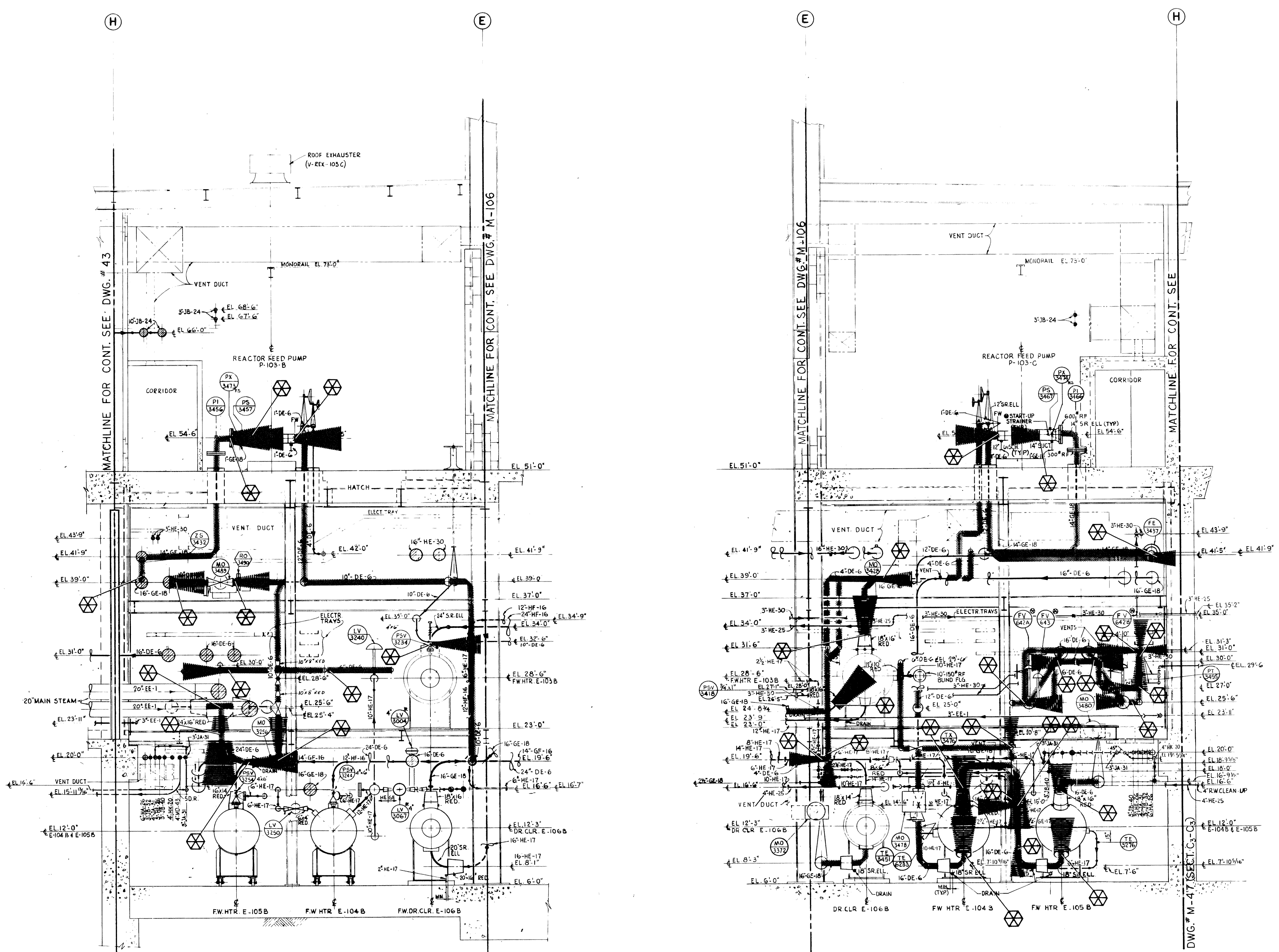
M-84



SECTION : $C_7 - C_7$.

SECTION : $D_7 - D_7$.

LEGEND:
⊗ – Pipe Break Point

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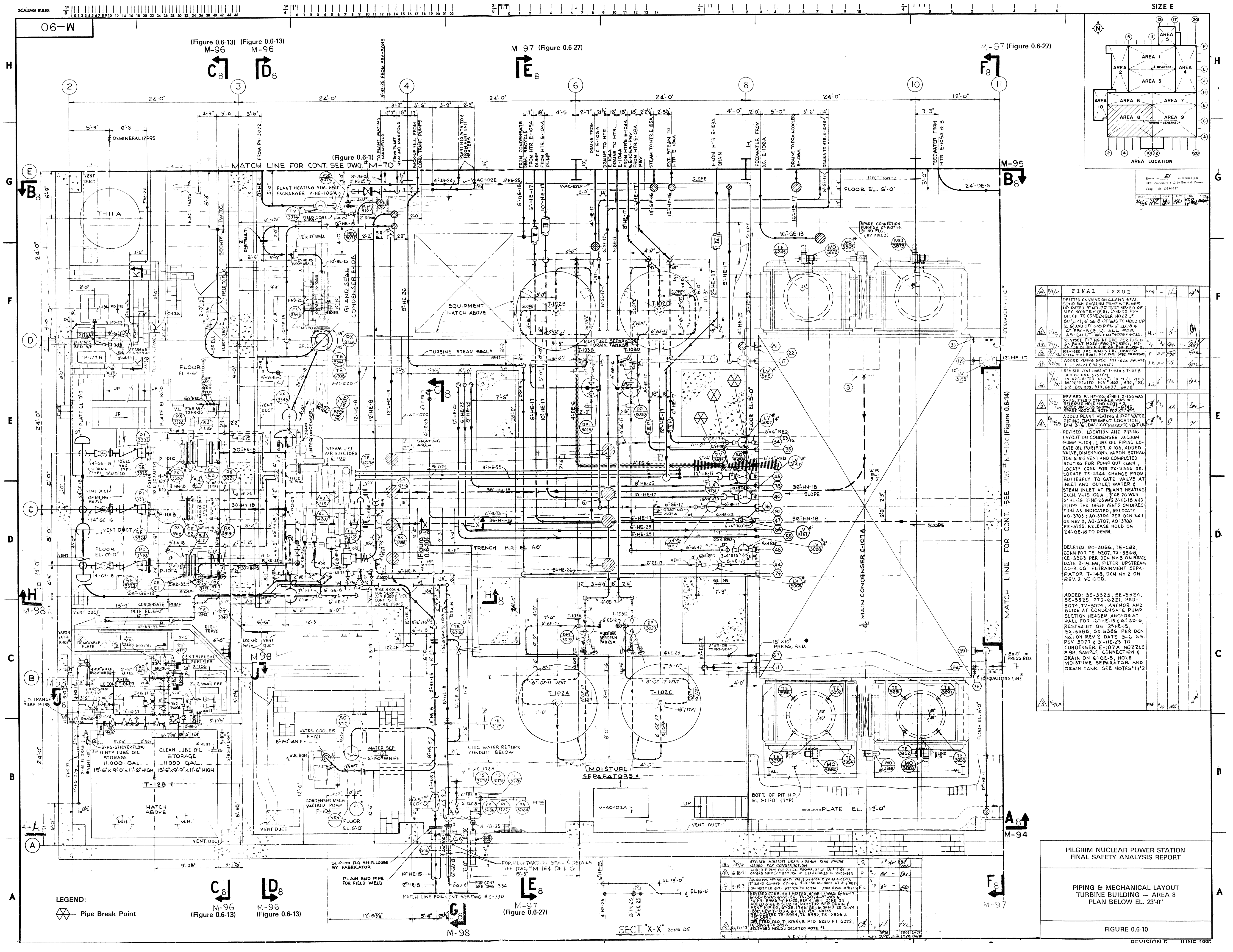
PIPING & MECHANICAL LAYOUT
TURBINE BUILDING — AREA 7
SECTIONS C_7 - C_7 & D_7 - D_7

FIGURE 0.6-9

REVISION 5 — JUNE 1985



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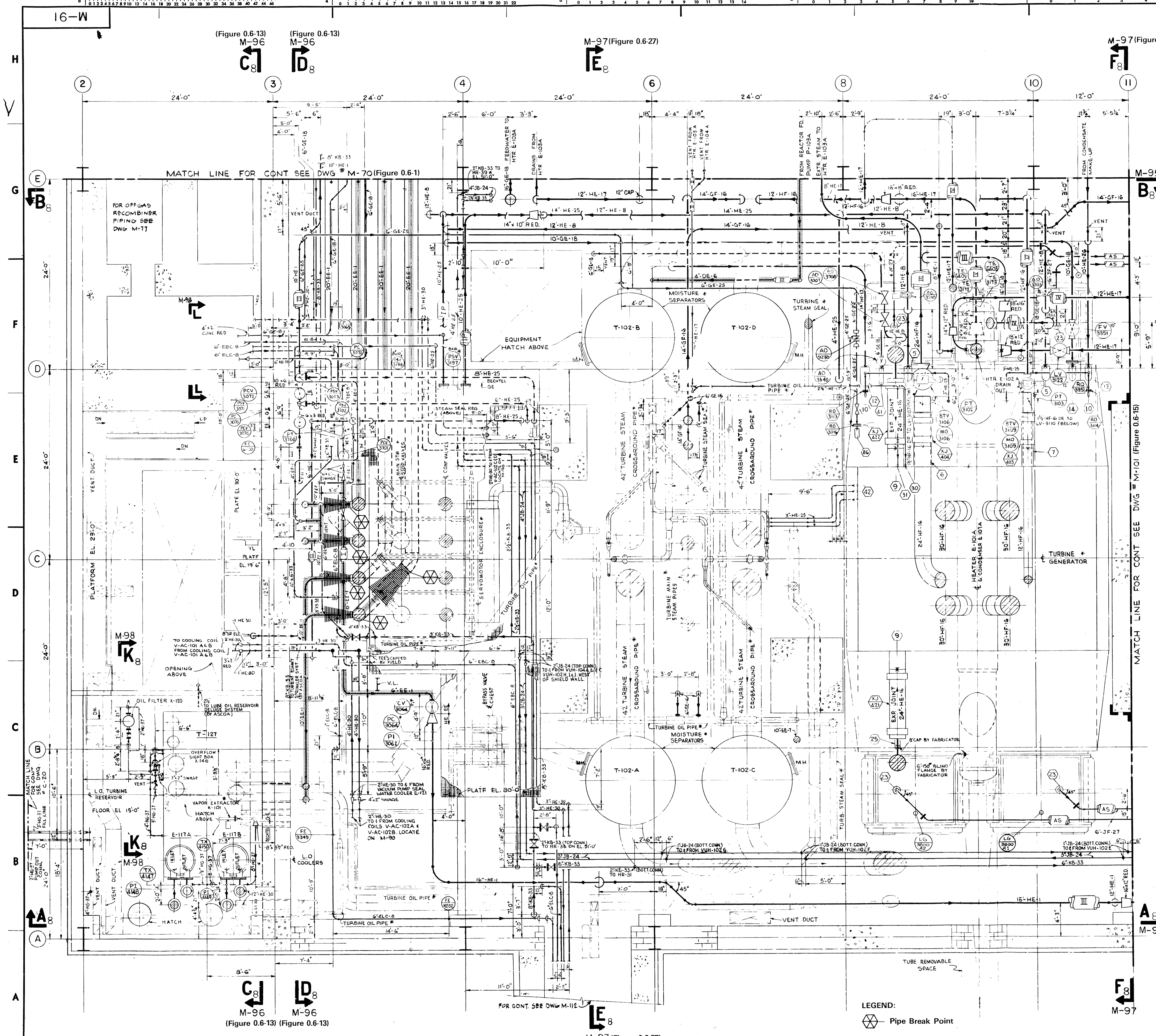
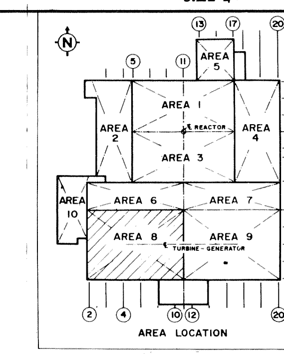
FINAL SAFETY ANALYSIS REPORT

PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 8
PLAN BELOW EL. 23'-0"

FIGURE 0.6-10





E-1	8/16/78	REV PER PDCR 77-52 DCN #1	RLT	—	RCE		WJM
13	5/3/74	FINAL ISSUE	RVM	—	—		SAG
12	4/22/74	UPDATED 3"-HE-30 CLOSING COOL'G WATER TO & FROM RETENTION & RECOMBINER BLDG (B,5 & F,6); 6"-EBC-8 & 6"-ELC-8 OFFGAS LINES (B,5 & F,5), 6"-KB-33 (D,7) FIRE LINE; 4"-EE-1 MAIN STEAM (E,7); 6"-GE-18 CONDENSATE TO & FROM RECOMB (F,6) & 6"-GE-25 OFFGAS COND. DRAIN (G,6) ALL PER AS-BUILT. INCORP. FCN # 629 & 772.	N.L.	—			
11	6/22/72	REVISED TYPE OF FE ON OFFGAS 6" ELC-8 PER LATEST GE SKETCH	AJP	P			
10	2/11/72	REVISED PIPE SPEC. ON OFFGAS. ADDED SECOND FE & NUMBERS TO FE'S.	P	AP			
9	1/17/72	ADDED PIPE SPEC OFF-GAS PIPING REV. PIPE PENETR. INTO RECOMB. ROOM INCORPORATED DCN #1 TO M-218 REV. 7 FCN # 806, 809, ADDED 8"-KB-33	J.R.	AP			
8	12/3/71	REVISED PIPE PENETRATION INTO RECOMBINER ROOM. REVISED 3" EE-1. ADDED (2) 6"-KB-33; (2) 3"-HE-30; 6"-GE-25 DUMP LINE; (3) 6" OFFGAS LINES. INCORPORATED FCN # 810 & 6036	J.R				
7	6-18-71	ADDED PIPING FOR OFF GAS RECOMBINER 6" GE 18 SUPPLY & RETURN 3" HE-30 SUPPLY & RETURN 4" GE-18 SUPPLY 6" HE-25 TO COND 3" GE-18 RETURN 4" HE-25 TO COND. 4" EE-1 SUPPLY PER FIELD MARK UP	P	AJP		—	
6	2-19-71	ADDED 3" EE 1 STM TO FUTURE UNIT (D-7) ADDED AO-3707 & 3708 (F-3) RELOCATED FROM DWG M-90	FC	AJP	JTA		
5	6/17/70	REVISED 10" & 14"-HE-25 WAS 10" & 14"-HE-1, 6"-GE-18 WAS 6"-GE-26, 2½" PIPE BY ASCOA WAS 4"-KB-33 PC-3070 WAS PT-3070. ADDED 6"-GE-16; PT-3700 DELETED 3"-HE-30.		AJP			
4	1/23/70	REVISED 3"-HE-30 LINE, PC-3072 WAS PCV-3072, ADDED 6" DIM & X-140 & DIM. AS SHOWN		AJP			
3	8/15/69	ADDED PLANT HEATING & FIRE WATER PIPING		FBF			
2	7/3/69	REVISED: STEAM TO AIR EJECTOR STATION PIPING, RELOCATE PSV-3702 & PCV-3070, SPEC CHANGE PER DCN No 2 ON REV. 1, COOLING WATER LINES & ADDED 3"-HE-30, 14"-GF-16 SR ELL TO L R ELL, RELOCATE INSTS. & ADDED NEW INSTS PER DCN No 4 ON REV. 1, FV-3351 TO 10" RE-MOVE REDUCERS, 3"-HE-25 WAS 3"-HE-17 DELETED: PCV-3061, PSV-3063 3"-HE-25 TO COND. & RELEASE HOLD WITH CHANGES AS INDICATED DIMENSION TO 4"-HG-37, PSV-3118 & 8"-HE-17 TO COND. NOZZLE # 25 PER DCN. No 3 ON REV 1, MO.3119 & CHANGE PIPE SHAPE ADDED 2"-HG-37 PUMP OUT CONN., VENT CONN ON 6"-GE-18 & 6"-EE-1, AO-3703 & AO-3710 PER DCN No 1 ON REV 1, 45° ON WATER BOXES VENTS.	FBF	AJP			

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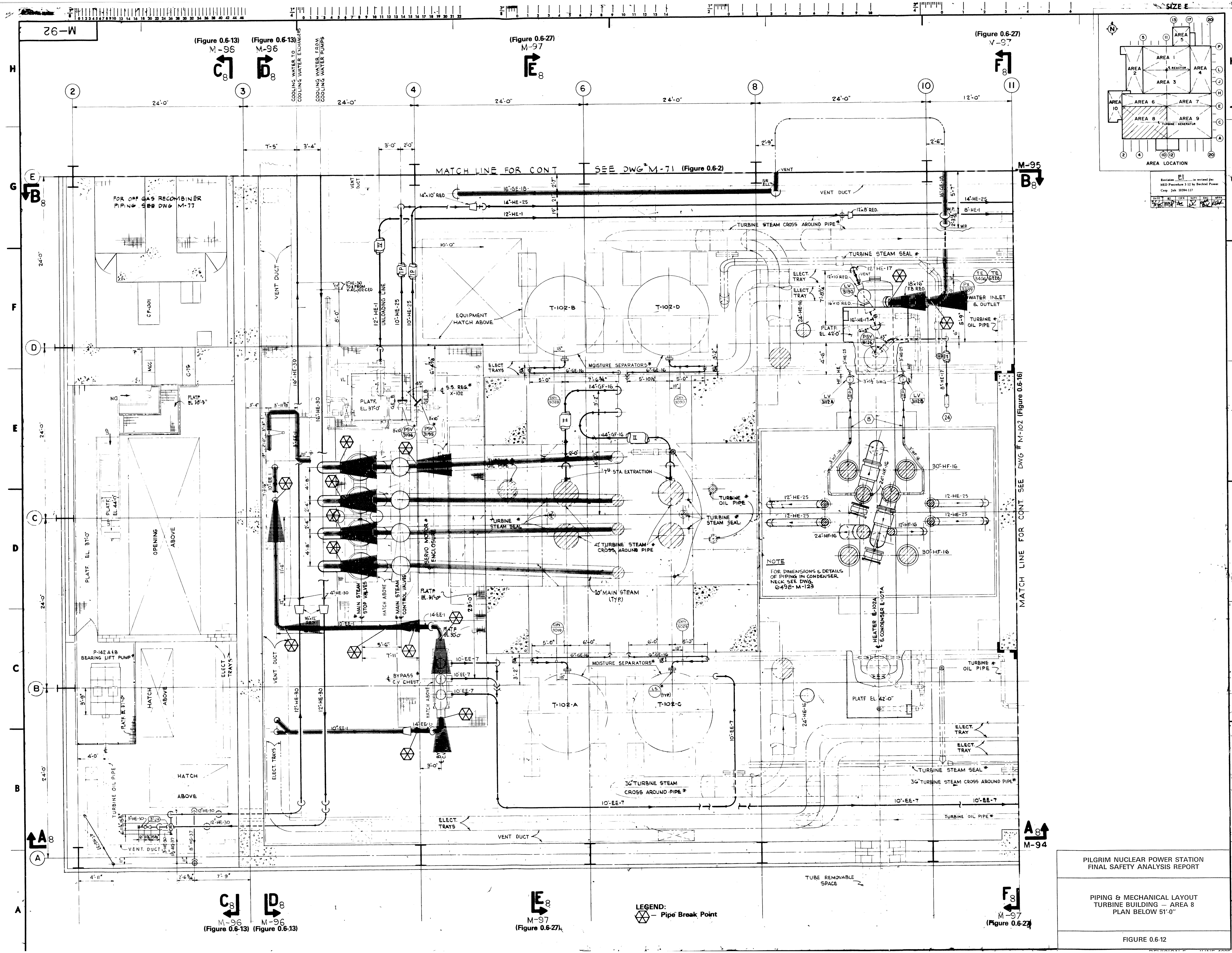
TURBINE BUILDING – AREA 8
PLAN BELOW EL. 37'-0"

