2 Applicable Codes and Regulations

The following publications, of the issues listed below, form a part of the applicable codes and regulations used in the design of the Pressure Suppression Containment System.

L.3.2.2.1 American Society of Mechanical Engineers (ASME)

Boiler and Pressure Vessel Code, Sections III, VIII, and IX, 1967 edition, and the particular requirements for Class B vessels as defined in paragraph N-132, Section III. Modifications were made using the same Code 1977 edition with Addenda up to Summer 1978 with particular requirements of Subsections NE for containment and NF for supports.

Boiler and Pressure Vessel Code, Section II, 1967 edition with all applicable addenda, for the following material specifications for original construction and Section II, 1977, with Summer 1978 Addenda for modifications.

Designation	Title
SA-194	Carbon and Alloy Steel Nuts for Bolts for High-Pressure and High-Temperature Service (Grade 4)
SA-240	Corrosion Resisting Chromium and Chromium Nickel-Steel Plate, and Strip for Fusion Welded Unfired Pressure Vessels
SA-312	Seamless and Welded Austenitic Stainless Steel Pipe

SA-320 Alloy-Steel Bolting Materials for Low-Temperature Service (Grade L7 or L43)

SA-333 Seamless and Welded Steel Pipe for Low-Temperature Service (Grade 1 or 6)

SA-350 Forged or Rolled Carbon and Alloy Steel Flanges, Forged Fittings, Valves and Parts for Low-Temperature Service (Grade LFI)

SA-516 Carbon Steel Plates of Intermediate Tensile Strength and Fusion Welded Pressure Vessels for Atmospheric and Lower Temperature

Service (Grade 70 Firebox Quality Aluminum Killed)

L.3.2.2.2 American Society for Testing and Materials Standards (ASTM)

Designation	Title
A-36	Structural Steel
A-53	Welded and Seamless Steel Pipe (Grade B)
A-106	Seamless Carbon-Steel Pipe for High-Temperature Service
A-300	Steel Plates for Pressure Vessels for Service at Low Temperatures

L.3.2.2.3 United States Steel Publication No. ADUSS 01-1205

T-1 Low-Carbon Constructional Alloy Steel

L.3.2.2.4 American National Standards Institute (ANSI)

B31.1.0 Power Piping

L.3.2.2.5 American Institute of Steel Construction (AISC)

"Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings" adopted April 1963.

L.3.2.2.6 The Commonwealth of Massachusetts

Board of Standards Building Code

Rules and Regulations for the Prevention of Accidents in Construction Operations (Industrial Bulletin No. 12)

L.3.2.3 Design Loadings

The loadings considered in the design of the drywell, suppression chamber, and interconnecting elements are shown on Tables L.3-3 and L.3-4.

A description of the loads and the various load combinations used in the design are presented in the following paragraphs.

L.3.2.3.1 Description of Loads

L.3.2.3.1.1 Pressures and Temperatures Under Normal Operating Conditions

During reactor operation the vessels will be subjected to temperatures up to 150°F at atmospheric pressure. The suppression chamber will also be subjected to the loads associated with the 84,000 cubic ft of water distributed uniformly within the vessel.

L.3.2.3.1.2 Pressures and Temperatures Under Accident Conditions

The drywell, suppression chamber, and the Vent System are designed for a maximum internal pressure of 62 psig coincident with a drywell wall temperature of 281°F and the suppression chamber will be subjected to the increased loads associated with the storage of 94,000 cubic feet of water.

L.3.2.3.1.3 Jet Forces

The drywell and closure head are designed to withstand the jet forces listed on Table L.3-3. These listed forces do not occur simultaneously. However, the jet force was assumed to occur concurrently with the design internal pressure of 56 psig and a temperature of 150°F. The jet forces consist of steam and/or water at 300°F. The drywell is largely enclosed within the structural and shielding concrete. There is a nominal 2 in gap between the vessel and the concrete except at the closure head and top flanges. Where the drywell shell is backed up by concrete, local yielding may take place due to jet force impingement; however, rupture will not occur.

Where the shell is not backed up by concrete, the primary stresses resulting from the jet force loads do not exceed 0.9 times the yield point of the material at 300°F. However, primary plus secondary stresses permitted are three times the allowable stress values given on Table N-421 of Section III, Section B, of the ASME Boiler and Pressure Vessel Code.

The suppression chamber and vent system are designed to withstand the vessel blowdown reactions associated with the design basis loss of coolant accident. The design forces on downcomer pipe are listed on Table L.3-4. Stresses resulting from these reactions are limited to ASME Code allowable stresses.

L.3.2.3.1.4 Gravity Loads to be Applied to the Primary Containment

The gravity loads consist of weight of the shell and appurtenances, personnel lock, equipment hatch, spray headers, equipment supporting structural members, water inside the torus, water up to El 116 ft (during refueling), air (during leak rate tests), and live and dead loads on welding pads. These loads are shown on Table L.3-3.

L.3.2.3.1.5 Lateral Loads-Wind

Wind loads prior to construction of the Reactor Building are in accordance with ASCE paper 3269 "Wind Forces on Structures". The wind loads are shown on Table L.3-3.

L.3.2.3.1.6 Seismic Loads

Seismic loads for the primary containment are due to the Operational Basis Earthquake horizontal ground acceleration of 0.8g percentage acting simultaneously with vertical acceleration of 0.53g percentage.

The customary 1/3 increase in allowable stresses is not used for seismic loads.

Loads due to the Safe Shutdown Earthquake are not governing since the margins between actual stresses and allowable increased stresses would be greater than those for the Operating Basis Earthquake, because stresses due to the seismic load produced by the Safe Shutdown Earthquake will increase by approximately, 1 percent while the allowable stresses may be increased by 50 percent or up to the yield stress, whichever is smaller. The seismic loads generally are insignificant when compared with the internal accident pressure loading. See Figures L.3-1 and L.3-2 for acceleration curves for construction and final stages, respectively.

L.3.2.3.2 Load Combinations Used in the Design of the Primary Containment Vessel

Tables L.3-5 and L.3-6 show the CB&I case numbers used in the design of the drywell and the suppression chamber. The right hand columns contain and relate load symbols used on Tables L.3-7 through L.3-12 to the loads shown on Tables L.3-5 and L.3-6.

The load symbols considered in the Design Summary of the containment include the following:

D = Dead load of the structure and related equipment plus any other permanent loads contributing stress, such as soil or hydrostatic loads; live loads expected to be present when the station is operating; and the loads due to thermal expansion under normal operating conditions. This load takes into account any deviations from normal operating conditions which are reasonably expected to occur during the design lifetime of the station.

R = Loads resulting from jet forces and pressure and temperature transients associated with rupture of a single pipe within the primary containment. This load is considered as indicated in the tables.

E = Loads due to the Operating Basis Earthquake (0.08 g horizontal ground acceleration; two-thirds of horizontal ground acceleration spectrum applied simultaneously for vertical seismic acceleration).

Flood = Loads due to flooding the drywell up to El 116 ft.

= Design wind loading conditions.

The following are the load combinations and corresponding allowable stress limits, as shown on Tables L.3-7 through L.3-12.

Load Combination Limits

1. D+E Stresses remain within normal code allowable stresses (AISC for structural steel, ACI for reinforced concrete, ASME Pressure Vessel Code Section III (Class B) for the primary containment). The customary increase in design stress for earthquake loadings is not permitted.

2. D+W Maximum allowable stresses may be increased one-third above normal code allowable stresses.

3. D+R+E Stresses remain within normal code allowable stresses (AISC for structural steel, ACI for reinforced concrete, ASME Pressure Vessel Code, Section III (Class B) for the primary containment). The customary increase in design stress for earthquake loadings is not permitted. In the case of jet impingement loading on the primary containment, where it is backed up by concrete, the general primary membrane stress, plus the primary bending stress, plus the secondary membrane, plus bending stress must be less than either twice the yield stress or 90 percent of the ultimate stress, whichever is lower. For jet impingement loading on the primary containment (including containment penetration assemblies), where the primary containment is not backed up by concrete, the primary stresses must not exceed 90 percent of the yield strength of the material at 300°F.

4. D+E +Flood — Local membrane stresses in the primary containment may exceed the yield point, but must not rupture.

L.3.2.4 Design Calculations

L.3.2.4.1 Introduction

A complete set of design calculations for the drywell, suppression chamber, interconnecting elements, nozzle reinforcements, and access openings have been prepared by CB&I and Teledyne Engineering Services and will be on file at the Boston Edison Company, Boston, Massachusetts. The analyses have taken into consideration all of the design loads, and load combinations shown on Tables L.3-3, L.3-4, L.3-5, and L.3-6. The maximum stresses computed are all within the ASME Boiler and Pressure Vessel Code allowables.

L.3.2.4.2 Drywell Design-Primary Membrane Stresses

The drywell is designed by membrane theory which is based on the principle that the thin shell resists the imposed loads by direct stresses only. To resist earthquake loads, the stabilizer assembly is provided at El 81 ft 6 1/4 in to transfer the seismic load on the internal structure through the shell and into the external concrete shield wall.

The seismic load on the shell and appurtenances is resisted jointly by the shell and by the stabilizer.

The shell acts as a beam of variable cross section fixed at embedment level El 9 ft 2 in, and simply supported at stabilizer level, El 81 ft 6-1/4 in. The stabilizer assembly is designed for loads due to seismic and jet forces on the internal structure, in addition to a stay force on the drywell shell. The magnitude of the forces is shown on Table L.3-3. The deflection due to the stay force is accommodated in the gap between the male and female parts of the stabilizer assembly. The stresses induced in the shell due to the stay force are extremely small, and they do not govern the design of the shell.

L.3.2.4.3 Drywell Design-Maximum Primary Membrane Stresses in the Shell

The maximum primary general membrane stresses in the shell result from the combination of an internal pressure of 62 psig, the dead load of the shell and appurtenances, lateral and vertical seismic loads, and gravity load on welding pads, which is Case 7, Table L.3-5, the accident condition. The internal pressure load causes by far the greatest stress.

The maximum primary membrane stress, shown on Table L.3-7, of 17.448 ksi is less than the 17.500 ksi allowed by the code. It occurs in the cylindrical portion of the drywell. Other stresses computed at other points along the drywell are shown on Table L.3-7.

Case 1 shown on Table L.3-5 is for the overload test conducted at a pressure of 70 psig, which is higher than the design internal pressure of 56 psig. Since this condition and pressure were temporary, an increase in the allowable membrane stress was allowed.

In addition to maximum stresses computed for the cylindrical and spherical portions of the drywell, stresses have been computed on the elliptical top closure head of the vessel, taking into account the effect of jet forces, since this portion of the vessel is not backed up by concrete. The maximum stress on the head has been found to be 30.24 ksi and results from jet forces combined with the design internal pressure of 56 psig. The design specification allowance for this loading combination is 30.33 ksi (0.9 Fy at 300°F).

L.3.2.4.4 Drywell Design-Discontinuity Stresses

Drywell discontinuity stresses at embedment, expansion joint, and vent-to-drywell shell have been accounted for and stress values included in the CB&I certified stress report. The following gives the actual and allowable stresses at these discontinuities:

Location	Maximum Actual Stress	Allowable Stress
Drywell embedment at accident condition	22.22 ksi	3 S_m =52.50 ksi
Expansion joint	15.20 ksi	19.80 ksi @ 300°F for ASME SA-240 Type 304
Vent-to-drywell shell	15.07 ksi	1.5 S_m =26.250 ksi

L.3.2.4.5 Drywell Design-Expansion of the Drywell Containment Vessel and Jet Forces

Design pressure for the drywell permits a relatively thin-walled steel vessel. However, the vessel has relatively little capability to resist concentrated jet forces. Such loads are, however, readily accepted by the massive concrete shield which surrounds the vessel. Accordingly, the space between the steel drywell vessel and the concrete shield outside has to be sufficiently small so that, although local yielding of the steel vessel can occur under concentrated forces, yielding to the extent causing rupture will be prevented. Space has been provided to allow the drywell to expand in its stressed condition in order for it to function as a pressure vessel. In addition, the vessel is subjected to thermal expansion caused by operating or possible accident condition temperatures significantly higher than ambient. Jet impingement force stresses are summarized on Table L.3-8.

In order to ensure that a steel shell could deflect up to 3 in locally without failure as a result of a concentrated load, CB&I has conducted a series of tests on a steel plate formed to simulate a portion of the drywell vessel. The tests were satisfactory and also provided data on loading required to produce a given deflection, and the strain at various points of the shell. In performing these tests, permanent deformation was not considered as failure.

L.3.2.4.6 Drywell Design-Flooded Condition

The primary containment was analyzed for its ability to withstand loading from post accident flooding of the drywell.

Under this condition, the drywell is flooded with water to El 116 ft. Other loads, such as internal pressure, temperature, live loads, and jet forces are not combined with the hydrostatic load since these loads will not occur simultaneously with flooding. However, the vessel was analyzed for earthquake loads combined with the hydrostatic loads.

Table L.3-7 summarizes the stresses in the shell under the flooded conditions and earthquake.

L.3.2.4.7 Drywell Design-Buckling Considerations

The drywell shell must be capable of resisting the compressive stresses resulting from the external pressure, the dead load of the shell and appurtenances, the live load on the access hatch and beam loads, the gravity loads on the weld pads, plus the seismic loads. These loads produce biaxial compressive stresses of varying magnitude at different points along the drywell shell.

The worst condition for drywell buckling is during the refueling condition, Case 6, Table L.3-5, combined with stresses due to seismic loading. The maximum compressive stress occurs at the drywell embedment and it is 0.81 of the allowable stress.

L.3.2.4.8 Drywell Design-Stabilizer Shear Lugs

Eight stabilizer mechanisms are designed to transfer at El 81 ft 6-1/4 in into the building the reaction due to seismic loads or seismic plus jet loads acting on the drywell, reactor, and shield. The loads are shown on Table L.3-3.

Each stabilizer mechanism is composed of four components: (1) the connection between the reactor stabilizer and the drywell shell, (2) the male lug, (3) the female lug, and (4) the concrete shear connectors. The geometry of the stabilizer mechanism allows for radial and vertical movements due to pressure and temperature. Computed stresses in the stabilizer mechanism are compared to either the AISC or ASME Code allowables, depending upon the component being analyzed. All components and welds which are attached directly to the drywell shell satisfy the ASME Code. The stresses in the remaining components are compared to AISC allowables. The allowed and computed stresses are summarized on Table L.3-9.

L.3.2.4.9 Suppression Chamber Design-Primary Membrane Stresses

The suppression chamber is supported on 16 pairs of equally spaced columns located on the inner and outer perimeters. Although the principal stresses computed on the suppression chamber were circumferential, detailed analyses have been performed to determine the magnitude of localized stresses at the points of column and downcomer supports and vents to determine the need for and to provide additional stiffeners and reinforcing as required. The computed stresses are summarized on Tables L.3-10 and L.3-11.

Due to the complexity of the analysis involved in the determination of maximum stresses under various loads and load combinations, Teledyne Engineering Services set up a computer program for each of the major loading combinations. These combinations were the initial and final condition at ambient temperature at the time of the acceptance test, and the accident condition at 281°F. In addition, the flooded condition was analyzed. The Teledyne Engineering Services calculations for the suppression chamber, including the printout sheets for the computer program, will be included in the certified stress report.

L.3.2.4.10 Suppression Chamber Design-Accident Condition

The maximum primary membrane stresses in the shell and ring girder result from a combination of downcomer thrusts of 21,000 lb each, an internal pressure of 62 psig at 281°F or an external pressure of 2 psig, dead load of shell and appurtenances, the load of the 84,0003 ft of water in the suppression pool, lateral and vertical seismic loads, and vent thrusts of 62 psig at 281°F.

The maximum primary membrane stresses in the shell and ring girder result from a combination of loads as shown on Table L.3-6. The principal stresses are shown on Table L.3-10. The maximum actual stresses calculated in the columns due to a combination of axial compression and bending are 0.897 and 0.795 of the allowable stress for the outside and inside columns, respectively.

Stresses are determined at critical points along the girder. The maximum stresses, 13.11 ksi acting in the plane of the shell, and 16.80 ksi acting in the ring girder flange, are for the accident condition and the ASME Code allowable of 17.5 ksi.

L.3.2.4.11 Suppression Chamber Design-Flooded Condition (Ring Section and Supports)

With the water level at elevation 116 ft in the drywell for the flooded condition, a computer analysis showed that the maximum stresses in the support ring are 16.24 ksi in the plan of the shell, and 22.94 ksi in the ring girder flange which are below ASME Code allowable of 1.33 X 17.5 ksi = 23.33 ksi.

The outside and inside column stresses were investigated at three locations 90 deg apart. The maximum stress due to a combination of axial compression and bending was calculated to be 0.809 and 0.549 of the allowable stress for the outside and inside columns, respectively.

The design of rods, column connections, plates, etc., have been analyzed for the flooded condition with earthquake and the stresses are less than the code allowable stresses.

L.3.2.4.12 Suppression Chamber Design-Header, Downcomer, and Vent Pipes

These components of the suppression chamber were analyzed and adequately sized for plate thickness and reinforcements as required, and in conformance with the ASME Code.

L.3.2.4.13 Containment System Design - Summary

All possible loads, as well as their combinations, have been taken into consideration and the maximum stresses computed are all within the design specification and the ASME Boiler and Pressure Vessel Code allowable stresses.

L.3.3 Penetration Nozzle Design

CB&I designed the penetration nozzles. The shell stresses, from loads on the nozzles, at the nozzle neck to shell junction, were analyzed by the methods outlined in Welding Research Council Bulletin No. 107. A computer program was written to perform the calculations outlined in the computation forms for a spherical and a cylindrical shell.

Unit loads were run on the computer to determine the stresses for various combinations of loads. The stress report by Teledyne Engineering Services includes the computer printout sheets listing the stresses for a 1,000 lb radial load, 1,000 in-lb moment and 1,000 lb shear. Using these coefficients, stresses were determined for combined loading conditions including thermal, earthquake, dead load, and pipe rupture loads.

The size and thickness of the nozzle neck and necessary reinforcement are computed from requirements listed in Section III, Section B, of the ASME Boiler and Pressure Vessel Code. The attachments are designed to provide the strength required by the ASME Code. The computed stresses and allowable stresses are summarized on Table L.3-12.

TABLE L.3-1

MATERIALS AND STRESSES

The vessels are constructed with the following materials:

Material Code Designation	Min Ult Tensile (ksi)	Min Yield (Ambient) (ksi)	Code Tensile Allow (To 650°F) (ksi)	Notes
Plate				
ASME-SA516 Gr 70 fabricated to ASTM-A300	70	38	17.5	Yield at 300°F = 33.7 ksi
ASME-SA240 TP 304	75	30	13.75	
Pipe				
ASME-SA333 Gr 1	55	30	13.75	Yield at 300°F = 26.6 ksi
or ASME-SA333 Gr 6	60	35	15.0	Yield at 300°F = 31.0 ksi
ASME-SA312 TP 304	75	30	13.75	
Forgings				
ASME-SA350 LF1	60	30	15	
Bolting*				
ASME-SA320-L7	125	105	25	Through 2 1/2 in ϕ
or ASME-SA320-L43	125	105	25	Through 4 in ϕ
ASME-SA194 Gr 4	-	-	-	Specification Req. Proof Test
Structural				
ASTM-A36	60	36	22	Not to be used for pressure part nor within 4 in of pressure part.
ASTM-A53 Gr B	60	35	21	
ASTM-A106 Gr B	60	35	15	
USS T-1	118	105	-	

*Excludes Gibbs Manway Cover Studs

TABLE L.3-2

PRIMARY CONTAINMENT DIMENSIONS AND DESIGN DATA

Drywell Design Data	
Cylindrical Section internal diameter	34 ft 2 in
Cylindrical Section height	38 ft 3 1/8 in
Spherical Section internal diameter	64 ft
Spherical Section height	56 ft 7 3/4 in
Spherical Shell to Cylindrical Neck height	5 ft 6 1/2 in
Free Air Volume	146,900 ft
Wall Plate Thickness	
Spherical Shell	13/16 in to 1 1/8 in
Spherical Shell to Cylindrical Neck	2 5/8 in
Cylindrical Neck	Varies, 0.670 to 1 7/16 in
Top Head	1 7/16 in
Vent System	
Number of Vent Pipes	8
Internal Diameter	6 ft 9 in
Break Area/Vent Pipe Area	0.019
Downcomer Pipes	
Number of Downcomer Pipes	96
Internal Diameter	2 ft
Submergence Below Suppression Pool Water Level	4 ft (approximately)
Pressure Suppression Chamber (Torus) Design Data	
Water Volume, maximum (test)	94,000 cu ft (105,000 cu ft test)
Free Air Volume, maximum	125,000
Chamber Internal Diameter	29 ft 6 in
Torus Major Diameter	102 ft

TABLE L.3-3

GENERAL DRYWELL DESIGN CONDITIONS

The vessel was designed and analyzed for the following conditions as specified in the Bechtel specification:

1. Design Pressures:

Internal:	Maximum	62 psig @ 281°F
	Design.	56 psig @ 281°F
	Operating	<2.5 psig @ 150°F
External:	Maximum	2 psig @ 281°F
	Design.	2 psig @ 218°F
	Operating	<2.0 psig @ 150°F

2. Earthquake:

   Horizontal: See Curves on Figures L.3-1 and L.3-2
   Vertical: 5.3% g

3. Weight of Compressible Material: None

4. Bellow Loads:

Inside:	Operating	60 lb/in
	Refueling	60 lb/in
Outside:	Operating	60 lb/in
	Refueling	60 lb/in

5. Loads to be Transferred Through Drywell:

   (A) At Bottom of Drywell Elevation = 9 ft 2 in
   (1) Vertical:
   Normal. . . . . 7,950,000 lb
   Refueling . . . . 8,250,000 lb
   (2) Horizontal: 1,450,000 lb + 190,000 lb seismic load on pumps
   (3) Moment: 66,000,000 ft-lb
   (B) At the Stayed Elevation = 81 ft 6-1/4 in
   (1) Without Jets 1,000,000 lb total load
   (2) With Jets 1,400,000 lb total load

6. Wind (prior to construction of the Reactor Building)

Structure	Wind Load (psf)	Shape Factor Used
Cylinder. . .	20	0.60
Sphere. . . .	15	0.45

TABLE L.3-3 (Cont)

7. Top of Refueling Water: To be @ elevation 116 ft 0 in

8. Miscellaneous Live Loads:

   1. 150 lb/ft^2 on Personnel Lock Floor
   2. 40,000 lb on Equipment Access Opening
   3. 2,350 lb on Upper Spray Header
   4. 3,850 lb on Lower Spray Header

   Weights of all appurtenances are estimated weights, and may be heavier than the actual weights.

9. Jet Forces:

Location	Jet Force (Max.) (kip)	Area Subjected To Jet Force (ft^2)
On Spherical part of drywell . . .	665	3.69
On Cylindrical part and sphere transition to cylinder	316	1.76
On Closure Head	32.6	0.18
*Steam and/or Water Temperature . .	300°F	
*Shell Temperature	150°F	

10. Stabilizer Loads

Seismic Force	370k
Seismic + Jet Forces.	520k
Seismic + Flooded Condition	370k

*Shell temperatures in the jet impingement target area are assumed to be the temperature of the impinging jet (i.e. 300°F). The remainder of the shell is assumed to be 150°F.

TABLE L.3-4

GENERAL SUPPRESSION CHAMBER DESIGN CONDITIONS

1. Design Pressures:

Internal:	Maximum	62 psig @ 281°F
	Design	56 psig @ 281°F
	Operating	<2.5 psig @ 50 - 100°F
External:	Maximum	2 psig @ 281°F
	Design	2 psig @ 281°F
	Operating	2.0 psig @ 50 - 100°F

2. Earthquake:
   - Horizontal: 12% g
   - Vertical: 5.3% g

3. Water Volumes:

Minimum Operating	84,000 ft^3
Normal Operating	84,000 ft^3
Accident	94,000 ft^3
Test	105,000 ft^3

4. Weld Pads:
   - D.L. = 250 lb/pad
   - L.L. = 750 lb/on one of any two adjacent pads

5. Catwalks: 75 lb/ft^2

6. Weights of appurtenances are estimated weights, and may be more than actual weight.

7. Jet Forces:

Location	Jet Force (Max.) (kip)	Area
Downcomer	21	24 in dia

TABLE L.3-5

DRYWELL LOADING COMBINATIONS

Loads	Construction	Overload Test	Final Test		Normal Operating		Refueling	Accident		Flooding	Included in Load Combination Symbol On Tables L.3-7-9-12
CB&I Case Number	C	1	2	3	4	5	6	7	8	9	
Dead Load, Vessel and Attachments	X	X	X	X	X	X	X	X	X	X	O
Pressure											
Positive		+70	56		2			56			R
Negative				2		2	-2		2		R
Contained Air		X	X	X							D
Lateral Load, Seismic or Wind	X	X	X	X	X	X	X	X	X	X	E
Vertical Seismic	X	X	X	X	X	X	X	X	X	X	E
Vent Thrusts		X	X	X	X	X					R
Welding Pads											
Dead Load			X	X	X	X	X	X	X		D
Live Load			X	X			X				D
Equipment Support Loads			X	X	X	X	X	X	X	X	D
Weight and/or Restraint of Compressible Material			X	X	X	X	X	X	X	X	D
Temporary Pressure or Unrelieved Deflection Due to Concrete Load			X	X	X	X	X	X	X	X	D
Personnel Lock											
Dead Load	X	X	X	X	X	X	X	X	X	X	D
Live Load					X	X	X				D
Equipment Hatch											
Dead Load	X	X	X	X	X	X	X	X	X	X	D
Live Load					X	X	X				D
Refueling Seal Loads					X	X					D
Water on Refueling Seals							X				D
Jet Forces								X	X		R
Hydrostatic Pressue due to Flooding										X	Flood

TABLE L.3-6

SUPPRESSION CHAMBER LOADING COMBINATIONS

Loads	Construction	Overload Test	Final Test		Normal Operating		Accident		Flooding	Included in Load Combination Symbol on Tables L.3-7-9-12
CB&I Case Number	C	1	2	3	4	5	6	7	8	
Dead Load, Vessel and Attachment	X	X	X	X	X	X	X	X	X	D
Suppression Pool Water		X	X	X	X	X	X	X	X	D
Pressure										
Positive		70	56		2		56			R
Negative				2		2		2		R
Seismic										
Vertical	X	X	X	X	X	X	X	X	X	E
Lateral	X	X	X	X	X	X	X	X	X	E
Vent Thrusts		X	X	X	X	X	X	X		R
Contained Air		X	X	X						D
Temporary Concrete Loads	X									D
Welding Pads										
Dead Loads	X	X	X	X	X	X	X	X	X	D
Live Loads	X	X	X	X	X	X	X	X		D
Live Load on Catwalks and Platforms	X				X	X				D
Jet Forces on Downcomer Pipes							X	X		R

TABLE L.3-7

DRYWELL MEMBRANE STRESSES

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress, (ksi)	Maximum Stresses, (ksi)				Location
The vessel is bulb shaped and houses the primary nuclear reactor vessel, the coolant recirculation lines, pumps etc. In case of an operating accident, the vessel must contain the steam released within the drywell, and conduct this steam to the suppression chamber. Structural steel plate material is ASME SA-516 fabricated to ASTM Designation: A 300, minimum service temperature 30°F, with Charpy impact requirements at maximum 0°F. Seismic design load includes load due to vertical acceleration equal to 2/3 of the horizontal ground acceleration. For other design criteria see Table L.3-3. After an accident, the drywell may be flooded up to El. 116 ft; stresses shall be below yield point (without seismic load), or may exceed yield but with a margin against rupture if seismic is considered. Accident load (R) includes pressure and temperature in the primary containment.	ASME Section III including Code Cases 1330-1 and 1177-5 and Addenda as of June 9, 1967 for Vessel Class "B." Stress intensities are defined per Code para. N-413 and their limits are per Code para. N-413. End conditions are found with methods described in the book Theory of Plates and Shells by Timoshenko.		Primary General Membrane PM = 17.5 @ 281°F	σ_t Circumferential	σ_l meridional	σ_r -radial	Critical Stress (total) P_m =	Section L.1.1
		D + R + E		7.74		-0.028	7.768	Head
		D + R + E			3.98	-0.028	4.008	Cylinder on head
		D + R + E		9.128		-0.028	9.156	Cylinder at top flange
		D + R + E		17.42		-0.028	17.448	Cylinder
		D + R + E		16.25		-0.028	16.278	Knuckle @ el 56 ft
		D + R + E		13.41		-0.028	13.438	Sphere
		D + R + E	Primary Local Membrane PL = 1.5 PM = 26.25 @ 281°F	16.52		-0.056	P_l =16.58	Cylinder at flange. (Much lower stresses are at other locations)
		D + R + E	Primary + Secondary + Bending Q = 3 PM 52.50 @ 281°F	18.03		-.056	Q=18.09	Knuckle (stresses are much lower at other locations)
		D + E + Flood	Yield 38.0 @ ambient Ultimate 70.0 Critical buckling 23.77 (meridional)	25.64	7.23		32.87	Embedment @ el. 9 ft

NOTE: Only additive stresses shown above for simplicity

TABLE L.3-8

JET IMPINGEMENT FORCE STRESSES

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress, ksi	Maximum Stress, ksi	Reference	Location
A jet force is assumed to occur in any direction within the drywell.	Find maximum load on shell prior to breaking	D + R	30.33	28.80	1/(Case 20)	Equipment door
		D + R	23.17	20.50	1/(Case 20)	Jet deflector support at vent
The force is calculater as 1250 psi pressure acting on the area equal to the cross section of	Compare with given jet load	D + R	90 (90% of yield of T-1 Steel at 300°F)	76.94	1/(Case 60)	Jet deflector baffle plate
ruptured pipe. See PDAR V2.3.6 and Amendment No. 2 to PDAR, Question No. 3		D + R	30.33	30.24	1/(Case 20)	Top closure head
The jet impingement force is considered		D + R	30.33	29.46	3	Cylinder above flanges
to act coincidentally with the design internal pressure and 150°F shell temperature.	Apply the smaller load of either of above; calculate deformation, limiting stresses to prevent a progressive deformation or	Experimental test in 1964	CB&I experimental investigation proved that 3/4" thick plate can deform 3" without failure.			Spherical Shell See Amendment No. 5 to PDAR Comment No. 11
Temperature of the shell and welds are assumed to be 300°F	strain as follows:	D + R	30.33	22.72	2	Cylindrical Shell
if hit directly by jet.	(a) $P_m + P_b \leq 0.9$ yield (b) $P_m + P_b + Q \leq 0.9$ (ultimate stress, S_u)	D + R	27.90	23.12		Upper spray header pipe
Local thermal effects and dynamic jet effects are disregarded.		D + R	21.23	17.7		Upper spray header weld
	Pm = general primary membrane stress Pb = primary bending stress	D + R	30.33	15.51		Upper spray header support

TABLE L.3-8 (Cont)

Description/Criteria	Method of Analysis	Load Combination	Maximum Allowable Stress, ksi	Maximum Stress, ksi	Reference	Location
There is 2" air gap between drywell shell and backup concrete.	Q = Secondary membrane + bending stress Yield taken from ASME Sec. III	D + R	27.90	27.23		Upper spray header inlet pipe
	Table N-424 at pertinent temperature	D + R	27.90	23.21		Lower spray header pipe
Material is ASME SA-516 Grade 70 fabricated to ASTM Designation: A-300 or T-1 by USS where noted	Assume shear type failure of weld, and its stress equal to 7/10 of parent material.	D + R	21.23	15.83		Lower spray header weld
		D + R	30.33	29.52		Lower spray header support
The load combination D+R is a lesser case of D+R+E. The effect of E is insignificant when compared to the effect of the jet impingement force.		D + R + E	63.0	47.65	4	Upper Beam Seats
		D + R + E	63.0	51.96	4	Lower Beam Seats

Reference: 1/Formulas for Stress & Strain by Roark
2/"Analysis of Shells of Revolution" by A. Kalnin, Journal of Applied Mechanics, September 1964
3/"Stresses from Radial Loads" by P. P. Bijlaard, Welding Journal, December 1954
4/Theory of Plates by Timoshenko

TABLE L.3-9

DRYWELL STABILIZER SHEAR LUGS

Description/ Criteria	Method of Analysis	Load Combination	Maximum Allowable Stresses, ksi	Maximum Stresses, ksi	Location
The stabilizer mechanism transfers into building the reaction due to seismic loads or seismic plus jet loads acting on the drywell, reactor and shield, or seismic, plus flooding of the drywell.	ASME Code Section III including addenda as of June 9, 1967, Vessel Class "B"	D + E	Male Lug σB = 17.5 (plate) σS = 12.25 (plate) σC = 15.8 (weld)	σB = 16.37 (plate) σS = 4.22 (plate) σC = 14.11 (weld)	Stabilizer Shear Lug for Drywell at el 81 ft - 6 3/4 in
		D + R + E	σB = 23.33 σS = 16.33 σC = 21.07	σB = 22.34 σS = 5.75 σC = 19.26	
The geometry of the stabilizer allows for radial and vertical movements due to pressure and temperature.	Formulas for Stress and Strain by Roark, Case 22 for plate.	D + E + Flood	σB = 23.33 σS = 16.33 σC = 21.07	σB = 19.16 σS = 4.93 σC = 16.51	
Materials: components attached to the drywell are ASME SA-516 Grade 70 fabricated to ASTM Designation: A300, per ASME Code Section III: components outside the drywell are ASTM Designation: A 36 per AISC-1963	σC = combined stress $\sqrt{(\sigma B+\sigma T)2= \sigma S}$	D + E	Female Lug σB = 22.0 σS = 14.5 σC = 15.8	σB = 17.25 σS = 3.74 σC = 13.6	
	σB = bending stress σT = tensile stress σS = shear stress σC = F_b (AISC)	D + R + E	σB = 29.33 σS = 19.33 σC = 21.07	σB = 23.54 σS = 5.1 σC = 18.56	
		D + E + Flood	σB = 29.33 σS = 19.33 σC = 21.07	σB = 20.18 σS = 4.37 σC = 15.92	
Stress increase by 1/3 is allowed for jet loading or flooding.					

TABLE L.3-10

STRESSES IN TORUS SHELL AND SUPPORTS

Description/Criteria	Method of Analysis	Load Combination*	Maximum Allowable Stress, ksi	Maximum Stresses, ksi	Location
The vessel is in the shape of a torus supported on 32 columns, and is located below the drywell; 8 vent pipes connect drywell and torus. The torus contains a large amount of water for condensation of an accident steam.	ASME Section III and addenda as of June 9, 1967 for Vessel Class "B"				L.2.19
		D + R + E	F_t = 17.50	f_t = 17.50	Torus top shell
		D + R + E	F_t = 17.50	f_t = 17.48	Torus bottom shell
		D + Flood	F_y = 38.0	f_t = 16.83	Torus bottom shell
		D + R + E	F_a = 20.93	f_a = 8.23	Torus outside column
		D + R + E	F_b = 22.80	f_b = 12.24	Torus outside column
		D + E + Flood	F_a = 27.84	f_a = 16.77	Torus outside column
		D + E + Flood	F_b = 30.32	f_b = 2.23	Torus outside column
Structural steel plate material is ASME SA-516 fabricated to ASTM A-300. Minimum service temp. 30°F, with Charpy impact requirements at max. 0°F.		D + R + E	F_t = 17.50	f_t = 16.80	Ring girder flange
		D + R + E	F_t = 17.50	f_t = 13.11	Ring girder shell
		D + E + Flood	F_t = 23.33	f_t = 22.94	Ring girder flange
		D + E + Flood	F_t = 23.33	f_t = 16.24	Ring girder shell
		D + R + E	F = 20.93	f_a = 8.23	Header supporting column
After an accident the drywell, vent system and torus may be flooded up to El. 116ft; stresses in shell shall be below yield point (without seismic load) or may exceed yield but with a margin against rupture if seismic is considered.		D + R + E	F_b = 22.80	f_b = 12.22	
		D + R + E	F_{bear} = 36.00	f_{bear} = 22.91	Pin header supporting Col.
			F_b = 24.00	f_b = 17.36	
			F_v = 16.00	f_v = 5.30	
Bellows stainless steel (2 ply) material is ASME SA-240 Type 304.		D + R + E	F_t = 19.80	f_t = 15.20	Bellow on Vent pipes
		D + R + E	F_t = 17.50	f_t = 6.41	Vent Header
	Kalvin's Shell Program (Yale Univ.)	D + R + E	S_m = 17.50 1.5 S_m = 26.25 3 S_m = 52.50	P_m = 12.88 P_l = 15.07 Q = 26.46	Vent Pipes Drywell

TABLE L.3-11

TORUS SEISMIC TIES

Description/Criteria	Method of Analysis	Load Combination*	Maximum Allowable Stresses, ksi	Maximum Stresses, ksi	Location
The saddles, located below the torus at 90° intervals are oriented so that either set of two opposite saddles will withstand an assumed 0.12 g horizontal seismic acceleration (E).	Structural Design in Metals by C. D. Williams and E. C. Harris, p. 315	D + E	F_v = 18.0	f_v = 4.58 shear	At el. (-)17ft-6in, below torus 5inϕPin
		D + E	F_b = 18.0	f_b = 8.58 bending	1 1/2in thick shear bar
		D + E	F_c = 1.0	f_c = .213 concrete bearing	1 1/2in thick shear bar
		D + E	F_t = 22.0	f_t = 13.1 tension	1 1/2in ϕ anchor bolt
		D + E	F_p = 27.0	f_p = 11.99 bearing	1 1/2in thick upper plate
		D + E	F_p = 27.0	f_p = 15.00 bearing	1 1/2in thick lower plate

Materials:
Concrete f'_c = 4,000 psi at 28 days, ACI 318-63

Pin AISI-1081 (F_Y - 45 ksi)

Bolts ASTM Designation: A 36

Plates ASTM Designation: A 283 Grade C

Vertical seismic load is carried by supporting columns, not by saddles.

Assume accident condition for water volume (94,000 ft^3)

Maximum stresses may not exceed normal code allowable values.

TABLE L.3-12

MAXIMUM STRESSES IN DRYWELL PENETRATION NOZZLES

Description/Criteria

Pipe penetrations for process lines (Type 1 see figure 5.2-9) must accommodate thermal movement and must resist relatively high thermal stress, therefore, expansion bellows are required. The process lines are anchored outside the containment to limit the movement relative to the containment. This design assures integrity of the penetration.

The penetration nozzle is welded to the drywell. The process line which passes through the nozzle is free to move axially, with the two-ply bellows joint accommodating the movement. A guard pipe which surrounds the process line is designed to protect the bellows should the process line fail within the penetration. The two-ply expansion joint permits periodic leak testing of the bellows during normal operation of the plant by pressurizing the annular gap between the two plies.

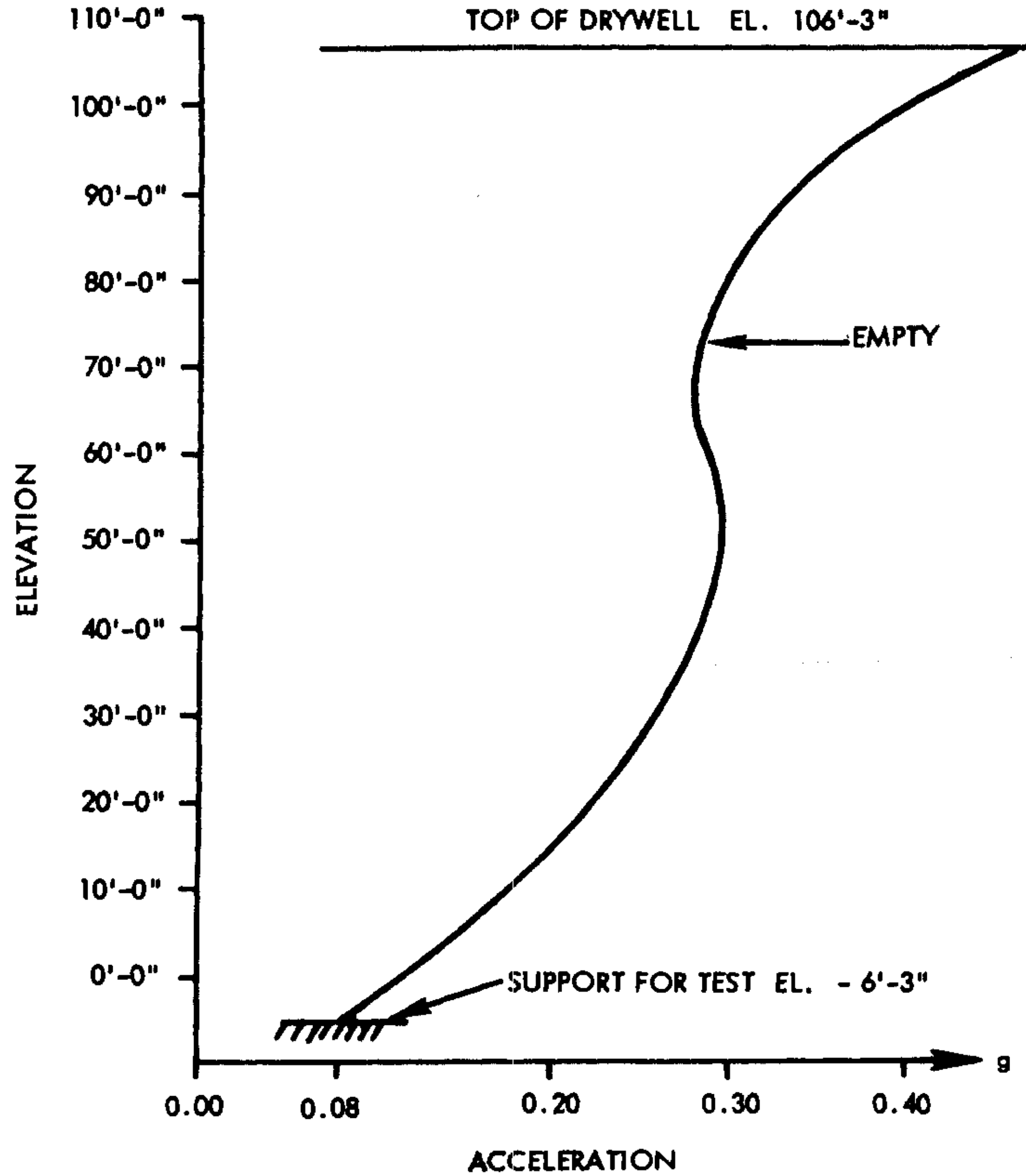
The design of the penetration takes into account the simultaneous stress associated with internal pressure, thermal expansion, dead loads, seismic loads, and loads associated with a loss of coolant accident. Restraint lugs on the guard pipe are provided to transfer any load associated with random failures of the process line directly to the vessel without causing any bending moment stresses The penetration nozzle design takes into account the jet force loading resulting from the failure.

Structural Steel Plate Material is ASME SA-516-70 FBX to ASTM A-300.

Maximum allowable stress is 0.90 of ultimate stress at 300°F.

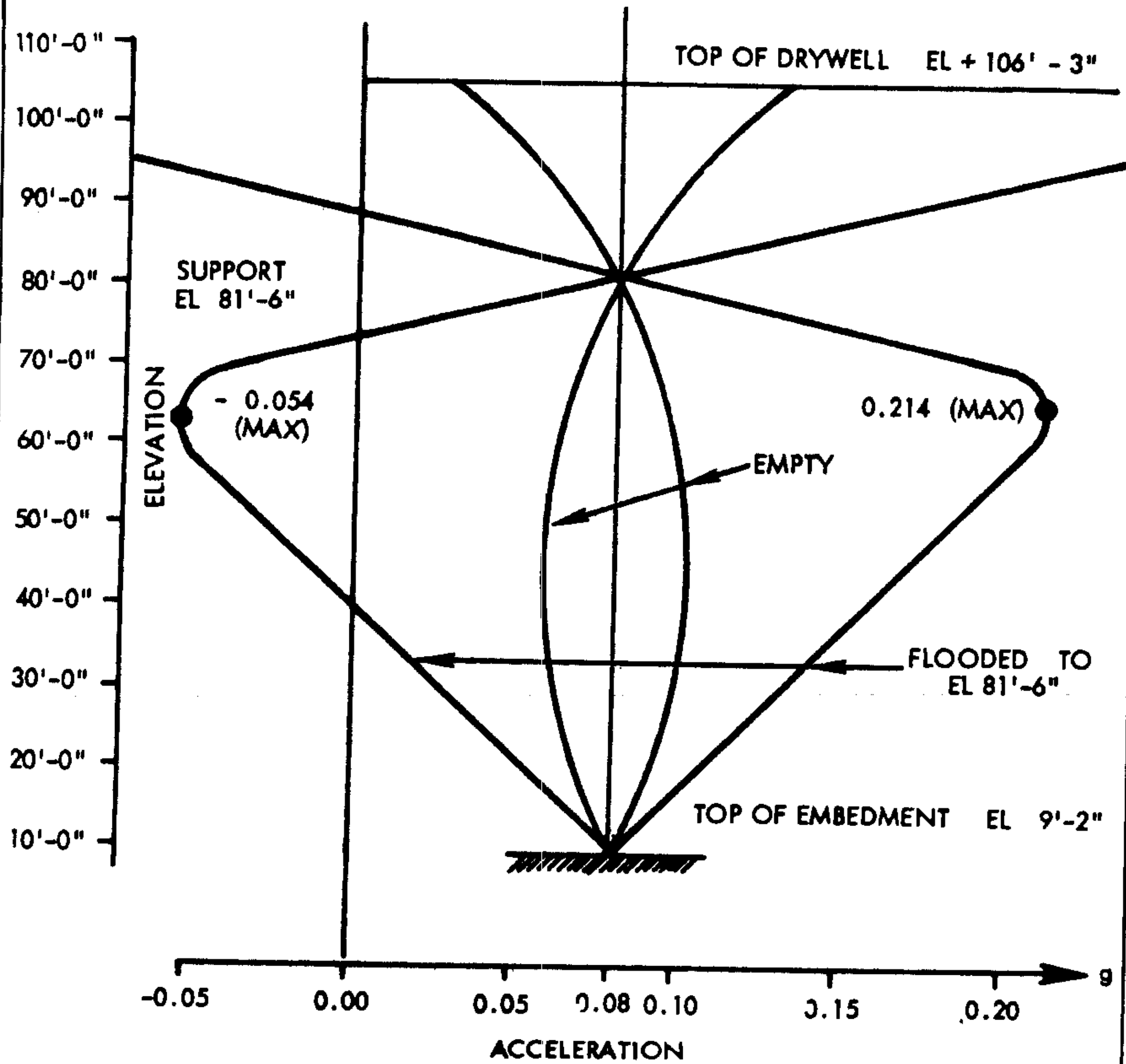
Method of Analysis	Load Combination	Maximum Allowable Stress, ksi	Maximum Stresses, ksi	See also Table L.2.1 Location and Service
Welding Research Council Bulletin	D + R	63.0	72.45* See Note	El. 25'-3 X-7A thru X-7D 27' -2 Primary Steam
	D + R	63.0	26.96	El. 23'-11 X-8 Steam Drain
	D + R	63.0	48.82	El. 31' -0X-9A thru X-9B Primary Feedwater
	D + R	63.0	104.43*See Note	El. 45' -0 X12 RHR Supply
	D + R	63.0	26.90	EL. 67' -6 X-14 Clean-up Supply
	D + R	63.0	33.92	El. 63' -10 X-16A & B Core Spray
	D + R	63.0	26.60	El. 87' -0 X-17 spare
				El. 30' -6 X-51 A&B RHR System Return
	D + R	63.0	32.67	El. 30' -6 X-52 HPCI Stm to Turbine
	D + R	63.0	28.86	El. 35' -6 X-53 RCIC Stm to Turbine
	D + R	63.0	27.01	

NOTE: *These are fictitious stresses which cannot occur because the shell plate is backed up by the concrete shield and can deflect only 2 in. Experimental investigation by CB&I showed that a 3/4 in thick shell plate with reinforced penetrations can deflect 3 in causing large, permanent strains but without failure.



CANTILEVER CONDITION: 0.08g GROUND ACCELERATION

FIGURE L.3-1
ESTIMATED DRYWELL
ACCELERATIONS FOR 1% DAMPING
CANTILEVER CONDITION
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT



OPERATING CONDITIONS; 0.08g GROUND ACCELERATION

NOTE: MAX. HORIZ. DISPLACEMENT OF SUPPORT @ EL. 81'-6" RELATIVE TO TOP OF EMBEDMENT @ EL. 9'-2" DUE TO BLDG. MOVEMENTS IS 40 MIL.

FIGURE L. 3-2
ESTIMATED DRYWELL ACCELERATIONS FOR 1% DAMPING-OPERATING CONDITIONS
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

L.4 INITIAL OVERLOAD AND LEAKAGE RATE TEST

The containment vessel was tested by CB&I for its structural integrity by an overload test and for leakage by the reference chamber method as described in the American Nuclear Society's publication ANS 7.60.

Tests showed that the leakage rate for the vessel was less than the 0.2 weight percent per day leakage rate limit as specified in the containment vessel specification (i.e., 0.0779 weight percent per day).

The containment vessel was tested for leak tightness after all the equipment was installed in the drywell and connected to the penetrations.

L.5 MANUFACTURER'S DATA REPORT FOR NUCLEAR VESSELS

Rev. 1 3/13/70

FORM N-1 MANUFACTURERS DATA REPORT FOR NUCLEAR VESSELS

As required by the Provisions of the ASME Code Rules

PTN 5/8/69

1. Manufactured by Chicago Bridge & Iron Co., Greenville, Pennsylvania (Name and address of Manufacturer)

2. Manufactured for Boston Edison Company, Plymouth, Mass. (Name and address of Purchaser)

3. Type Vert. (Horiz. or Vert.) Kind Tank (Tank, Jacketed, Heat Ex.) Vessel No. (G 1403) (Mfrs. Serial No.) (______) (State & State No.) Nat'l Bd. No. ______ Yr. Built 1969

Items 4-8 incl. to be completed for single wall vessels, jackets of jacketed vessels, or shells of heat exchangers.

DRYWELL AND VENT SYSTEM

4. Shell: Material SA 516 Gr 70 FBX to A300 (Kind & Spec. No.) T.S. 70,000 (Min. of range specified) Nominal Thickness .25,.375,.4375,.50, .670,.8125,1.0625,1.25,1.4375,2.5, 2.625 in. Corrosion Allowance 0 in. Diam. 64 ft. 0 in. Length 109 ft. 1 5/8 in.

5. Seams: Long Dbl. Butt Weld H.T. See Note 1 Below X.R. Yes - 100% Efficiency 100 %

Girth Dbl. Butt Weld H.T. See Note 1 Below X.R. Yes - 100% No. of Courses 10 (If Class B)

6. Heads: (a) Material SA 516 Gr 70 FBX to A300 T.S. 70,000 (b) Material SA 516 Gr 70 FBX. to A300 T.S. 70,000

Location (Top, bottom, ends)	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Press. (Convex or Concave)
(a) Bottom	.615,1.125					32.0'		Concave
(b) Top	1.4375			2:1				Concave

If removable, bolts used SA320-L7 (125,000) 2.5"∅ (76) (Material, Spec. No., T.S., Size, Number) Other fastening ______ (Describe or attach sketch)

7. Jacket Closure ______ (Describe as ogee & weld, bar, etc. If bar give dimensions, describe or sketch)

8. Constructed for (See Note #5) operating press.[2] ______ psi at Max. temp. ______ °F Charpy Impact 20 ft-lb at temp. of 0 °F. Pneumatic ~~Hydrostatic~~ ~~Combination~~ Test Pressure 70 psi — See Note 2 Below

Items 9 and 10 to be completed for tube sections.

9. Tube Sheets: Stationary. Material ______ (Kind & Spec. No.) Diam. ______ in. (Subject to press.) Thickness ______ in. Attachment ______ (Welded, Bolted)

Floating. Material ______ (Kind & Spec. No.) Diam. ______ in. Thickness ______ in. Attachment ______

10. Tubes: Material ______ (Kind & Spec. No.) O.D. ______ in. Thickness ______ inches or gage Number ______ Type ______ (Straight or U)

Items 11 to 14 incl. to be completed for inner chambers of jacketed vessels, or channels of heat exchangers.

SUPPRESSION CHAMBER

11. Shell: Material SA516 Gr 70 FBX to A300 (Kind & Spec. No.) T.S. 70,000 (Min. of range specified) Nominal Thickness .568,.629,1.125,1.750 in. Corrosion Allowance 0 in. Major Diam. 102 ft. 0 in. Minor Dia. 29 ft. 6 in.

12. Seams: Long Dbl. Butt Welded (Welded, Dbl., Single) H.T. See Note 1 Below (Yes or No) X.R. Yes - 100% Efficiency 100 % (If Class B)

Girth Dbl. Butt Welded H.T. See Note 1 Below X.R. Yes - 100% No. of Courses 16

13. Heads: (a) Material ______ T.S. ______ (b) Material ______ T.S. ______ (c) Material ______ T.S. ______

Location	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Press. (Convex or Concave)
(a) Top, bottom, ends								
(b) Channel								
(c) Floating								

If removable, bolts used (a) ______ (b) ______ (c) ______ (Material, Spec. No., T.S., Size, Number) Other fastening ______ (Describe or attach sketch)

14. Constructed for specified (See Note #5) operating press.[2] ______ psi at Max. temp. ______ °F Charpy Impact 20 ft-lb at temp. of 0 °F Combination Test Pressure 70 psi in air space (See Note 2 Below)

[1] If Postweld Heat-Treated.

[2] List other internal or external pressures with coincident temperatures when applicable.

Note 1: Vessel Sub-Assemblies were PWHT as follows:
A. Knuckle, upper & lower 34'-2"∅ flange assemblies - field PWHT.
B. All nozzles were preassembled into shell plates and insert plates, and category D joints were PWHT (except 298 1"∅ stainless steel nozzles in drywell, two 1/2"∅ couplings in suppression chamber, and one 1/2"∅ coupling in a vent line) in the shop.

Note 2: The lock barrel and outer bulkhead are subjected to an overload test (line 8) in the shop. Test is certified in Certificate of Shop Inspection. During completed vessel overload test in field, outer door of the lock is open, inner door is closed and subjected to test pressure.

Note 3: All material shipped from the Greenville, Pa., shop is included in the Certificate of Shop Inspection. All material shipped from the New Castle, Del., shop is included in the Certificate of Field Assembly Inspection.

Note 4: All drywell field seams were Halide tested up to Elev. 9'-8. Shop seams in drywell vent assembly were field Halide tested up to elevation 7'-11.

Note 5: Design Internal Pressure: 56 psig at 281°F
Maximum Internal Pressure: 62 psig at 281°F
Design external pressure (max. containment external pressure) 2 psig in accordance with N-1312(3). Paragraph UG-28(f) does not apply.

Note 6: Temperature +30°F minimum.

PNPS

FORM N-1 (back)

Items below to be completed for all vessels where applicable.

15. Safety Valve Outlets: Number NONE Size Location

16. Nozzles:

Purpose (Inlet, Outlet, Drain) | Number | Diam. or Size | Type | Material | Thickness | Reinforcement Material | How Attached

See Dwg. 2 Rev. 9 & Dwg. 5 Rev. 10 Contract 9-8014 for a complete list of nozzles. All penetrations were welded into insert assemblies and PWHT in the shop except as noted in note 1. Category A & B welds were radiographed. All other welds were inspected by the magnetic particle or liquid penetrant method.

17. Inspection Openings: Manholes, No. 1 Size 24" Location Drywell Top Head

Manholes, No. 2 Size 48" Location Suppression Chamber

No. Size Location

18. Supports: Skirt Yes (Yes or No) Lugs (Number) Legs 32 (Number) Other Columns (Describe) Attached Welded to Suppression Chamber Shell (Where & How)

19. Remarks: Pressure suppression containment system including a bulb shaped containment vessel (drywell) to house the primary nuclear reactor vessel, the coolant recirculating pumps, control rod drives, etc. and a torus shaped vessel (suppression chamber) surrounding the drywell to store a water pool to condense steam which may be released in the event of an operating accident. The vent lines & header channeling the steam from the drywell to the suppression chamber are a part of this system. Expansion joints are designed and fabricated in accordance with Case Interpretations 1177-5 and 1330-1.

(Brief description of service for which vessel was designed)

CERTIFICATION OF DESIGN

Design information on file at Chicago Bridge & Iron Company - Memphis, Tennessee

Stress analysis report on file at Chicago Bridge & Iron Company - Memphis Tennessee.

Design specifications certified by Darrell W. Halligan Prof. Eng. State Mass. Reg. No. 21790

Stress analysis report certified by Wilfred W. LaRiviere Prof. Eng. State Ill. Reg. No. 25612

We certify that the statements made in this report are correct and that all details of material, design, construction, and workmanship of this pressure vessel conform to the ASME Code for Nuclear Vessels.

Date 4-10 19 70 Signed Chicago Bridge & Iron Co. (Manufacturer) By Howard L. Ardel

Certificate of Authorization Expires 12-31-70

CERTIFICATE OF SHOP INSPECTION

VESSEL MADE BY Chicago Bridge & Iron Company at Greenville, Penna.

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State of PA. and employed by HARTFORD S.B.I.&I. Co. HARTFORD CONN. have inspected the pressure vessel described in this manufacturer's data report on [illegible] 5-12 19 69, and state that to the best of my knowledge and belief, the manufacturer has constructed this pressure vessel in accordance with the ASME Code for Nuclear Vessels.