FIGURE 0.6-11



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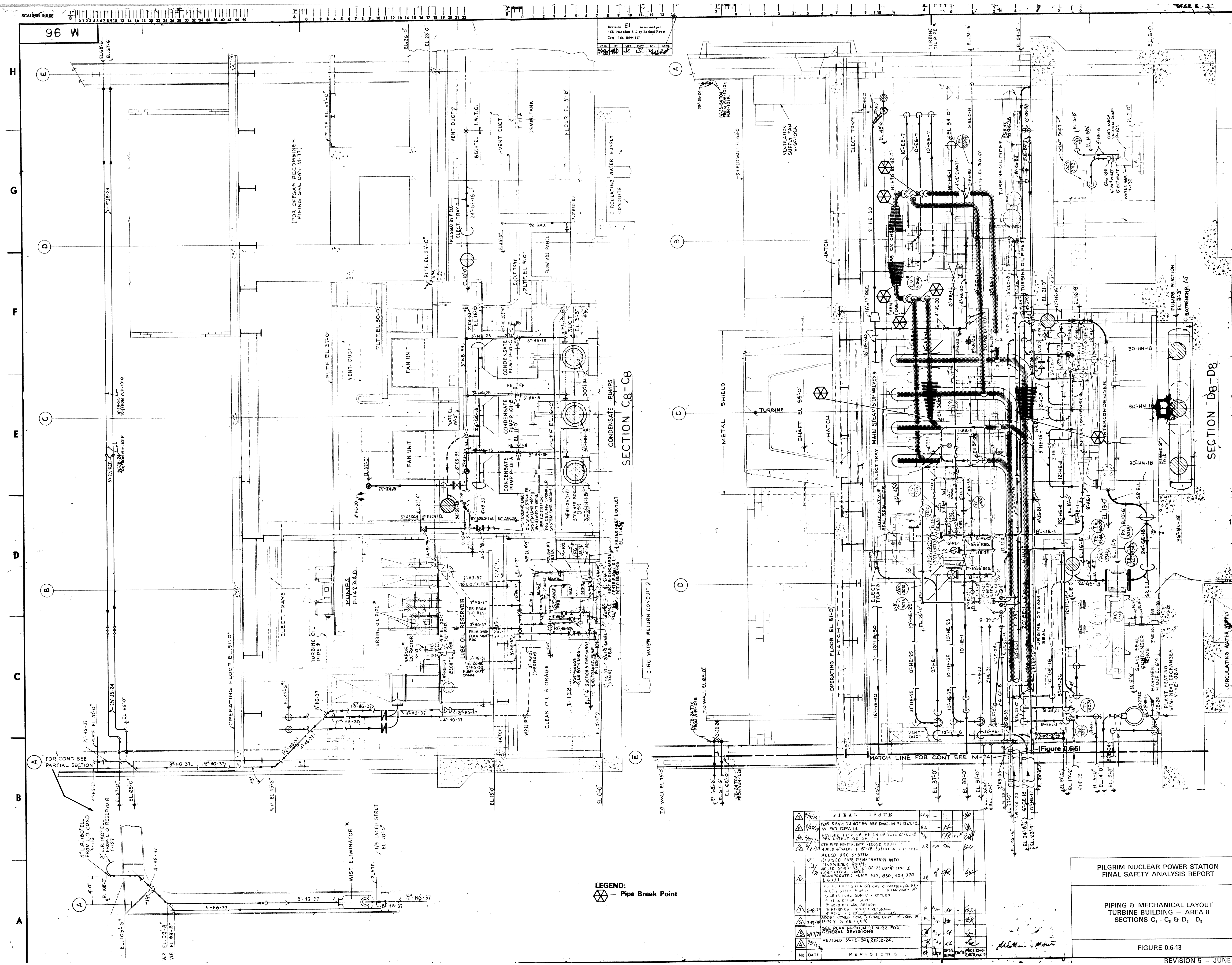
PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 8
PLAN BELOW 51'-0"

FIGURE 0.6-12



No.	DATE	REVISIONS	BY	CH'K	DFTG. SUPVR	ENG'R	PROJ. ENG'R	CHIEF ENG'R
12	5/3/74	FINAL ISSUE	R.V.M.					
11	4/22/74	FOR REVISION NOTES SEE DWG M-91 REV. 12, M-90 REV. 14.	N.C.					
10	6/21/72	REVISED TYPE OF FE ON OFFGAS 6"ELC-8 PER LATEST GE SKETCH	AJP					
9	2/1/72	REV PIPE PENETR. INTO RECOMB ROOM ADDED 6" VALVE & 8"-KB-33 OFFGAS PIPE SPEC	J.R.	AP				
8	12/3/71	ADDED URC SYSTEM REVISED PIPE PENETRATION INTO RECOMBINER ROOM. ADDED 6"-KB-33, 6"-GE-25 DUMP LINE & (2) 6" OFFGAS LINES INCORPORATED FCN # 810, 830, 909, 970 & 6037	JR					
7	6-18-71	ADDED PIPING FOR OFF GAS RECOMBINER PER FIELD MAKE UP 4" EE-1 STEAM SUPPLY 6" GE-15 COND SUPPLY & RETURN 4" GE-8 OFFGAS SUPPLY 3" GE-8 OFFGAS RETURN 3" HE-30 CW SUPPLY & RETURN 4" HE-[illegible] HE-25 TO CONDENSER	P	AJP		-		
6	2-19-70	ADDED CONNS FOR FUTURE UNIT 4"-GE-8 (F-2) & 3" EE-1 (E-3)	F	AJP		-		
5	6/17/70	SEE PLAN M-90, M-91, M-92 FOR GENERAL REVISIONS		AJP				
4	1/23/70	REVISED 3"-HE-30 & 2½"-JB-24.		AJP				

PILGRIM NUCLEAR POWER STATION
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PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 8
SECTIONS C_8 - C_8 & D_8 - D_8

FIGURE 0.6-13

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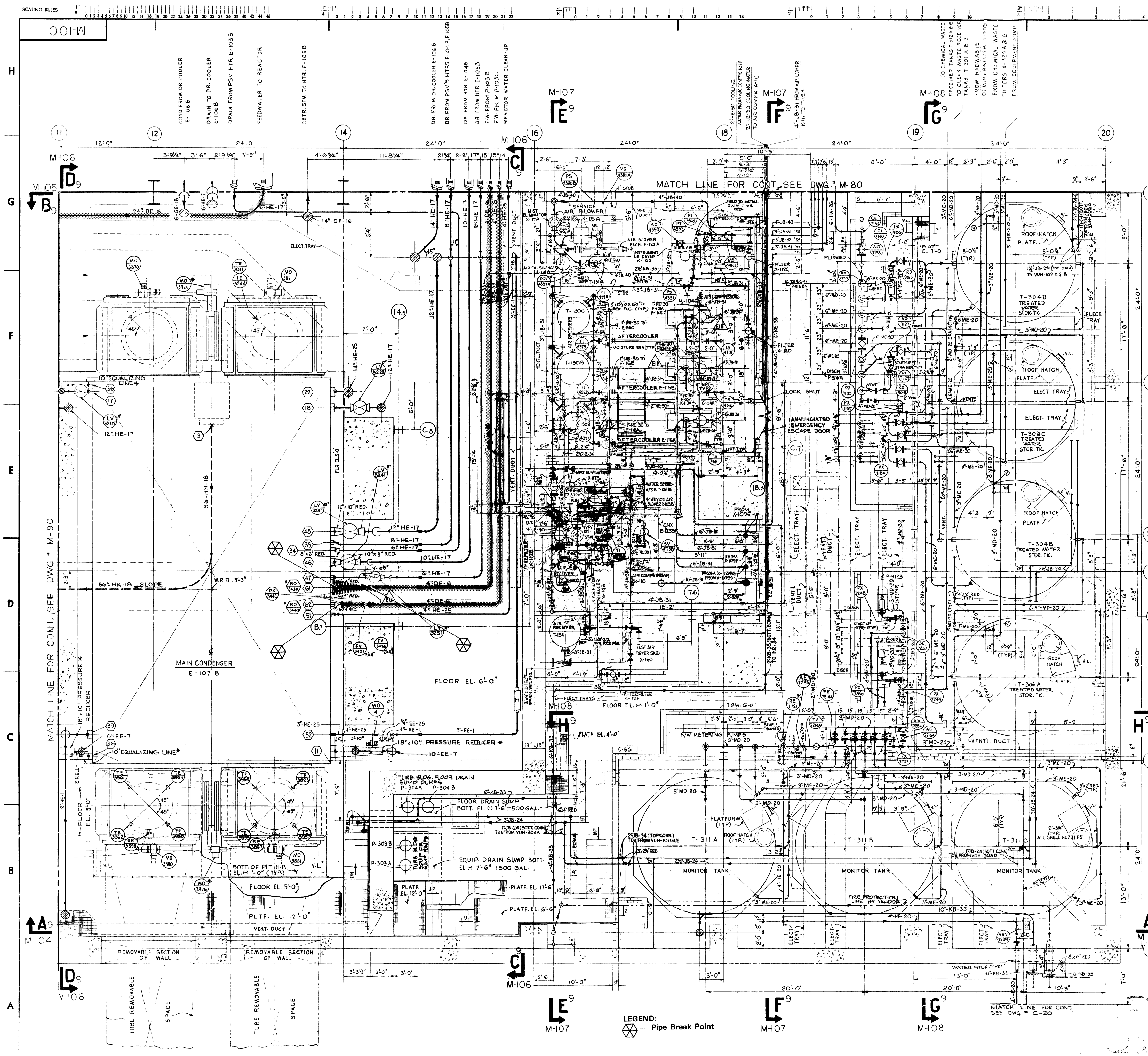

Rev.	Date	Description					
12	3/3/74	FINAL ISSUE	RVM	—			SAG
11	4/13/7?	INCORP. FCN # 676 & 973.	RVM.	—			
10	2/8/72	INCORPORATED FCN # 406, 416, 477, 607, 805, 934, 6014 & 6021	J.R.	AJP			
9	6-4-71	ADDED X-151; K-109; E-126; T-154 AND 6"-JB-31, 5"-JB-31 & 3"-JB-31 PIPING CONNECTED TO EXIST. 3"-JB-31 @ ℄ EL. 7'-3 1/8". REVISED SAMPLE CHAMBER RE-1721.		AJP		DRK	
8	1-18-71	FIELD REVISIONS TO PIPING AS FOLLOWS 4" HE-30 WAS 2 1/2" HE 30 (E-5). DELETED SX-7131 (F3) ADDED CE-7133 (F-3) RELOCATED 4" MD 20 (E-3) ADDED RO-7127 (F-3) & RO-9015 (E-3) REVISED 6" ME 20 (F-3) (E-3) ADDED PX-7155 (E-3). REV 4" MD-20 (C-3). VENDOR PIPING WAS 6"-KB-33 (A 3, A, 5)	EH/FC	AJF		—	
7	7/14/7?	ADDED SERVICE AIR BLOWER K-105B. ADDED 4" JB 40 PIPING TO RWFB FILTERS. ADDED BYPASS TO EXIST'G 4" JB 40. ADDED 2 1/2" HE-30 PIPING REVISED SERVICE AIR BLOWER EQUIP NOS. K-105A WAS K-105, ETC. ADDED TE-3811, SE-38?7 WAS TX-3897, SE-3898 WAS TX-3898, SE-7224 WAS SX-7224, 14" HE 25 WAS 14" HE 1. FIELD CHANGES AS FOLLOWS. SX-7131 RELOCATED (F-4) PX'S ADDED TO RMPS P-312A & B & P-316A & B 1" BYPASS ADDED TO 4" HE-20 (C-4) 1 1/2" JB 24 CONNS ADDED (G-2) REV 4" JA 31 (G-4). PS-80A WAS PS 80 ADDED PS 80 B (G-5) REV EL. E-116A (SEE SECT. E-E)	FC	AJP			
6	2/27/70	REVISED 6" KB-33, 2" MD-20 WAS 4" MD-20, REV. LOCATIONS TE-3894 & TE-3960 FV-7214A 2" WAS 2 1/2"		FBF	KB		
5	1/23/70	REVISED 4"-JA-31; 3"-MD-20, PI-7130 WAS PI-7133B, ADDED 3"-JA-31; PT-4352; PS-4365; TE-6244 & EXTRA C.W.S. FLG'D NOZZLE.		AJP	KB		
4	8/25/69	ADDED PLANT HEATING AND FIRE WATER PIPING AND INSTRUMENTS AND VENT LOCATIONS		FBF			
3	7/3/69	REVISED: COMPRESSED AIR SYSTEM PIPING, PCV-4397 WAS PCV-4307 AND RELOCATE COOLING WATER TO E-123, TI-4321, TI-4322 & TI-4323 WAS TE-4321, TE-4322 & TE-4323, RELOCATE 12"-HE-1. DELETED: PSV-4371, PCV-4370, AO-4353, AO-4358, RO-3067, 4"-JB-32, S.R. ELL ON 12"-HE-1, INSULATING FLANGES FOR LINES GOING IN AND OUT OF BUILDING. ADDED: DIMENSIONS FOR PIPING AROUND COMPRESSOR AND AIR BLOWER SEE SECTIONS FOR ELEVATION CHANGES AND ADDITIONS, 6"-KB-33, 6"-KC-33, MS-4363, INSTRUMENT LOCATIONS, 8"x6" RED. ON 6"-KB-33 SPEC. CHANGE AND DIM AS SHOWN, BREAKOUT FLANGES ON 4"-JB-31	FBF	AJP		AT	

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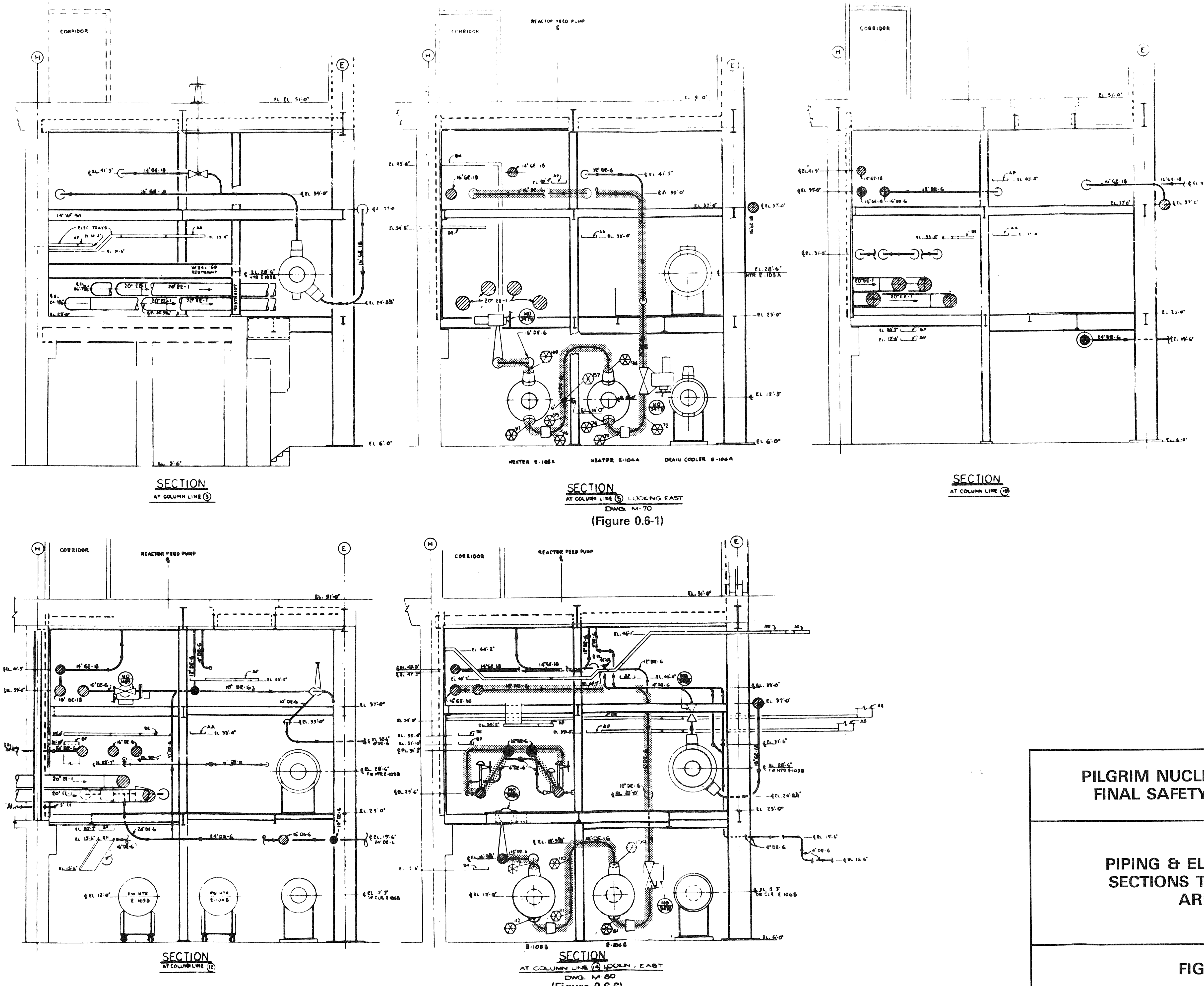
PIPING & MECHANICAL LAYOUT
TURBINE BUILDING – AREA 9
PLAN BELOW EL. 23'-0"

FIGURE 0.6-14



PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

PIPING & ELECTRICAL LAYOUT
SECTIONS TURBINE BUILDING
AREAS 7 & 8

FIGURE 0.6-17

0.7 Environmental Qualification of Electrical Equipment

Safety related electrical equipment must be able to perform its safety functions throughout its installed life in accordance with 10CFR50.49. The purpose of equipment qualification is to provide tangible evidence that the equipment will operate on demand.

Equipment must remain operable during the following exposure to harsh environmental conditions (e.g., temperature, pressure, humidity, chemical sprays, radiation and submergence) resulting from a design basis accident. The harsh environments are generally defined by limiting conditions resulting from the complete spectrum of postulated break sizes, break locations and single failures.

0.7.1 Environmental Qualification Program

Boston Edison is responsible to maintain a program to qualify the electrical equipment important to safety. This is defined as:

1) Safety related electrical equipment;

2) Non-safety related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of a safety function; and

3) Certain post-accident monitoring equipment.

The Equipment Qualification Master List (EQML) identifies the electrical equipment important to safety that must be environmentally qualified. The EQML also contains information concerning the location and mission time of the equipment.

The location of a component determines whether or not it may be subjected to harsh environments. The areas of PNPS which are considered to be subject to harsh environments are designated on the M631 series of drawings.

0.7.2 Environmental Parameters

10CFR50.49 (e) requires specific environmental effects to be addressed. The governing parameters are: temperature, pressure, humidity, chemical effects, radiation and submergence. Specific values and/or profiles of these parameters may be obtained from various documents including the M632 series of drawings.

0.7.3 Qualification Documentation

A record of qualification is maintained in an auditible form as required by 10CFR50.49 (j). The qualification records include Equipment Qualification Data Files (EQDF) and Reference Files. The EQDFs document the reports, analyses, etc., required to prove qualification.

APPENDIX P

Deleted

APPENDIX Q

SUPPLEMENTAL RELOAD LICENSING REPORTS

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SUPPLEMENTAL RELOAD LICENSING REPORTS

Q.1 INTRODUCTION

The latest plant supplemental reload licensing report is "Suppleemental Reload Licensing Report for Pilgrim Nuclear Power Station Reeload 20/Cycle 21", 000N9841-SRLR, Revision 0, Class I, Feebruary 2015 (ECH-NE- 15-00008, Revision 0). This report provides the cycle core loading pattern and the results of the cycle specific nuclear transient and vessel overpressurization analyses. This report also addresses the applicability of generic stability and accident analyses. Refer to this report for reload information. All other sections of Appendix Q have been removed from the FSAR. References for previous cycle specific reports are as follows:

RELOAD SUBMITTAL REFERENCES

Reload No.	Reference
1	"Reload 1 Licensing Submittal of Pilgrim Nuclear Power Station", NEDO-20286, Revision 1, February 1974.
2	"General Electric Boiling Water Reactor Reload No. 2 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1", NEDO-20855, June 1975.
	"Reload No. 2 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 with Bypass Flow Holes Plugged", NEDO-20855-01, September 1975.
3	"General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Pilgrim Nuclear Power Station Unit 1", NEDO-21460-01, May 1977.
4	"Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4", NEDO-24224, November 1979.
	"Supplement 1 to Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4", NEDO-24224-1, Supplement 1, March 1980.
	"Supplement 2 to Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 4 (Load Line Limit Analysis Reverification)", NEDO-24224-2, April 1981.
5	Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 5", Y1003J01A28, Revision 2, February 1983.

Reload No.	Reference
6	"Supplement Reload Licensing Submittal for Pilgrim Nuclear Power Station Unit 1 Reload 6", 23A1694, March 1984.
7	"Supplemental Reload Licensing Submittal for Pilgrim Nuclear Power Station Reload 7", 23A4800, December 1986.
8	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 8, Cycle 9," 23A7101, March 1991.
	(Note: the generator load reduction without bypass analyzed in the above licensing report is updated in another analysis (BECo SUDDS 91-44). All results presented here reflect this updated analysis).
9	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 9, Cycle 10" 23A7195, February 1993.
10	“Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 10, Cycle 11”, 24A5172, Revision 0, February 1995.
11	“Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 11/Cycle 12”, J11-03014SRL, Revision 0), February 1997.
12	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 12/Cycle 13", J11-03474-10 SRLR, Revision 0, Class I, April 1999 (SUDDS RF99-142).
13	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 13/Cycle 14", J11-03878-10 SRLR, Revision 0, Class I, February 2001 (SUDDS/RF 00-112).
14	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 14/Cycle 15", 0000-0008-6613-SRLR, Revision 1, Class I, March 2003 (SUDDS RFO258).
15	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 15/Cycle 16", 0000-0030-7302-SRLR, Revision 0, Class I, February 2005.
16	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 16/Cycle 17", 0000-0056-6173-SRLR, Revision 0, Class I, February 2007.
17	"Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 17/Cycle 18", 0000-0083-7478-SRLR, Revision 0, Class I, February 2009.

18 "Supplemeental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 2019/Cycle 210", 0000-0147-0084-SRLRN9841-SRLR,, Revision 01, Class I, February 2015April 2013.

Sections Q.2, Q.A, Q.B, Q.C, and Q.D have been removed.

Please refer to "Supplemental Reload Licensing Report for Pilgrim Nuclear Power Station Reload 19/Cycle 20", 0000-0147-0084-SRLR, Revision 1, Class I, April 2013 (ECH-NE-13-00002, Revision 0), and "Updated Loading Pattern GESTAR II Licensing Assessment for Pilgrim Cycle 20", GNF S-0000-0160-1647, Revision 1, April 2013 (ECH-NE-13-00002, Revision 0).

APPENDIX R

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APPENDIX R

INITIAL CORE STATION SAFETY ANALYSIS

R.1 INTRODUCTION

The following is the safety analysis which was performed for the initial core loading. Included in these analyses are the abnormal operational transients (R.2), design basis accidents (R.3), and special events (R.4). The analytical methods used and evaluation using the standard NRC approach for dose analyses are given in R.5 and R.6, respectively.

R.2 ANALYSES OF ABNORMAL OPERATIONAL TRANSIENTS (INITIAL CORE)

R.2.1 Events Resulting in a Nuclear System Pressure Increase

Events that result directly in significant nuclear system pressure increases are those that result in a sudden reduction of steam flow while the reactor is operating at power. A survey of the station systems has been made to identify events within each system that could result in the rapid reduction of steam flow. The following events were identified:

1. Generator trip (turbine control valve fast closure)

2. Turbine trip (turbine stop valve closure)

3. Closure of the main steam line isolation valves

4. Failure of the turbine bypass valves to open when required

5. Loss of main condenser vacuum

6. Pressure regulator malfunction

A consideration of the last three varieties of events shows that turbine bypass valve failure, loss of condenser vacuum, and pressure regulator malfunction are specific cases of the first three event types. A failure of the turbine bypass valves to open when required is analyzed as the most severe form of turbine or generator trip. A loss of condenser vacuum causes turbine stop valve closure and turbine bypass valve closure; thus, loss of vacuum is a turbine trip without bypass. Pressure regulator malfunctions that result in turbine steam flow shutoff and a nuclear system pressure rise are mild forms of a generator trip.

R.2.1.1 Turbine Control Valve Fast Closure

All automatic generator trips initiate turbine stop valve closure for which transient analyses are discussed in Section R.2.1.2. Control valve fast closure is provided as a backup. Turbine control valve fast closure would occur without stop valve closure only when the main generator breakers are manually opened (operator error) or in the event of a failure which prevents automatic (relay) tripping. These conditions produce the following transient sequence:

1. Turbine generator acceleration relay (in the Hydraulic Control System) is actuated to initiate turbine control valve fast (about 0.20 sec) closure.

2. Actuation of the acceleration relay is sensed from pressure switches by the Reactor Protection System (RPS), which initiates a scram (for initial power levels above 45 percent).

3. The turbine bypass valves are opened simultaneously with turbine control valve closure.

4. Reactor vessel pressure rises to the relief valve setpoints, causing them to open for a short period. The steam passed by the relief valves is discharged into the suppression pool.

5. The Turbine Bypass System controls nuclear system pressure after the relief valves close.

Figure R.2-1 shows the transient from 1,998 MWt conditions, which is the worst case. Neutron flux and fuel surface heat flux do not rise above their nominal values.

Although below 45 percent power, the acceleration relay scram is disabled, high flux or high pressure scram is adequate to protect the reactor.

Below 25 percent power the full amount of generated steam is passed to the condenser and no scram occurs.

R.2.1.2 Turbine Trip (Turbine Stop Valve Closure)

A variety of turbine or reactor system malfunctions can initiate a turbine stop valve closure, normally called a turbine trip. This event represents the fastest possible steam flow shutoff and therefore, the potential for the most severe pressure-induced transient. The most serious transient occurs when the reactor is operating at 1,998 MWt, and is shown on Figure R.2-2.

The sequence of events for a turbine trip is very similar to that for a generator load rejection. However, the valve closure is faster, occurring in about 0.1 sec. Position switches mounted on the stop valves provide the means of sensing the trip and initiating immediate reactor scram. The bypass valves are also opened immediately. The relief valves open for a short time to help relieve the pressure. The fuel thermal transient is mild, never exceeding the initial fuel surface heat flux.

Peak neutron flux is held to 106 percent by the fast action of the stop valve scram. The peak pressure in the vessel dome is 1,128 psia, and at the location of the safety valves it is 1,126 psia, well below their setpoint of 1,255 psia.

Turbine trips from lower initial power levels decrease in severity; below 45 percent, the trip scram is not initiated because flux and pressure scram are adequate to protect the reactor.

R.2.1.2.1 Loss of Condenser Vacuum

If the main condenser vacuum is lost while the station is operating at 1,998 MWt the following trips would occur:

Alarm at	15-27 in Hg vacuum
Turbine stop valve closure scram at	22.4 in Hg vacuum
Turbine bypass valve closure at	7 in Hg vacuum

The worst case is if the vacuum should be lost instantaneously. In this case, the transient becomes identical to a turbine trip with bypass failure, and more severe than a full load rejection with bypass failure because the turbine stop valve closure is faster than the turbine control valve closure. Slower losses of condenser vacuum will produce less severe transients because the scram on loss of condenser vacuum will precede the stop valve closure and some bypass flow will be permitted.

Figure R.2-3 presents the instantaneous loss of vacuum transient from 1,998 MWt conditions. Because this is the most severe reactor isolation, it is used to select the dual purpose relief/safety valve capacity sufficient to remove enough energy from the reactor in order to prevent the primary system spring safety valves from lifting. Peak neutron flux reaches 107 percent of nominal, however the fuel surface heat flux does not exceed its initial value and the minimum critical heat flux ratio (MCHFR) remains unchanged. Therefore, no damage to the fuel results from this transient. The relief valves open fully to limit the pressure rise to 1,176 psia in the vessel dome and at the valve location. This is considerably below the 1,255 psia point of the safety valves, demonstrating the ability of the design relief valve capacity of 30 percent of 1,998 MWt steam flow to prevent spring safety valve actuation.

R.2.1.3 Main Steam Line Isolation Valve Closure

Automatic circuitry or operator action can initiate closure of the main steam isolation valves. Position switches on the valves provide reactor scram if the valves are closed and reactor pressure is above 600 psig. However, Protection System logic does permit the test closure of one valve without initiating scram from the position switches. Inadvertent closure of one or all of the isolation valves from reactor scrammed conditions (such as operating states C or E) will produce no significant transient. Closures during reactor heatup (operating state D) will be less severe than the maximum power cases (maximum stored and decay heat) which follow.

R.2.1.3.1 Closure of All Main Steam Line Isolation Valves

Figure R.2-4 shows the changes in important nuclear system variables for the simultaneous isolation of all main steam lines while the reactor is operating at 1,998 MWt conditions. Reactor scram is initiated by the isolation valve position switches before the valves have traveled more than 10 percent from the open position. A 3 sec nonlinear valve closure was simulated, which is the fastest closure attainable. Scram is initiated before any significant flow interruption occurs; therefore, no neutron flux or fuel surface heat flux peaks occur. No reduction in MCHFR occurs. The nuclear system relief valves begin to open when pressure reaches the lowest set point (1,090 psig for design conditions) at about 4 sec after the start of the isolation. They close sequentially as the stored heat is dissipated and will continue to intermittently discharge the decay heat. The peak pressure in the main steam line near the spring set point safety valves is 1,116 psig, well below their lowest set point (1,240 psig).

R.2.1.3.2 Closure of One Main Steam Line Isolation Valve

Closure of one main steam isolation valve is desirable for testing purposes. Therefore, the protection system logic from the valve position switches permits full shutoff of any steam line without initiating scram. Normal procedures for such a test will require an initial power reduction to about 80-90 percent of design conditions in order to avoid high flux or pressure scram. Figure R.2-5 graphically shows the changes of important nuclear system variables during the simulated 3 sec closure of one isolation valve from design power conditions. The steam flow disturbance raises vessel pressure and reactor power causing a high neutron flux scram. The peak surface heat flux is 102.5 percent of initial conditions and peak center fuel temperature increased 49°F; however, MCHFR remained above 1.8 showing that no fuel damage occurs. Peak pressures remain below the setting of the lowest relief valves.

R.2.2 Events Resulting in a Reactor Vessel Water Temperature Decrease

Events that result directly in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the vessel or reduce the temperature of water being delivered to the vessel. The events that result in the most severe transients in this category are the following:

1. Loss of feedwater heater

2. Shutdown cooling Residual Heat Removal System (RHR) malfunction-decreasing temperature

3. Inadvertent pump start

R.2.2.1 *Loss of a Feedwater Heater*

Figure R.2-6 shows the response of the nuclear system to the loss of 100°F of the feedwater heating capability. This represents the maximum expected single heater or group of heaters which can be tripped or bypassed by a single event. The reactor was assumed to be at maximum power conditions on automatic flow control when the heater was lost. Note that the Flow Control System responds to the power

increase by reducing core flow so that steam flow from the vessel to the turbine remains essentially constant throughout the transient. Neutron flux increases above the initial value in order to produce the same steam flow with the higher inlet subcooling. The reactor settles out at 111 percent power and with core flow reduced to about 94 percent. The fuel clad barrier is not threatened by these conditions; fuel surface heat flux reaches 110 percent but the increase core inlet subcooling helps keep the lowest MCHFR above 1.5. Therefore, no clad damage occurred.

This transient is less severe from lower power levels for two main reasons; lower initial power levels will have initial MCHFR values greater than the 1.9 limiting value assumed here, and the magnitude of the power rise decreases with the initial power condition. Therefore, transients from other reactor operating states or lower power levels within operating state F will be less severe.

R.2.2.2 Shutdown Cooling (Residual Heat Removal System) Malfunction-Decreasing Temperature

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHRS heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the reactor were critical or near critical (operating states B or D), a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron flux reactor scram from the intermediate range monitor channels would terminate the transient without fuel damage and without any measurable nuclear system pressure increase.

R.2.2.3 Inadvertent Pump Start

Several systems are available for providing high pressure supplies of cold water to the vessel for normal or emergency functions. The Control Rod Drive System and the Makeup Water System, normally in operation, can be postulated to fail in the high flow direction introducing the possibility of increased power due to higher core inlet subcooling. The same type of transient would be produced by inadvertent startup of either the Reactor Core Isolation Cooling (RCIC) System or the High Pressure Coolant Injection (HPCI) System. In all of these cases, the normal feedwater flow would be correspondingly reduced by the water level controls. The net result is simply a replacement of a portion of the 365°F feedwater flow (at design power operations) by flow having a temperature of approximately 100°F.

The severity of the resulting transient is highest for the largest of these abnormal events, the inadvertent startup of the HPCI System.

Since the startup of the steam turbine driven pump takes approximately 25 sec, the transient which occurs is very similar to the loss of feedwater heater transient given above. As in that case,

the most threatening transient would occur where minimum initial fuel thermal margins exist (maximum power within reactor operating state F). The HPCI startup transient is clearly less severe than the loss of feedwater heater case because its effect on mixed feedwater temperature will produce a smaller temperature change than the 100°F feedwater temperature change previously analyzed. For this reason, no fuel clad barrier damage will result for the malfunction or inadvertent startup of any of these auxiliary cold water supply systems.

R.2.3 Events Resulting in a Positive Reactivity Insertion

Events that result directly in positive reactivity insertions are the result of rod withdrawal errors and errors during refueling operations. The following events result in a positive reactivity insertion:

1. Continuous rod withdrawal during power range operation
2. Continuous rod withdrawal during reactor startup
3. Control rod removal error during refueling
4. Control curtain removal error during refueling
5. Fuel assembly insertion error during refueling

R.2.3.1 Control Rod Withdrawal Errors - General

Design features provided to minimize the probability of inadvertent continuous control rod withdrawals and to limit potential power transients in the event they should occur include the following:

1. Normal control rod operation is a step (notch) at a time. Two control switches must be held open at the same time to withdraw a control rod continuously.
2. The continuous control rod withdrawal rate is limited by the Control Rod Drive System Hydraulic Control System to 3 in/sec.
3. Interlocks prevent control rod withdrawal if the Nuclear Instrumentation System monitors are not in a condition to provide the required protection, or if the control rod withdrawal timing relay should fail.
4. Preplanned withdrawal patterns and procedural controls are used to prevent abnormal configurations giving high control rod worths.
5. A control rod worth minimizer will back up operating procedures in maintaining control rod patterns which minimize individual control rod worths.

6. Intermediate and power level scrams limit power excursions during operation at low power levels. During power operation, the Incore Nuclear Instrumentation System monitors alarm to warn the reactor operator if local neutron flux levels approach preset limits. When the bulk power reaches preset limits, an incore monitor averaging circuit blocks control rod withdrawal and initiates reactor scram.

7. Control Rod Position Indicating System

R.2.3.2 Continuous Rod Withdrawal During Power Range Operation

Control rod withdrawal errors resulting from single operator errors or from single equipment failures are considered over the entire power range. Examples of single rod withdrawal errors are as follows:

1. An increase in average core power above the normal operating power - flow limit resulting from excessive rod withdrawal while using normal rod withdrawal sequences.

2. An increase in average core power and core power peaking factor resulting from the complete withdrawal of any rod from the expected rod pattern.

3. An increase in average core power and core power peaking factors resulting from erroneous withdrawal of a single control rod from an abnormal withdrawal sequence.

Rod withdrawal errors of type 1 above could result in an average core power level which exceeds the normal allowable power level for the existing core flow condition; however, core power distributions would remain normal. Operator errors resulting in this type of rod withdrawals from an initial full power condition would be expected to terminate upon actuation of low power range monitor (LPRM) or average power range monitor (APRM) high flux alarm signals. These six APRM channels would independently block control rod withdrawal if average flux level reaches 108 percent of the full power flux level under rated core flow conditions. The APRM control rod block action occurs at reduced average flux levels if core flow is reduced. The APRM control rod block action occurs before MCHFR decreases to 1.5 for control rod withdrawal errors of type 1 initiated within the power operating range.

Rod withdrawal errors of type 2 above could result in an increase in both core average power level and core power peaking factors. The increase in core average power is limited by the reactivity worth of the single withdrawn rod.

The complete withdrawal of the central control rod from an expected rod pattern was analyzed as representative of this type of rod withdrawal error. The most reactive condition of the fuel was assumed (no xenon or samarium poisoning and no fuel burnup) to maximize the reactivity worth of the central control rod in the

assumed rod pattern. The reactor was assumed to be at full power with an initial core power peaking factor of 2.13 and an initial MCHFR of 2.32. The complete withdrawal of the center control rod increased the core power peaking factor to a maximum value of 2.42. The average core power level increased to approximately 105 percent and MCHFR decreased to about 1.13 for this case.

Rod withdrawal errors of type 3 above could result in an increase in average core power combined with an increase in core power peaking factor. The abnormal control rod patterns considered are those which might be established during normal operation because of the following special operations:

1. Interchange of normal control rod patterns. The control rod patterns are periodically interchanged during normal operations to obtain more uniform fuel burnup throughout the core loading.

2. Establishment of special control rod patterns as an aid for identifying core regions having failed fuel assemblies.

3. Establishment of special control rod patterns resulting from Control Rod Drive System malfunctions.

Special operations, such as those above, which could result in abnormal control rod patterns are permitted only after detailed written operating procedures have been prepared and then reviewed and approved by the Station Operations Review Committee. The operating power level will be limited during such special operations so that MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of a single operable control rod from the approved abnormal rod pattern. The actual performance of special operations such as 1 or 2 above will be directly supervised by Station Supervisory Personnel even though controlled by approved written procedures. Normal operating limitations on local flux level (LPRM) indications are observed and any manual rod withdrawal terminated upon actuation of an LPRM high flux alarm. The extensive procedural and administrative controls provide the principal protection against fuel damage resulting from rod withdrawal errors associated with abnormal control rod patterns. These controls are supplemented by the Rod Block Monitor System which will block manual rod withdrawal when LPRM detectors indicate local high flux conditions.

R.2.3.3 Continuous Rod Withdrawal During Reactor Startup

Control rod withdrawal errors are considered when the reactor is at power levels below the power range. The most severe case occurs when the reactor is just critical at room temperature and an out of sequence rod is continuously withdrawn. The rod worth minimizer would normally prevent withdrawal of such a rod. In the intermediate and power range instrumentation, high neutron flux scrams are located at the top of each instrument range. Interlocks prevent control rod

withdrawal if the instruments are not reading on scale or if the Intermediate Range Monitor System (IRMS) is not set on the lowest range.

The scaling arrangement of the intermediate range monitors (IRM) is such that a scram signal is generated before the detected neutron flux has increased by more than a factor of twenty assuming that the IRM channels are in the worst conditions of allowed bypass. In addition, a high neutron flux scram is generated by the APRMs at 120 percent of rated power.

The analysis was performed for a 2.5 percent control rod withdrawal at the maximum rod drive speed of 3 in/sec starting from an average moderator temperature of 82°F and assuming no intermediate level scram. The excursion was assumed turned only by the inherent negative Doppler reactivity effect from fuel heating. Only then was the reactor scram, at over 120 percent of full power, assumed to be effective. In actuality, the reactor would have been scrammed at a much lower power level.

The results of these analyses indicate a maximum fuel temperature of 2,650°F, well below the melting point of UO_2 and a maximum fuel clad temperature of 460°F which is less than the normal operating temperature of the clad. The possible failure of the fuel clad due to strain was analyzed using the following conservative assumptions:

1. The total volume expansion of UO_2 is in the radial direction.

2. There is no thermal expansion of the fuel cladding.

3. The fuel is assumed to be incompressible.

The results of these analyses indicate a maximum radial strain of 0.6 mils which is much less than the postulated cladding damage limit of approximately one percent plastic strain which corresponds to approximately 3.3 mils radial expansion. Thus, no fuel damage will occur due to a continuous rod withdrawal during reactor startup.

R.2.3.4 Control Rod Removal Error During Refueling

The nuclear characteristics of the core assure that the reactor is subcritical even in its most reactive condition with the most reactive control rod fully withdrawn during refueling.

When the mode switch is in refuel, only one control rod can be withdrawn. Selection of a second rod initiates a rod block, thereby preventing the withdrawal of more than one rod at a time.

Therefore, the Refueling Interlocks will prevent any condition which could lead to inadvertent criticality due to a control rod withdrawal error (using the Control Rod Drive Hydraulic System) during refueling.

In addition, the design of the control rod, incorporating the velocity limiter, physically prohibits the upward removal of the control rod without the simultaneous or prior removal ~~of the control rod without the simultaneous or prior removal~~ of the four adjacent fuel bundles thus eliminating any hazardous condition due to rod removal from above the core.

R.2.3.5 Control Curtain Removal Error During Refueling

The mechanical design of the control curtains is such that they cannot be removed without the prior removal of the adjacent fuel bundles. Thus no hazardous condition can result from the erroneous removal of the control curtains.

If a fuel bundle should be inserted without the adjacent control curtain being in its proper location, the error would be detected by the reduced shutdown margin indication during the control rod functional check performed after inserting the four fuel bundles adjacent to the control rod. Procedural restrictions also require verification of proper control curtain installation prior to fuel bundle insertion.

R.2.3.6 Fuel Assembly Insertion Error During Refueling

The core is designed such that it can be made subcritical under the most reactive conditions with the strongest control rod fully withdrawn. Therefore, any single fuel bundle can be positioned in any available location without violating the shutdown criteria providing all of the control rods are fully inserted. The Refueling Interlocks, backed up by procedural restrictions, require that all control rods must be fully inserted before a fuel bundle may be inserted into the core.