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the pressure vessel described in this manufacturer's data report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date April 16 19 70

[illegible] Inspectors Signature Commissions PA. 1395 National Board or State and No.

CERTIFICATE OF FIELD ASSEMBLY INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State of MASS. and employed by HARTFORD S.B.I.&I. Co. of CONN. have compared the statements in this manufacturer's data report with the described pressure vessel and state that parts referred to as data items 5, 12, 18, not included in the certificate of shop inspection have been inspected by me and that to the best of my knowledge and belief the manufacturer has constructed and assembled this pressure vessel in accordance with the ASME Code for Nuclear Vessels. The described vessel was inspected and subjected to a hydrostatic test of 70 psi.

↑Pneumatic

By signing this certificate neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the pressure vessel described in this manufacturer's data report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date 9/16 19 70

John P. Tomis Jr. Inspectors Signature Commissions MASS 755 National Board or State and No.

REVISED COPY; - NOTE #5

INSPECTOR WM. O'NEIL SIGNING FOR INSPECTOR

A. M. VUKSAN PA 1762

FORM N-2 MANUFACTURERS' PARTIAL DATA REPORT
A Part of a Nuclear Vessel Fabricated by One Manufacturer for Another Manufacturer
As required by the Provisions of the ASME Code Rules

1. (a) Manufactured by Chicago Bridge & Iron Co., Chicago, Illinois
(Name and address of manufacturer of part)

(b) Manufactured for Chicago Bridge & Iron Co., Greenville, Pa.
(Name and address of manufacturer of completed nuclear vessel)

2. Identification-Manufacturer's Serial No. of Part G1403-13
Work orders 8,9,10,11,12,13

(a) Constructed According to ~~XXXXXX~~ 9-8014 Drawing Prepared by Chicago Bridge & Iron Co.

(b) Description of Part Inspected knuckle sub-assemblies

3. Remarks: The parts are four (4) knuckle plate assemblies and will serve as a portion of the pressure shell of a nuclear containment vessel and are designed and constructed under the rules of ASME III Class B.

(Brief description of service for which vessel part was designed)

We certify that the statements made in this report are correct and that all details of material, design, construction, and workmanship of this pressure vessel conform to the ASME Code for Nuclear Vessels.

Date 7-17 1968 Signed Chicago Bridge & Iron Co. By Louis Baucher
(Manufacturer)

Certificate of Authorization Expires 12-31-70

CERTIFICATION OF DESIGN

Design information on file at Chicago Bridge & Iron Co., Memphis, Tennessee

Stress analysis report on file at Chicago Bridge & Iron Co., Memphis, Tennessee

Design specifications certified by Darrel W. Halligan Prof. Eng. State Mass. Reg. No. 21970

Stress analysis report certified by Wilfred W. LaRiviere Prof. Eng. State Ill Reg. No. 25612

CERTIFICATE OF SHOP INSPECTION

I, the undersigned, holding a valid commission issued by the National Board of Boiler and Pressure Vessel Inspectors and/or the State of [illegible] and employed by [illegible] of [illegible] have inspected the part of a pressure vessel described in this manufacturer's partial data report on 1-2-[illegible] 1968, and state that to the best of my knowledge and belief, the manufacturer has constructed this part in accordance with the ASME Code for Nuclear Vessels.

By signing this certificate, neither the Inspector nor his employer makes any warranty, expressed or implied, concerning the part described in this manufacturer's partial data report. Furthermore, neither the Inspector nor his employer shall be liable in any manner for any personal injury or property damage or a loss of any kind arising from or connected with this inspection.

Date 7-17 1968

[illegible]
Inspectors Signature

Commissions [illegible]
National Board or State and No.

60 181

FORM N-2 (back)

Items 4-8 incl. to be completed for single wall vessels, jackets of jacketed vessels, or shells of heat exchangers.

4. Shell: Material (A300) A516 GR70 (Kind & Spec. No.) T.S. 70,000 (Min. of Range specified) Nominal Thickness 2 5/8 in. Corrosion Allowance 0 in. Diam. — ft. — in. Length — ft. — in.

5. Seams: Long DBL.Butt Weld H.T.[1] — X.R. Complete Efficiency 100 % (If Class B) —

Girth — H.T.[1] — X.R. — No. of Courses

6. Heads: (a) Material None T.S. (b) Material T.S.

Location (Top, bottom ends)	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Press. (Conv. or Conc.)
(a)								
(b)								

If removable, bolts used (Material, Spec. No., T.S., Size, Number) Other fastening (Describe or attach sketch)

7. Jacket Closure: None (Describe as ogee and weld, bar, etc. If bar give dimensions, if bolted, describe or sketch).

8. Constructed for specified operating pressure[2] 56 psi at max. temp. 281 °F Charpy Impact 20 ft-lb at temp. of 0 °F

Items 9 and 10 to be completed for tube sections.

9. Tube Sheets: Stationary. Material (Kind & Spec. No.) Diam (Subject to pressure) Thickness in. Attachment (Welded, Bolted)

Floating. Material Diam Thickness in. Attachment

10. Tubes: Material O.D. in. Thickness inches or gage. Number Type (Str. or U)

Items 11-14 incl. to be completed for inner chambers of jacketed vessels, or channels of heat exchangers.

11. Shell: Material (Kind & Spec. No.) T.S. (Min. of Range specified) Nominal Thickness in. Corrosion Allowance in. Diam ft. in. Length ft. in.

12. Seams: Long H.T.[1] X.R. Efficiency % (If Class B)

Girth H.T.[1] X.R. No. of Courses

13. Heads (a) Material T.S. (b) Material T.S.

Location	Thickness	Crown Radius	Knuckle Radius	Elliptical Ratio	Conical Apex Angle	Hemispherical Radius	Flat Diameter	Side to Pressure (Convex or Concave)
(a) Top, bottom, ends								
(b) Channel								

If removable, bolts used (a) (b) (c) Other fastening (Describe or attach sketch)

14. Constructed for specified operating pressure[2] psi at max. temp. °F Charpy Impact ft-lb at temp. of °F

Items below to be completed for all vessels where applicable.

15. Safety Valve Outlets: Number Size Location

16. Nozzles:

Purpose (Inlet, Outlet, Drain)	Number	Diam. or Size	Type	Material	Thickness	Reinforcement Material	How Attached

17. Inspection Openings: Manholes, No. Size Location
Handholes, No. Size Location
Threaded, No. Size Location

18. Supports: Skirt (Yes or No) Lugs (Number) Legs (Number) Other (Describe) Attached (Where & How)

[1] If Postweld Heat-Treated.

[2] List other internal or external pressure with coincident temperature when applicable.

APPENDIX M

REACTOR PRESSURE VESSEL DESIGN REPORT

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REACTOR PRESSURE VESSEL DESIGN REPORT

M.1 INTRODUCTION TO THE REPORT

The reactor pressure vessel for the Pilgrim Nuclear Power Station was fabricated, inspected, and tested in accordance with the American Society of Mechanical Engineers' Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels" 1965 edition and addenda, plus the Nuclear Code Cases applicable. There were no deviations to the formal codes throughout the design, fabrication, inspection, and testing of the reactor pressure vessel.

The ASME Code and Cases were used as bases:

1. To establish minimum thickness of shell, head, flange, and nozzle materials,

2. To establish inspections and tests required by the Commonwealth of Massachusetts and any local governing bodies

Additional design rules, inspections, and tests not covered by the ASME Code and Cases are defined in the vessel specification.

A series of exhibits constitute the main body of this report. These exhibits present the purchase specifications, inspection report, fabrication test program, summary of tensile tests of special steels, and earthquake analysis of the reactor pressure vessel. These exhibits support the statement made in the opening paragraph. Reactor vessel stresses and analyses are summarized in Section C.3.4.1 of Appendix C. Stress analysis requirements and load combinations for the reactor vessel have been evaluated for the primary loading and cyclic conditions expected throughout the vessel life, with the conclusion that ASME code limits are satisfied.

A comparison of tentative thicknesses for the shell and heads with the formulas contained in Article I-1 follows:

Per Article I-1 of Section III of the ASME Boiler and Pressure Vessel Code, the minimum tentative thicknesses for the cylindrical shell and hemispherical heads are as follows:

A. Shell - $t = \frac{pR}{S_m - 0.5p}$ + Corrosion allowance

$$t = \frac{1250(113.812)}{26{,}700-0.5(1250)} + 0.062 = 5.5185 \text{ in}$$

Drawing minimum thickness = 5.53125 in

B. Heads - $t = \dfrac{pR}{zs_m - p}$ + Corrosion allowance

$t = \dfrac{1250(112.344)}{2(26,700)-1250}$ + 0.062 = 2.755 in

Drawing minimum thickness = 3.750 in

Figures M.1-1 and M.1-2 are provided as a part of this introduction to show the vessel physical dimensions and characteristics.

A description of the nozzle safe ends and weld pads on the Pilgrim reactor vessel is included in Section 4.2.5.1.

Figure M.1-1 and M.1-2 have been removed.

Please refer to BECo Controlled Drawings 1979-216-6(C.E.)
and 1979-185-3(C.E.) respectively.

M.2 SUMMARY

M.2.1 Purchase Specifications

The General Electric purchase specifications define the design specifications, control drawings, and responsibilities of Buyer and Seller imposed on Combustion Engineering, Inc., by the Atomic Power Equipment Department.

The original Purchase Specification 22A1110, September 11, 1967[1] provides detailed engineering requirements for the reactor pressure vessel. The Purchase Specification Data Sheet 21A1110AB, Revision 13,[2] sets forth the revisions to the original specification. The specifications are certified as conforming to ASME Code, Section III, paragraph N-140.

M.2.2 Manufacturer's Inspection Report

The Manufacturer's Inspection Report for the 224 inch id Boiling Water Reactor Vessel and Components dated January 26, 1970[3] is in response to GE Contract No. GE205-B-1171 Pilgrim I and Combustion Engineering, Inc. Contract No. 21466. The report includes pertinent certifications concerning Reactor Vessel No. 66107, and Closure Head No. 66207. These documents certify the results of various tests and inspections of the vessel and head, and demonstrate conformance to applicable ASME Codes and Cases, purchase specifications, etc. This report includes a temperature log showing the final post weld heat treatment vessel metal temperatures and the cross references between the manufacturer's piece numbers, and the material manufacturer's heat numbers. All results were certified to be satisfactory.

M.2.3 Fabrication Test Program

The Fabrication Test Program is reported in Fabrication Test Program for Pilgrim 224 - inch id BWR. Combustion Engineering, Inc., August 12, 1968.[4] This report included certifications concerning special welding, chemical and physical analyses, results of Charpy V notch impact testing, and tensile data.

M.2.3.1 Reactor Vessel Material Surveillance Program

The reactor vessel material surveillance program monitors changes in the fracture toughness properties of the ferritic materials in the reactor vessel beltline region resulting from their exposure to the neutron irradiation and thermal environment.

Pilgrim began with 3 surveillance capsules located circumferentially along the reactor vessel inside radius at the 30-degree, 120-degree and 300-degree azimuths and axially at the reactor vessel core mid-plane. Each surveillance capsule consists of three flux wires made of Copper, Iron and Nickel. The 30-degree capsule was withdrawn in 1980 after 4.17 Effective Full Power Years (EFPYs) of operation. The flux wire measurements derived from the Pilgrim surveillance capsule removed from the Pilgrim reactor vessel during the 1980 refueling outage, and the neutron transport calculations performed in 1985 form the bases of the calculations for projected fluence values used to predict future adjustments to the reactor vessel pressure-temperature limits.

Pilgrim is a participant in the Boiling Water Reactor (BWR) Vessel Internal Project (VIP) Integrated Surveillance Program (ISP) and Supplemental Surveillance Program (SSP). BWRVIP ISP/SSP is an alternative to individual plant-specific RPV surveillance program within the scope of paragraph III.C of Appendix H of 10 CFR 50. BWRVIP ISP/SSP is described in BWRVIP- 86A Report. The NRC has approved BWRVIP ISP/SSP for plant-specific use. Under the NRC approved program, the two Pilgrim specimens are deferred and representative specimens from host plants are selected to provide the required data for compliance with Appendix G and H requirements. The Pilgrim representative samples and withdrawal schedules are described in BWRVIP-86-A, "BWR Vessel and Internal Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan". Pilgrim will continue to follow the BWRVIP ISP/SSP program to demonstrate fracture toughness requirements and P-T limits to comply with Appendices G and H of 10 CFR 50. Pilgrim will provide future fluence calculations and P-T curves based upon the NRC approved methodology prescribed in R.G.1.190, R.G.1.99, Rev. 2, and dosimetry data obtained from BWRVIP ISP/SSP.

M.2.4 Tensile Testing

Tensile test results are reported in Tensile Test of SA533B Steel Specimens , Fritz Engineering Laboratory Report No. 200.68.414.3, certified by Lehigh University, March 10, 1969.[5] The test results on six specimens are tabulated on table entitled: Summary of Test Data (sheet 1 of the report). This table is supported by detailed reports on each specimen. These tests, or large diameter tensile test specimens, verify that small diameter tensile data are representative of tensile properties of thick plate sections. These reports also include data on movement of loading head stress curves and graphs and photographs of fractured surfaces and the extensometer arrangement used in these tests.

M.2.5 Earthquake Analyses

The method used to determine maximum accelerations, shears, and moments of the Pilgrim reactor pressure vessel and internals is reported in Pilgrim Station No. 500 Boston Edison Company Seismic Analysis to Reactor, August 1970.[6] The results of the analysis for those components with significant seismic loads are given on Table 1 (sheet 2 of the report). A discussion of results and supporting reference material are given on sheets 15, 16, and 17 of the report.

M.2.6 References

1. Purchases Specification No. 21A1110

2. Purchase Specification Data Sheet 21AI110AB

3. Inspection Report

4. Fabrication Test Program

5. Tensile Tests of SA533B Steel Specimens

6. Earthquake Analysis

APPENDIX N

EMERGENCY PLAN

N.1 EMERGENCY PLAN

An Emergency Plan which describes the Boston Edison Company's plans for coping with emergencies that may arise at the Pilgrim Nuclear Power Station. This plan has been prepared to meet the requirements of 10CFR50.34(b) and 10CFR50 Appendix E.

The plan is entitled "Pilgrim Nuclear Power Station Emergency Plan" and is maintained in a document separate from this FSAR.

APPENDIX O

ANALYSIS OF THE CONSEQUENCES OF HIGH ENERGY PIPING FAILURES OUTSIDE THE PRIMARY CONTAINMENT

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ANALYSIS OF THE CONSEQUENCES OF HIGH ENERGY PIPING FAILURES OUTSIDE THE PRIMARY CONTAINMENT

O.1 INTRODUCTION

This Appendix to the Pilgrim Nuclear Power Station (PNPS) Final Safety Analysis Report (FSAR) contains a description of the analysis performed to assess the potential consequences of postulated high energy piping failures outside the primary containment. The analysis was conducted in accordance with the considerations presented in the attachment to the AEC letter of December 18, 1972,[1] as modified by the errata dated January 19, 1973,[2] and incorporated the assumptions given in Section O.2.

As a result of this analysis, certain modifications were planned for incorporation into the station design to enlarge the available safety margins. Pending completion and AEC acceptance of these modifications, the AEC issued interim surveillance requirements to the PNPS Technical Specifications on December 20, 1974,[3] to ensure that the facility could withstand the consequences of postulated high energy line ruptures outside containment without losing the capability to achieve and maintain safe shutdown.

Boston Edison Company believes that the quality of the design and installation of the high energy piping systems in the Pilgrim Nuclear Power Station, Unit 1, makes their failure an extremely unlikely event. However, based on our review of the potential consequences of the postulated high energy line failures outside the containment, certain modifications have been incorporated into the station design to enlarge the available safety margins.

The modifications committed to have been incorporated into the station design and are summarized in Section O.6 and Table O.6-3.

O.1.1 References

1. AEC Letter from A. Giambusso to James M. Carroll, December 18, 1972.

2. AEC Letter from John F. Stolz to James M. Carroll, January 19, 1973.

3. AEC Letter from Dennis L. Ziemarm to Maurice J. Feldman, December 20, 1974.

0.2 ANALYSIS ASSUMPTIONS

1. The assumed modes of piping failure are:

   a. Circumferential breaks are those breaks which are perpendicular to the pipe axis. The break area is taken to be the same as the internal cross-sectional area of the pipe unless the pipe is adequately restrained to prevent relative motion of the two sides of the break. Dynamic forces resulting from such breaks are assumed to separate the piping axially and cause whipping in any direction normal to the pipe axis, unless the pipe is adequately restrained to prevent such motion.

   b. Longitudinal breaks are those breaks which are parallel to the pipe axis. The break area equals the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis, unless the pipe is adequately restrained to prevent such motion.

2. Circumferential and longitudinal breaks have been assumed to occur in each piping run or branch run of Class I or seismically analyzed piping. Only a single break has been assumed to occur. Their locations are:

   a. Terminal ends

   b. Any intermediate locations between terminal ends where the stresses calculated in accordance with ANSI B31.1.0-1967 under the loadings associated with seismic events and operational plant conditions exceed 0.8 (S_h + S_A)* or the expansion stresses exceed 0.8 S_A.

   c. Two intermediate locations in addition to those determined above, selected on the basis of highest stress determined by taking the sum of normal operation stresses and seismic stresses.

   Based on the requirements of MEB 3.1, Revision 2, these additional two intermediate breaks need not be postulated for the RWCU piping from penetration X-14 to the regenerative heat exchanger.

---

*S_h is the allowable stress at operating temperatures as given in ANSI B31.1.0-1967, Code for Pressure Piping.

S_A is the allowable stress range for expansion stress calculated by the rules of the Code for Pressure Piping, ANSI B31.1.0-1967.

d. A critical size crack has been postulated to occur at any location along the length and at any point around the circumference of a pipe carrying high energy fluid. Critical size cracks are taken to be one-half the pipe diameter in length and one-half the wall thickness in width.

Break locations and types postulated for various pipe sizes are listed in Table 0.2-1.

3. Other conditions assumed coincident with the assumed piping failure are:

   a. The postulated failure has been conservatively assumed to occur during normal steady-state operating conditions at rated power.

   b. Loss of offsite ac power has been assumed to occur concurrently with the postulated failure of the high energy pipe. This is, in general, a conservative assumption. However, in some cases, loss of normal ac power actually mitigates the effects of the postulated pipe failure, e.g., failure in the feedwater system. In those cases, it has been conservatively assumed that offsite power is not lost.

   c. No other accident has been assumed to occur concurrently with the pipe failure outside the primary containment.

   d. A single failure of an active component has been assumed to occur simultaneously in analyzing the accident and the ability to safely shut down the plant.

   e. A whipping pipe cannot fail another pipe of the same size or larger diameter with the same or larger wall thickness.

   f. Plastic hinge points for a postulated whipping pipe were considered to be at the next indicated breakpoint as this is the next highest stressed point. The position of the hinge is maintained unless the pipe in its movement contacts an item of equivalent or higher strength and shows that the contacted item can withstand the imparted energy (See Figures O.2-1 and O.2-2).

   g. Operator action is relied upon to mitigate the potential consequences of postulated piping failures outside the primary containment when such action is a planned response to station emergencies and when sufficient time is available for the initiation of appropriate operator responses. Actions which may be required in addition to automatic system responses include evaluating affected areas, isolating piping failures if the failures are not automatically isolated and scramming the reactor. Existing station emergency procedures will be followed after the postulated piping failure in the Turbine Building or in the Reactor Building. These procedures require operator evaluation of the abnormal conditions resulting from the postulated failures.

4. Water flooding was considered for each area affected by a postulated high-energy line break. The following assumptions were made:

   a. For a feedwater line break in the main steam tunnel (el 23 ft), flashing steam escapes through the blowout panel into the turbine building. For other postulated breaks, all flashing steam is condensed at atmospheric conditions within the compartment where the break occurs.

b. Only the gravity induced flow paths of water are considered.

c. The flood levels are maximized based upon quasi-static analysis of flow between relatively "still" rooms/areas. Wave fronts and reflections from walls and objects are deemed insignificant.

d. Only the limiting break in each compartment is considered.

e. Floor hatches with covers are considered water tight.

f. No credit is taken for floor drainage.

g. No credit is taken for shallow low points, such as floor slopes, or tapered walls.

h. Leak rates and duration of leakage are based upon total flows. Durations represent isolation valve closing times per Technical Specifications. Leak rates represent critical flows for the line losses from the fluid reservoir up and downstream of the break location to the break location.

   The leak rate and duration for a feedwater line break in the main steam tunnel represent actual system configuration, main steam isolation valve closure upon high steam tunnel temperature, makeup to condenser and hotwell normal content.

5. A conservative analysis was employed for short-term depressurization of the various compartments. The following assumptions were made:

   a. No heat transfer takes place between the escaped fluid and its surroundings and all of the energy in a compartment after pressurization is dissipated by the pressure drop across the open vents.

   b. Energy content in the escaped fluid remains constant throughout the accident.

   c. Openings in walls, floors and ceilings are treated as square edged orifices.

   d. Steam and air are homogeneously mixed during the pressurization of a compartment.

   e. Back pressure existing is equal to atmospheric pressure outside the compartment.

   f. The steam-air mixture obeys the perfect gas law.

TABLE O.2-1

POSTULATED BREAK LOCATIONS
AND TYPES OF BREAKS FOR VARIOUS PIPE SIZES

Break Location	<1"	1" to 3-1/2"	≥ 4"
Terminal ends at equipment, branch connection to main run and at anchor points	NONE	Circumferential	Circumferential & Longitudinal
Location per "minimum two between terminal ends" criteria	NONE	Circumferential	Circumferential & Longitudinal
Tees	NONE	Small Branch Weld	Smaller Branch Weld
Elbows	NONE	Circumferential at Both Welds	Both Longitudinal & Circumferential Welds Unless Circumferential Weld is specifically shown only. Longitudinal On Flat Sides Only

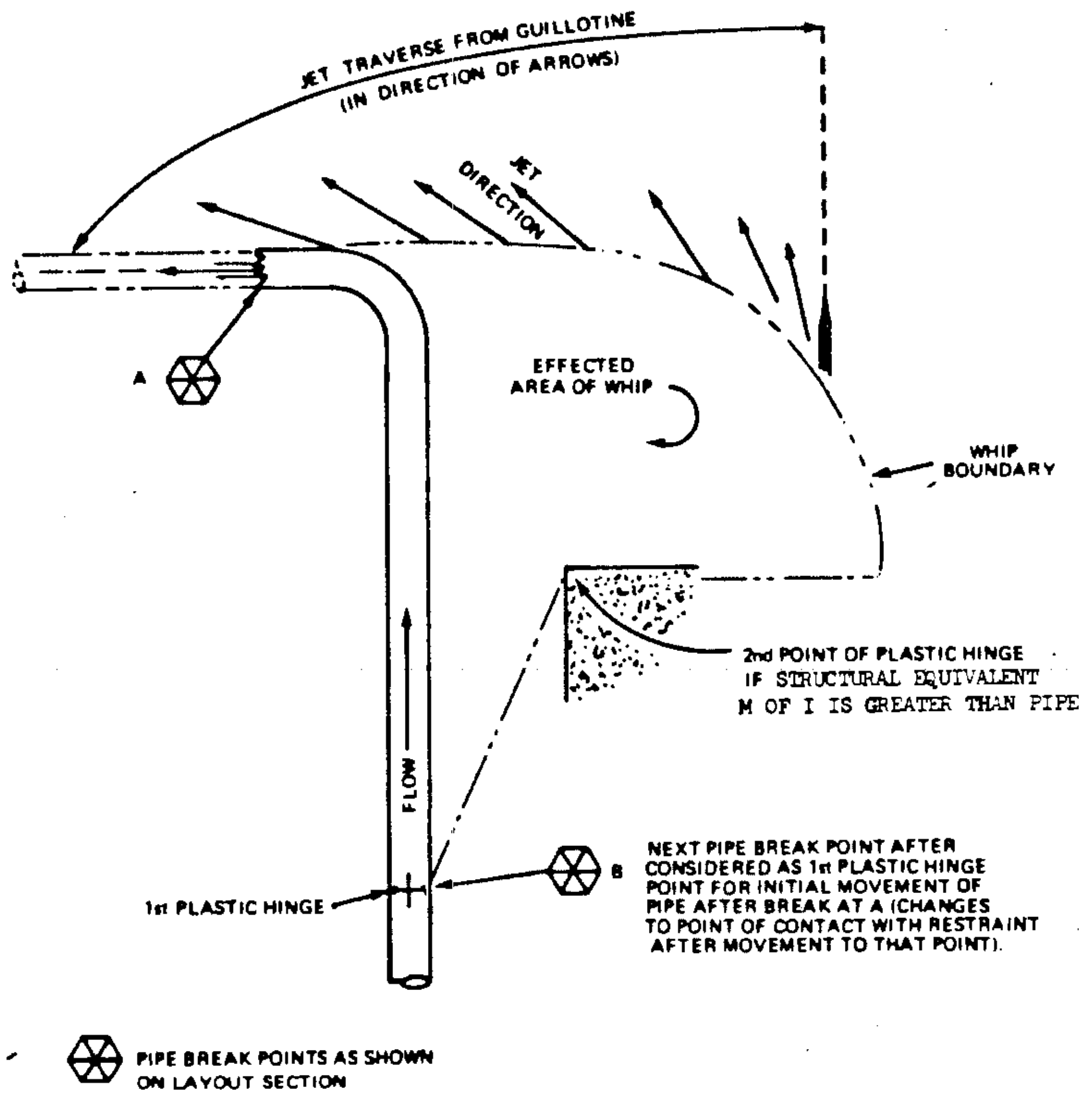
NOTES:

1. Bechtel Design Guide for Pipe Break Protection, Draft December 1980, does not postulate Longitudinal breaks at terminal ends.

2. Bechtel Design Guide does not postulate longitudinal breaks at locations selected by the "minimum two between anchor points" criteria.

3. Bechtel Design Guide clarification for unspecified orientation, in Amendment 34.

Longitudinal and circumferential jet zones of influence of the upstream pipe break end are shown for the Reactor Building only. Only circumferential upstream jet zones of influence are shown for the Turbine Building. All jets are shown to expand at ten degree half angles. Jet lengths shown are arbitrary. Axial views of jets, jet zones on whip paths and small crack jet zones are not shown for clarity. Jet interferences by equipment and structures are disregarded.



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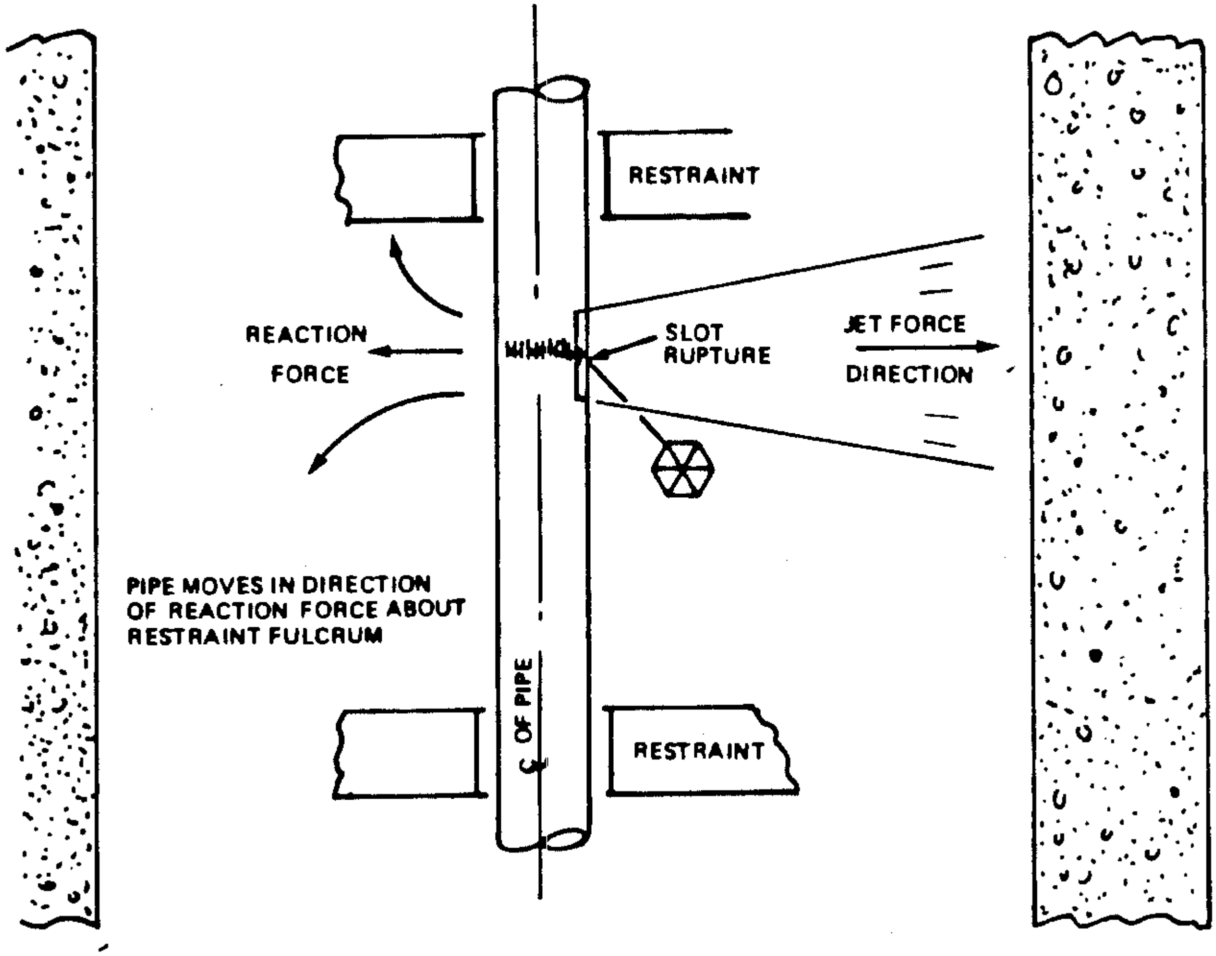
FINAL SAFETY ANALYSIS REPORT

PIPE WHIP MODEL FOR GUILLOTINE FAILURE

FIGURE 0.2-1



PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPE WHIP MODEL
FOR SLOT FAILURE

FIGURE 0.2-2

O.3 ANALYSIS APPROACH

The procedure used to conduct the analysis is outlined below:

1. The high energy piping considered in this review is defined as piping containing fluid at a temperature above 200°F coincident with a pressure above 275 psig during normal operation. The systems containing high energy piping are as follows:

    a. Main steam,
    b. Feedwater,
    c. RCIC,
    d. HPCI,
    e. RWCU and
    f. Sampling

2. High energy line failures are assumed to result in pipe whip, jet impingement, missiles, water flooding, steam flooding and compartment pressurization, as applicable. For each high energy line area considered, a detailed list of safety-related systems and components that could potentially be affected by the postulated failure was prepared. This list included equipment in adjacent compartments that could be affected by the environmental conditions caused by steam venting to those compartments. This list was reviewed to determine the effects of items assumed to be lost on reactor shutdown capabilities. Corrective measures were developed where reactor shutdown capabilities might be affected.

3. Compartment pressurization analyses were conducted to determine the individual maximum compartment temperatures and pressures resulting from the various postulated high energy line failures. The results of these analyses are summarized in Table O.6-2. The individual compartment structures were analyzed to assess their structural integrity under the loadings resulting from the compartment pressurizations determined. In each area containing a postulated high energy piping failure, the walls, roof, and floor were analyzed to ensure that these structures, where required, would withstand the forces due to compartment pressurization, jet force and impact load due to pipe whip, as applicable.

0.4 STRUCTURAL LOADING ANALYTICAL TECHNIQUE

0.4.1 General

The structural loading resulting from a postulated break in a high energy piping system outside of the containment is the subject of this section. The pipe rupture can be classified as guillotine, slot or crack. This rupture may cause a jet impingement or a pipe whip loading on structures whose integrity is important to safe shutdown of the plant.

A simplified analysis and design approach was chosen based on conservative assumptions in material and structural behavior. The basic reference in determining the structural response from the effects of impact and impulse loadings resulting from pipe failure is Bechtel Corporations BN-TOP-2, Design for Pipe Break Effects, September 1972. The material presented herein supplements BN-TOP-2 in the following areas:

1. Expands the material given in BN-TOP-2 and demonstrates the analytical techniques to cover plastic impact.
2. Gives the equations necessary to evaluate resistance and yield displacement for several typical systems.
3. Lists ductility ratios for various materials and structural systems.
4. Uses dynamic strength of materials in evaluating structural capacity.
5. Gives examples illustrating the analytical techniques.

0.4.2 Loading

The type of loading is a function of the mode of failure, direction of flow and the location of the rupture. Some examples are shown on Figure 0.4-1. For a jet, a ten degree half angle is used to simulate dispersion.

The required strength U shall be at least equal to:

$$U = 1.0L + 1.0D + 1.0F + 1.0P + 1.0T \qquad (4.2\text{-}1)$$

Where:

L = live load (lb)
D = dead load (lb)
P = environmental pressure load (lb)
T = temperature load (lb)
F = rupture load which may be due to a jet impingement or a whipping pipe (lb)

In most cases, D and L are negligible, F and P do not peak simultaneously, and T affects only the surface.

When pipes under pressure fail, a fluid jet is created and a force is exerted on the piping. The force exerted on the piping system has a rapid buildup and decreases to zero as the system blows down. This force time relationship is illustrated on Figure 0.4-2. The dynamic solution for a system subjected to this rupture force may be simplified if the force is conservatively assumed to build up instantaneously and then remain constant as shown on Figure 0.4-2. The assumption that the force remains constant with time also means that it remains a constant with respect to displacement and simplified analytical techniques can be used which are shown in Section 0.4-3.

0.4.3 Analytical Techniques

0.4.3.1 General

To obtain a solution for the actual complex system, the structural system is converted to an equivalent one or two degrees-of-freedom system. Of particular interest are the responses to an impulse by a fluid jet and an impulse-impact from a whipping pipe. These structural responses will be treated in subsequent sections.

0.4.3.2 Impulse Loading

To illustrate the concept of impulse loading, consider the simple system shown on Figure 0.4-3. The work done by the dynamic load is Fx and the energy absorbed by the equivalent system equals the area under the resistance-displacement curve given by:

$$Fx = R(x - \frac{x_y}{2}) \tag{4.3-1}$$

Where:

R = resistance force (lb)
x_y = displacement at yield (in)
F = jet force (lb)

Equating the work done to the energy absorbed leads to the following general solution for a constant magnitude impulse load acting on an elastoplastic system:

$$R = \frac{Fx}{x - \frac{x_y}{2}} \tag{4.3-2}$$

Defining the ductility ratio by

$$\mu = \frac{x}{x_y} \tag{4.3-3}$$

the solution can be expressed as

$$R = \frac{F_\mu}{\mu - \frac{1}{2}} \tag{4.3-4}$$

As an example, if a system is allowed to displace to three times yield, i.e., μ equals 3, then R equals $1.2F$. This requires that the resistance or capacity must be at least 20 percent higher than the dynamic load F to stop the system within a displacement of three times yield. For a brittle failure within the elastic range, if the ductility ratio (μ) is taken to equal 1, then R equals $2F$. Therefore, if the response is to remain elastic, the resistance must be twice the applied load since the solution assumed the load was suddenly applied.

The preceding solution for an impulse load is applicable to situations where the fluid jet is impinging on a structural element or the rupture force is acting on a piping system in contact with its supports or restraints.

O.4.3.3 Impact Combined with Impulse Load

An impact problem is here defined to be the collision of two bodies with an energy transfer during impact. A common situation associated with this type of loading occurs when the ruptured piping system travels across a gap to strike a structural element. An equivalent simplified system is shown on Figure O.4-4.

If the effective masses are not easily evaluated or a very conservative solution is desired, then the following equation can be written.

$$\underbrace{Fx}_{\text{Work Done}} = \underbrace{R_p\left(x - \frac{x_p}{2}\right)}_{\text{Energy Absorbed by Pipe}} + \underbrace{R_r\left(x - x_g - \left[\frac{x_r - x_g}{2}\right]\right)}_{\text{Energy Absorbed by Restraint}} \tag{4.3-5}$$

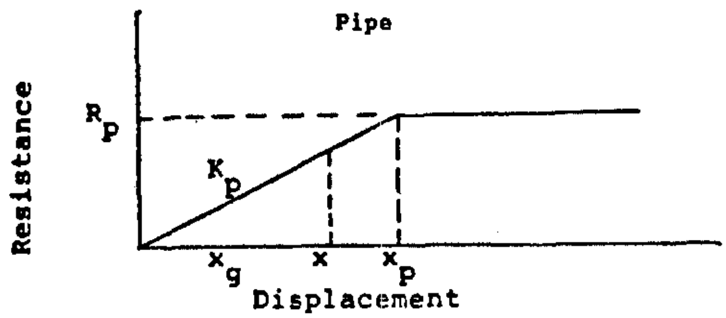
This equation assumes a full transfer of energy to the restraint after the pipe system crosses the gap.

The only unknown in Equation 4.3-5 is x and its solution represents the pipe displacement while the restraint displacement is given by $(x - x_g)$.

The preceding solution is valid when:

$x_g > x_p$
$x > x_p$
$x > x_r$

Consider next the case where the pipe remains elastic in the previous problem, then the pipe resistance curve on Figure 0.4-4 becomes:



and the solution is:

$$Fx = K_p \frac{x^2}{2} + R_r \left[(x - x_g) - \frac{x_r - x_g}{2} \right] \qquad (4.3\text{-}6)$$

for the limits $(x > x_g,\ x < x_p,\ x < x_r)$.

Since there are many different solutions depending on the displacement relationships, it is important to understand the analysis technique and handle each problem on a case-by-case basis, if necessary.

The previous solutions assumed a full energy transfer upon impact, which is very conservative. A more realistic solution involves a fully plastic impact which assumes that a portion of the mass of the pipe remains in contact with the restraint mass after impact. This fully plastic impact concept is valid since large forces remain on the piping system after impact and since the pipe deforms locally.

The concept of plastic impact utilizing the conservation of momentum can be illustrated by considering the two masses shown below:

Before Impact   After Impact

The conservation of momentum gives:

$$mv = (m + M)V \tag{4.3-7}$$

and

$$V = \frac{m}{M + m} v \tag{4.3-8}$$

The initial kinetic energy before impact is:

$$KE = \frac{1}{2} mv^2 \tag{4.3-9}$$

and after impact is:

$$KE' = \frac{1}{2} (m + M) \left[\frac{m}{M + m} v\right]^2$$

$$KE' = \frac{1}{2} mv^2 \left(\frac{m}{M + m}\right) = KE \frac{m}{M + m} \tag{4.3-10}$$

Therefore, the kinetic energy after impact is $\left(\frac{m}{m + M}\right)$ times the initial kinetic energy. The energy which appears lost has been absorbed in the deformation of the masses during impact.

Applying the concept of plastic impact to the system shown on Figure 0.4-4 gives:

$$\underbrace{\left[\frac{m_p}{m_p + m_r}\right]\left[Fx_g - R_p\left(x_g - \frac{x_p}{2}\right)\right]}_{\text{External Energy Transferred During Plastic Impact}} + \underbrace{F(x - x_g)}_{\text{Work Done After Impact}} \;\underset{\text{Equals}}{=} \qquad (4.3\text{-}11)$$

$$\underbrace{R_p\left(x - x_g\right)}_{\text{Energy Absorbed By Pipe After Impact}} + \underbrace{R_r\left(x - x_g - \frac{x_r - x_g}{2}\right)}_{\text{Energy Absorbed By Restraint}}$$

where:

m_p = pipe mass (lbm)

m_r = restraint mass (lbm)

Again, x is the only unknown in Equation 4.3-11 and its solution represents the pipe displacement while the restraint displacement is given by $(x - x_g)$. This solution is valid when:

$x > x_p$
$x > x_r$
$x_g > x_p$

In summary, the force-time relationship was conservatively simplified so that the force is a constant with respect to displacement. This enables a solution by equating work done to energy absorbed and eliminates the necessity of solving an equation of motion for the system, which would yield the same results.

0.4.3.4 Resistance-Displacement Relation

The previous section showed the necessity of having a resistance versus displacement curve for each resisting element in the system. As an illustration of the determination of a resistance displacement curve,

consider the fully fixed beam shown on Figure 0.4-5. For the load at the center of the span, applying the virtual work principle to the lowerbound limit analysis mechanism gives the following relationship between the applied load P and the yield moment M_u:

$$P\delta = 4M_u\theta \tag{4,3-12}$$

since:

$$\delta = \theta \frac{L}{2} \tag{4.3-13}$$

then:

$$P = \frac{8M_u}{L} \tag{4.3-14}$$

Since the failure mechanism is associated with a fully hinged condition, the load P will be equivalent to the resistance or capacity at yield:

$$R = P = \frac{8M_u}{L} \tag{4.3-15}$$

For a steel beam:

$$M_u = 0.9f_yS_p \tag{4.3-16}$$

where:

f_y = yield stress (psi)
S_p = plastic section modulus (in^3)
0.9 = capacity reduction factor

For a reinforced concrete member, M_u will be based on the strength principle of ACI-318-71. This will be covered in more detail in Section 0.4.4.

Since R is the load at yielding, its application to the fully fixed elastic beam equation will give a minimum estimate of the displacement at yield:

$$x_y = \frac{R_L{}^3}{192\ EI} \qquad (4.3\text{-}17)$$

where:

E = modulus of elasticity (psi)
I = section moment of inertia (in^4)

With R and x_y determined, the elastoplastic curve can be constructed as shown on Figure 0.4-5.

0.4.4 Criteria

0.4.4.1 Resistance

The resistance of any structural component shall be based on its minimum strength. As an example, for a concrete slab subjected to an impulse loading, yield line theory is applied, and slab strength shall be evaluated based on its flexural, shear and punchout capacity. The lower of the capacities will determine its resistance. The capacity of the restraint elements shall be based on the dynamic material strength which is defined by:

Allowable dynamic stress = allowable static stress x DIF (4.4-1)

where:

DIF = the Dynamic Increase Factor given on Table 0.4-1.

The following codes shall be used where applicable to determine the cross-sectional strengths of various members:

1. Reinforced and Prestressed Concrete

   Building Code Requirements for Reinforced Concrete, (ACI-318-71) American Concrete Institute

2. Structural Steel

   AISC 7th Edition 1970 - Manual of Steel Construction

3. Steel Pressure Vessels

ASME Boiler and Pressure Vessel Code, Section III - 1971, Nuclear Power Plant Components

4. Piping Systems

ANSI B31.1.0-1967 - Code for Pressure Piping

Tables O.4-2 and O.4-3 are provided to determine the resistance capacity based on cross-sectional moment capacity and member dimensions.

O.4.4.2 Ductility, Yield and Maximum Displacement

Tables O.4-2 and O.4-3 provide yield displacement approximations. The resistance and displacement values presented in these tables are only applicable to systems whose flexural strength defines minimum capacity. In evaluating the yield displacement with the usual elastic analysis, an adjustment must be made on the moment of inertia to account for cracking of the concrete. The empirical relation to be used shall follow Section 5-8 of Structures to Resist the Effects of Accidental Explosions, TM5-1300, June 1969. The average moment of inertia (I_a) which should be used in calculating deflection is:

$$I_a = \frac{I_g + I_c}{2} \qquad (4.4\text{-}2)$$

where:

I_g = moment of inertia of gross concrete cross section of thickness t about its centroid (neglecting steel areas)
= $t^3/12$ (in^4/in)

I_c = moment of inertia of the cracked concrete section of width b
= Fd^3 (in^4/in) (see Figure O.4-6 for the values of coefficient F)

This value of I_a shall be used in the displacement equations given on Tables O.4-2 and O.4-3 for all reinforced concrete members.

In situations where the energy input is essentially unlimited, such as a jet (impingement) loading acting on a wall, the allowable ductility value will be limited to 3. This provides a resistance of at least 1.2F based on Equation (4.3-4) where μ equals 3. This limit will only be applied to the primary resisting component, such as the wall. Any secondary resisting components, such as the cracked pipe, may exceed the limit of 3.

In a situation where pipe restraints are provided to limit pipe motion, the restraints are considered as the primary load resisting components and the piping is the secondary component.

When the loading consists of both impact and impulse loading, such as a pipe whip situation, the maximum allowable displacement shall be based on the ductility ratio given on Table O.4-4. In this case, the minimum resistance will also be limited to 1.2F for the primary load resisting component.

For the case of plastic impact, the mass ratio in Equation (4.3-11), $m_p/(m_p + m_r)$, can be written in terms of the weight ratio, $W_p/(W_p + W_r)$. W_p shall be taken to be equal to one-half of the whipping pipe weight bounded by the nearest plastic hinge(s). W_r for a slab or wall shall be the weight in the volume:

$$V_w = (d+t)^2(t) \qquad (4.4\text{-}2)$$

where:

d = striking pipe diameter (in)
t = slab or wall thickness (in)

W_r for a beam shall be the weight in the volume:

$$V_b = (d+2t)^2(t) \qquad (4.4\text{-}3)$$

where:

d = striking pipe diameter (in)
t = depth of the beam (in)

Both the wall and beam effective volume dimensions are shown on Figure O.4-7.

For restraints subjected to compressive loadings, the resistance-displacement curves shall include any reduction due to either elastic or inelastic instability. Flexural loading on restaints and piping may cause lateral instability in the beam compression flange or local buckling in the pipe, and these effects shall be considered in developing the resistance-displacement curves.

0.4.5 Structural Adequacy

0.4.5.1 General

The following general steps are necessary to determine the adequacy of the structural system in resisting the applied loading.

1. Determine the magnitude and location of the peak dynamic force F, according to the methods given in Section 0.5.1. In a simplified form, $F = C_D P A_F$ where P is fluid stagnation pressure, A_F is the effective cross-sectional flow area of the fluid jet and C_D is a factor evaluated according to the methods given in Section 0.5.1, depending on the geometry of break, pipe dimensions, friction effects and fluid phase.

2. Evaluate the resistance versus displacement curve to the maximum allowable displacement which is given by μx_y for all resisting elements.

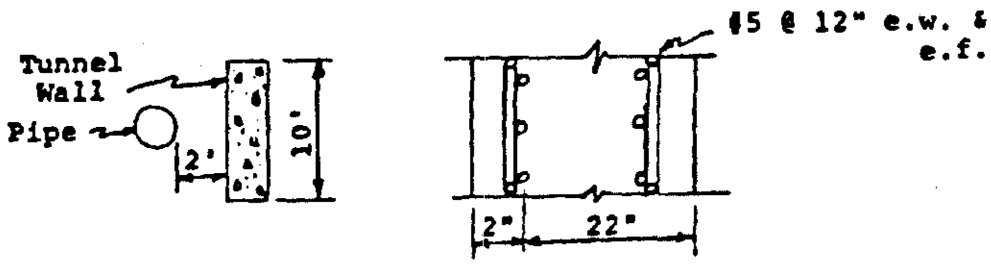
3. Using the applicable impulse or impulse and impact solution technique, evaluate the maximum displacement of the structural element under consideration.

4. If the displacement found in Item 3 is less than the maximum allowable displacement of Item 2, then the system has successfully resisted the load; otherwise, failure is predicted.

0.4.5.2 Examples of Impulse Loading and Impact Combined with Impulse Loading

The following example will illustrate a situation where a pipe located 2 ft from a concrete wall may rupture. The pipe is supported at 30 ft intervals and it is postulated that a slot failure occurs at mid-span. If the rupture occurs on the side of the pipe nearest that wall, then the case becomes one of jet impingement of the wall. On the other hand, the rupture may be on the side of the pipe away from the wall, in which case the pipe will be subjected to a whipping action, and the pipe will subsequently impact against the wall.

0.4.5.2.1 Example of Impulse Loading - Jet Impinges at Center of One Face of a Concrete Tunnel



1. Structural Sizes and Material Constants*

Concrete Wall

Reinforcement	Concrete
$A_s = 0.31\ in^2/ft$	$f'_c = 3$ ksi
#5 @ 12 in	$b = 12$ in
$f_y = 1.2(40) = 48$ ksi	$d = 22$ in

Where 1.2 is the Dynamic Increase Factor (see Table 0.4-1).

$E_s = 30 \times 10^6$ psi

$n = 10$

$\rho = \frac{A_s}{bd} = 0.00117$

$E_c = 3{,}000$ ksi

Piping System

r_p = Mean Pipe Radius = 6 in
t = Wall Thickness = 0.5 in
ID = Inside Diameter = 11.75 in
A_f = Fluid Area = 108.4 in^2
y_p = 1.2(Yield Strength) = 1.2(30) = 36 ksi

Where 1.2 is the Dynamic Increase Factor,

p = Fluid Pressure = 1,000 psi
C_D = Dynamic Fluid Factor = 1.25

*Assumed for illustration purposes only. Actual design calculations are done by calculating thrust force F or the factor C (sonic velocity) according to the methods given in Section 0.5.1.

2. Evaluation of Wall Resistance Function

Section Strength - M_u determination. M_u is to be based on the strength principle of ACI-318-71* (see Section 0.4.3.4). Check that

$\rho \leq 0.75\ \rho_b$

where:

$$\rho_b = \frac{0.85K_1f'_c}{f_y}\left(\frac{87000}{87000 + f_y}\right) \text{ where } K_1 = 0.85 \qquad (4.5\text{-}1)$$

$$= \frac{(0.85)^2(3)}{48}\left(\frac{87000}{135000}\right) = 0.029$$

Checking 0.00117 < 0.75(0.029) OK

$$a = \frac{A_sf_y}{0.85f'_cb} = \frac{0.31(48)}{0.85(3)(12)} = 0.486 \qquad (4.5\text{-}2)$$

$$M_u = \phi A_sf_y(d-0.5a) = 0.9\left[0.31(48)(22-\frac{0.486}{2})\right]$$

$$= 291.4 \text{ K-in/ft} = 24.28 \text{ K-ft/ft} \qquad (4.5\text{-}3)$$

From Table 0.4-3 Case (2)

$$R_r = 4\pi M_u = 4\pi(24.28) = 305K \qquad (4.5\text{-}4)$$

Also From Table 0.4-3 assuming a 10 ft x 10 ft clamped wall panel,

$$X_y = \frac{\alpha Ra^2}{12E_cI_a}(1-\nu^2) \qquad (4.5\text{-}5)$$

*The notations used herein conform to Appendix B of ACI-318-71.

where

$\alpha = 0.0671$

$a = 10 \times 12 = 120$ in

$\nu = 0.2$

$t = 24$ in

$E_c = 3000$ ksi

$$x_y = \frac{0.0671(305)(120)^2\ (0.96)}{12(3000)I_a} = \frac{7.859}{I_a}$$

This form of the equation is used so that the value $t^3/12$ (moment of inertia of gross concrete section) may be averaged with the cracked section moment of inertia.

$$I_g = \text{Gross } I = \frac{(24)^3}{12} = 1150 \text{ in}^4/\text{in} \qquad (4.5\text{-}6)$$

Determine cracked section moment of inertia I_c.

For

$n = 10$
$p = 0.00117$ .... $F = 0.0085$ (see Figure 0.4-6.)

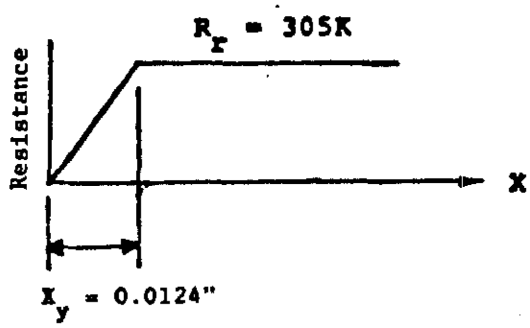
$$I_c = Fd^3 = 0.0085(24)^3 = 118 \text{ in}^4/\text{in} \qquad (4.5\text{-}7)$$

$$I_a = \frac{I_g + I_c}{2} = \frac{1150 + 118}{2} = 634 \text{ in}^4/\text{in}$$

Substituting I_a into the previous equation gives:

$$X_y = \frac{7.859}{634} = 0.0124 \text{ in} \qquad (4.5.8)$$

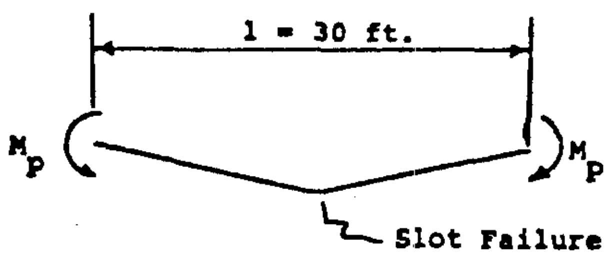
This establishes the following wall resistance function:



This completes the analysis of the wall resistance function.

3. Evaluation of Pipe Resistance Function

The following is a simplification which neglects stresses in the pipe due to pressure, temperature, dead load, etc. Assume the pipe is supported at 30 ft intervals and slot failure occurs at mid-span.



S_p = Plastic Section Modulus = $4r_p^2t = 4(6)^2(0.5) = 72 \text{ in}^3$ (4.5-9)

$I = \pi r_p^3 t = 340 \text{ in}^4$ (4.5-10)

Find M_p to yield the pipe:

Use 0.9 as Capacity Reduction Factor.

$$M_p = (0.9)y_pS_p$$
$$M_p = (0.9)(36)(72) = 2,333 \text{ K-in} \qquad (4.5.11)$$

From the assumption that plastic hinges form at the support points and at mid-span (see Table 0.4-2 Case 3).

$$R_p = \frac{8M_p}{L} = \frac{8(2333)}{12(30)} = 52K \qquad (4.5\text{-}12)$$

Find the maximum displacement at mid-span for yielded condition.

$$X_p = \frac{R_pL^3}{KEI} \qquad (4.5\text{-}13)$$

When:

$$\left.\begin{array}{ll} K \text{ for simple supported span} = & 48 \\ K \text{ for fixed supports} = & 192 \end{array}\right\} \begin{array}{l}\text{average} \\ K = 120\end{array}$$

Then:

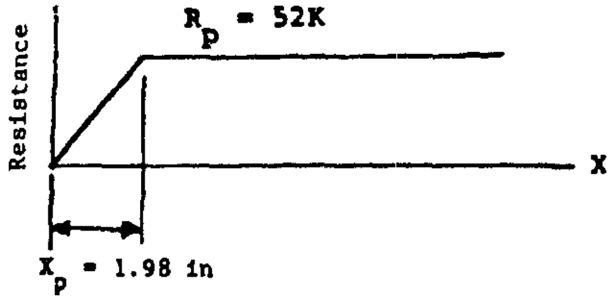
$$X_p = \frac{52(30)^3(12)^3}{120(30,000)(340)} = 1.98$$

μ_p = Maximum ductility ratio for pipe = 30

Therefore, the maximum allowable pipe deflection is:

$x = 1.98(30) = 59$ in

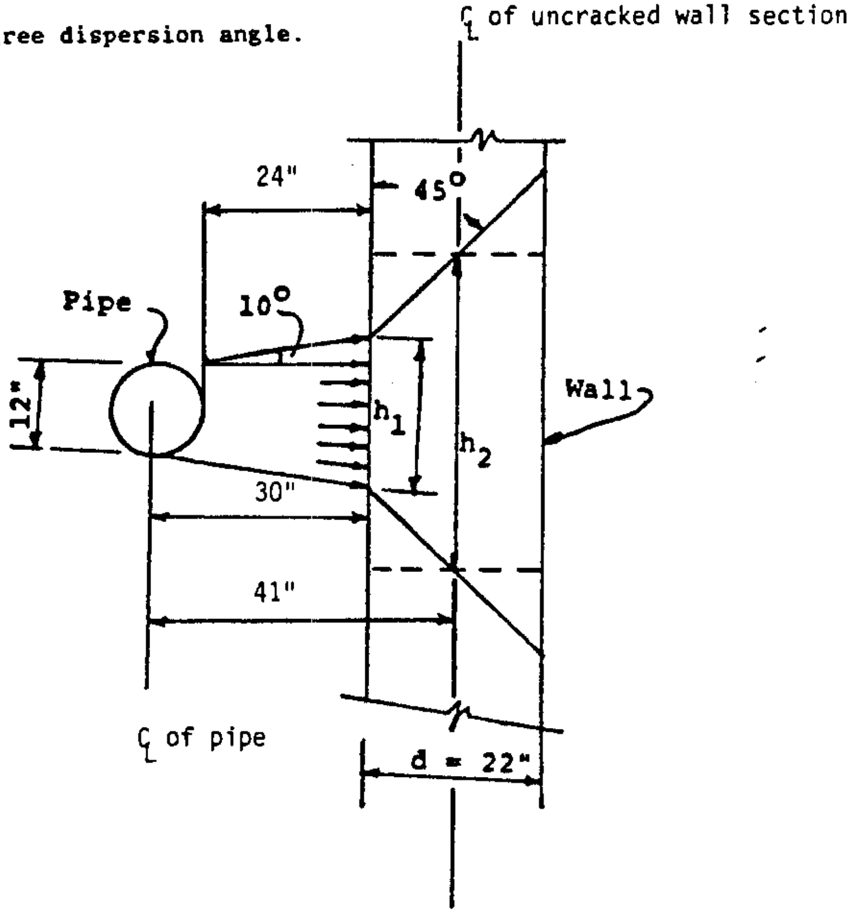
This estabiishes the following pipe resistance function.