R.2.4 Events Resulting in a Reactor Vessel Coolant Inventory Decrease

Events that result directly in a decrease of reactor vessel coolant inventory are those that either restrict the normal flow of fluid into the vessel or increase the removal of fluid from the vessel. Five events are identified as causing the most severe transients in this category:

1. Pressure regulator failure

2. Inadvertent opening of a relief or safety valve

3. Loss of feedwater flow

4. Loss of offsite power to station auxiliaries

5. Opening of turbine bypass valves

R.2.4.1 Pressure Regulator Failure

The turbine pressure regulator can be assumed to fail in either of two ways; calling for zero output or maximum output.

In the first case, the backup regulator will take over control of the turbine control valves as soon as the failed regulator attempts to close the valves and pressure begins to rise. The transient is similar to a pressure setpoint increase as shown in Section 7.17, Nuclear System Stability Analysis.

If either regulator fails in a wide open direction, the maximum control plus bypass valve demand is limited by the Turbine Control System to 110 percent of the steam flow at 1,998 MWt. Figure R.2-7 shows the transient which occurs when this malfunction occurs at 1,998 MWt. Vessel pressure drops 80 psi in the first 12 sec. Neutron flux is decreased significantly as the pressure drop increases the void content of the core. When the steam line pressure at the turbine throttle valves decreases 100 psi (at about 12 sec) closure of the main steam isolation valves is initiated. Scram occurs when the isolation valves have closed from the 100 percent to 90 percent open position. The depressurization is stopped as soon as the isolation becomes effective and the reactor is shut down with pressure rising. Eventually the relief valves operate to limit the pressure rise.

R.2.4.2 Inadvertent Opening of a Relief Valve

The opening of a relief valve allows steam to be discharged into the primary containment. The sudden increase in the rate of steam flow leaving the reactor vessel causes the reactor vessel coolant (mass) inventory to decrease.

The result is a mild depressurization transient. Figure R.2-8 shows the transient resulting from the opening of a relief valve with the capacity to pass 10 percent of rated nuclear system steam flow. An initial power level corresponding to 1,998 MWt conditions is assumed.

The pressure regulator senses the nuclear system pressure decrease and closes the turbine control valves far enough to maintain constant turbine throttle pressure. Reactor power settles out at nearly the initial power level. Automatic recirculation flow control (assumed to be active) increases recirculation flow to the maximum. Because the recirculation flow cannot satisfy the additional load demand, the pressure regulator setpoint is automatically reduced to its lower limit, and nuclear system pressure decreases. No reduction in MCHFR occurs (MCHFR 1.9), and no fuel damage results from the transient. Because pressure decreases throughout the transient, the nuclear system process barrier is not threatened by high internal pressure. The small amounts of radioactivity discharged with the steam are normally contained inside the primary containment; the situation is not significantly different, from a radiological viewpoint, than that normally encountered in cooling the station using the relief valves to remove decay heat.

R.2.4.3 Loss of Feedwater Flow

The transient response of the station to a feedwater controller malfunction-demanding closure of the feedwater control valves is shown on Figure R.2-9. The initial power level is 1,998 MWt. The valves are closed at their maximum rate of 25 percent/sec. The station response to simultaneous tripping or accidental loss of all feedwater pumps is very similar to this transient. The station response to loss of one feedwater pump is described in Section R.2.4.3.1.

The reactor water level decreases rapidly due to the mismatch between the steam flow out of the vessel and the shutoff feedwater flow. Low water level scram occurs after about 7 sec. The recirculation flow motor generator sets are automatically run down to 20 percent speed demand when the feedwater flow drops below 20 percent (this interlock is used to protect the recirculation drive pumps from steady state net positive suction head problems). The decrease in moderator subcooling slightly decreases the neutron flux until scram occurs on low level and completely shuts down the reactor. Vessel steam flow closely follows the decay of fuel surface heat flux. The analysis of the transient was discontinued after about 25 sec since the model is not programmed to handle the situation when core inlet subcooling becomes negative. The MCHFR remains above 1.5 throughout the transient. Subsequent events would be complete drive motor trip, main steam isolation valve closure, and initiation of the RCICS and HPCIS, all occurring when the water level drops to the low-low level setpoint, 48 in below the bottom of the separator skirt. The time when this will occur, estimating from the established rate of level decrease, is about 43 sec. Pressure will then rise following the isolation.

Water inventory loss from the vessel from the time the transient calculations were discontinued until the time the transient isolation valves would be closed (approximately 46 sec) was conservatively estimated to be less that 352 ft^3 of saturated water. Accounting for this conservative inventory loss and assuming the recirculation pumps would be tripped off upon isolation, an estimate of the final water level was made. All steam existing as carry under and as voids in the core, upper plenum, standpipes, and separators at the time of discontinuation was allowed to condense, and the volume of water discharged to the scram dump tanks was removed from the vessel. Neglecting any inventory makeup from auxiliary cooling systems, the calculations made showed that greater than 4.5 ft of water would remain above the active core.

R.2.4.3.1 Loss of One Feedwater Pump

The station is designed to withstand the loss of one of its three electrically driven feedwater pumps without experiencing a low water level scram. Figure R.2-10 depicts the transient if the loss occurs while the station is operating at 1,998 MWt.

A loss of one pump will result in a run out of the two remaining pumps given a final feedflow of 77 percent of that at normal 1,998 MWt conditions. Sensed vessel water level falls until it reaches the low level alarm (22 1/2 in above the bottom of the separator skirts) at about 7 sec. The low level alarm signal and a signal indicating the pump loss initiate a runback of the recirculation pumps to about 50 percent speed. This is to reduce the power level to within the capacity of the remaining feed pumps and eliminate the steamflow/feedflow mismatch. Level continues to fall until nearly 12 sec, reaching a lowest level of 18 in above the skirts. Then, after the power reduction becomes effective, it begins to increase. Since scram occurs when sensed water level is 9 in above the skirts, ample margin has been demonstrated. At no time during the transient did the surface heat flux exceed 100 percent, and the station will ultimately settle out at a power corresponding to the recirculation flow which pertains at the reduced pump speed.

R.2.4.4 Loss of All Offsite Power to Station Auxiliaries

The basic function of the auxiliary electrical power system is to provide for station auxiliaries during startup, operation, and shutdown and to provide highly reliable power sources for station loads which are important to its safety. Auxiliary power is supplied during normal power operation by the unit auxiliary transformer with the startup transformer for backup. This transient analysis assumes that both the unit auxiliary transformer and the startup transformer are lost simultaneously. The transmission system and generator are assumed to initially remain in operation. The station is initially at 1,998 MWt conditions and in the manual recirculation flow control mode. Refer to Figure R.2-11.

The loss of all offsite power to auxiliaries causes an immediate trip of both recirculation motor generator set drive motors and a trip of all the feedwater pump motors. The coast down of the recirculation pumps is as described in Section R.2.5.2 and a coast down time of 8 sec was assumed for the feedwater pumps. The initial portion of the transient is thus similar to a two-pump trip and the MCHFR is nearly identical to the value obtained for that transient. The power loss causes the RPS motor generator sets to coast down, and after about 5 sec the voltage has decayed sufficiently to permit the relays on the RPS to drop out. This initiates reactor scram and closure of the isolation valves. A 3 sec isolation was assumed. It is expected that scram due to loss of condenser vacuum would also occur about 5 sec after the power loss. At no time will the loss of auxiliary power prevent scram, because stored pneumatic energy and reactor pressure are the means of driving in the control rods. Following the scram, the reactor power and water level drop quickly.

It is expected that closure of the turbine main stop valves due to loss of condenser vacuum will occur after isolation valve closure because the coast down of the main condenser cooling water will be offset by the decreasing turbine steam flow which follows the decreasing reactor power. Nevertheless, this transient assumes the

turbine stop valves close 6 seconds after the loss of power, which is pessimistic because at this time the isolation valves have only closed about 26 percent.

The rapid closure of the turbine stop valves (0.1 second) results in the bypass valves opening by 6 seconds. Thus the vessel steam flow dips momentarily, then increases; however, a rapid decrease of flow ensues, because the isolation valve closure, begun at 5 seconds, is now nearly complete. By 8 seconds the condenser vacuum has decreased to 7 inches resulting in closure of the bypass valves. At this time, also, the isolation valves have fully closed. Thus, by 8 seconds after the loss of auxiliary power the flow of steam from the pressure vessel will be completely stopped. Pressure rises rapidly to the pressure relief valve setpoint, 1,105 psia. The reactor water level initially rises due to the swell caused by the recirculation pump trip, but the feedwater flow drops to zero by 5 seconds and the level rise terminates. The scram brings about a subsequent rapid decrease in level; the lowest sensed level of 37 inches below the separator skirts occurs 40 seconds after the power loss. The RCIC and HPCI Systems would not actuate unless the sensed level reached 4 ft below the separator skirts. At the lowest point, nearly 7 ft of water remains above the core, so the level transient is not as severe as that resulting from the loss of all feedwater pumps alone. Figure R.2-12 shows the calculated long term vessel water level transient conservatively considering RCIC operation only.

R.2.4.5 Opening of Turbine Bypass Valves

An assumed transient in which a control malfunction results in simultaneous full opening of all steam bypass valves was analyzed at each of the following operating levels:

1. 25 percent thermal power with 35 percent core flow
2. 30 percent thermal power with 35 percent core flow
3. 50 percent thermal power with 100 percent core flow
4. 75 percent thermal power with 100 percent core flow
5. 1,998 MW (100 percent thermal power with 100 percent flow)
6. Hot standby

As is indicated by the attached transients (see Figures R.2-13 through R.2-18) the reactor will scram only at low power levels for the malfunction analyzed. This is predominantly due to the low 25 percent bypass limit specified for this plant. For powers higher than approximately 30 percent, the pressure regulator control will adjust vessel steam flow to its initial value avoiding a scram.

Figure R.2-13 shows the transients for the 25 percent power and 35 percent flow case. Here bypass flow was continuously larger than steam line flow even though the pressure regulator attempted to correct the malfunction. The turbine control valve was completely closed by the regulator in approximately 2 seconds. Reactor vessel blowdown through the bypass valves then occurred, and at approximately 47 seconds the low steam line pressure set point (895 psia)

was reached, effecting main steam line isolating valve closure and stream.

Figure R.2-14 shows the transients for the 30 percent power and 35 percent flow case. As is shown on Figure R.2-14 pressure regulation drives turbine steam flow to zero in approximately 3.5 sec during the transient; however, steam line flow is large enough to cause turbine steam to flow again at a low value, thereby avoiding reactor vessel blowdown and subsequent scram.

The transients for the 50 percent power and 100 percent flow case are shown on Figure R.2-15. Steam line flow is sufficiently large in this case for the pressure regulator to keep turbine steam flowing during the entire transient, more definitely avoiding a blowdown and scram.

In the next two cases, namely 75 percent power (Figure R.2-16) and 1,998 MWt power (Figure R.2-17), steam line flow is still larger resulting in a larger margin for pressure regulation and the avoidance of blowdown and scram.

Since some of the cases considered had core flows larger than 65 percent, an automatic flow mode was permissible. However, this analysis assumed manual flow for simplification. For the 100 percent core flow cases, had the automatic mode been considered, pressures would have been lower by approximately 30 psia due to the pressure setpoint adjuster attempting to raise turbine steam flow to satisfy load demand. Steam line pressure, however, would still not have decreased to the low pressure set point to cause isolation valve trip scram.

The Pilgrim Nuclear Power Station has a 25 percent bypass system and should it spuriously fully open when the reactor is being held in the hot standby condition, the level rise associated with the depressurization would give a high level trip of the main steam line isolation valves approximately 5 sec after the event.

The only power being generated when the reactor is at hot standby is decay heat. The active core remains covered by reactor liquid and thus the fuel is at all times adequately cooled.

R.2.5 Events Resulting in a Core Coolant Flow Decrease

Events that result directly in a core coolant flow decrease are those that affect the Reactor Recirculation System. Transients beginning from operating state F are the most severe since only in this state do power levels approach fuel thermal limits. The following events result in the most significant transients in this category:

1. Recirculation flow control failure decreasing flow

2. Trip of one recirculation pump

3. Trip of two recirculation pumps

4. Recirculation pump seizure

R.2.5.1 Recirculation Flow Control Failure, Decreasing Flow

A failure in one of the motor generator set speed controllers could cause the scoop tube positioner of the fluid coupler to move at its maximum speed in the direction of decreasing pump speed and flow. In the transient analyzed (Figure R.2-19), the failed speed controller moved its positioner down to zero coupling at about 20 percent/sec. The resulting transient is similar to a one pump trip; however the recirculation flow and thermal power decay is less severe than that which results from tripping one of the recirculation drive motors. Therefore, the MCHFR during this transient is higher than for the one pump trip.

A malfunction of the master flow controller giving zero speed demand would produce a less severe transient even though it controls both speed controllers because each speed control loop is rate limited, which would produce a slower flow cutback than an individual speed controller failure.

R.2.5.2 Trip of Two Recirculation Pumps

The two loop trip provides an evaluation of the thermal margins provided by the rotating inertia of the recirculation drive equipment. The decrease in flow causes additional void formation in the core which decreases reactor power. The time constants of the fuel cause the surface heat flux to lag behind the flow decay, and the mismatch between reactor thermal power and recirculation flow brings about a decrease in the critical heat flux ratio of the reactor.

Using the rotating inertia available from the recirculation drive equipment, the MCHFR was analyzed by employing a power distribution which gives an initial MCHFR of 1.9. The lowest MCHFR of 1.47 was found to occur at about 2 sec after the trip. This shows that adequate inertia has been provided in the recirculation drive equipment. See Figure R.2-20.

R.2.5.3 Trip of One Recirculation Pump

The results of this transient from full power and flow are obviously less severe than the trip of both drive motors; therefore, the thermal margins during this transient are greater. Figure R.2-21 shows the expected transient. The lowest MCHFR of 1.69 occurs at about 1.58 sec after the trip. Flow increases in the live loop, finally providing about 69 percent of the initial flow coming through the active loop jet pump diffusers. This flow splits in the lower plenum with about 85 percent of it going through the core providing 59 percent of the initial core flow and the remainder providing reverse flow up the jet pumps of the tripped loop. A small amount of

forward flow will still be induced in the tripped drive loop due to the static pressure difference between the downcomer and jet pump throat.

R.2.5.4 Recirculation Pump Seizure

The transient response of the station to an instantaneous stoppage (seizure) of one recirculation pump shaft from 1,998 MWt conditions is shown on Figure R.2-22. This case represents the most rapid decrease of flow in a single drive loop. Jet pump diffuser flow in this loop reverses at about 0.8 sec after the seizure. The steady state flow pattern is similar to the one pump trip, but it is reached more quickly. The MCHFR for this transient analysis remained above 1.0 and occurred about 1 sec after the seizure.

R.2.6 Events Resulting in a Core Coolant Flow Increase

Events that result directly in a core coolant flow increase are those that affect the Reactor Recirculation System. The following events result in the most significant transients in this category:

1. Recirculation flow control failure increasing flow

2. Startup of idle recirculation pump

R.2.6.1 Recirculation Flow Control Failure - Increasing Flow

The most severe case is the failure of one of the motor generator set speed controllers. It is capable of increasing its loop flow faster than a master flow controller failure could change flow because of the speed control loop rate limits. Response of the station to this transient is shown on Figure R.2-23. The initial conditions were 60 percent power 40 percent flow, which would give the most severe transient. Both flow control couplers are initially at about 15 percent position for the assumed initial conditions. The operative coupler remains near this position, while the failed loop coupler reaches full stroke by 5.5 sec. The failed speed controller moves its positioner to full coupling at about 20 percent/sec. Diffuser flow in the failed loop quickly increases. It reaches 140 percent by 6 sec and levels out. Diffuser flow in the opposite jet pumps is decreased by the greater core P and actually reverses at about 4.5 sec. Core inlet flow increases rapidly to about 70 percent causing neutron flux to increase and scram the reactor at 3.6 sec. The neutron flux reached over 400 percent. However, peak fuel surface heat flux was only 82 percent and MCHFR remained above 1.9.

R.2.6.2 Startup of an Idle Recirculation Pump

The transient response of the station to the starting of an idle recirculation loop without warming the drive loop water is shown on Figure R.2-24. The initial conditions are:

1. One drive loop is shut down and contains cold water (100°F).

2. The active recirculation pump is operating at a speed producing about 80 percent of rated drive flow in the 10 active jet pumps.

3. The core is receiving 43 percent of its normal flow, while the remainder of the flow is reversed up the 10 active jet pumps.

4. Reactor power is 70 percent of 1,998 MWt, the most severe imaginable condition for startup of the dead loop because it is the rod block power corresponding to 43 percent core flow.

5. The drive pump suction and discharge bypass valves are removed, cut, and capped.

6. The fluid coupler scoop tube in the dead recirculation loop is at a position giving approximately 46 percent generator speed demand, which is about that required to break the pump away.

The startup transient sequence is:

1. The drive motor breaker is closed at t=0

2. The drive motor reaches near synchronous speed quickly, while the generator approaches its top speed attainable corresponding to the initial coupler position within about 5 sec.

3. At 5 sec, the generator field breaker is automatically closed, loading the generator and starting the pump. Generator speed is drawn down as it tries to free the stopped rotor of the pump. Pump acceleration and then controlled speed is shown on Figure R.2-24. The coupler demand is automatically programmed back to 20 percent speed.

4. The pump discharge valve is manually started open as soon as the drive motor breaker is closed. (Normal procedure would delay valve opening to separate the two portions of the flow transient and make sure the drive loop is properly mixed with vessel-temperature water). A nonlinear 30 sec valve opening characteristic was used.

Neutron flux shows a fairly sharp peak (107 percent) shortly after the actual pump excitation due to the peak incore inlet flow (about 47 percent). Core flow subsequently increases slowly to its final value of 48 percent. Throughout the transient, diffuser flow in the startup loop jet pumps is either reversed or less than about 10 percent of rated. For this reason, the cold water does not significantly effect the reactor.

Peak fuel surface heat flux of 78 percent occurs at about 11.5 sec which corresponds to a MCHFR of 1.48.

R.2.7 Event Resulting in a Core Coolant Temperature Increase

An event which can cause directly a reactor vessel water temperature increase is one in which hotter water is returned to the reactor vessel without changing the coolant flow rate. This event is loss of Residual Heat Removal (RHR) shutdown cooling.

R.2.7.1 Loss of Residual Heat Removal Shutdown Cooling

The loss of RHR shutdown cooling can only occur during the low pressure portion of a normal reactor shutdown and cooldown. At this time the RHRS is operating in the shutdown cooling mode which occurs only in states A, B, C, and D.

For most single failures which could result in loss of shutdown cooling no unique safety actions are required; in these cases shutdown cooling is simply reestablished using other, normal shutdown cooling equipment. In the cases where the RHRS shutdown cooling suction line becomes inoperative a unique requirement for cooling arises. In states A and B, in which the reactor vessel head is off, either half of the RHRS LPCI mode can be used to maintain water level to assure continued core cooling. In states C and D, in which the reactor vessel head is on and the system can be pressurized, the low pressure cooling systems, relief valves (manually operated), and RHRS suppression pool cooling mode can be used to maintain water level and remove decay heat.

R.2.8 Event Resulting in Excess of Coolant Inventory

An event which can cause directly an excess of coolant inventory is one in which makeup water flow is increased without changing other core parameters. This event is feedwater controller failure-maximum demand.

R.2.8.1 Feedwater Controller Failure - Maximum Demand

The transient response of the station to a maximum feedwater flow demand of 110 percent of 1,998 MWt feed flow by the feedwater controller is shown on Figure R.2-25. The transient was initiated from the analytical lower limit of the automatic flow control range with reactor power at 65 percent of 1,998 MWt and recirculation flow at 47 percent. The low initial power level results in a more severe steam/feed flow mismatch and level transient.

The maximum rate of level rise is about 2 in/sec. Cooler core inlet flow produces an increase in reactor power up to approximately 76 percent before the sensed water level reaches the high level turbine trip at 10 sec. Closure of the turbine stop valves produces a simultaneous shutoff of turbine steam flow, opening of the Steam Bypass System, and reactor scram. Since the bypass is slightly slower than the stop valve closure and of 25 percent capacity, pressure rise is produced. The stop valve closure scram limits the peak neutron flux to 84 percent of 1,998 MWt level. Peak surface

heat flux equals 70 percent at 10.1 sec. However, inlet subcooling is over 45 Btu/lb preventing any significant decrease in thermal margins.

Following a dip during the scram, level continues to rise as the high feedwater flow continues to fill the vessel until the pumps can be tripped or switched to manual control. The Bypass System and scram limit the peak vessel dome pressure 1,091 psia; the bypass valves begin closing shortly after they are fully open to take over pressure control.