4. Jet Impingement on Wall

Consider a 12 in dia pipe located 2 ft from the face of the wall. Total force of jet load, F, equals $C_D p A_f$ or 1.25(1) (108.4) equals 136K. See Section 0.4.5.1.

Use a ten degree dispersion angle.



$h_1 = 12 + 2(24) \tan 10^\circ = 20.5$ in

$h_2 = h_1 + d = 42.5$ in

shear area* $= \pi h_2 d = \pi(42.5)(22) = 2{,}940$ in^2

shear capacity $= (\text{shear area } 4\phi\sqrt{f'c})/1000\text{lb/K}$
$= [2{,}940(4)\phi\sqrt{f'c}]/1{,}000$
$= [2{,}940(4)(0.85)\sqrt{3{,}000}]/1{,}000$
$= 548$K

*In this case, the slot break was assumed to be a circular area since this has a minimum perimeter and is conservative for punch out shear evaluation.

Since shear capacity (548K) is greater than flexural capacity (305K), then flexural capacity is the governing condition, and the previously calculated load displacement (see Section 2 above) is valid. For impulse loading and elastoplastic resistance, an allowable ductility ratio μ must be considered.

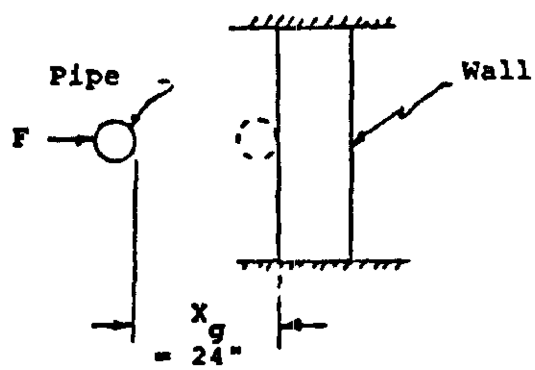
$\mu = 3 =$ Allowable ductility ratio value (see Section 0.4.4)

$$R(\text{required}) = \frac{F_{\mu}}{\mu - 1/2} = 1.2F = 1.2(136) = 163K$$

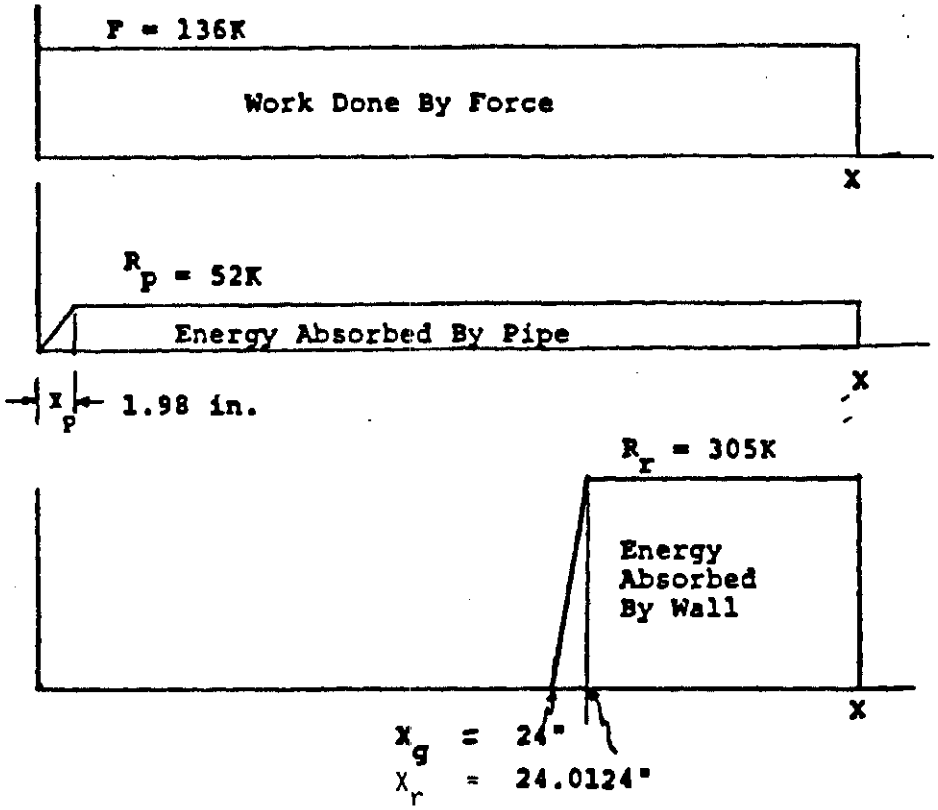
Since $R_r = 305 > 163$, then the minimum requirement for R has been met.

0.4.5.2.2 Pipe Whip Effect

Here the pipe traverses the 24 in gap (X_g) before striking the wall (see Section 0.4.5.2.1, item 4) and the jet reaction (136K) continues to act after the pipe strikes the wall. The resulting flexural behavior of the wall is evaluated and the wall displacement is limited by a maximum allowable ductility ratio of 30. Note that maximum μ equals 10/0.1 equals 100, but use the lower limit of 30.



1. First assume full energy transfer for conservative solution (no plastic impact credit).



$$\underbrace{136X}_{\text{Work Done}} = \underbrace{52\left(X-\frac{1.98}{2}\right)}_{\text{Energy Absorbed By Pipe}} + \underbrace{305\left(X-24-\frac{0.0124}{2}\right)}_{\text{Energy Absorbed By Wall}}$$

Solving for X:

$X = 33.4$ in

The solution is valid since it satisfies the original assumptions.

i.e., $X > X_g$ + Wall Yield.....33.4>24+0.0124

and $X >$ Pipe Yield........33.4>1.98

Wall Displacement = 33.4 - 24.0 = 9.4 in

Maximum Allowable Displacement = $\mu(X_r - X_g)$ = 30(0.0124) = 0.372 in
of wall

Therefore, failure is indicated.

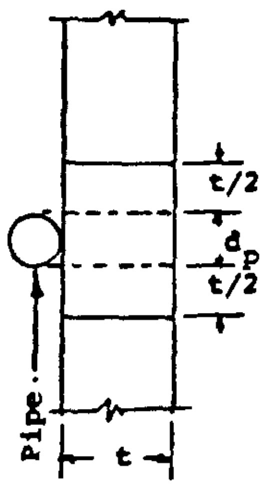
2. Pipe Whip Effect with Credit for Plastic Impact

This calculation is made since failure was indicated in 1 above without credit for plastic impact.

Pipe weight per ft = (1) (3.14) (1) (20) = 62.8 lb

Water weight per ft = $0.785(1)^2$ (62.5) = 49.0 lb
Total = 111.8 lb

Half of 30 ft pipe span shall be considered to be effective in plastic impact, i.e., W_p = 15 W = 1,680 lb.



The sketch indicates the volume of wall effectively entering into the plastic impact.

$$\text{Vol.} = (d_p + t)^2 t = (1+2)^2 (2) = 18 \text{ cu.ft.}$$

$$W_w = 18(150) = 2700 \text{ lbs.}$$

Plastic Impact Factor =

$$\frac{W_p}{W_p + W_w} = \frac{1680}{1680 + 2700} = 0.384$$

The appropriate equation when considering impact energy is Equation (4.3-11) with m_p and m_r replaced by W_p and W_w.

$$\underbrace{\frac{W_p}{W_p + W_w}\left[FX_g - R_p\left(X_g - \frac{W_p}{2}\right)\right]}_{\text{Reduced Energy Transferred From Plastic Impact}} + \underbrace{F(X-X_g)}_{\text{Work Done After Impact}} \underset{\text{Equals}}{=}$$

$$\underbrace{R_p(X-X_g)}_{\text{Energy Absorbed By Pipe After Impact}} + \underbrace{R_r\left[X-X_g-\frac{X_r-X_g}{2}\right]}_{\text{Energy Absorbed By Wall Restraint}} \qquad (4.5\text{-}17)$$

where:

$\frac{W_p}{W_p + W_w} = 0.384$ $\qquad R_p = 52K \qquad X_r - X_g = X_y + 0.0124$ in

$F = 136$ K $\qquad X_p = 1.98$ in

$X_g = 24$ in $\qquad R_r = 305K$

$0.384\ [136(24) - 52(24-0.99)] + 136(X-24)$

$= 52(X-24) + 305(X-24-0.0052)$ $\qquad$ solving for X,

$X = 27.6$ in

The solution is valid since it satisfies the original assumptions, i.e.,

$X > X_g$ + Wall Yield ..... 27.6>24.0 + 0.0124

and $X >$ Pipe Yield ...27.6>1.98

Wall Displacement = $X - X_g$ = 27.6 - 24 = 3.6 in

Maximum Allowable Displacement = μX_y = 30 (0.0124) = 0.372 in

Therefore, failure is still indicated for this example even with credit for plastic impact.

TABLE 0.4-1

DYNAMIC INCREASE FACTOR
(DIF)

	DIF
I. Reinforced or Prestressed Concrete	
a. Concrete	
Compression	1.25
Diagonal Tension & Direct Shear (Punch Out)	1.00
Bond	1.00
b. Reinforcing Steel	
Tension	1.20
Compression	1.20
Diagonal Tension & Direct Shear (Stirrups)	1.00
II. Structural Steel	
Flexure & Tension	1.20
Compression	1.20
Shear	1.00
III. Piping	
Flexure & Tension	1.20
Compression	1.20
Shear	1.00

Reference: Structures to Resist the Effects of Accidental Explosions, TM5-1300, June 1969.

TABLE 0.4-2

RESISTANCE-YIELD DISPLACEMENT VALUES FOR BEAMS

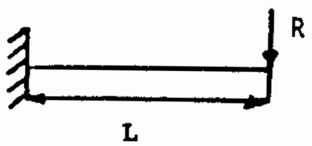
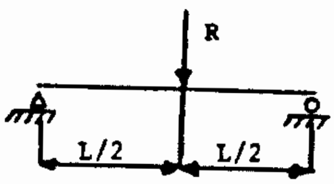
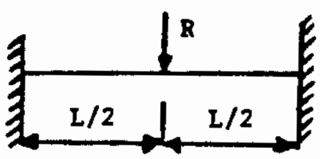
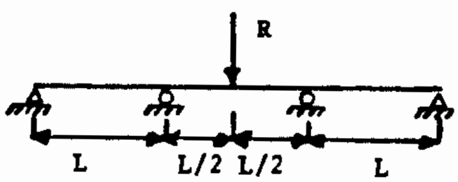
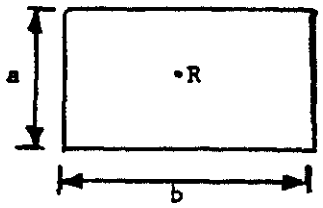
Description	Resistance	Yield Displacement
(1) Cantilever		
	$R = \frac{M_u}{L}$	$x_y = \frac{RL^3}{3EI}$
(2) Simply Supported		
	$R = \frac{4M_u}{L}$	$x_y = \frac{RL^3}{48EI}$
(3) Fixed Supports		
	$R = \frac{8M_u}{L}$	$x_y = \frac{RL^3}{192EI}$
(4) Multi-Span		
	$R = \frac{8M_u}{L}$	$x_y = \frac{0.011RL^3}{EI}$

TABLE 0.4-3

RESISTANCE YIELD DISPLACEMENT VALUES FOR SLABS

Description	Resistance[1]	Yield Displacement[2]
(1) Simply supported on all 4 sides with load at center	$R = 2\pi M_u$	$X_y = \frac{\alpha R a^2}{12EI_a}(1-\upsilon^2)$



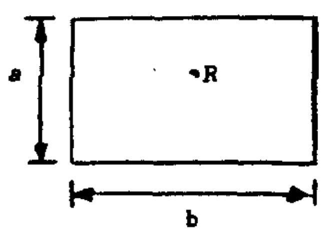
b/a	1.0	1.1	1.2	1.4	1.6	1.8	2.0	3.0	∞
α	0.1390	0.1518	0.1624	0.1781	0.1864	0.1944	0.1981	0.2029	0.2031

(2) Fixed supports on all 4 sides with load at center

υ = Poisson's ratio

t = Thickness (in)

E = Modulus of Elasticity (psi)



$R = 4\pi M_u$ $x_y = \frac{\alpha R a^2}{12EI_a}(1-\upsilon^2)$

b/a	1.0	1.2	1.4	1.6	1.8	2.0
α	0.0671	0.0776	0.0830	0.0854	0.0864	0.0866

Notes:

(1) The resistance functions are taken from:

K.W. Johansen, Pladeformler, Polyteknish Forening, Copenhagen, 2nd Edition, 1949.

K.W. Johansen, Pladeformler Formelsamling, Polyteknish Forening, Copenhagen, 2nd Edition, 1954.

TABLE 0.4-3 (Cont.)

Notes (Cont.)

(2) The displacement equations are taken from:

S. Timoshenko & S. Woinowsky-Krieger, Theory of Plates and Shells, McGraw Hill, 1959.

TABLE 0.4-4

DUCTILITY RATIOS

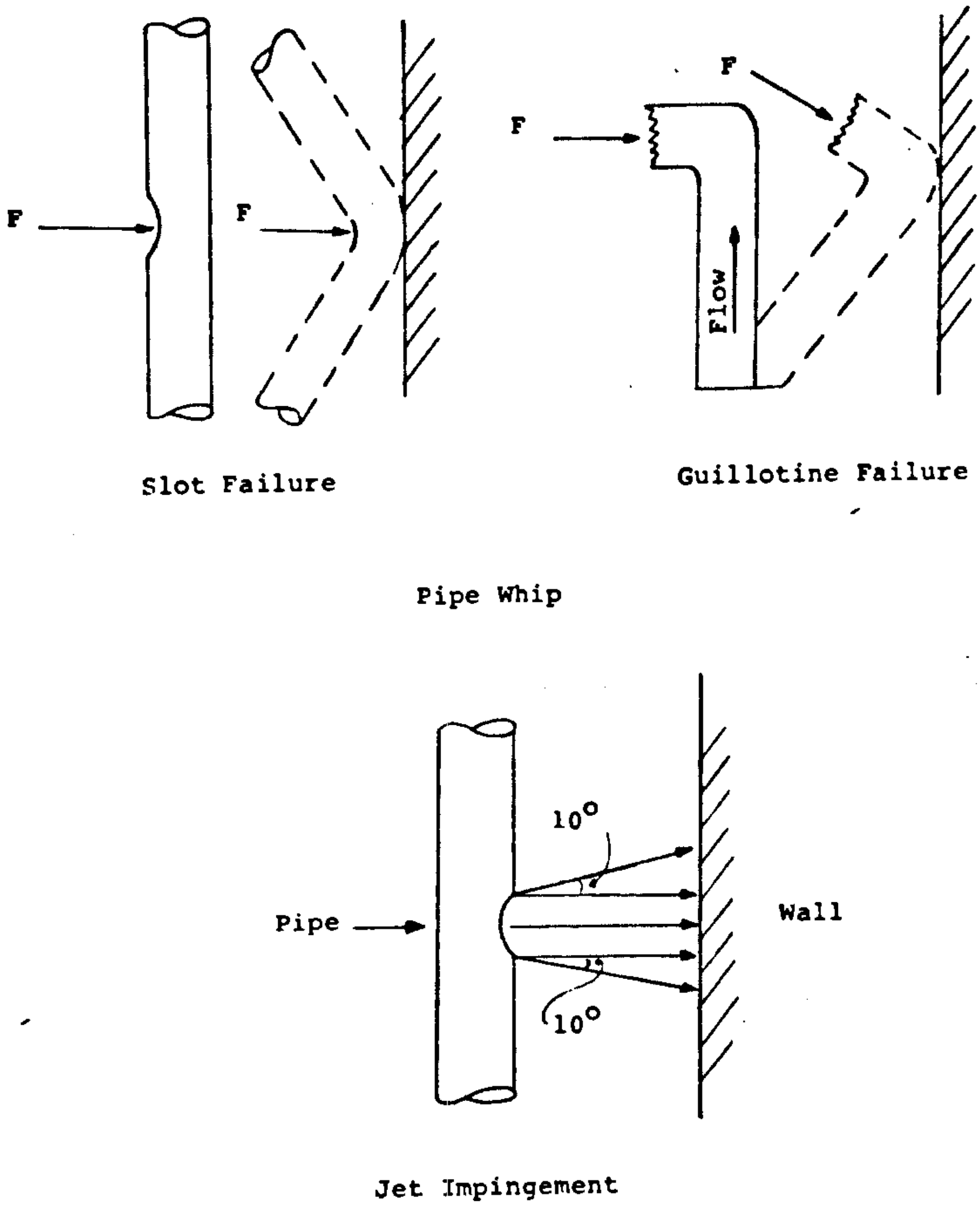
Reinforced Concrete	Max. Value of
Slabs & Beams (Tension steel only)	$\frac{10}{p} \leq 30$
Slabs & Beams (Tension and compr. steel)	$\frac{10}{p-p'} \leq 30$
Walls & Columns	
Compression	1.3
Composite T Beam	8

*p is the percent of tensile reinforcement.
p' is the percent of compressive reinforcement.

$*p = \frac{A}{bd}(100)$ and $p \leq 1.5\%$

Steel Components	
Steel Beams (lateral load)	26
(Note: To develop this ductility, the flanges must be thick enough to prevent local plastic buckling.)	
Steel Beams (lateral and axial loads)	8
Welded Portal Frames (vertical load)	6-16
Piping	26

Reference: Air Force Design Manual AFSWC-TDR-C2-138, Dec. 1962

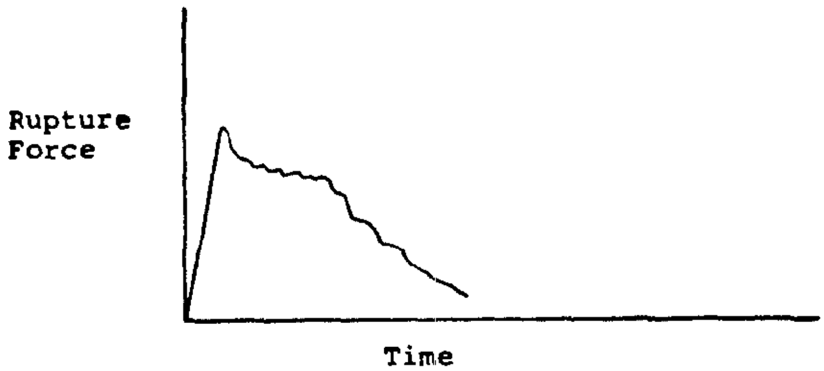


PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

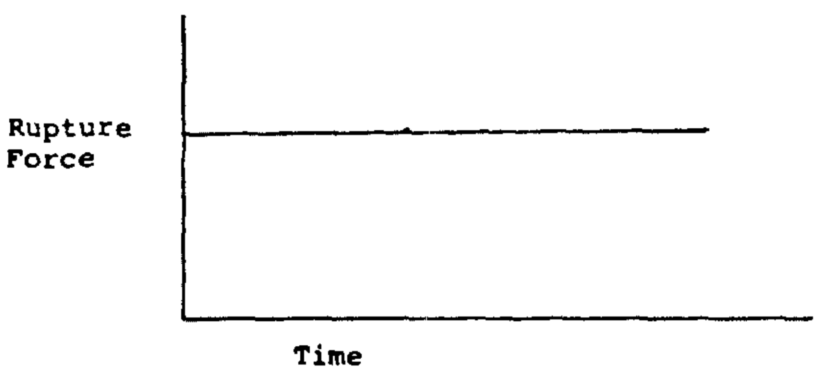
PIPE RUPTURE EXAMPLES

FIGURE 0.4-1

REVISION 5 — JUNE 1985



Actual Force Time History



Simplified Conservative Force Time History

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RUPTURE FORCE VS.
TIME RELATIONSHIP

FIGURE 0.4-2

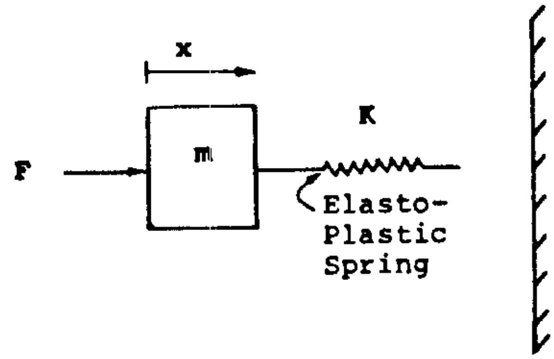
R = Resistance

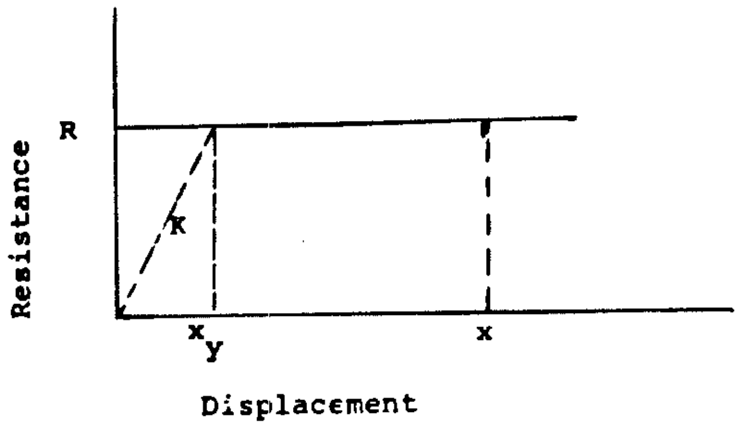
K = Stiffness

m = Mass

x = Displacement

F = Jet Force





R = Resistance Force

x_y = Yield Displacement

F = Jet Force

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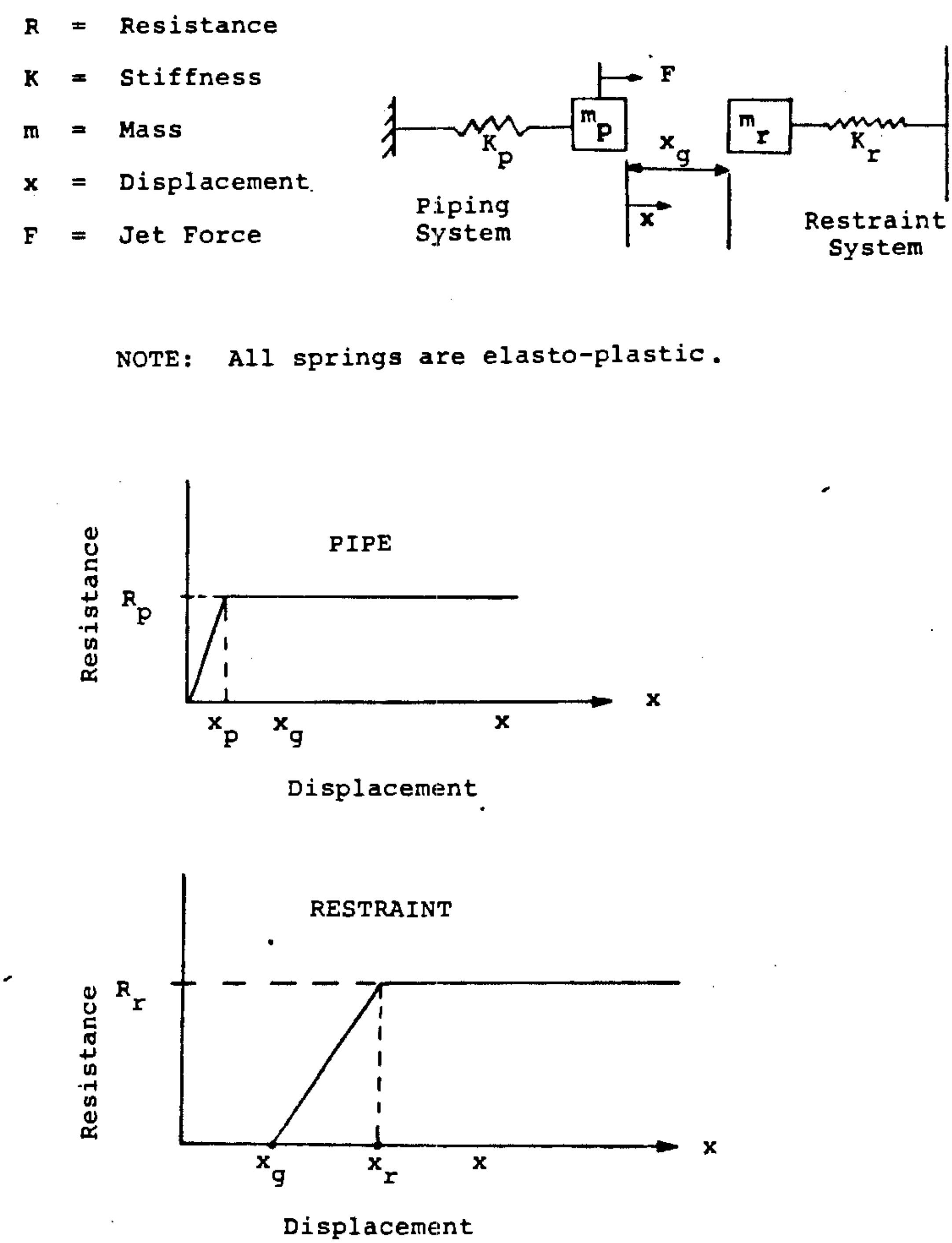
IMPULSE SYSTEM EXAMPLE

FIGURE 0.4-3



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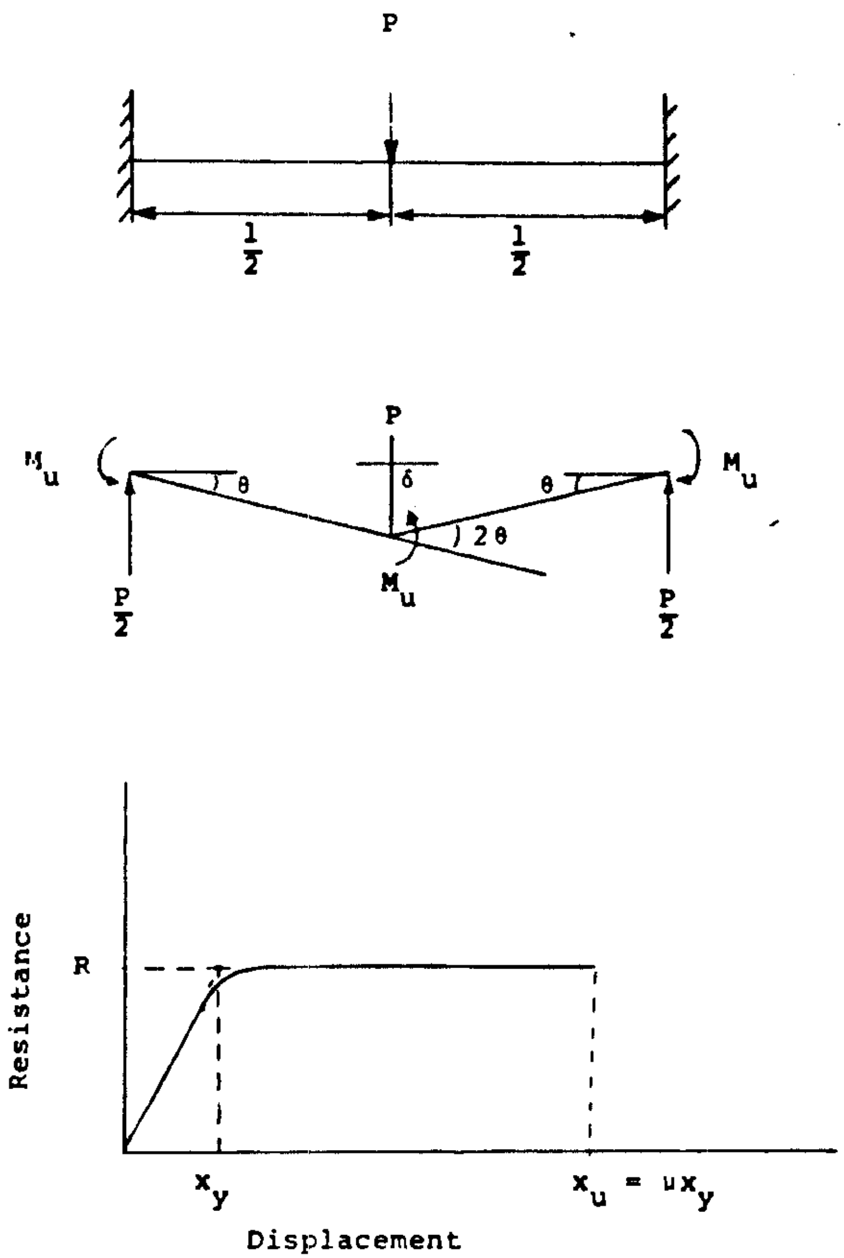
IMPACT + IMPULSE
SYSTEM EXAMPLE

FIGURE 0.4-4



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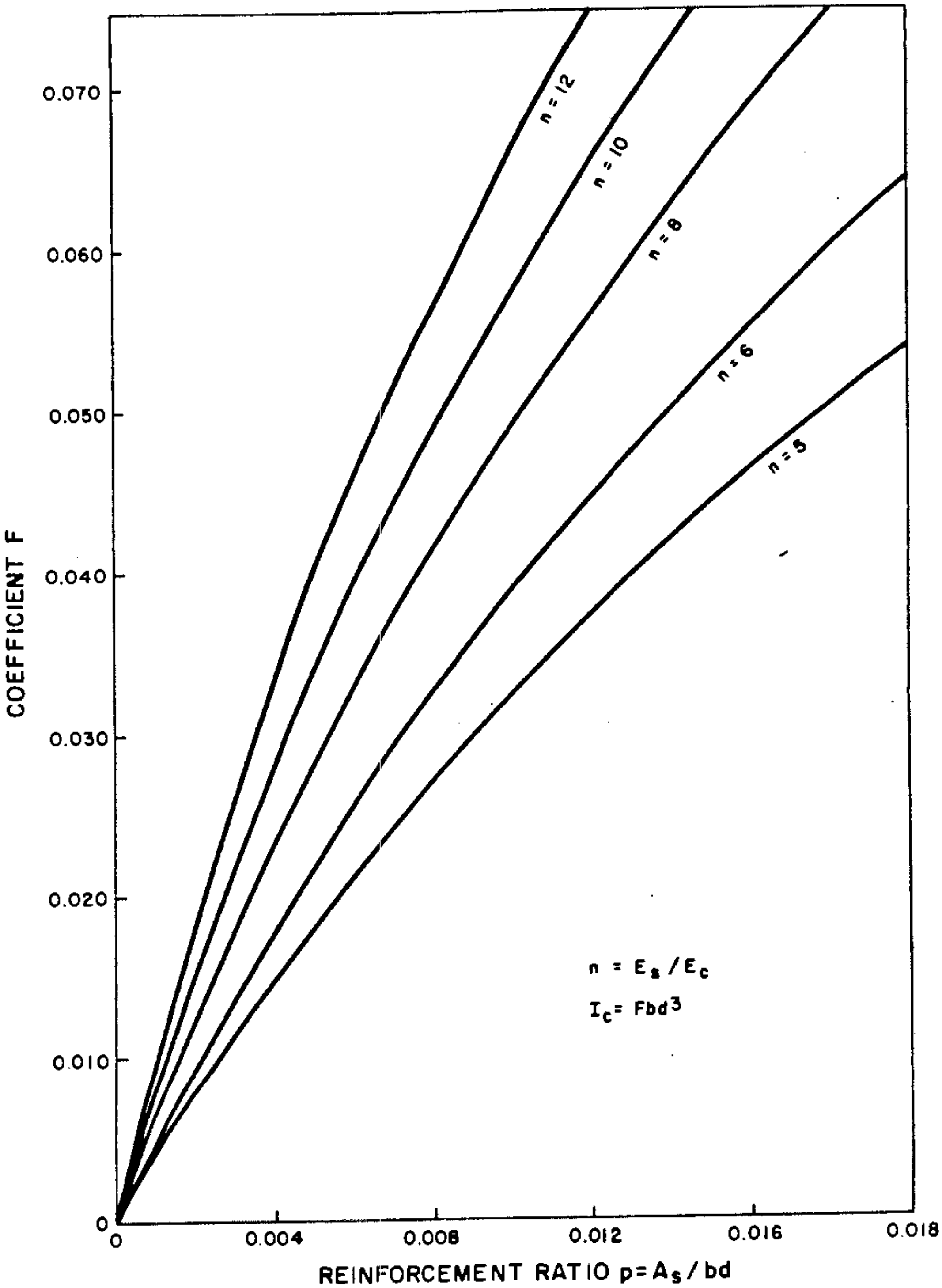
RESISTANCE-DISPLACEMENT EXAMPLE

FIGURE 0.4-5



REFERENCE: "STRUCTURES TO RESIST THE EFFECTS OF ACCIDENTAL EXPLOSIONS", TM5-1300, JUNE 1969.

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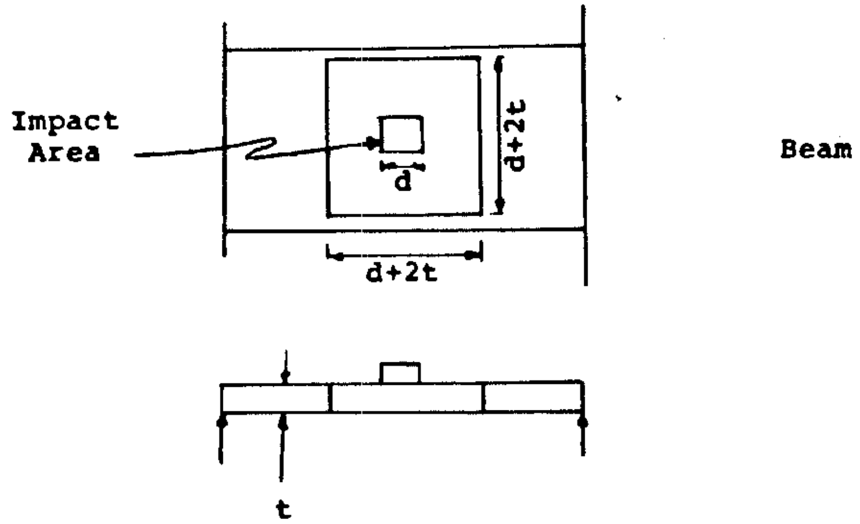
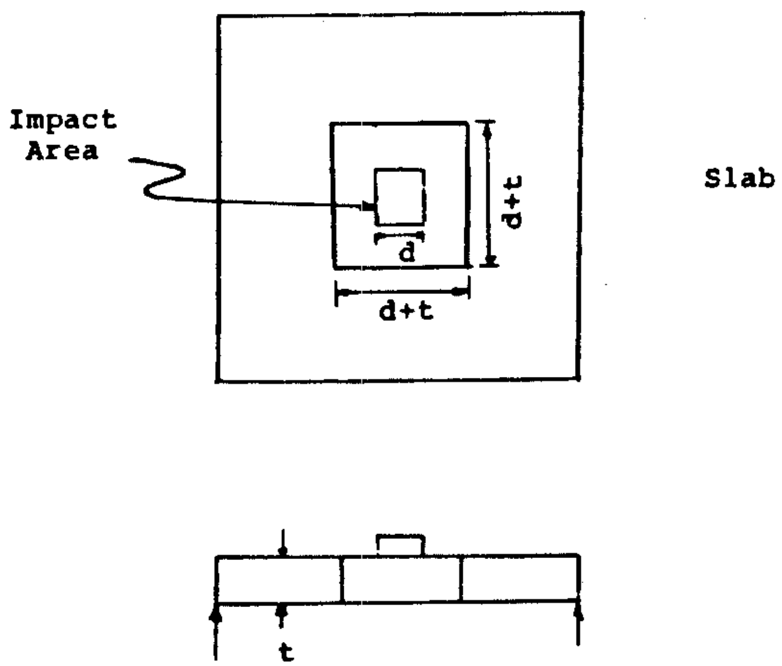
COEFFICIENTS FOR MOMENT OF INERTIA OF CRACKED SECTIONS WITH TENSION REINFORCEMENT ONLY

FIGURE 0.4-6





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VOLUME FOR PLASTIC IMPACT THRUST PARAMETER VS. FRICTION PARAMETER

FIGURE 0.4-7

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0.5 JET AND FLUID FORCES ANALYTICAL TECHNIQUES

0.5.1 Jet Thrust Forces

Jet thrust forces that accompany pipe break are both steady-state and time dependent in nature. The steady-state forces are due to the blowdown of fluid from some system stagnation pressure, P_o. The time-dependent forces are due to the propagation of pressure disturbances in the fluid immediately following pipe break. Both types of forces must be considered in calculating pipe break.

0.5.1.1 Steady-State Thrust Calculation

The generalized steady-state thrust equation as developed by Shapiro[1] is:

$$F = \dot{m}V_e + (P_e - P_a)\,A_e \qquad (5.1\text{-}1)$$

where:

m = fluid mass flow rate (lb/sec)

V_e = fluid exit velocity (ft/sec)

g_c = gravitational constant (lb ft/lb sec^2)

P_e = fluid exist pressure (psf)

P_a = ambient pressure (psf)

A_e = exit area (ft^2)

A convenient nondimensional thrust can be defined by dividing through by P_o and A_e obtaining:

$$F/P_oA_e = \frac{\dot{m}V_e}{P_oA_eg_c} + \frac{P_e - P_a}{P_o} \qquad (5.1\text{-}2)$$

One-dimensional continuity, $\dot{m} = \rho VA$ and the definition, $G = \frac{\dot{m}}{A}$ can be

used with Equation 5.1-2 to obtain the alternate expression:

$$F/P_oA_e = \frac{GV_e}{g_c\ P_o} + \frac{P_e - P_a}{P_o} \qquad (5.1\text{-}3)$$

and

$$F/P_oA_e = \frac{G^2}{\rho_e g_c P_o} + \frac{P_e - P_a}{P_o} \tag{5.1-4}$$

where:

ρ_e exit mass density (lb_m/ft^3).

Four blowdown situations are considered for rupture of steam and water lines. They are:

1. Blowdown of steam from superheated conditions
2. Blowdown of a steam-water mixture
3. Blowdown of cold water
4. Blowdown of water with flashing from subcooled water conditions

0.5.1.2 Superheated Steam

Superheated steam is usually treated as an ideal gas with the gas constant, R, equal to 85.75 $ft\text{-}lb_f/^oFlb_m$ and the ratio of specific heats, γ, equal to 1.3.[2]

If the flow is further considered isentropic, the thrust parameter becomes[2]:

$$F/P_oA_e = 1.26 - \frac{P_a}{P_o} \tag{5.1-5}$$

Friction effects can be considered by assuming the flow process follows the Fanno line as described by Shapiro[1]. The Fanno analysis predicts the thrust parameter will be a function of the pipe friction parameter fL/D, as shown on Figure 0.5-1. For the case of fL/D = 0 the Fanno analysis reduces to the inviscid flow case (Equation 5.1-5). Flow restrictions will tend to decrease flow rates and can be included using Figure 0.5-2.

0.5.1.3 Steam-Water Mixtures

An equilibrium, a two-phase flow model has been developed by Moody[3] which can be used to predict blowdown of mixtures of steam and water. Moody provides plots of G_{max} as a function of stagnation conditions for friction parameters between 0 and 100. He also provides a plot which can be used to determine exit conditions. For frictionless flows (fL/D = 0),

Moody's model gives approximately the same results as Equation 5.1-5.

Fauske[4] has proposed a second model which includes nonequilibrium effects. He compares his model with equilibrium models and concludes that for low steam qualities ($x < 2\%$) and short pipes, equilibrium models may not be conservative. For these conditions Fauske's analysis should be used to predict jet thrust forces. As an alternative, cold water relationships, discussed below, can be used to obtain a quick conservative estimate of thrust for short pipes and low quantities. Again, Figure 0.5-2 can be used to include flow restriction effects.

0.5.1.4 Cold Water Flow

Blowdown of cold water can be treated as flow of an incompressible fluid ($\rho e = \rho o$). For inviscid flow, the exit pressure becomes the ambient pressure, P_a, and the exit velocity becomes:

$$V_e = \sqrt{\frac{2\,(P_o - P_a)\,g_c}{\rho o}} \qquad (5.1\text{-}6)$$

The trust parameter is then[2]:

$$F/P_oA_e = \frac{2-2P_a}{P_o} \qquad (5.1\text{-}7)$$

Friction effects can be included using the approximate expression:

$$F/P_oA_e = \frac{2}{(f\,\frac{L}{D}) + 1} \qquad (5.1\text{-}8)$$

0.5.1.5 Subcooled Water Flow

Subcooled water blowdown is characterized by flashing of the fluid near the pipe exit. This flashing tends to cause lower thrust levels than those predicted using nonflashing incompressible flow theory. The nonequilibrium model developed by Fauske[4] is applicable in the subcooled region and can be used to predict subcooled water blowdown. The cold water thrust Equation 5.1-7 can be used to obtain a quick conservative estimate of subcooled thrust.

0.5.1.6 Unsteady Flow Thrust

Definition of the unsteady thrust requires an examination of the interaction between the propagation of pressure disturbances and the initiation of blowdown. An approximation can be made by assuming the thrust to be defined by:

$$F/P_oA_b = 1.0 \qquad (5.1\text{-}9)$$

over the period, $0 \leq t \leq t_1$

where:

$$t_1 = \frac{2L}{C} \qquad (5.1\text{-}10)$$

and

L = length of pipe from break to pressure vessel

or

= length of pipe from break to flow restriction

C = sonic velocity

= 1600 ft/sec for steam

= 4000 ft/sec for water

For $t > t_1$, thrust is equal to the steady-state value.

0.5.2 Fluid Jet Impingement Forces

In the event of a pipe break, the fluid flowing through the pipe emerges out as a jet impinging at nearby structures or equipment. Various blowdown situations considered here are described in Section 0.5.1.

When the fluid emerges from the break point, the jet undergoes free rapid expansion to the ambient pressure at relatively short distance, i.e., a few diameters of break area. For this asymptotic distance, momentum and shear interactions with jet environment can reasonably be neglected. As such, applying forward momentum conservation, the total jet force, F_j, is constant throughout its travel; and, therefore, as assumed by Moody[2]:

$$F_j = F \qquad (5.1\text{-}11)$$

where:

F = the total thrust force defined in Section 0.5.1.

For the purpose of this report it is further assumed that F_j remains constant for all distances beyond the asymptotic area. This assumption is conservative. Therefore, the jet pressure at any location along the axis of the jet is given by:

$$P_j(x) = F_j/A_j(x) \qquad (5.1\text{-}12)$$

where:

A_j (x) = the expanded jet area at location x along the jet axis.

See Figure 0.5-3 for system geometry.

Moody[2] has developed a simple analytical model for estimating the asymptotic jet area for steam, saturated water and steam/water blowdown situations. Experimental results from LOFT[5] tend to indicate that for subcooled water and steam blowdown situations, the jet area expands uniformly at a half angle of about 15 degrees, whereas, steam/water blowdown expands much more rapidly because of large scale water flashing. Results of Moody's analytical analysis agree, at least qualitatively, with LOFT results. In addition, Moody's analytical analysis predicts results of other experiments.[2]

In this report, an empirical approach has been adopted combining Moody's analytical model with the uniform half angle approach, as shown on Figure 0.5-3. The half angle is conservatively assumed to be ϕ = 10 degrees.

According to this empirical model, the distance of jet travel is divided into three regions. Region 1 extends to the asymptotic area, at which point jet expansion area is calculated according to Moody's method; in

region 2, jet area remains constant; and in region 3, the jet expands at half angle ϕ = 10 degrees.

For subcooled water blowdown, this model assumes the half angle approach, ϕ = 10 degrees, uniformly in all three regions, since Moody's model is not applicable for this case.

To follow Moody, extent of region 1 is taken as:

$$x_1 = 5\ D_e \qquad (5.1\text{-}13)$$

and the jet area at location x is given by the equation:

$$A_j(x_1) = \pi R_j{}^2 1 = (A_e G)^2\ v1/G_c F_j \qquad (5.1\text{-}14)$$

where:

D_e = Equivalent diameter of pipe break area. See Figure 0.5-4.

A_e = Pipe break area

R_j1 = Radius of the expanded jet at location x. R_j1 is constant in region 2.

F_j = F, thrust force (Equation 5.1-11)

v = Specific volume. v_1 is calculated as described in Moody.[2]

Other terms in Equation 5.1-14 are described in Section 0.5.1. Region 2 extends to the location x_2 given by:

$$A_j\ (x_1) = A_j(x),\ x = x_2 \qquad (5.1\text{-}15)$$

where A_j (x) is the jet area in region 3 and is calculated by any one of the following equations:

1. <u>Guillotine break</u>

$$A_j\ (x) = A_e\ (1 - \frac{2x}{D_e} \tan \phi)^2 \qquad (5.1\text{-}16)$$

where:

ϕ = 10 degrees is the half angle of jet expansion

2. <u>Longitudinal (slot) break</u>

$$A_j\ (x) = A_e\ (1 + \frac{2x}{l} \tan \phi)(1 + \frac{2}{w} \tan \phi) \qquad (5.1\text{-}17)$$

where l and w are slot dimensions.

3. <u>Circumferential crack</u>

$$A_j\ (x) = A_e\ (1 + \frac{2x}{w} \tan \phi)(1 + \frac{x}{\ell}\ (1 + 2 \tan \phi)) \qquad (5.1\text{-}18)$$

In region 1, an additional conservative assumption is made that the jet area increases uniformly from A_e at x = o, to $A_j\ (x_1)$ at $x=x_1$,

or:

$$A_j\ (x) = A_e \left[1 + \frac{x}{x_1} \left[\frac{R_j 1}{R_e} - 1\right]\right]^2, \text{ for } o \leq x \leq x_1 \qquad (5.1\text{-}19)$$

where:

$R_e = D_e/2 = \sqrt{A_e/\pi}$ and $R_j 1$ is given by Equation (5.1-14)

Impingement Loads on Targets

Once the jet area A_j is calculated by the method described above, the jet pressure is readily calculated according to Equation 5.1-12, i.e.,

$$P_j = F_j A_j \qquad (5.1\text{-}20)$$

and the jet impingement load on the target is given by:

$$FT = P_j A_{te} \qquad (5.1\text{-}21)$$

where:

A_{te} = the effective target area.

Calculation of A_{te} for various geometries is outlined below:

1. Flat surface

   If the target with physical area A_t cancels all the fluid momentum in the jet, then:

   $A_{te} = A_t$

   For the case where target is oriented at angle θ with respect to the jet axis and there is no flow reversal,

   $A_{te} = A_t \sin\theta$

2. Pipe surface

   Let:

   Dp = Diameter of pipe, and

   Dj = Diameter of jet impinging on pipe

   $= \sqrt{\frac{4/Aj}{\pi}}$

Then,

For $Dp > Dj$

$$A_{te} = CA_t$$

where C is pipe curvature factor and $C = 2/\pi$

for $D_e < D_j$

$$A_{te} = CA_t$$

where $A_t = DpDj$ (conservative approximation)

3. Deflecting surface

Effective target surface area, A_{te} , for the targets which deflect the jet rather than totally cancel the fluid momentum in the jet, is calculated as described by Moody.(2)

0.5.3 References

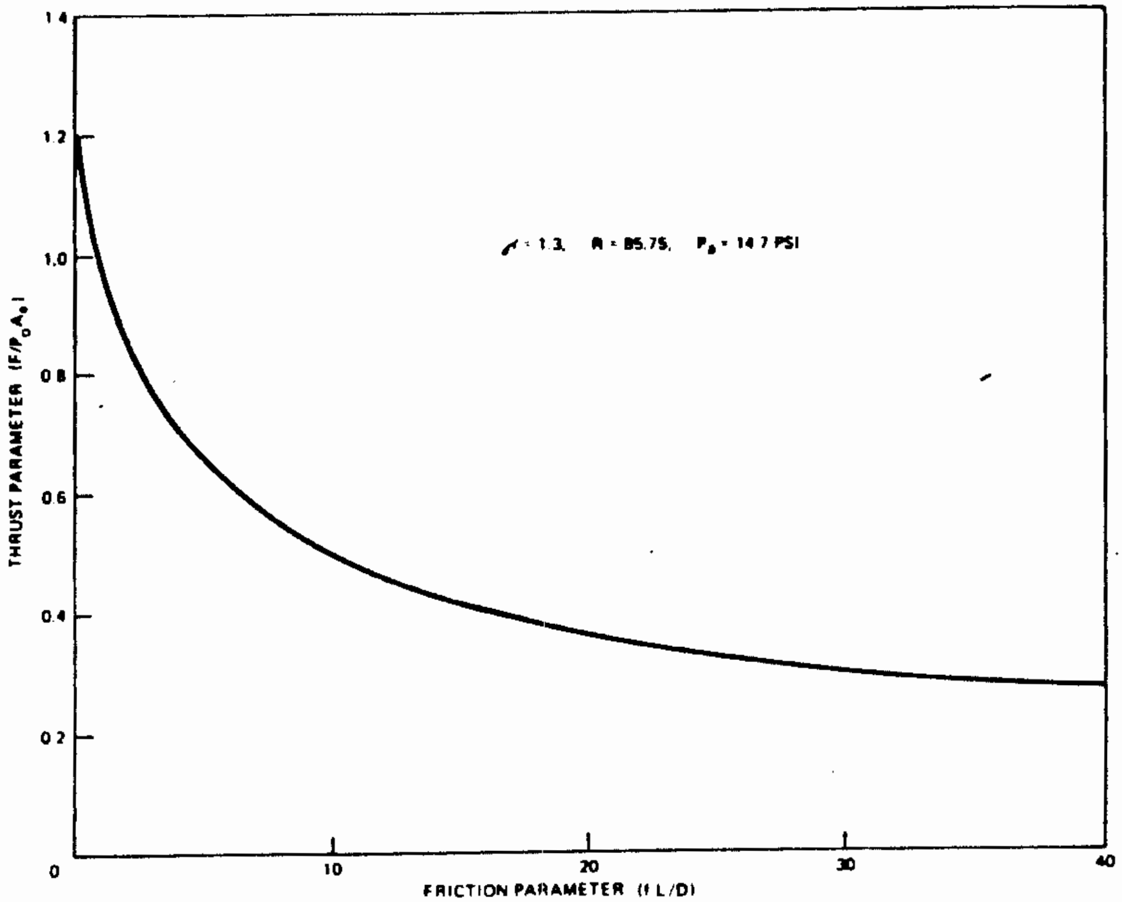
1. Shapiro, A. H., The Dynamics and Thermodynamics of Compressible Fluid Flow, Vol. 1, Ronald Press, New York, N.Y., 1953.

2. Moody, F. J., Prediction of Blowdown Forces and Jet Thrusts, ASME Publication, June 1969.

3. Moody, F. J., Maximum Two Phase Vessel Blowdown from Pipes, GE Report, APED - 4827, April 1965.

4. Fauske, Hans K. and Henry, Robert. The Two Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices, and Short Tubes, ASME Transactions, Journal of Heat Transfer, May 1971.

5. Semiscale Blowdown Test Program, IDO 17242, 4th Quarter 1967.



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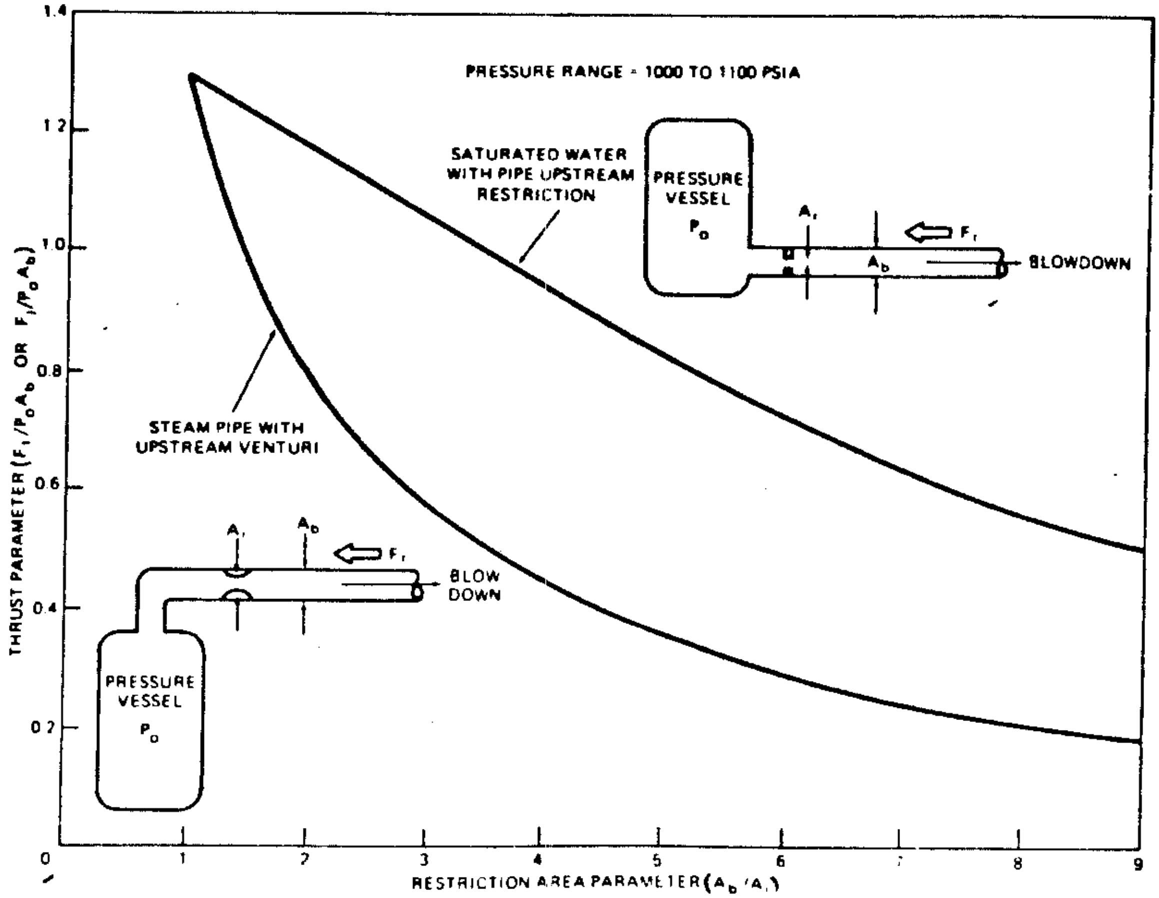
THRUST PARAMETER VS.
FRICTION PARAMETER FANNO FLOW
ROUTINE IDEAL GAS FLOW

FIGURE 0.5-1



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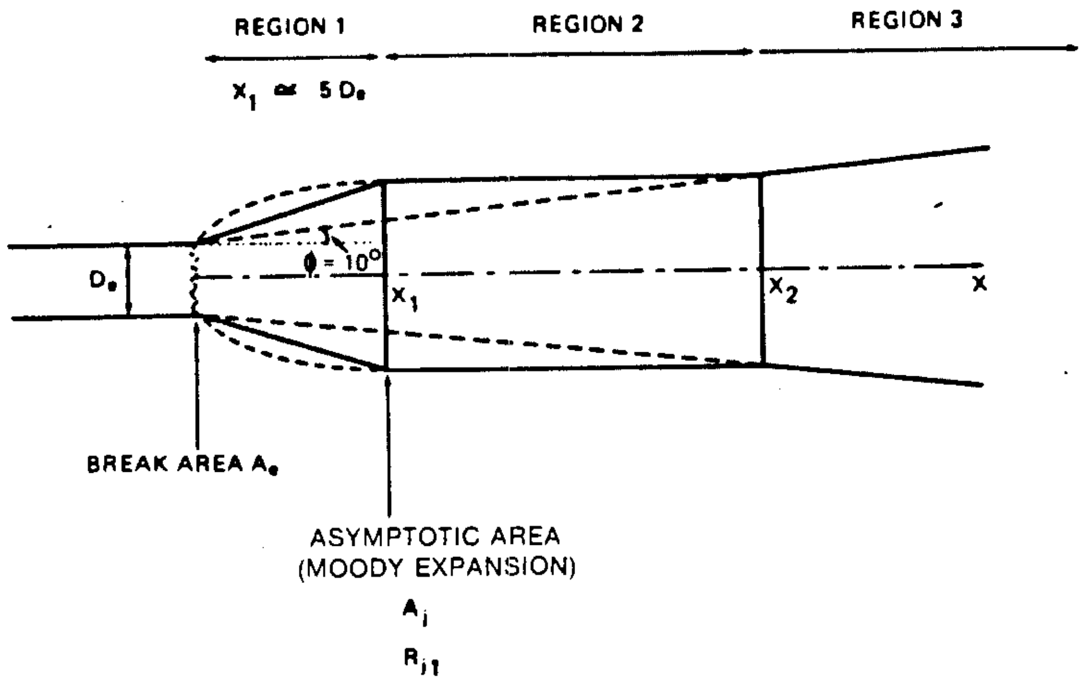
EFFECT OF AREA RESTRICTION ON
STEAM-WATER BLOWDOWN STEADY THRUST

FIGURE 0.5-2



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FLUID JET GEOMETRY

FIGURE 0.5-3

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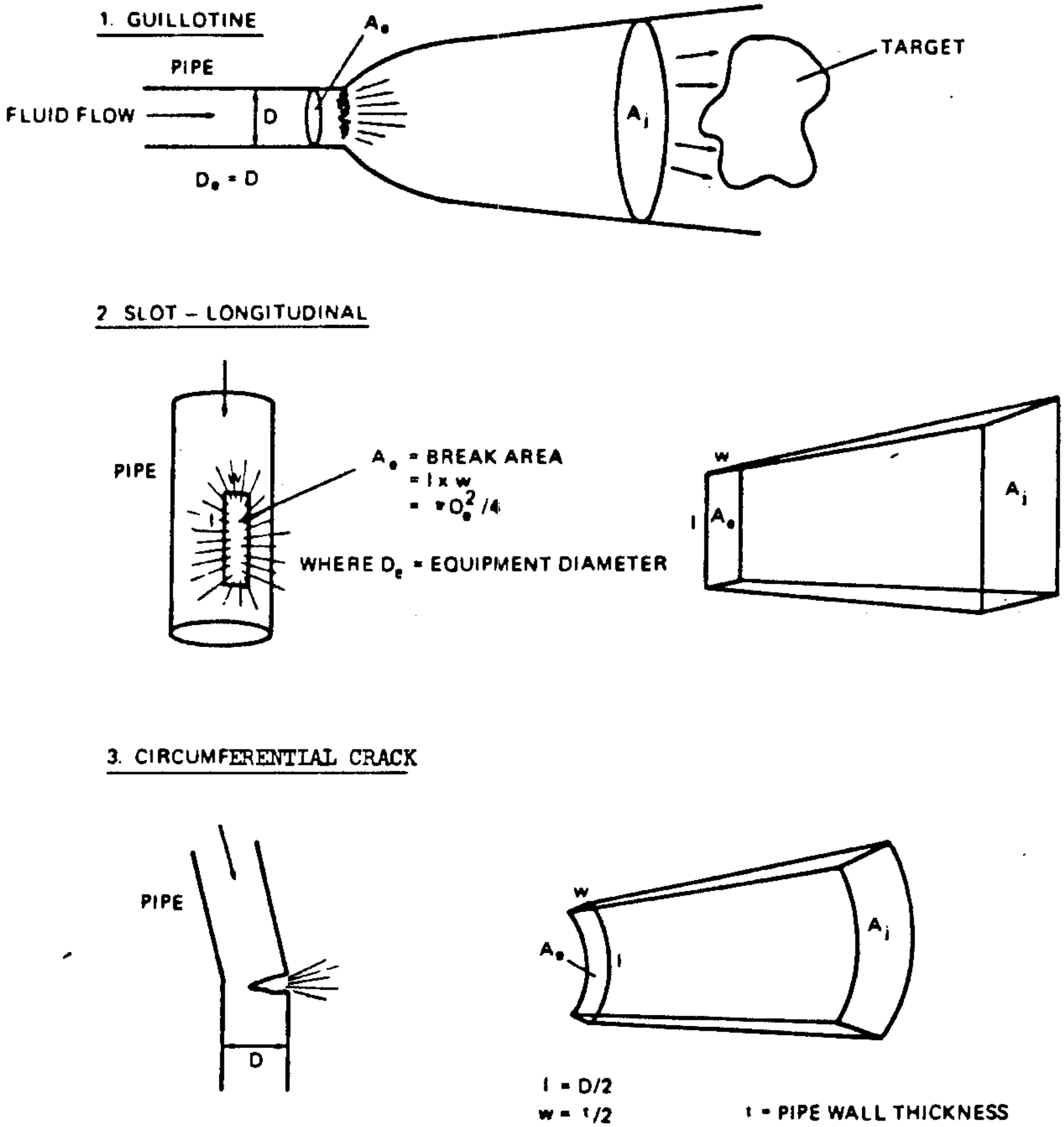

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PIPE BREAK GEOMETRIES

FIGURE 0.5-4

0.6 DETAILED SYSTEM ANALYSES

Postulated piping failures in each high energy system are discussed in the following sections.