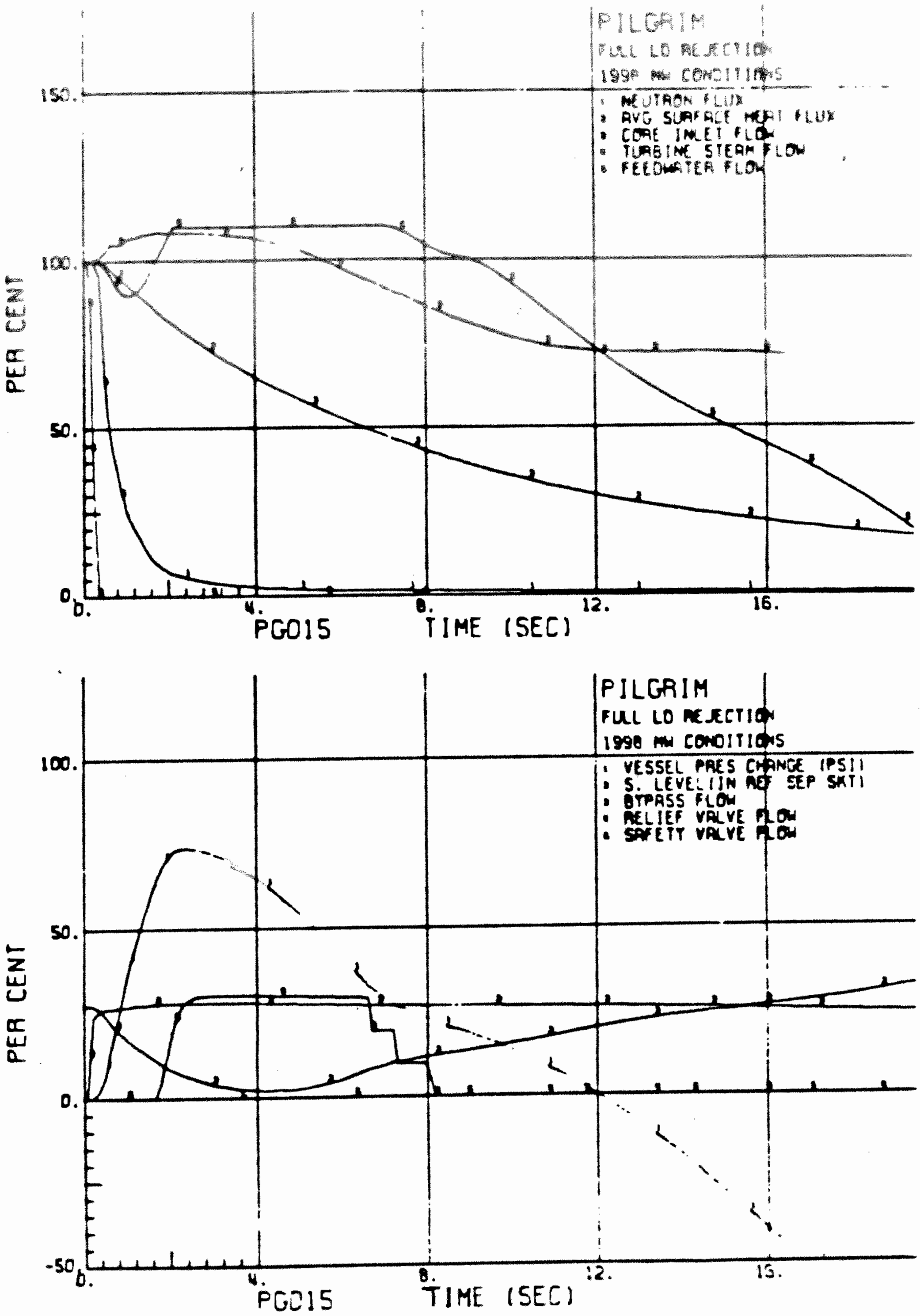


Figure R.2-1. Initial Core Full Load Rejection Pilgrim Nuclear Power Station Final Safety Analysis Report

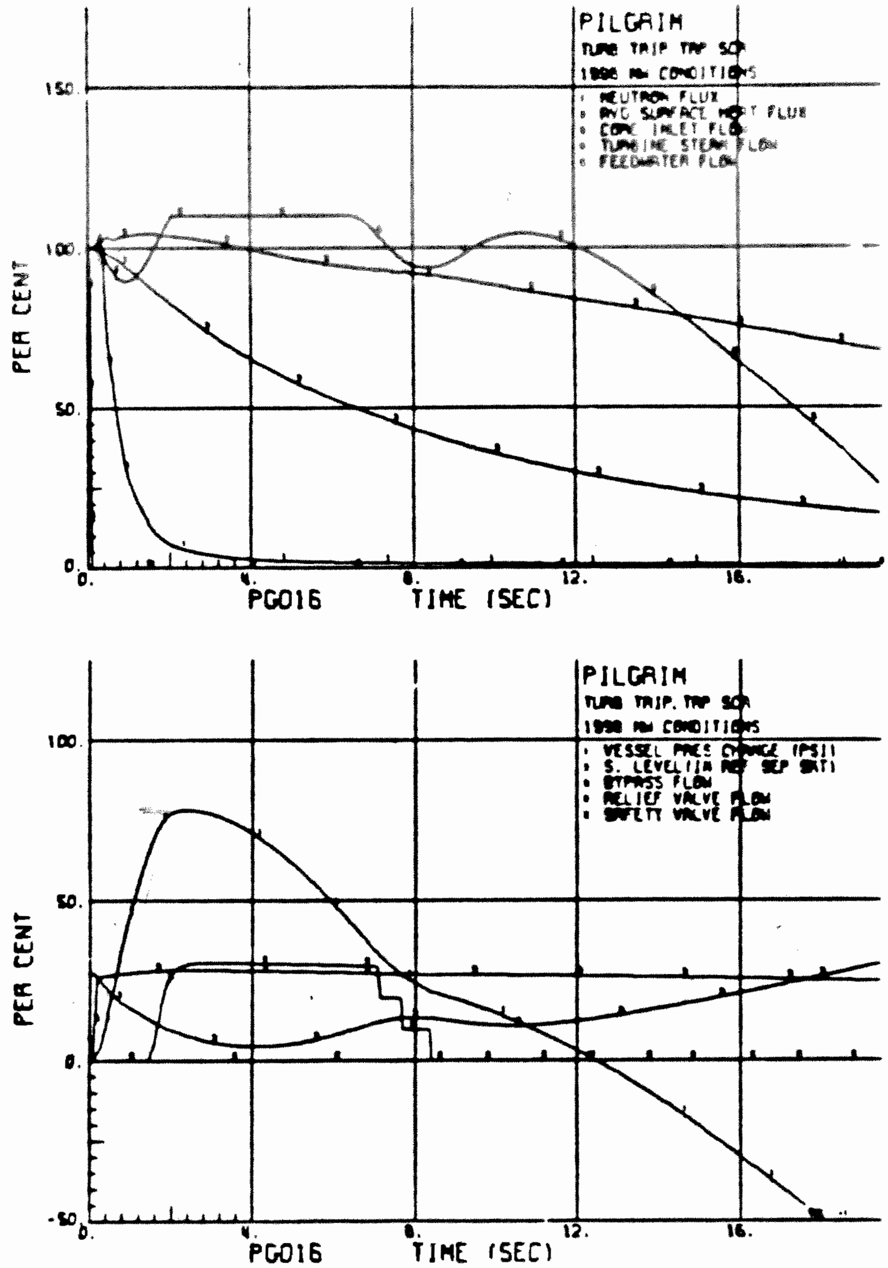


Figure R.2-2. Initial Core Transient Results Nominal Turbine Trip From High Power With Bypass

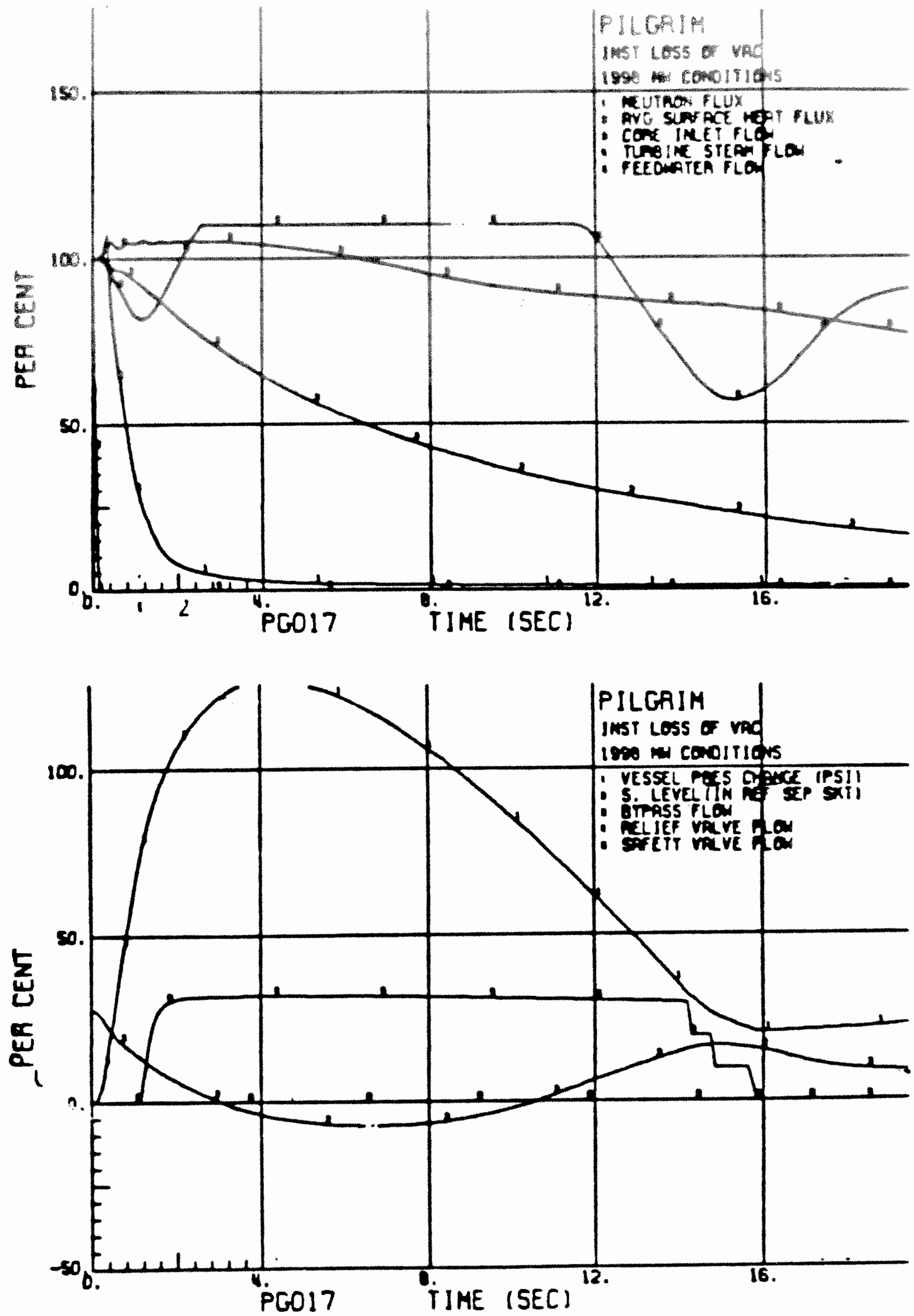


Figure R.2-3. Initial Core Instantaneous Loss of Condenser Vacuum

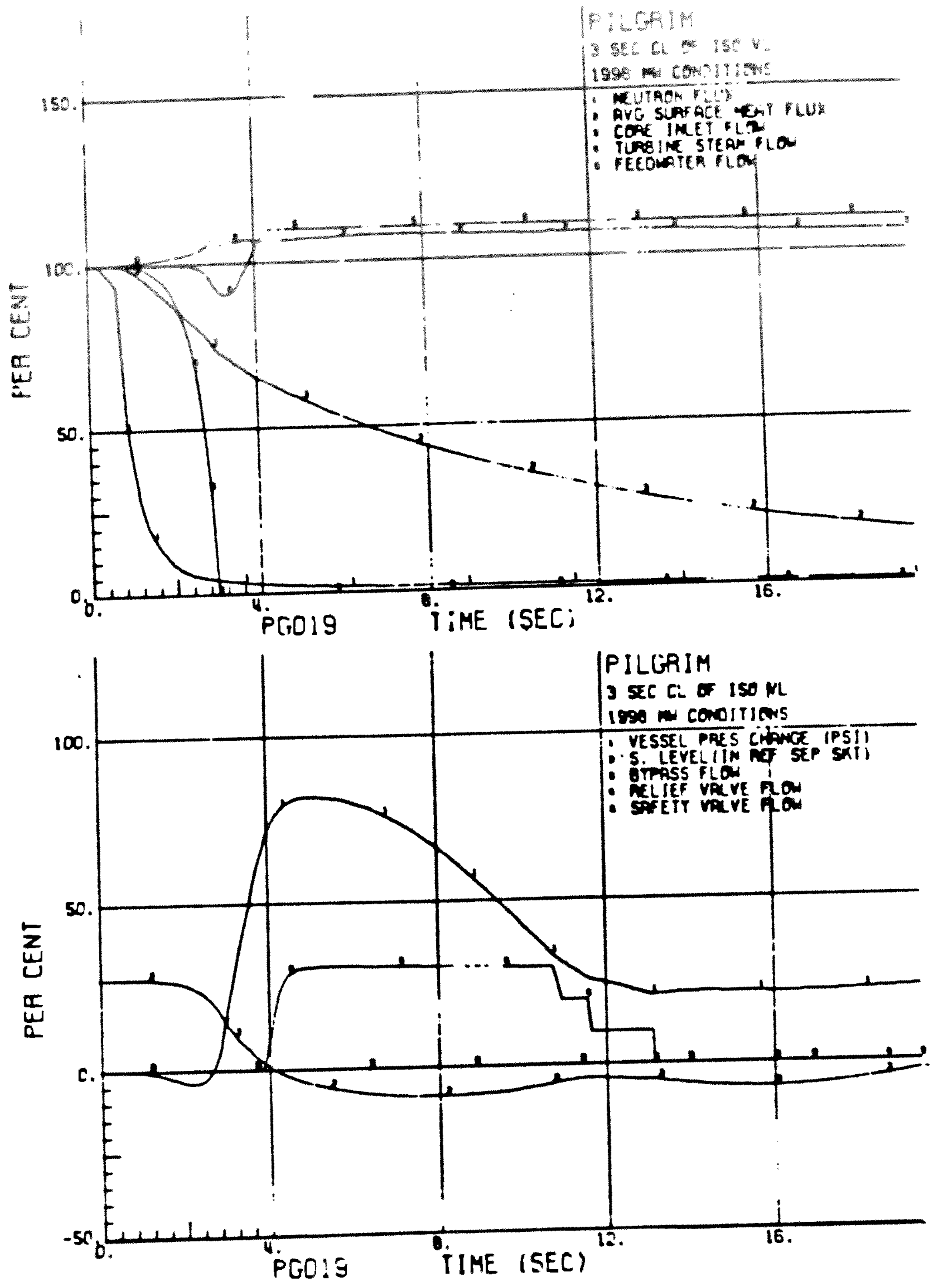


Figure R.2-4. Initial Core Closure of all Main Steamline Isolation Valves

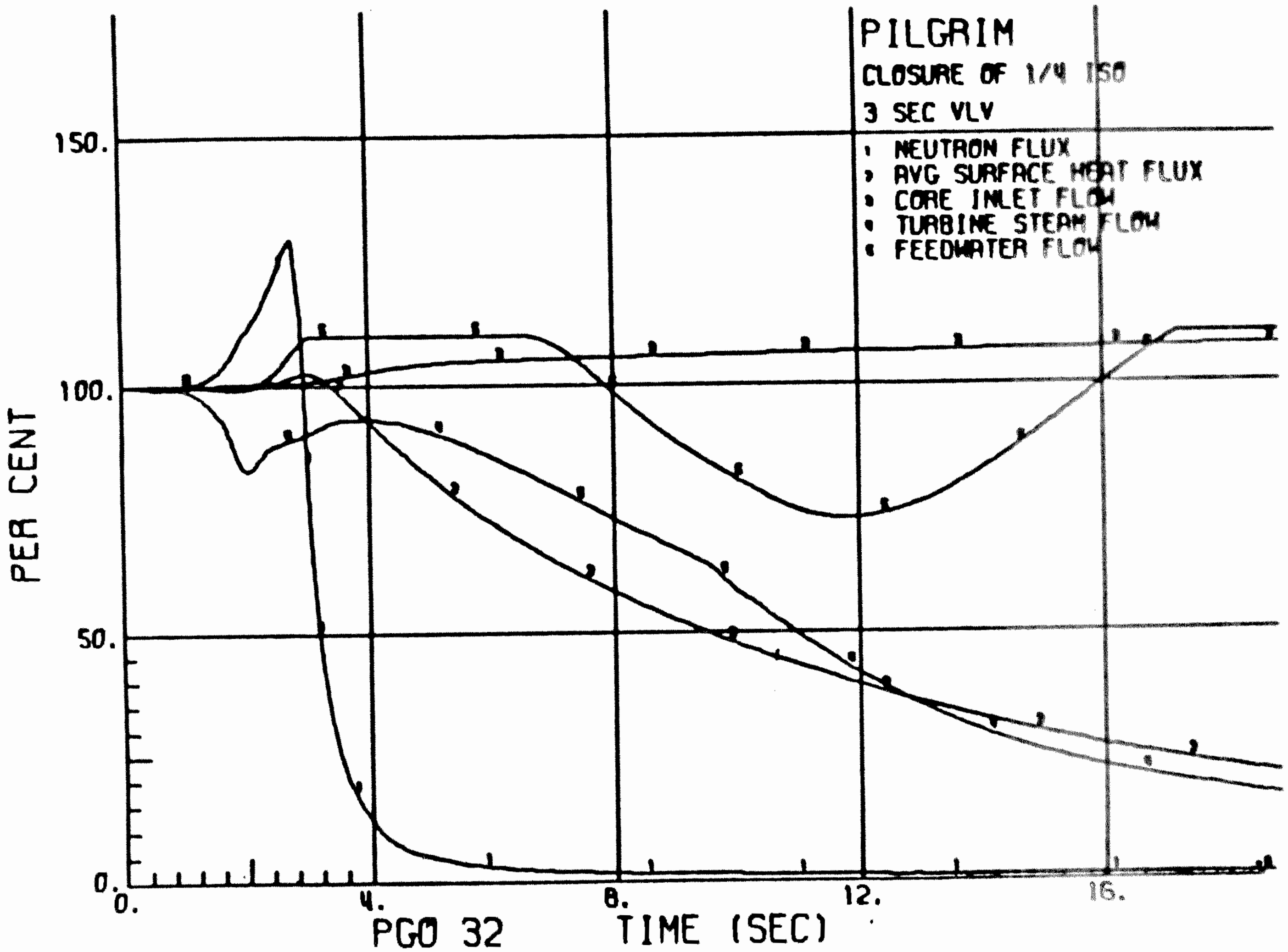


Figure R.2-5. Initial Core Transient Results, Closure of One Main Steam Isolation Valve