0.6.1 Main Steam and Feedwater Systems

0.6.1.1 Condenser Compartment

The main steam and feedwater high energy line potential breakpoints are listed in Table 0.6-1. The physical location of these breakpoints is shown with the installed restraint locations on Figures 0.6-1 through 0.6-17 and 0.6-27.

Since the failure of a feedwater line is less severe than the failure of a main steam line, a break in the main steam line within the condenser compartment will result in the greatest temperature and pressure impact on this compartment. Backflow from the reactor vessel to the break is prevented by closure of the feedwater check valves coincident with flow reversal. Thus, flow through the break would be from the feed pumps only. It is assumed that water from all three feed pumps would discharge through the break. Should the postulated break occur, the discharge pressure of the pumps would decrease and the flow would increase until pump runout occurs. The motor driven feed pumps will continue to run until the condenser hotwells are empty or operator action trips the pumps. This assumption implies that offsite ac power is not lost for this accident. If offsite ac power is lost, the pumps would lose power almost immediately and only a small quantity of water would be discharged through the break.

The maximum calculated overpressure and temperature due to the failure of a main steam line in the condenser compartment is shown on Table 0.6-2. To achieve those conditions, a total of 506 ft^2 of vent area is required. As noted in Table 0.6-3, Section D, the required vent area was provided by the removal of the concrete shield plugs in the turbine deck equipment hatch at el 51 ft (between column lines 4 and 5 and E and D), and by modifications to the main turbine combined intermediate valve (CIV) enclosures to ensure these enclosures will fail at small overpressures and relieve to the turbine deck.

Using the criteria in Section 0.2, item 2, 114 potential piping failure locations in the condenser compartment could be postulated. Detailed analysis of each of these postulated failures to assure that no unacceptable interactions with safety-related electrical cable exist would be a prohibitively time consuming task. A more direct means of providing assurance that the effects of postulated failures are acceptable is to relocate certain potentially vulnerable safety-related electrical cable to areas remote from high energy lines. This approach has been adopted. Table 0.6-3, Section A, includes a listing of electrical cables located in the condenser compartment which have been relocated. The cables requiring relocation were identified by a detailed protection sequence analysis.

0.6.1.2 Main Steam Tunnel

Main steam and feedwater piping failures are postulated to occur at seven locations in the main steam tunnel. See Table 0.6-1 and Figure 0.6-18.

A main steam line failure in the main steam tunnel results in a maximum overpressure and temperature as shown in Table 0.6-2. No failure mode of the main steam tunnel floor has been identified that can result in damage to safety-related equipment. However, jet loadings on the main steam tunnel floor resulting from main steam line or feedwater line slot failures could result in loadings which exceed the structural capacity of the floor. Additional beams and bracing have been installed to increase the main steam tunnel floor capability as noted in Table 0.6-3, Section D.

A main steam or feedwater line rupture in the main steam tunnel will result in steam and/or water flooding into the adjacent access room (residual heat removal(RHR) valve station) at el 23 ft, and eventually through the blowout panel (which relieves at 0.5 psid) into the condenser compartment. The steam or water flooding in these adjacent areas will not affect safe reactor shutdown, allowing for a single active component failure. The equipment in these areas is designed to withstand the pressures and temperatures resulting from the postulated break in the main steam tunnel. The RHR shutdown cooling isolation valve and motor operator located in the RHR Valve Station is protected from submergence following a feedwater line rupture by a seismic Class I flood barrier.

The overpressure in the Turbine Building condenser compartment due to this break is less than that which results from the case of a main steam line failure in the condenser compartment as discussed in Section 0.6.1.1.

A feedwater line failure in the main steam tunnel could result in loss of function of the high pressure coolant injection (HPCI) system, due to loss of integrity of the HPCI path. However, sufficient redundancy remains to achieve and maintain safe reactor shutdown allowing for a single active component failure.

Based on the above discussion, it is concluded that no high energy line break in this area can preclude safe shutdown of the reactor.

0.6.2 Reactor Core Isolation Cooling (RCIC) System

0.6.2.1 RCIC Piping Failure in Valve and TIP Compartments

A RCIC steam line break in the RCIC valve compartment, at el 23 ft in the Reactor Building, would result in discharge of steam until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the break. Additionally, high temperatures resulting from a RCIC steam line failure will also automatically isolate the RCIC system and initiate an alarm in the main control room. To assure the operability of the RCIC outboard steam isolation valve, MO-1301-17, pipe whip restraints have been provided as noted in Table 0.6-3, Section D.

Postulated RCIC steam piping failure locations are noted in Table 0.6-1 and shown on Figures 0.6-18 and 0.6-26. The maximum overpressure and temperature calculated to occur in this room are shown on Table 0.6-2. The ultimate structural capability limit of this compartment is governed by the reinforced block wall. Since the structural capability is greater than the calculated overpressure, no structural changes are required.

A RCIC steam line break is assumed to cause the loss of the electrical cables which result in the loss of operation of RHR containment spray valves MO-1001-23B and 26B; main steam temperature switches 261-15, A, B, C and D; and HPCI temperature switches TS-2370D and 2372D. However, redundant means of shutdown remain available.

0.6.2.2 RCIC Piping Failure in the Suppression Chamber Compartment

A RCIC steam line break in the suppression chamber compartment would result in discharge of steam external to the suppression chamber until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the break. Additionally, high temperatures resulting from a RCIC steam line failure in the suppression chamber compartment will also automatically isolate the RCIC System and initiate an alarm in the main control room. Postulated RCIC steam piping failure locations are defined in Table 0.6-1 and shown on Figures 0.6-19 and 0.6-25.

Compartment pressurization analysis indicates that no additional vent area is required in the suppression chamber compartment in order to prevent exceeding the external pressurization limits of the suppression chamber for the postulated RCIC steam line failure. Steam venting to the Reactor Building corner rooms would occur, but equipment in these adjacent areas would be unaffected by the slight pressurization and steam flow.

Failure and whip of the RCIC steam line in the suppression chamber compartment could result in failure of suppression pool water temperature indicator TE-502204. This indicator is a resistance temperature element and its failure (i.e., fail open) would result in a reading of 50°F (the lowest point on the scale) on the TE-502204 readout device in the main control room. If, concurrent with the steam line failure, suppression pool water temperature element TE-502102 failed, the operators would be without direct suppression pool water temperature indication. However, for the case of the RCIC steam line failure under consideration, sufficient time is available (approximately 4 hours) for the operators to observe the loss of direct suppression pool temperature information in the main control room and to establish operation of the RHR system in the torus cooling mode.

Once this system is placed into operation, torus water temperature information is available in the main control room from the RHR heat exchangers inlet water temperature recorder. Thus, although there is a potential loss (considering a single failure) of direct suppression pool water temperature indication in the event of the postulated RCIC steam line break in the suppression chamber compartment, this potential loss is judged acceptable since operator information from other information channels and sufficient time are available for appropriate operator response. RCIC steam line whip around the potential breakpoints identified in the suppression chamber compartment (see Figure 0.6-25) cannot result in sufficient impact force on the suppression chamber wall to cause loss of suppression chamber integrity.

0.6.2.3 RCIC Piping Failure in the RCIC Pump Compartment

A RCIC steam line break in the RCIC pump compartment in the southwest Reactor Building corner would result in a discharge of steam until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the failure. Additionally, high temperatures resulting from a RCIC steam line failure in the RCIC pump compartment will also automatically isolate the RCIC system and initiate an alarm in the main control room. Postulated break locations are shown on Figure 0.6-19 (Part Plan D), and Figure 0.6-25 and Table 0.6-1.

The maximum overpressure and temperature calculated to occur in this compartment are shown on Table 0.6-2. The structural capability in this area is adequate to withstand the resulting overpressurization from a RCIC steam line break.

O.6.3 High Pressure Coolant Injection (HPCI) System

O.6.3.1 HPCI Valve Compartment

A HPCI steam line failure in the HPCI valve compartment at el 23 ft in the Reactor Building would result in the discharge of steam until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the break. Additionally, high temperatures resulting from a HPCI steam line failure in the HPCI valve compartment will also automatically isolate the HPCI system and initiate an alarm in the main control room. To assure the operability of the HPCI outboard steam isolation valve, MO-2301-5, pipe whip restraints have been provided as noted in Table O.6-3, Section D. The postulated high energy line piping failure location is shown on Table O.6-1 and Figure O.6-18.

Maximum compartment pressurization and temperature analysis results are shown on Table O.6-2. The ultimate structural capacities of the reinforced concrete walls and floor slab in this area exceed the calculated maximum overpressure. A portion of the southwest concrete block wall of this area has been reinforced to withstand this maximum overpressure, as noted in Table O.6-3, Section D. Adjacent Reactor Building areas are subject to slight overpressure which will have no effect on Reactor Building structures.

A HPCI steam line piping failure in the HPCI valve compartment is assumed to result in the failure of electrical cables which power RHR injection valve MO-1001-29B. Sufficient redundancy remains to achieve and maintain safe shutdown.

O.6.3.2 HPCI Piping Failure in the Suppression Chamber Compartment

A HPCI steam line failure in the suppression chamber compartment would result in the discharge of steam until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the break. Additionally, high temperatures resulting from a HPCI steam line failure in the suppression chamber compartment will also automatically isolate the HPCI system, and initiate an alarm in the main control room. Pipe whip restraints have been provided as necessary (see Table O.6-3, Section D) in the suppression chamber compartment in order to assure the operability of the HPCI outboard steam isolation valve, MO-2301-5, and to provide added assurance that HPCI steam line whip around the potential breakpoints identified in the suppression chamber compartment cannot result in loss of suppression chamber integrity. See Figure O.6-20. The restraints further serve to protect the RHR to salt service water inter-tie line, shown on Figure 4.8-3, and the 10 inch GB-10 RHR containment spray line upstream of MO-1001-23A.

A HPCI steam line failure in the suppression chamber compartment could potentially damage control circuitry to ADS valve SV-203-3D. In order to preserve existing depressurization capability safety margins, this valve control circuitry in the suppression chamber compartment has been protected, as noted in Table O.6-3, Section D. Postulated high energy line piping failure locations in the suppression chamber compartment are shown on Table O.6-1 and Figure O.6-20.

A compartment pressurization analysis for the postulated HPCI steam line failure indicates that no additional vent area is required in the suppression chamber compartment, in order to prevent exceeding the external pressurization limits of the suppression chamber. This structural response analysis followed the methodology given in paragraph UG-28 and Figure UCS-28.2 of the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1. Steam venting to the Reactor Building corner rooms would occur, but equipment in these adjacent areas would be unaffected by the slight pressurization and steam flow.

O.6.3.3 HPCI Piping Failures in the HPCI Pump Compartment

A HPCI steam line failure in the HPCI pump compartment would result in discharge of steam until automatic isolation is achieved. Automatic isolation and alarm is initiated by high steam flow resulting from the failure. Additionally, high temperatures resulting from a HPCI steam line failure will also automatically isolate the HPCI System and initiate an alarm in the main control room. The postulated breakpoint is shown on Figure O.6-21 and listed on Table O.6-1.

As a result of the postulated HPCI steam line piping failure, the compartment could reach a pressure in excess of the compartment structural capability. To keep the pressure below the limiting ceiling pressure, a vent path has been provided through the existing equipment hatch and blowout panel at el 3 ft and the existing equipment hatch at the 23 ft elevation in the reactor auxiliary bay (see Table O.6-3, Section D). A vent path from the 23 ft area is provided through the existing truck access door (rollup) which will reach yield stress failure and allow significant venting at approximately 0.5 psid. A concrete hatch plug at el 3 ft was removed and a partition erected around the hatch opening at el 3 ft to separate the vent path from the remainder of the room at el 3 ft. A ventilation barrier (blowout panel) has been provided at el 3 ft designed to rupture at approximately 0.25 psid. This barrier separates secondary containment (the HPCI pump compartment) from the other areas. The removable concrete block wall between column line L and M at el 3 ft has been reinforced to withstand the resulting pressure, as noted in Table O.6-3, Section D. With the additional vent area, the ultimate capacity of the concrete walls in this compartment is in excess of the loads induced by compartment pressurization or jet force. The maximum compartment pressure and temperatures resulting after completion of the HPCI pump compartment modification are shown on Table O.6-2.

Safety-related equipment that could be affected by pipe whip and/or jet impingement from a HPCI steam line break are HPCI System related only. Their loss will not affect the capability to achieve and maintain safe shutdown. The indicated loss of the Reactor Building closed cooling water loop "B" (RBCCW B) results from the potential damaging effect of a HPCI steam line pipe whip on the RBCCW piping in the HPCI pump compartment. In this compartment RBCCW loop "B" piping is providing cooling water to the equipment area cooling

system unit coolers. This failure could potentially drain RBCCW loop "B" and result in loss of function (until the break is manually isolated) of this cooling loop. While not required for immediate response to the postulated HPCI piping failure, RBCCW manual isolation valves presently

located adjacent to the equipment area cooling units located within the HPCI pump compartment have been backed up by new manual isolation valves (see Table 0.6-3, Section D) installed in areas remote from the postulated HPCI line failure in the HPCI pump compartment in order to provide for early restoration of the RBCCW system. However, sufficient redundancy remains to provide immediate initial core cooling. Either the LPCI or Core Spray Systems utilized in conjunction with the ADS have the capability to restore and maintain initial core cooling. Extended core cooling is available in sufficient redundancy, taking credit for operator action as discussed in Section 0.2, item 3. It is, therefore, concluded that no further modifications in the HPCI pump compartment, other than those associated with establishing environmental qualification, are required.

0.6.4 Reactor Water Cleanup (RWCU) System

0.6.4.1 RWCU Pump and Heat Exchanger Compartments

A postulated Reactor Water Cleanup (RWCU) high energy line failure in the cleanup system pump or heat exchanger compartments at el 51 ft in the Reactor Building results in a single ended piping failure until isolation is achieved. A check valve in the return line immediately upstream of the connection into the feedwater piping will immediately close to prevent excessive backflow from the return side of the break. Automatic isolation and alarm of the cleanup system is initiated by high flow in the supply header. Additionally, high temperatures resulting from an RWCU system piping failure in the cleanup system will automatically isolate the cleanup system and initiate an alarm in the main control room. To assure the operability of the RWCU supply line isolation valve (MO-1201-5) and the RWCU return line check valve (1201-81), pipe whip restraints have been provided as noted in Table 0.6.3, Section D.

Postulated piping failure locations in this area are shown on Figures 0.6-22, 0.6-23, 0.6-24, and 0.6-26, and are listed on Table 0.6-1. A postulated high energy piping failure would result in a maximum pressure and temperature as shown on Table 0.6-2.

As a result of a reroute of the RWCU supply piping from penetration X-14 to the regenerative heat exchangers, postulated pipe break locations were modified as shown on revised Figures 0.6-22 and 24.

The ultimate capacity of the reinforced concrete block walls in these compartments are in excess of the loads induced by pressurization or jet forces. To prevent removable concrete blocks in compartment wall openings from being ejected due to pressurization or jet forces, additional structural reinforcement has been provided as noted in Table 0.6-3, Section D. Adjacent compartments would not be affected by the slight pressurization resulting from the postulated RWCU high energy piping failure.

Electrical cables to motor operated system RHR "A" drywell spray injection valves MO-1001-23A and MO-1001-26A (normally closed) could potentially be damaged by pipe whip resulting from the postulated failure of a high energy RWCU system line. This loss is acceptable since RHR system operation in the drywell spray mode is not required in the event of the postulated RWCU system piping failure. Electrical cable to motor operated Core Spray "A" system injection valves MO-1400-24A (normally open) and MO-1400-25A (normally closed), could potentially be damaged by pipe whip resulting from the postulated failure of a high energy RWCU system line. However, sufficient redundancy remains to achieve and maintain safe shutdown in the event of a failure to open of motor operated valve MO-1400-25A.

0.6.4.2 RWCU System Piping Failure in the RHR Valve Compartment

A postulated RWCU high energy line failure in the RHR valve compartment at el 40 ft in the Reactor Building would result in a single ended piping failure until isolation is achieved. Postulated high energy piping failure locations are shown on Figures 0.6-22 and 0.6-26 and are listed on Table 0.6-1. A RWCU line break in the area would lead to a maximum pressure and temperature as shown on Table 0.6-2. The postulated break location inboard of the RWCU check valve (see Figure 0.6-26) was bounded by a feedwater line break in the steam tunnel, and thus was not considered in this evaluation of compartment temperature and pressure.

The ultimate capacity of the reinforced concrete walls and floor slab in this compartment in excess of the loads induced by either compartment pressurization or jet force. However, the reinforced concrete block wall on Reactor Building column line H.5 has been strengthened, as noted in Table 0.6-3, Section D. This break would tend to lead to slight pressurization of the steam tunnel. However, this is a lesser case than the postulated failure of a main steam line in the steam tunnel as discussed in Section 0.6.1.2.

Electrical cables to motor operated RHR System "A", LPCI valves MO-1001-28A (normally open), and MO-1001-29A (normally closed), could potentially be damaged by pipe whip resulting from the postulated failure of a high energy RWCU System line in this compartment. This loss is acceptable because sufficient redundancy remains to achieve and maintain safe shutdown. Additionally, electrical cables to normally closed motor operated valves MO-1001-47 and MO-1001-50 could potentially be damaged by the postulated RWCU high energy line failure. This damage could result in loss of function of the shutdown cooling mode of the RHR System. This loss is acceptable because the reactor can be brought to safe shutdown conditions using the ADS and the RHR Systems in the suppression pool cooling mode. Electrical control cable to the ADS valve SV-203-3A could potentially be damaged by pipe whip resulting from the postulated failures. However, this loss is acceptable because sufficient redundant depressurization capability remains to achieve and maintain safe shutdown.

A postulated high energy line failure in the RWCU system return line in the RHR valve compartment at Reactor Building el 40 ft could potentially damage electrical control and power cable to RWCU isolation motor operated valve MO-1201-80. Protection is provided for the outboard isolation (check) valve by restraints added to the return line. Protection is also afforded to the cable to the inboard isolation valve (MO-1201-2) by structural protection of the tray carrying the cable, as noted in Table 0.6-3, Section C.

0.6.5 Sampling System

Seven high energy sample lines were reviewed. The review indicated that no sample line failure could affect the ability to achieve or maintain safe shutdown.

Postulated break locations for sample lines reviewed were chosen at the following locations:

1. The connection of the sample line to the process line being sampled.

2. The two intermediate locations of highest stress in each of the lines.

3. The connection of the sample line to the isolation valve in the sample station.

The rupture of a sample line will be a single ended failure. These postulated ruptures will not result in any unacceptable effects on structures, systems, or components required for shutdown. Pipe whip and jet impingement effects would be negligible due to the small size of the sample lines. Environmental effects resulting from sample line failure were considered in the review. No unacceptable effects were identified.

These indications available to the operator that a sample line break has occurred are:

1. High space temperature alarm and isolation from the RWCU Leak Detection System. Temperature detectors located in the RWCU equipment cells and in the exhaust ducts provide this function as indicated on Figure 4.9-3.

2. Reactor Building high radiation alarm. A sample line failure will result in higher than normal radiation levels in the Reactor Building ventilation exhaust.

3. Area radiation monitor alarms. There are area radiation monitors in the vicinity of each sample station.

4. Visual observation by plant operating or maintenance personnel. Sample line damage would be reported to control room personnel and appropriate action would be taken to manually isolate the leak.

TABLE 0.6-1

NUMBER AND LOCATION OF GUILLOTINE/SLOT FAILURES IN HIGH ENERGY PIPING

System	Line Size, in.	No. of Break Points	Location of Break Points	FSAR Reference Figure
Feedwater	18 -GE-18	1	Condenser Compartment	0.6-12
Feedwater	18 -DE-6	5	Condenser Compartment	0.6-17
Feedwater	16 -GE-18	10	Condenser Compartment	0.6-1, 0.6-2, 0.6-6, 0.6-7 0.6-12, 0.6-16, 0.6-17
Feedwater	14 -GE-18	13	Above Turbine Deck and Condenser Compartment	0.6-2, 0.6-3, 0.6-4, 0.6-7, 0.6-9
Feedwater	12 -DE-6	10	Condenser Compartment	0.6-1, 0.6-2 0.6-7, 0.6-8
Feedwater	16 -DE-6	17	Condenser Compartment	0.6-1, 0.6-2, 0.6-7, 0.6-17, 0.6-18
Feedwater	24 -DE-6	4	Condenser Compartment	0.6-1, 0.6-6, 0.6-9
Feedwater	16 -DE-6	2	Steam Tunnel	0.6-18
Feedwater	6 -DE-6	18	Condenser Compartment	0.6-1, 0.6-6, 0.6-9
Feedwater	10 -OE-6	9	Condenser Compartment	0.6-1, 0.6-2, 0.6-6, 0.6-7, 0.6-15
Main Steam	20 -EE-1	5	Steam Tunnel	0.6-18
Main Steam	20 -EE-1	2	Turbine Deck	0.6-26
Main Steam	20 -EE-1	9	Condenser Compartment	0.6-1, 0.6-11, 0.6-12, 0.6-13, 0.6-27

TABLE 0.6-1 (Cont.)

NUMBER AND LOCATION OF GUILLOTINE/SLOT FAILURES IN HIGH ENERGY PIPING

System	Line Size, in.	No. of Break Points	Location of Break Points	FSAR Reference Figure
Main Steam	14 -EE-1	2	Condenser Compartment	0.6-12
Main Steam	10- EE-1	10	Condenser Compartment	0.6-11, 0.6-12
Condensate	4 - DE-6	12	Condenser Compartment	0.6-2, 0.6-6, 0.6-7, 0.6-8, 0.6-9, 0.6-10, 0.6-14
HPCI	10 -EB-23	1	HPCI Pump Compartment	0.6-21
HPCI	10 -EB-23	1	HPCI Valve Compartment	0.6-18
HPCI	10 -EB-23	2	Torus Compartment	0.6-20
RCIC	3 -EB-13	2	Torus Compartment	0.6-19, 0.6-25
RCIC	3 -EB-13	2	RCIC Pump Compartment	0.6-19, 0.6-25
RCIC	3 -EB-13	4	RCIC Valve and Tip Compartment	0.6-18, 0.6-26
RWCU	4 -DC-12	5	Pump and Heat Exchanger Compartments	0.6-22, 0.6-24
RWCU	6 -EA-12	4	Pump and Heat Exchanger Compartments	0.6-22, 0.6-24
RWCU	4 -EA-12	8	Pump and Heat Exchanger Compartments	0.6-22, 0.6-24
RWCU	4 -DC-12	4	Pump and Heat Exchanger Compartments	0.6-22, 0.6-24

TABLE 0.6-1 (Cont)

NUMBER AND LOCATION OF GUILLOTINE/SLOT FAILURES IN HIGH ENERGY PIPING

System	Line Size, in.	No. of Break Points	FSAR Location of Break Points	Reference Figure
RWCU	6-DL-12	1	RHR Valve Compartment (El. 40'-9")	0.6-23, 0.6-26
RWCU	4-DA-12	2	RHR Valve Compartment (El. 40'-9")	0.6-23, 0.6-26

SUBTOTAL	
Feedwater	89
Main Steam	28
Condensate	12
HPCI	4
RCIC	8
RWCU	24
TOTAL	165

Table O.6-2

SUMMARY OF MAXIMUM PRESSURES AND TEMPERATURES IN VARIOUS COMPARTMENTS DUE TO POSTULATED HIGH ENERGY PIPING FAILURES

Location	Maximum Compartment Pressure (psig)	Maximum Compartment Temperature (°F)
Main Steam Tunnel	6.8	298
Condenser Compartment	2.3	195
RHR Valve Station	5.3	252
HPCI Pump Compartment	2.2	302
HPCI Valve Station	13.3	303
Torus Compartment	1.0	257
RCIC Valve Station	0.6	264
RCIC Pump Compartment	0.7	229
RWCU Heat Exchanger Compartment	2.8	250

TABLE 0.6-3

MODIFICATIONS PERFORMED AS A RESULT OF HELB ANALYSES

A. Relocated Cables

Scheme Cable No.	Circuit Function	Applicable Design Drawings (Location)		Affected Tray/Conduit
SAA 503A	Pump P203A Feeder	E-329	(F-G, 2-3)	A1320
		E-328	(G-2)	A1320
		E-333 Sh. 1	(A-F, 5)	A1320
		E-333 Sh. 1	(F, 5-7)	A1338
		E-298 Sh. 4	(C-E, 3-4)	A1338
		E-7	(B-C, 7)	N/A
SAA 506A	Pump P203C Feeder	E-329	(F-G, 2-3)	A1327
		E-328	(G-2)	A1327
		E-333 Sh. 1	(A-F, 5)	A1327
		E-333 Sh. 1	(F, 5-7)	A1339
		E-298 Sh. 4	(C-E, 3-4)	A1339
		E-7	(B-C, 7)	N/A
SAA 507A	Pump P215A Feeder	E-329	(F-G, 2-3)	A1318
		E-328	(G-2)	A1318
		E-333 Sh. 1	(A-F, 5)	A1318
		E-333 Sh. 1	(F, 5-7)	A1342
		E-298 Sh. 4	(C-E, 3-4)	A1342
		E-7	(B-C, 6)	N/A
SAA 509AA	Diesel Generator A Feeder	E-329	(F-G, 2-3)	A1321
		E-328	(G-2)	A1321
		E-333 Sh. 1	(A-F, 5)	A1321
		E-333 Sh. 1	(F, 5-7)	A1340
		E-298 Sh. 4	(C-E, 3-4)	A1340
		E-7	(A-C, 5-6)	N/A
SAA 509AB	Diesel Generator A Feeder	E-329	(F-G, 2-3)	A1322
		E-328	(G-2)	A1322
		E-333 Sh. 1	(A-F, 5)	A1322
		E-333 Sh. 1	(F, 5-7)	A1341
		E-298 Sh. 4	(C-E, 3-4)	A1341
		E-7	(A-C, 5-6)	N/A
SAA 509F	Breaker A509A Auto-Close Control	E-329	(G, 3-4)	A1323
		E-329	(G, 3-4)	A1324
		E-328	(G, 4)	A1324
		E-333 Sh. 1	(B-F, 7)	A1324
		E-298 Sh. 4	(C, 3)	A1324
		E-40		N/A

TABLE O.6-3 (Cont.)

A. Relocated Cables (Cont.)

Scheme Cable No.	Circuit Function	Applicable Design Drawings (Location)		Affected Tray/Conduit
SAB 103AA	MCC-B15 Feeder	E-329	(E-F, 4)	AS07
SAB 103B		E-298 Sh.2	(B, 5-7)	A1331
		E-310	(D-E, 8)	A1331
		E-9	(F-G, 8)	N/A
SAB 103AB	MCC-B15 Feeder	E-329	(E-F, 4)	AS07
		E-329	(E-F, 3)	AX03
		E-329	(F-G, 2-3)	A1332
		E-328	(G, 2)	A1332
		E-333 Sh.1	(A-F, 5)	A1332
		E-333 Sh.1	(F, 5-8)	A1343
		E-298 Sh.4	(D-H, 3-4)	A1343
		E-298 Sh.3	(B-G, 3-8)	A1343
		E-298 Sh.1	(B-F, 2-5)	A1343
		E-298 Sh.2	(B-H, 5-7)	A1343
		E-310	(D-E, 8)	A1343
		E-9	(F-G, 8)	N/A
SAB 104A	MCC-B17 Feeder	E-329	(F-G, 2-3)	A1319
SAB 104B		E-328	(G, 2)	A1319
		E-333 Sh.1	(A-F, 5)	A1319
		E-333 Sh.1	(F, 5-8)	A1345
		E-298 Sh.4	(D-H, 3-4)	A1345
		E-298 Sh.3	(B-C, 3)	A1345
		E-9	(F-G, 8)	N/A
SAD 404A	Diesel Generator Control DC Power Supply	E-329	(F-G, 4)	AW06
		E-329	(F-G, 3-4)	AW05
		E-329	(F-G, 3)	A1325
		E-329	(G, 3-4)	A1324
		E-328	(G, 4)	A1324
		E-333 Sh.1	(B-F, 7)	A1324
		E-298 Sh.4	(C, 3)	A1324
		E-203 Sh.1A		N/A
SAG 18B	Diesel Generator Compressor No. 1 Control	E-329	(G, 2-3)	A1326
SAG 20B		E-329	(G, 3-4)	A1324
		E-328	(G, 4)	A1324
		E-333 Sh.1	(B-F, 7)	A1324
		E-298 Sh.4	(C, 3)	A1324
		E-27	(E, 3-4)	N/A

TABLE O.6-3 (Cont.)

A. Relocated Cables (Cont.)

Scheme Cable No.	Circuit Function	Applicable Design Drawings (Location)		Affected Tray/Conduit
SBA 607A	Pump P215B Feeder	E-320	(G, 4)	B1275
		E-7	(B-C, 3)	N/A
SBA 609F	Breaker A 609 Close/Trip Control	E-328	(F-G, 3-5)	B1279
		E-320	(E-G, 4-5)	B1279
		E-40		N/A
SBB 204A	MCC-B14 Feeder	E-320	(G, 5)	B1276
		E-9	(F-G, 5)	N/A
SBB 204B	MCC-B14 Feeder	E-320	(G, 5)	B1277
SBB 204C		E-9	(F-G, 5)	N/A
SBB 205A	MCC-B18 Feeder	E-320	(G-5)	B1278
SBB 205B		E-9	(F-G, 5)	N/A
SBTE 502204B	Torus Water TE-502204B	E-297	(F-G, 6-7)	B1286
SXNS 12XF	ADS Valve SV203-3C Control	E-298 Sh.2	(B, 3)	X503
		E-298 Sh.4	(B, 5-8)	X503
SXNS 12XH		E-298 Sh.4	(B-C, 3-5)	X1153
		E-333 Sh.1	(D-F, 6-8)	X1153
		E-128	(A-B, 5-8)	N/A

B. Replaced Cables

Scheme Cable No.	Circuit Function	Applicable Design Drawings (Location)		Affected Tray/Conduit
SATE 502102B	Torus Water TE-502102B	E-297	(E-F, 3-4)	A415
SBTE 502204B	Torus Water TE-502204B	E-297	(F-G, 6-7)	B1286

TABLE O.6-3 (Cont)

MODIFICATIONS PERFORMED AS A RESULT OF HELB ANALYSES

C. Protected Raceway

Affected Tray/Conduit No.	Function	Applicable Design Drawings (Location)	
X406	Control for ADS Valve SV203-3D	C-801 C-802	(C-G, 3-8) (A-G, 4-8)
X166	Control for ADS Valve SV203-3C	C-804	(A-G, 4-8)
A302	Control and Power to RCIC MO-1301-17	C-804	(A-G, 4-8)
BP03 B1172 B1038	Control and Power to RCIC MO-1301-16	C-804	(A-G, 4-8)
B023	Control and Power to HPCI MO-2301-5	C-801 C-802	(C-G, 6-8) (C-G, 1-4)
A357	Control and Power to RWCU MO-1201-5	C-805	(A-G, 1-8)
XQ02 XQ03	Control to RWCU MO-1201-2	C-803	(A-G, 1-8)

D. Equipment/Structural Modifications

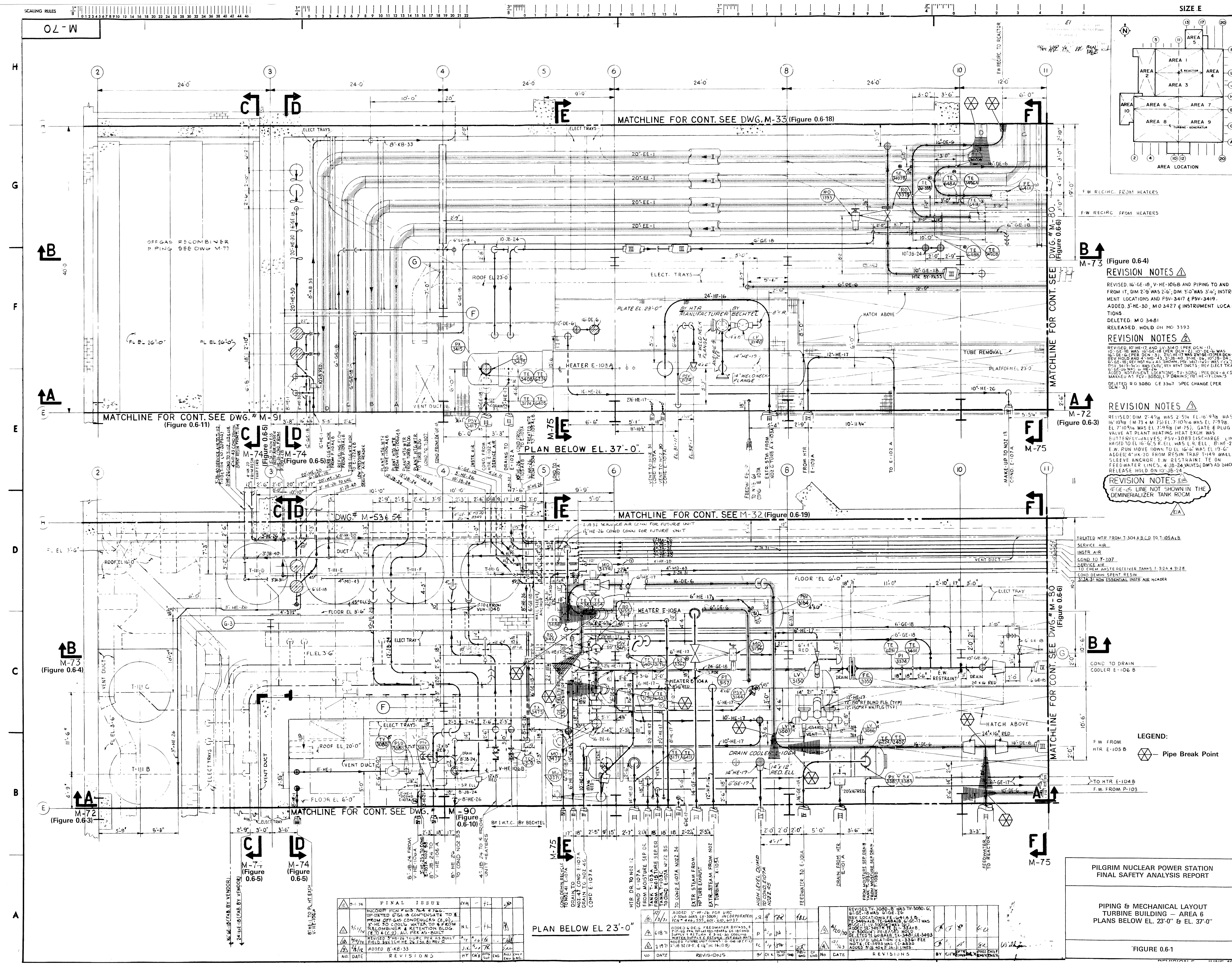
Modification Description	Applicable Design Drawings (Location)	
Provided additional compartment vent area by the removal of concrete hatch plug at El 51'-0" on the west end of the Turbine Building	C-234 C-212 C-217	(C-H, 1-3) (F, 6-7) (F, 5-6)
Provided a vent path from the HPCI pump room through the equipment hatch and blowout panel at El 3 ft, and the existing equipment hatch at El 23'-0" in the Reactor Auxiliary Bay	C-807 C-806	(A-H, 1-8) (A-G, 1-8)
Modified the main turbine combined intermediate valve enclosures at El 51'-0" in the Turbine Building to ensure blowout at 0.25 psid	C-235 C-212 C-217	(A-E, 6-8) (D-F, 4-5) (F, 5)
Provided structural reinforcement for the main steam tunnel floor	C-800	(A-H, 2-8)
Provided structural reinforcement for the south wall in the RHR system valve compartment at El 23'-0" in the Reactor Building	C-808	(A-G, 1-8)

TABLE 0.6-3 (Cont)

MODIFICATIONS PERFORMED AS A RESULT OF HELB ANALYSES

D. Equipment/Structural Modifications (Cont)

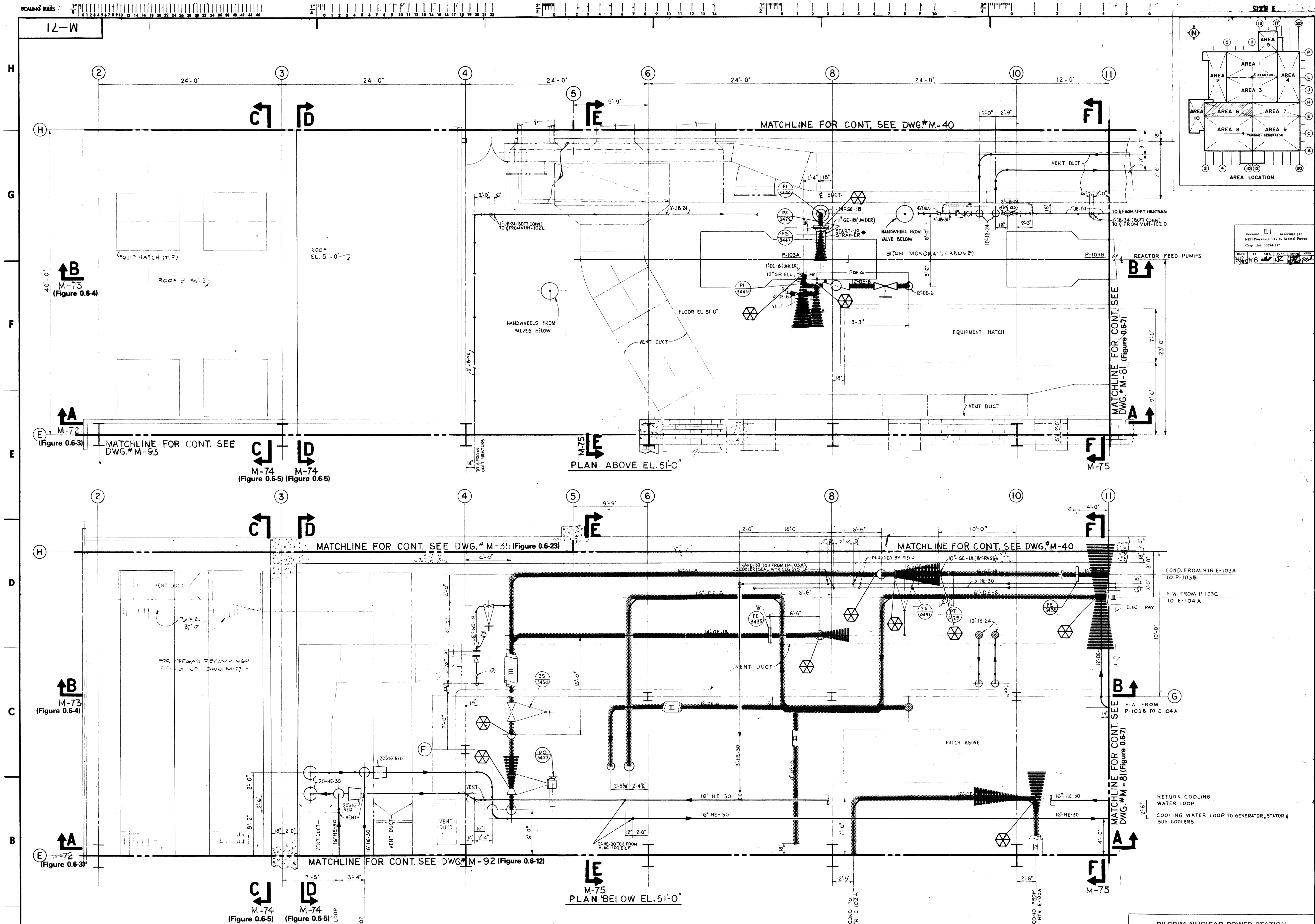
Modification Description	Applicable Design Drawings (Location)	
Provided structural reinforcement for the removable block wall on the north side of the RWCU System pump and heat exchanger compartments at El 51'-0" in the Reactor Building	C-805	(C-G, 1-3)
Provided structural reinforcement for a portion of the HPCI System valve compartment west wall at El 23'-0" in the Reactor Building	C-801	(A-G, 6-8)
Reinforced two doors at El 6'-0" in the east condenser compartment wall	C-226 C-191 A-7 A-37 A-36 A-43	(A-D, 1-4) (E, 4) (B-D, 4) (E, 8) (E-H, 5-6)
Provided pipe whip restraints to ensure the operability of HPCI outboard isolation valve MO-2301-5 in the torus compartment and the HPCI valve compartment	C-801 C-802	(C-G, 4-8) (A-G, 1-8)
Provided pipe whip restraints to ensure the operability of RCIC outboard isolation valve MO-1301-17	C-804	(A-G, 4-8)
Provided pipe whip restraints to ensure the operability of RWCU outboard isolation valve MO-1201-5	C-805	(A-G, 1-8)
Upgraded insulation on dc motors for isolation valves MO-2301-5, MO-1301-17, and MO-1201-5 to Class "H"	Data Sheet M-131 Data Sheet M-105	
Installed backup manual RBCCW valves for the equipment area cooling units located within the HPCI pump compartment	M-215 M-1004 M-1004	(E, 7-8) Sheet 29 Sheet 30



PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 6
PLANS BELOW EL. 23'-0" & EL. 37'-0"

FIGURE 0.6-1



LEGEND:

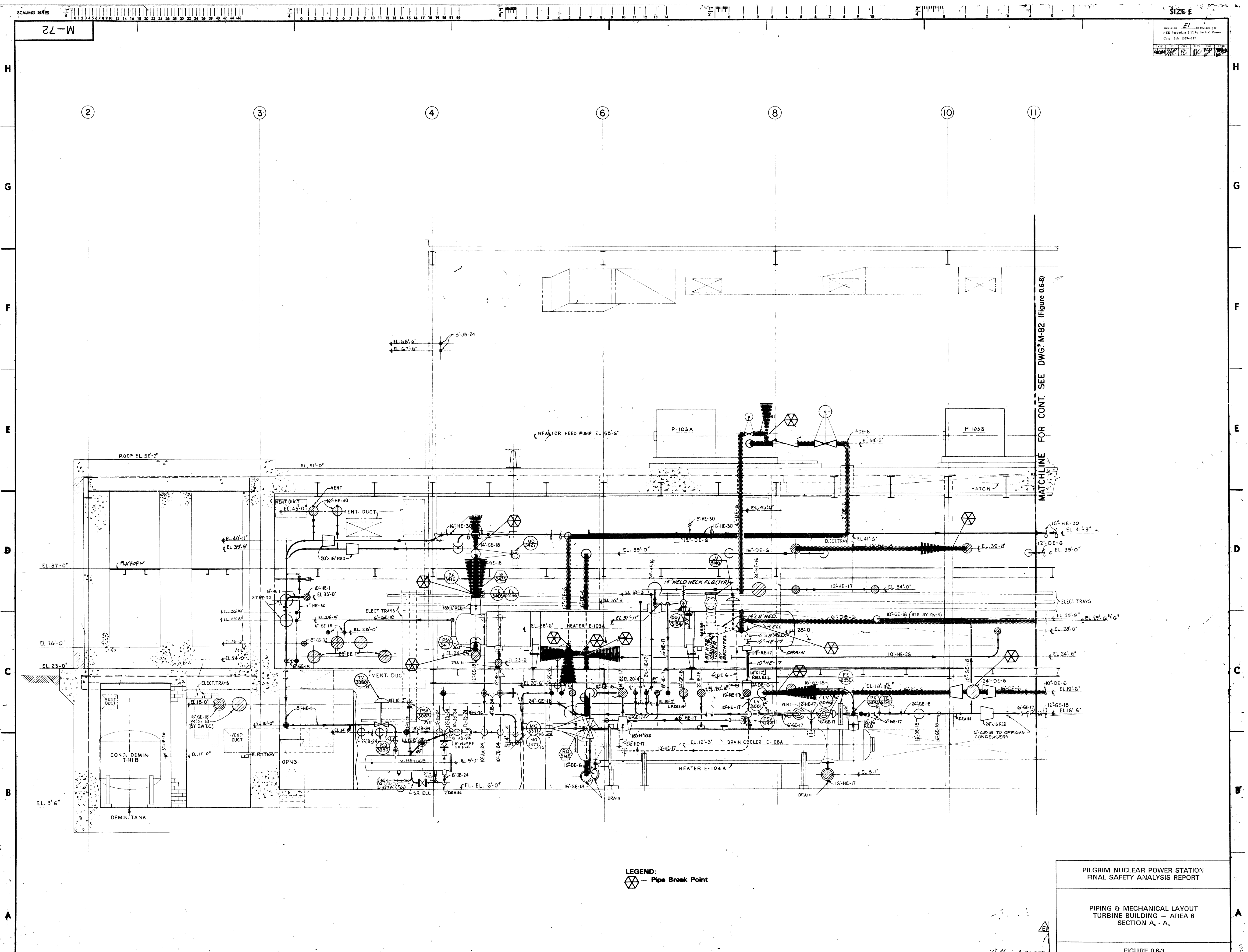
(symbol) – Pipe Break Point

No.	DATE	REVISIONS	BY	CH'K	DFTG SUPVR	ENG'R	PROJ ENG'R	CHIEF ENG'R
8	5/1/74	FINAL ISSUE	RVM	–			SAG	
7	4/17/74	REVISED 6"-GE-18 PER AS-BUILT (C,6)	N.L.	–				
6	6-18-71	ADDED PIPING 6"GE-18 FOR OFF GAS RECOMBINER	P	AJP		–		
5	2-19-71	ADDED 6" GE 18 FUTURE CONN (D 6)		AJP		–		
4	4/20/70	REVISED 1½"-HE-30 CONN. WAS 1"-HE-30 DELETED 1"-HE-30 CONN ADDED VENT; 2"-HE 30 CONN. TO & FROM V-AC-102 E & F; NOTES. L.O COOLER, TO GENERATOR, STATOR & BUS COOLERS; PLUGGED BY FIELD.				–		

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 6
PLANS BELOW EL. 51'-0" & ABOVE EL. 51'-0"

FIGURE 0.6-2

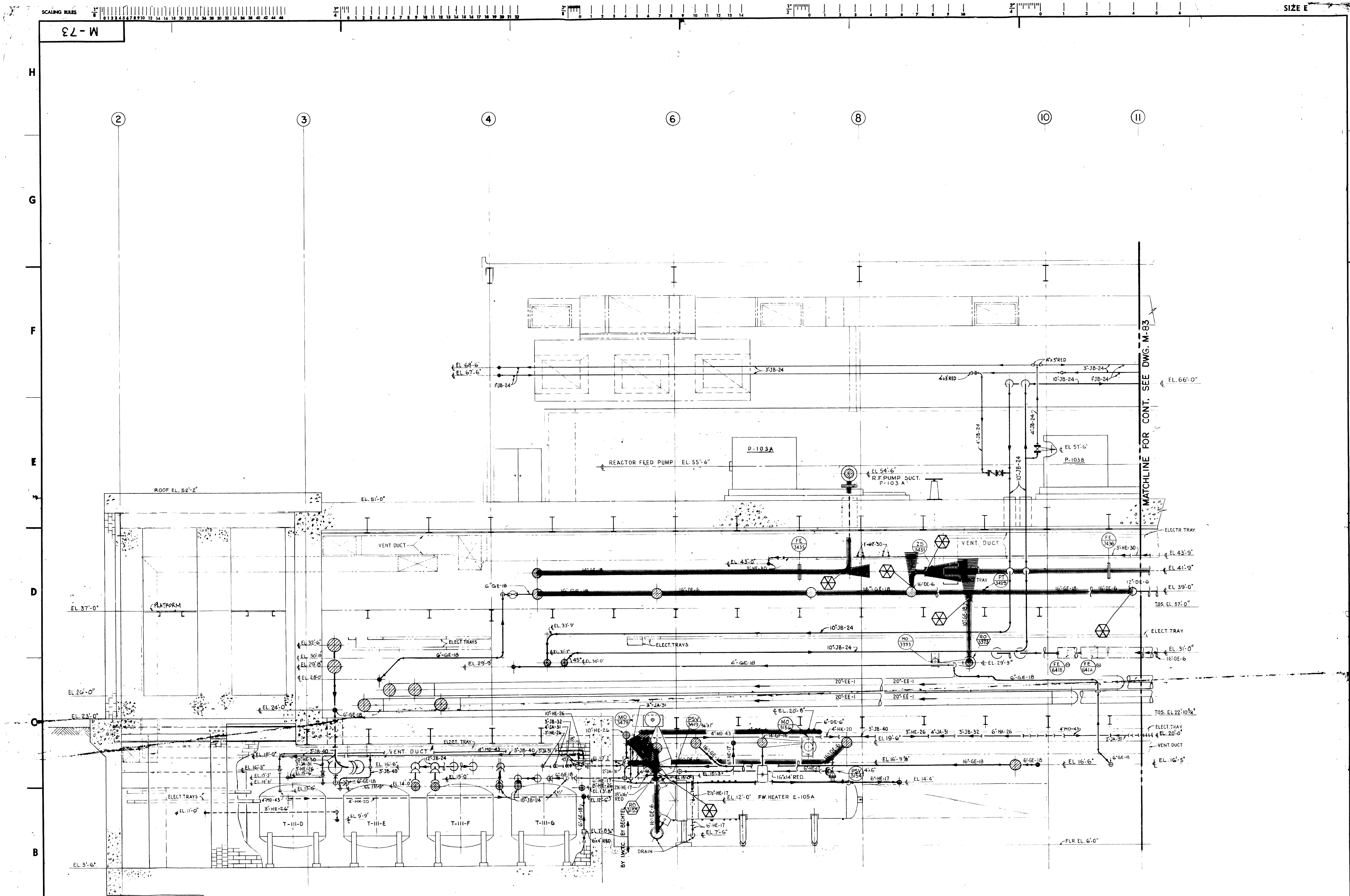


LEGEND:
— Pipe Break Point

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 6
SECTION A_6 - A_6

FIGURE 0.6-3



LEGEND:

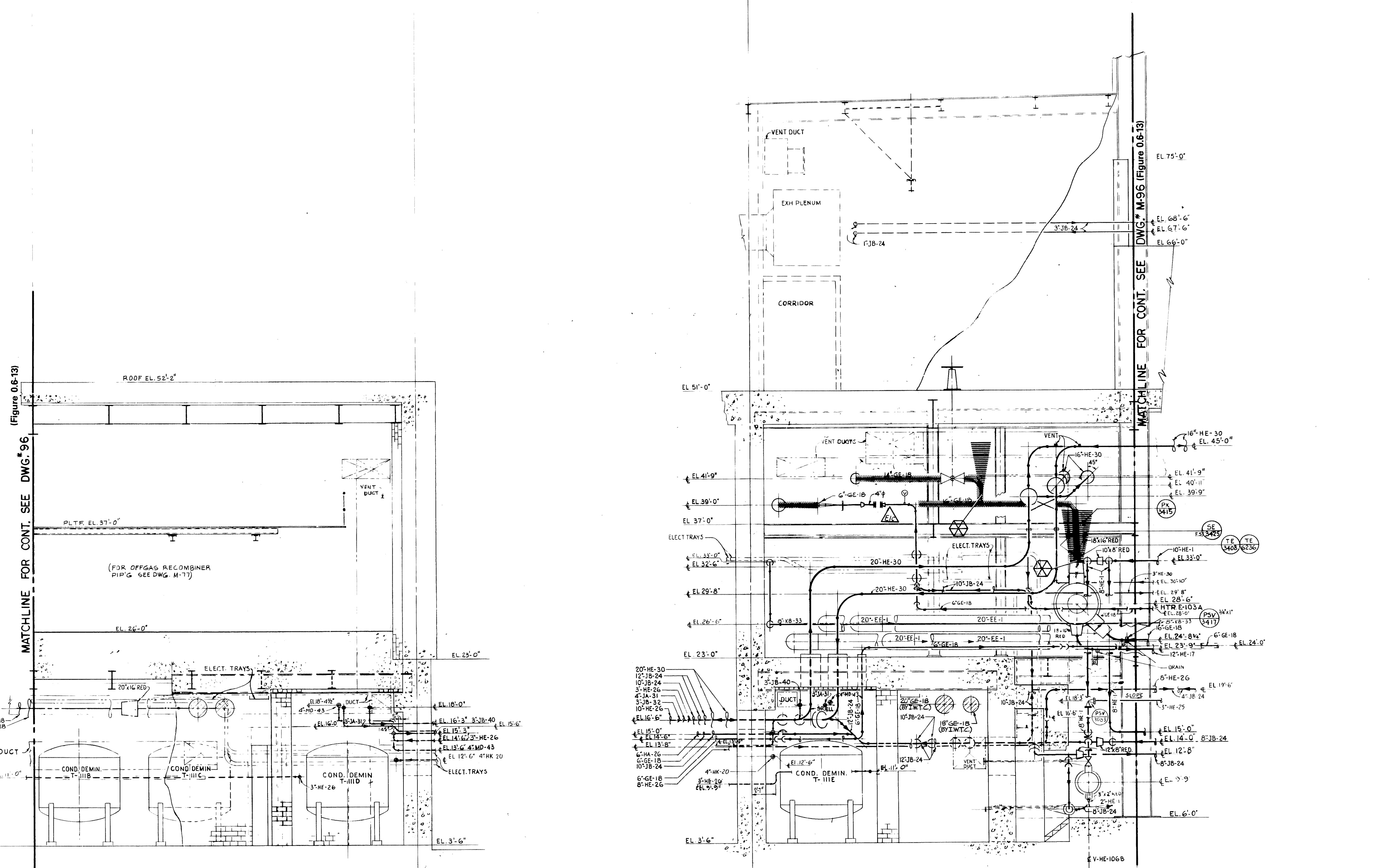
— Pipe Break Point

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING — AREA 6
SECTION B_6 - B_6

FIGURE 0.6-4

M-74



SECTION : C_6 - C_6

M-70171 (Figures 0.6-1 & 0.6-2)

SECTION : D_6 - D_6

M-70171 (Figures 0.6-1 & 0.6-2)

LEGEND:
— Pipe Break Point

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING — AREA 6
SECTIONS C_6 - C_6 & D_6 - D_6

FIGURE 0.6-5