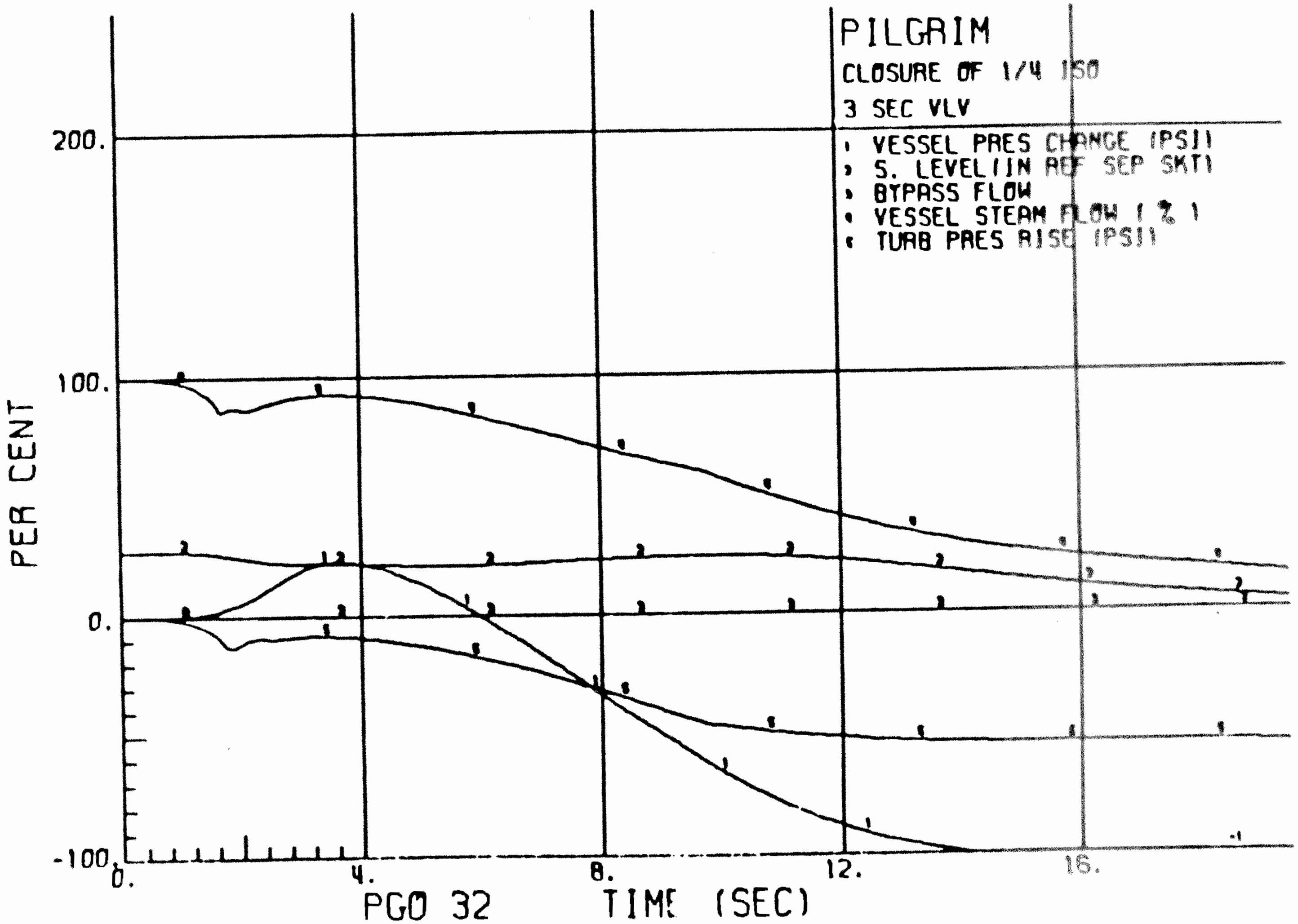
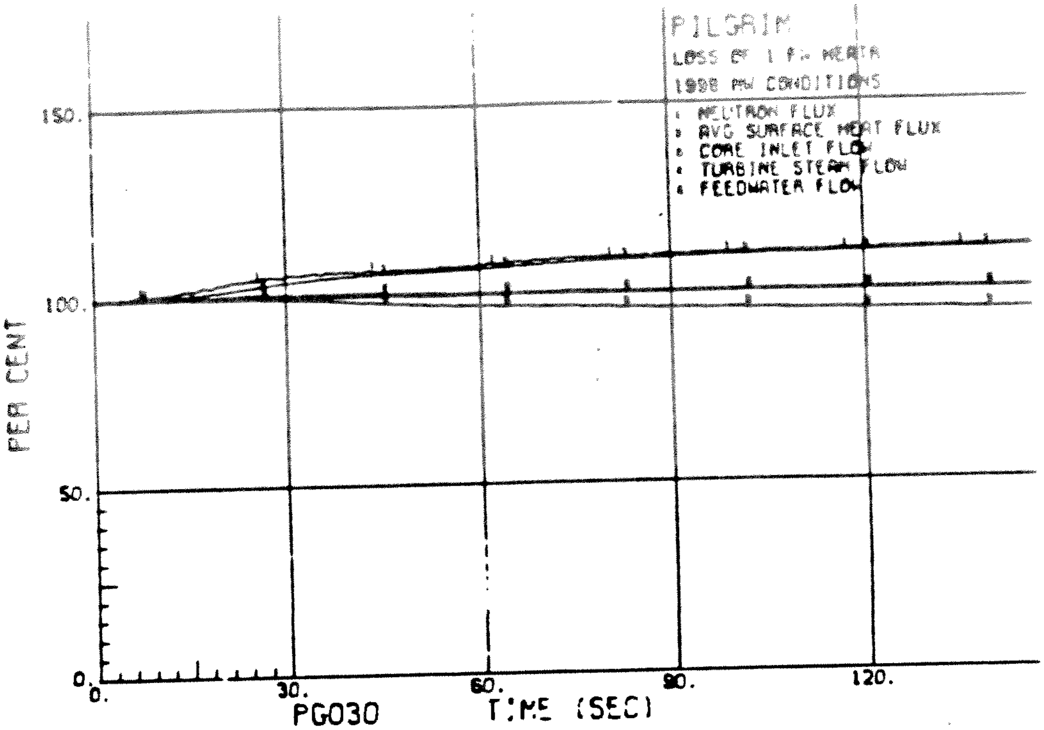


Figure R.2-5. Initial Core Transient Results, Closure of One Main Steam Isolation Valve (Continued)



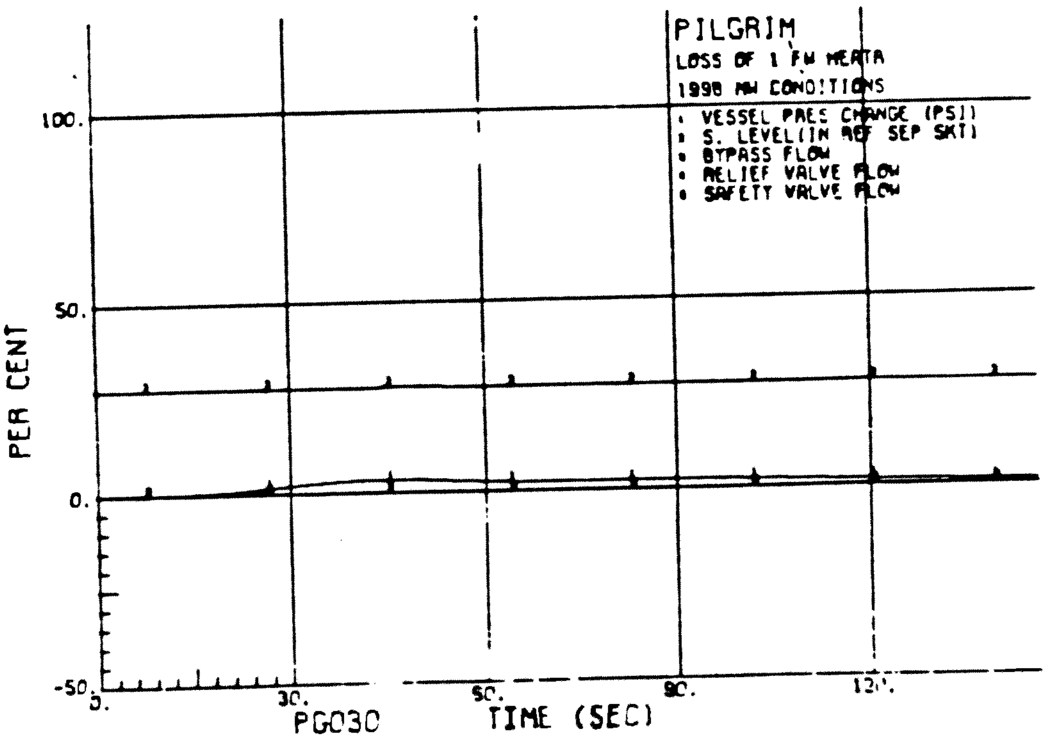
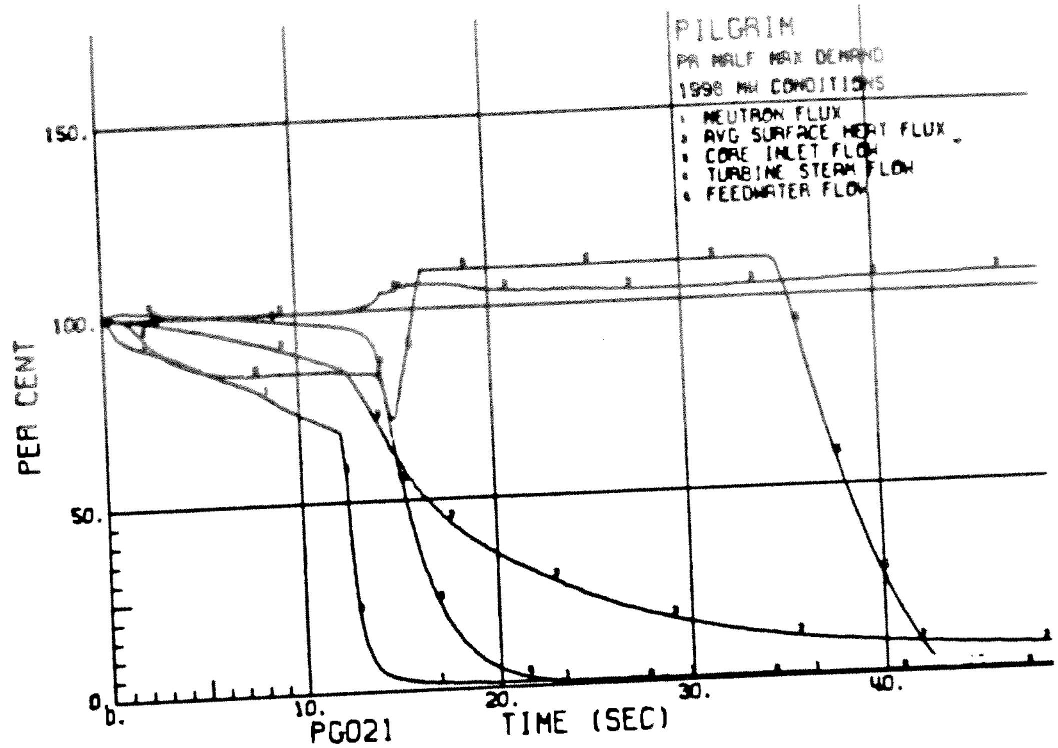


Figure R.2-6. Initial Core Loss of 100°F Feedwater Heating



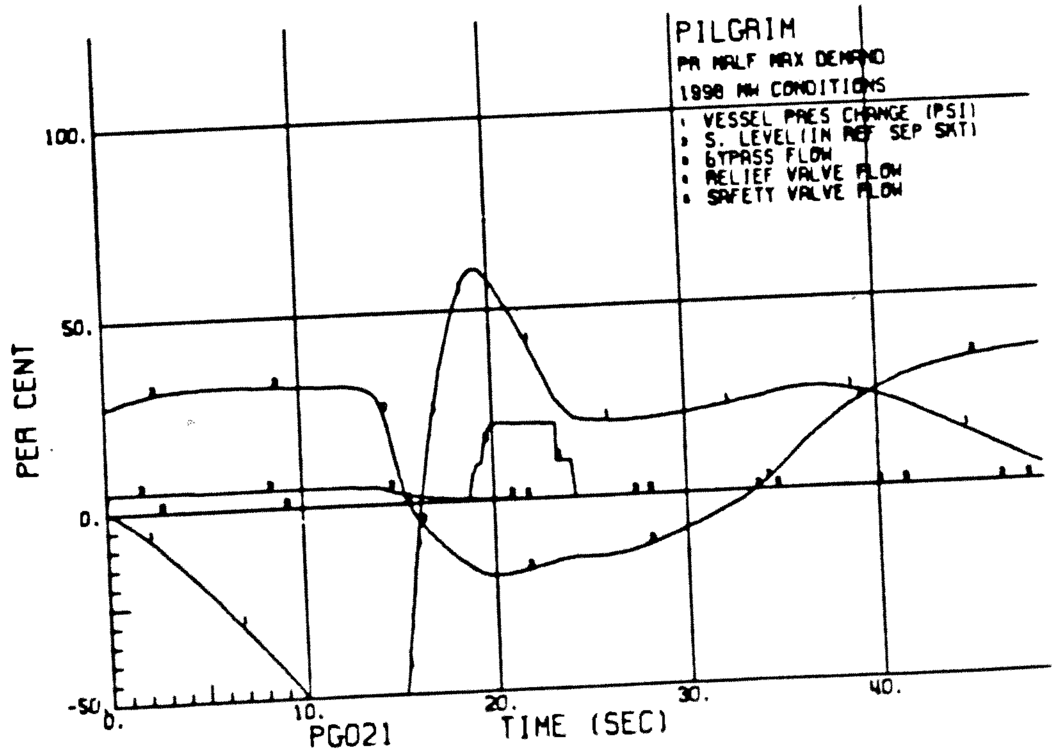


Figure R.2-7. Initial Core Pressure Regulator Malfunction Calling for Maximum Steamflow

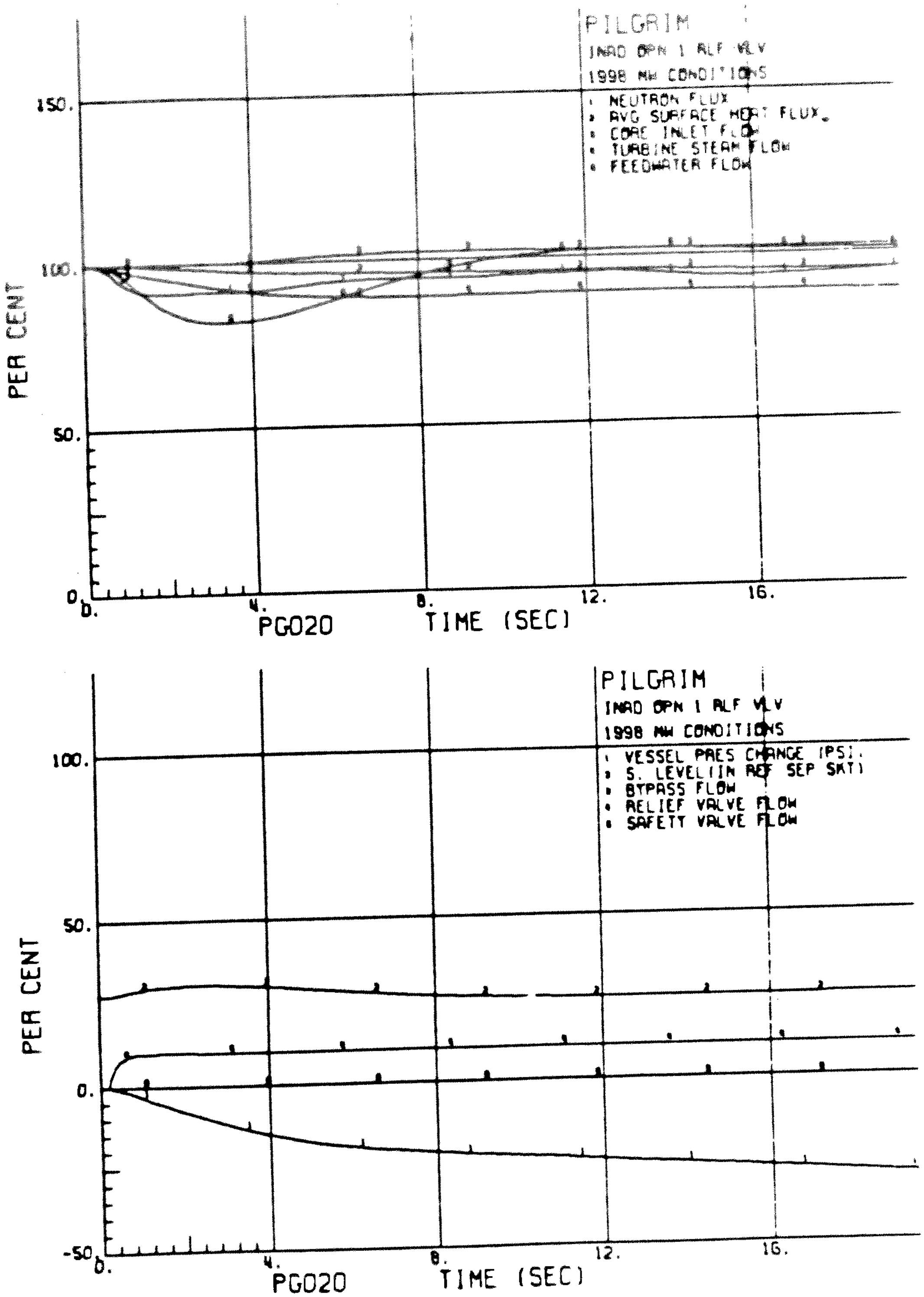
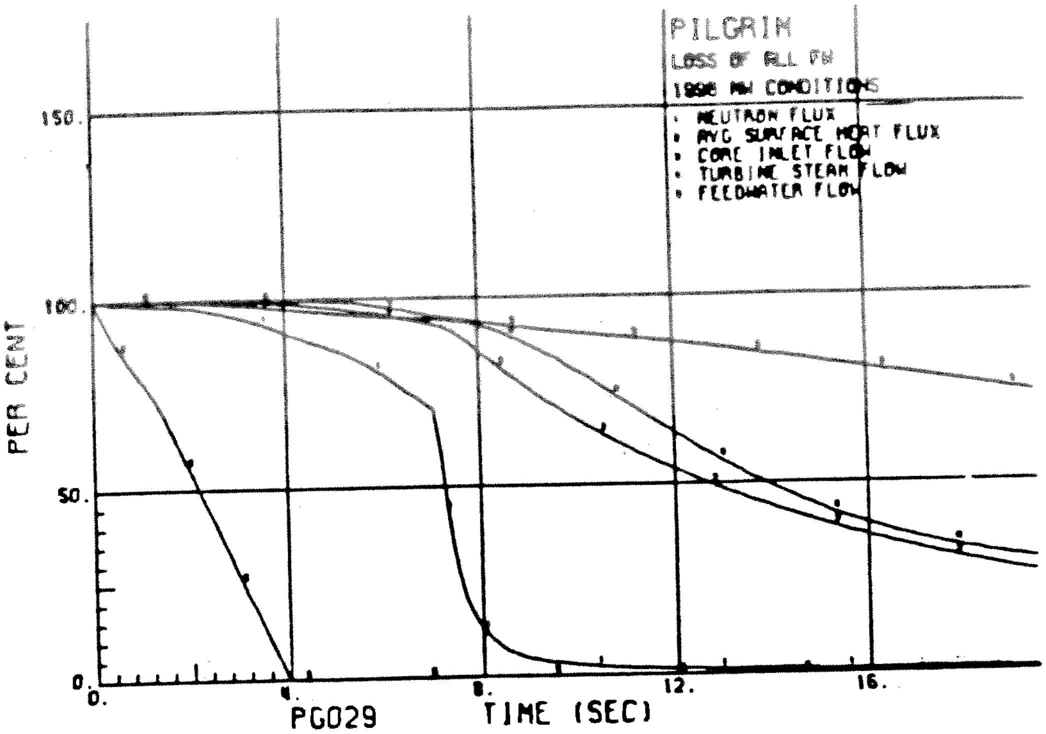


Figure R.2-8. Initial Core Inadvertent Opening of One Relief Valve



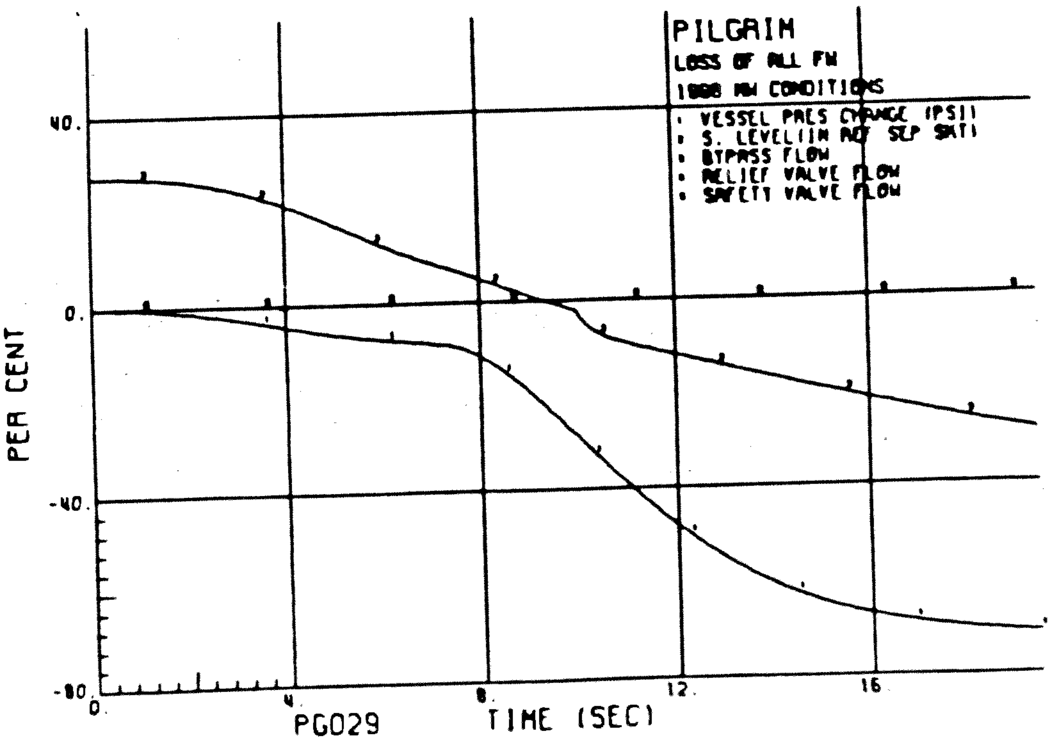


Figure R.2-9. Initial Core Loss Of All Feedwater

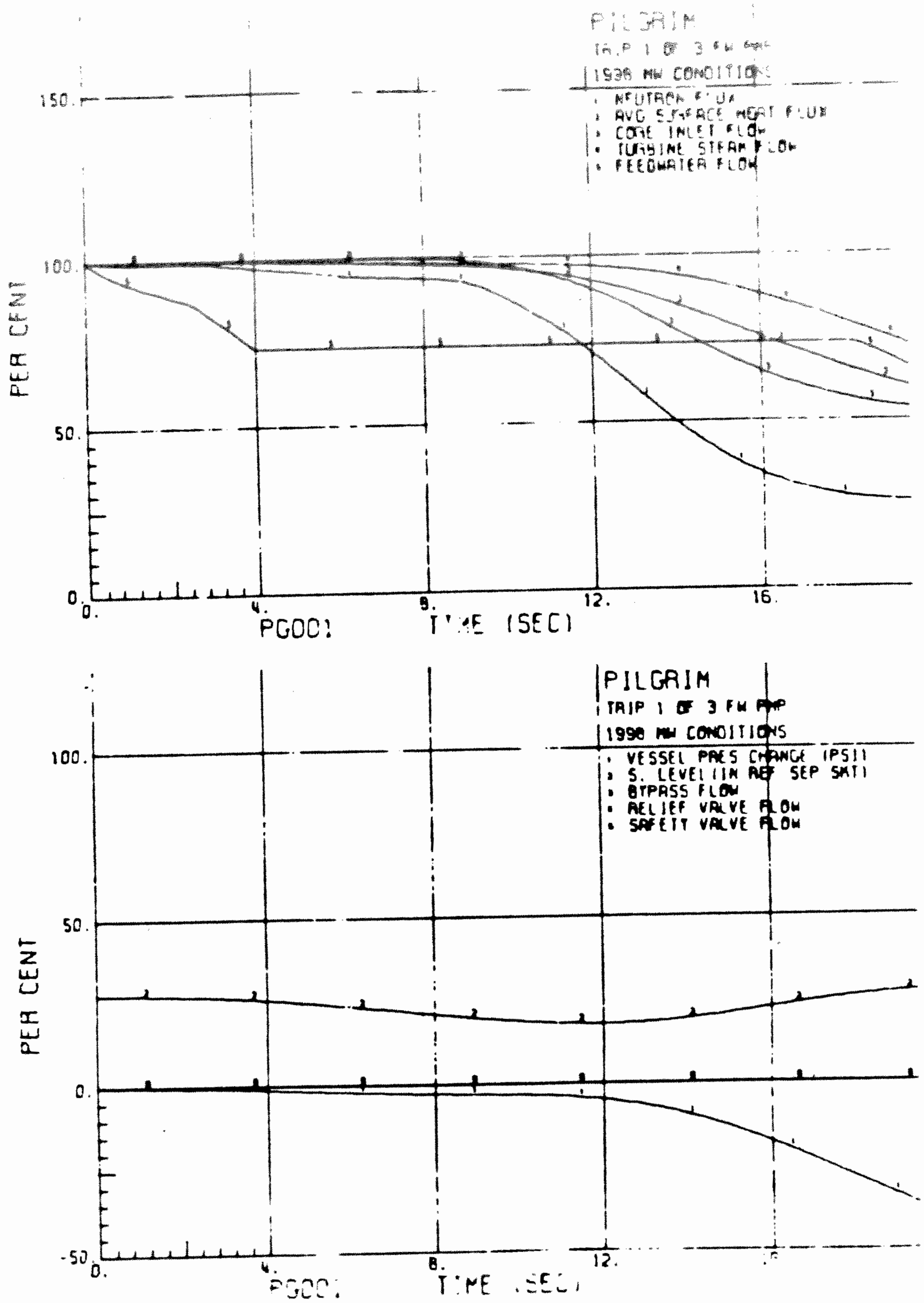
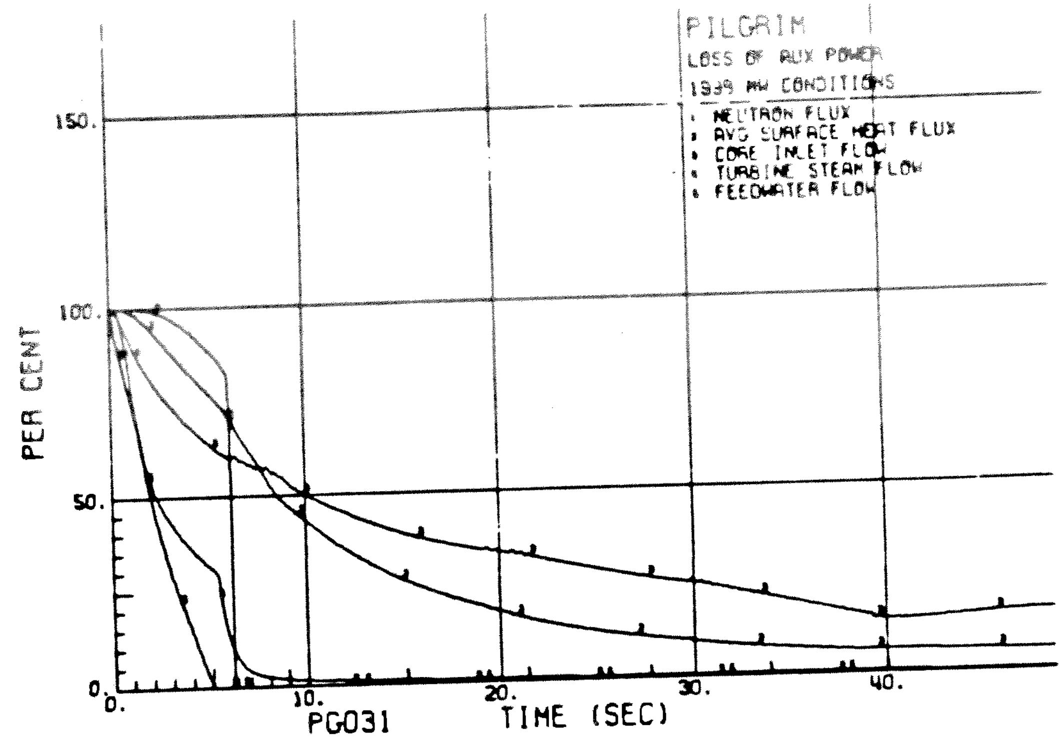


Figure R.2-10. Initial Core Trip of 1 of 3 Feedwater Pumps



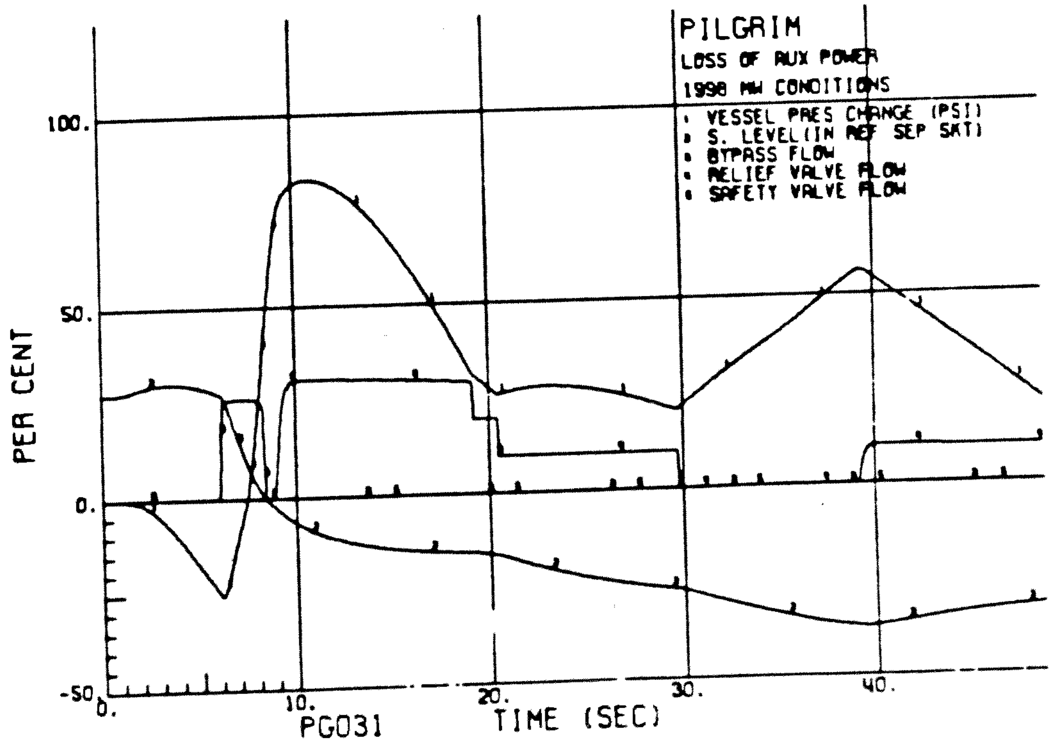


Figure R.2-11. Initial Core Loss of All Offsite Power to Station Auxiliaries

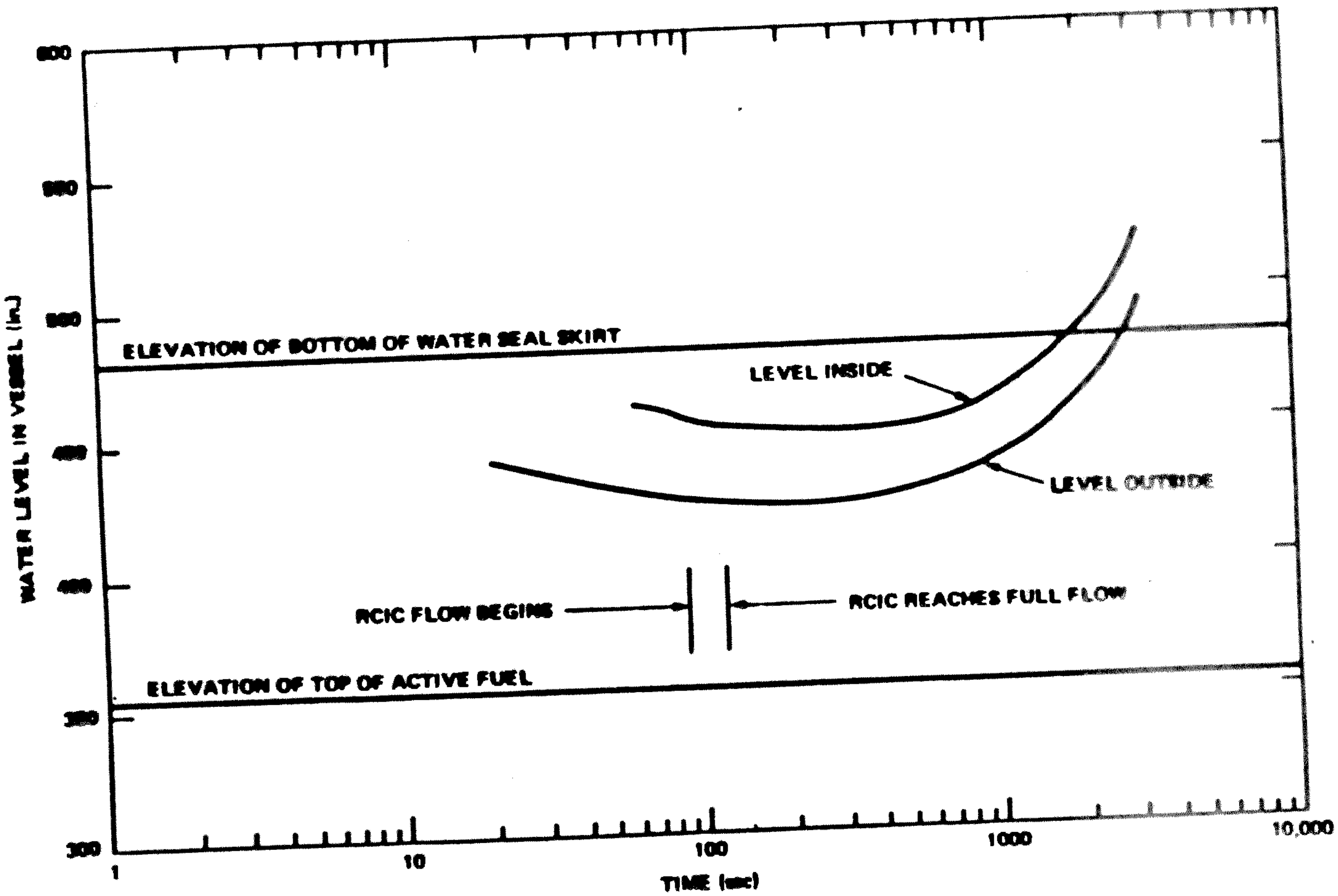


Figure R.2-12. Initial Core Water Level Versus Time Following Loss of Off-Site Power Assuming Only RCIC Operation

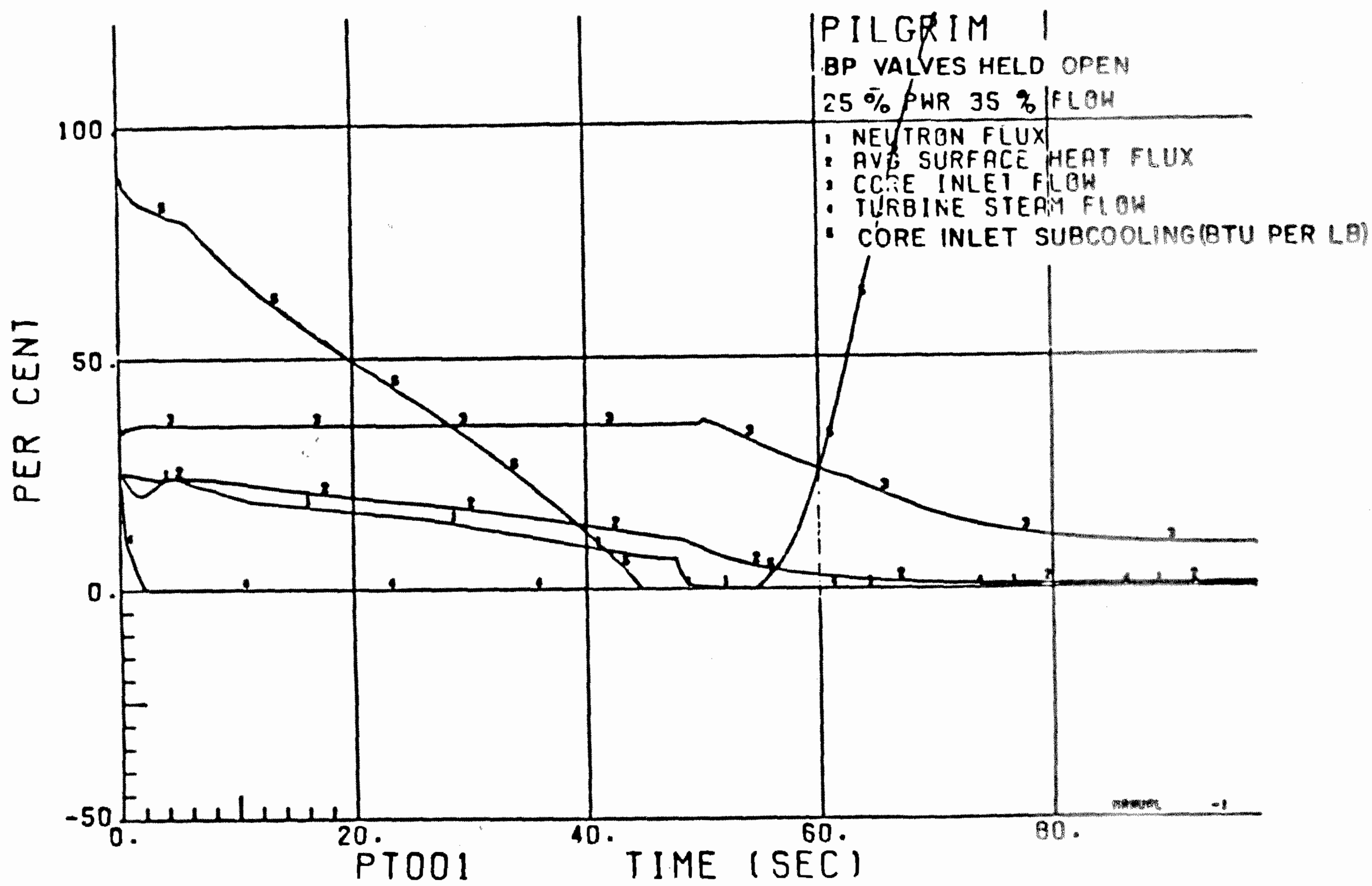


Figure R.2-13. Initial Core Inadvertent Full Bypass Opening: 25% Power, 35% Flow,

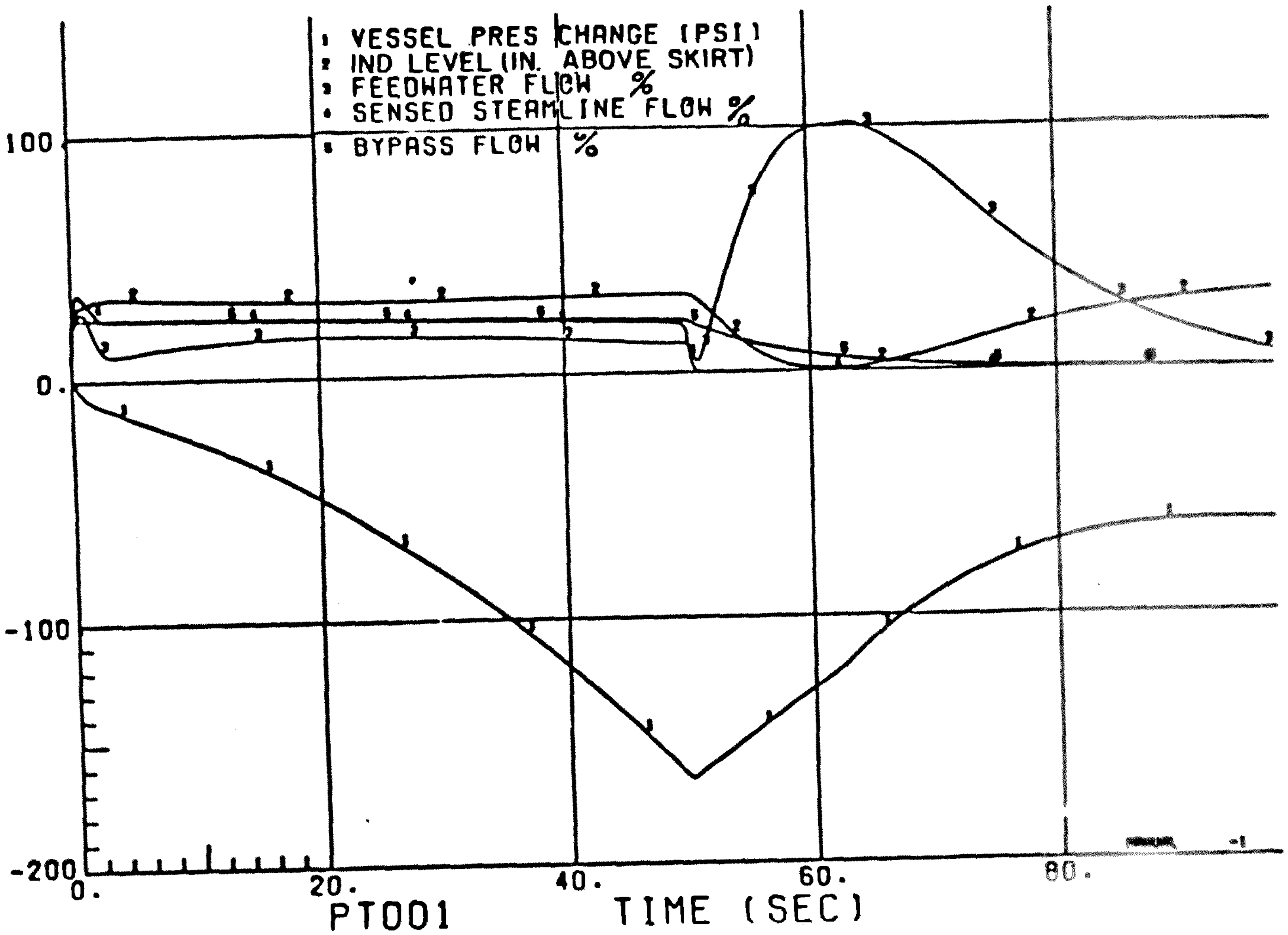


Figure R.2-13. Initial Core Inadvertent Full Bypass Opening: 25% Power, 35% Flow, (Continued)

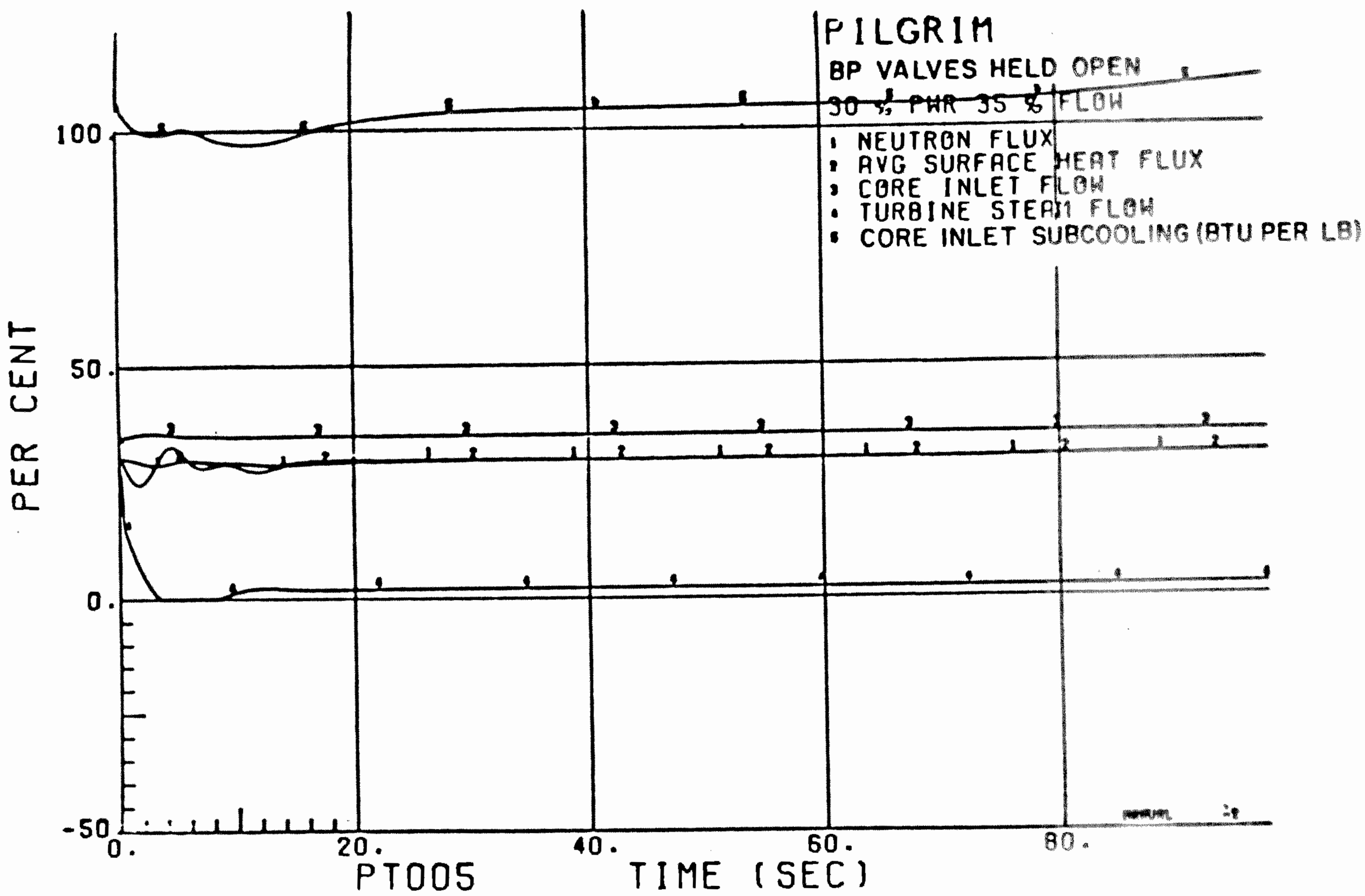


Figure R.2-14. Initial Core Inadvertent Full Bypass Opening: 30% Power, 35% Flow,

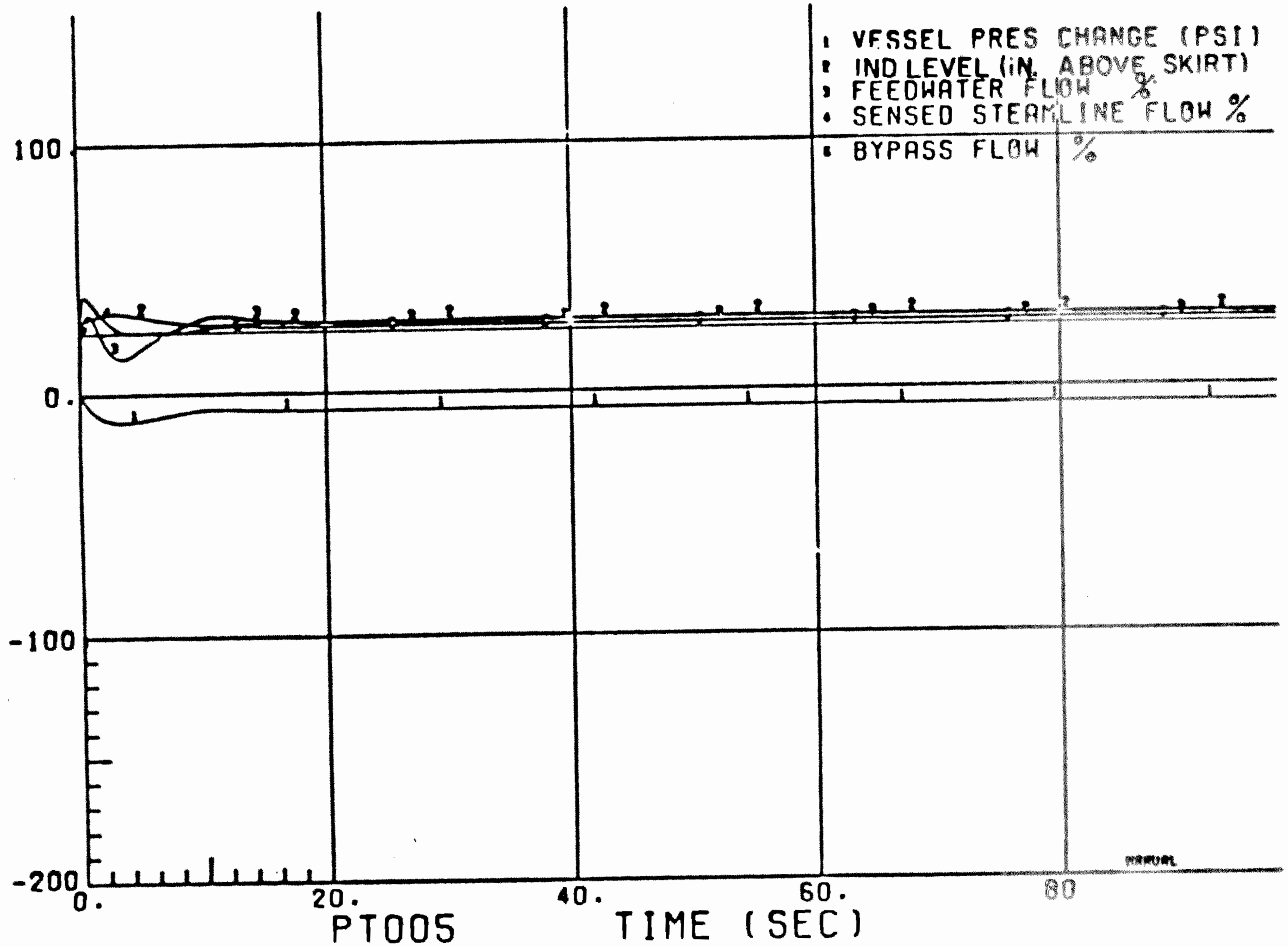


Figure R.2-14. Initial Core Inadvertent Full Bypass Opening: 30% Power, 35% Flow, (Continued)

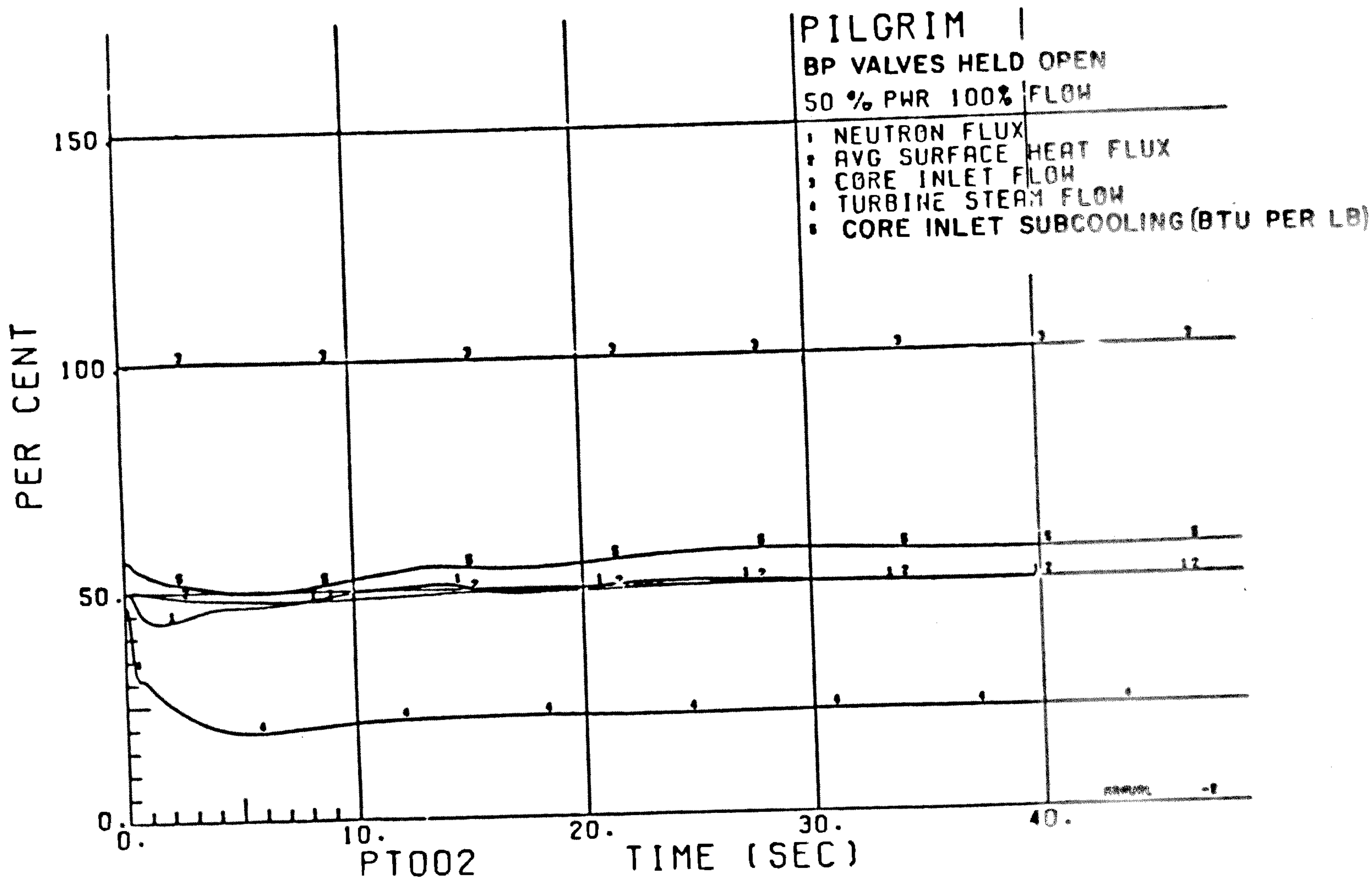


Figure R.2-15. Initial Core Inadvertent Full Bypass Opening: 50% Power, 100% Flow.

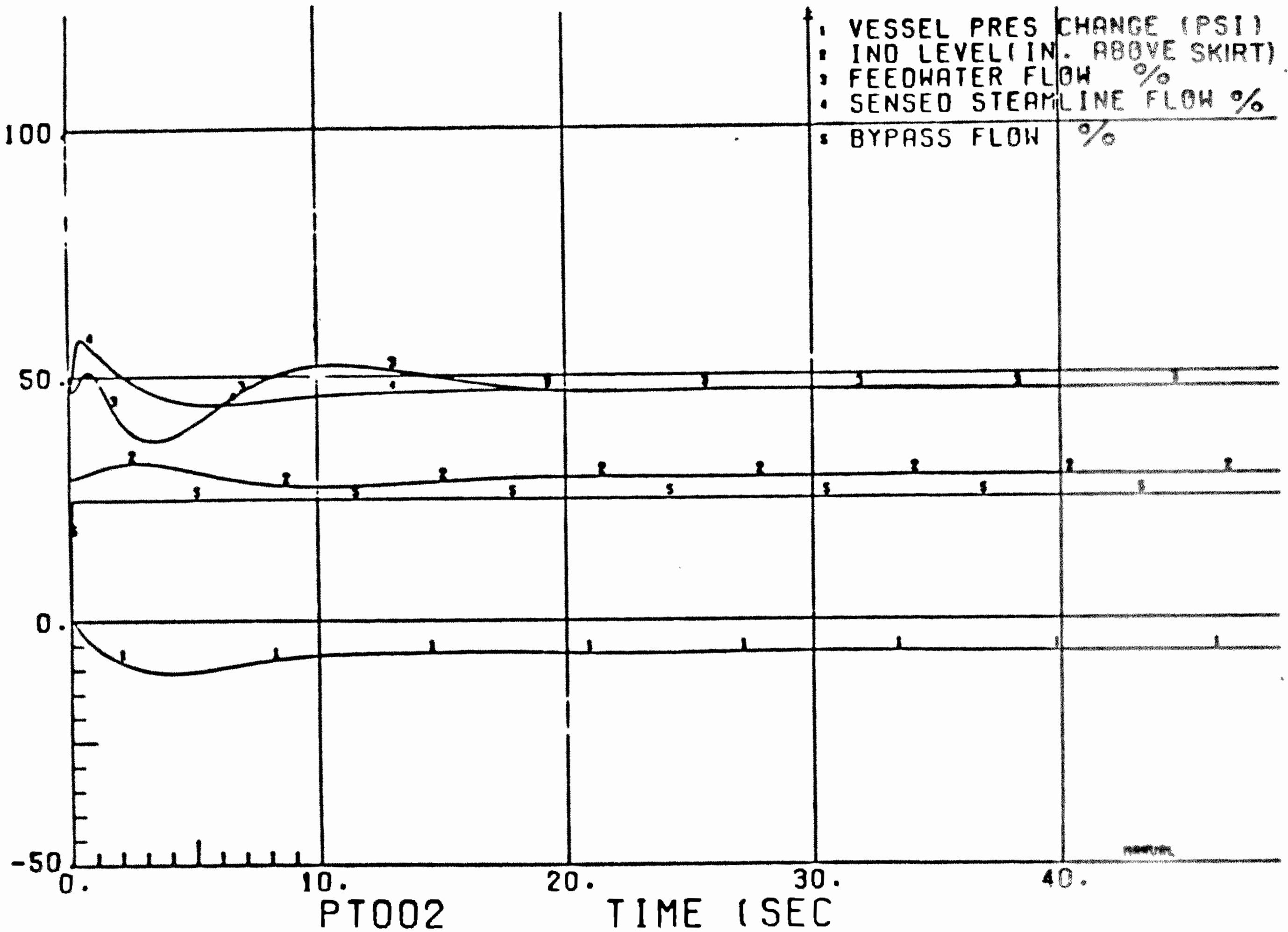


Figure R.2-15. Initial Core Inadvertent Full Bypass Opening: 50% Power, 100% Flow, (Continued)

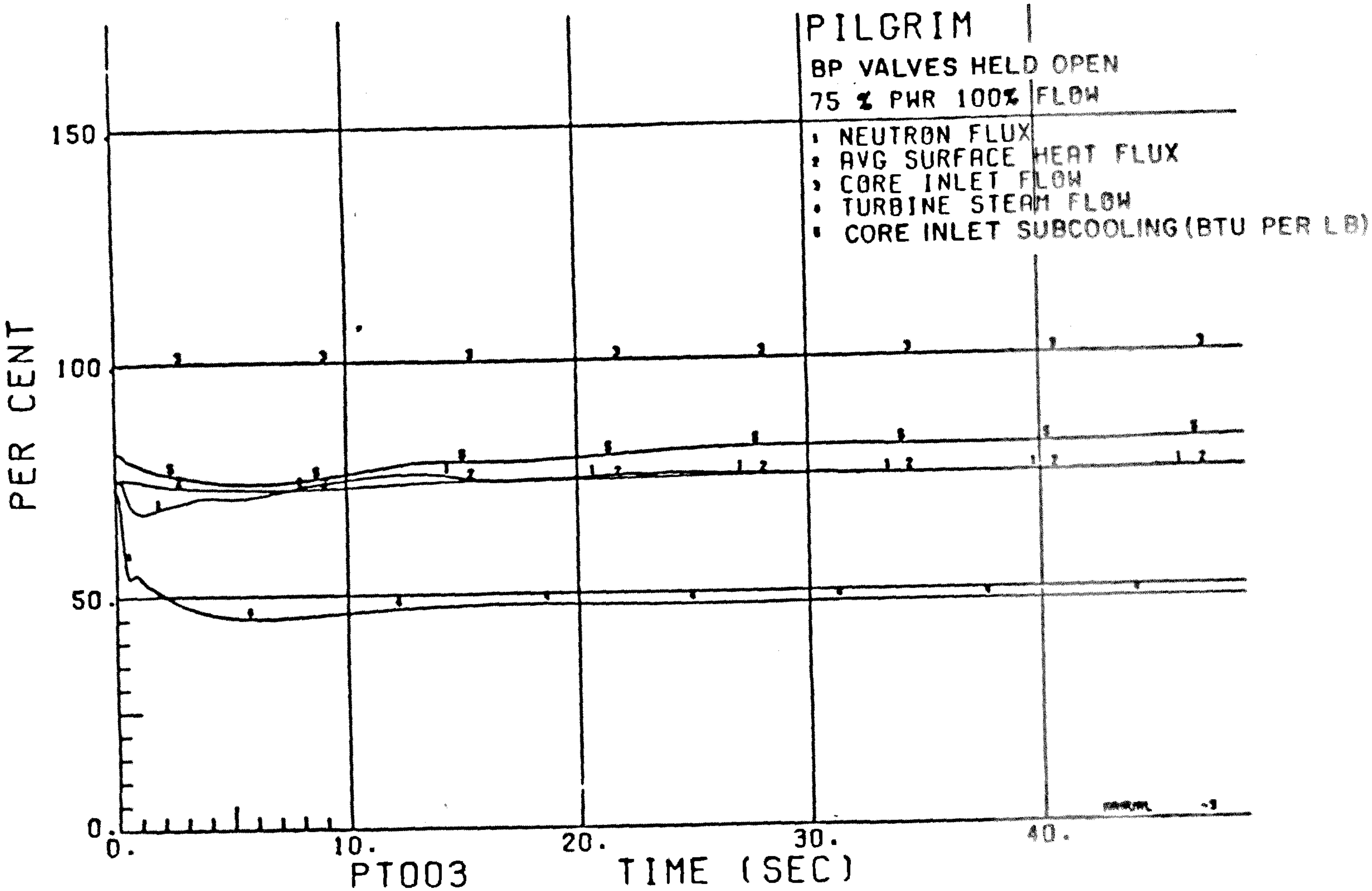


Figure R.2-16. Initial Core Inadvertent Full Bypass Opening: 75% Power, 100% Flow,

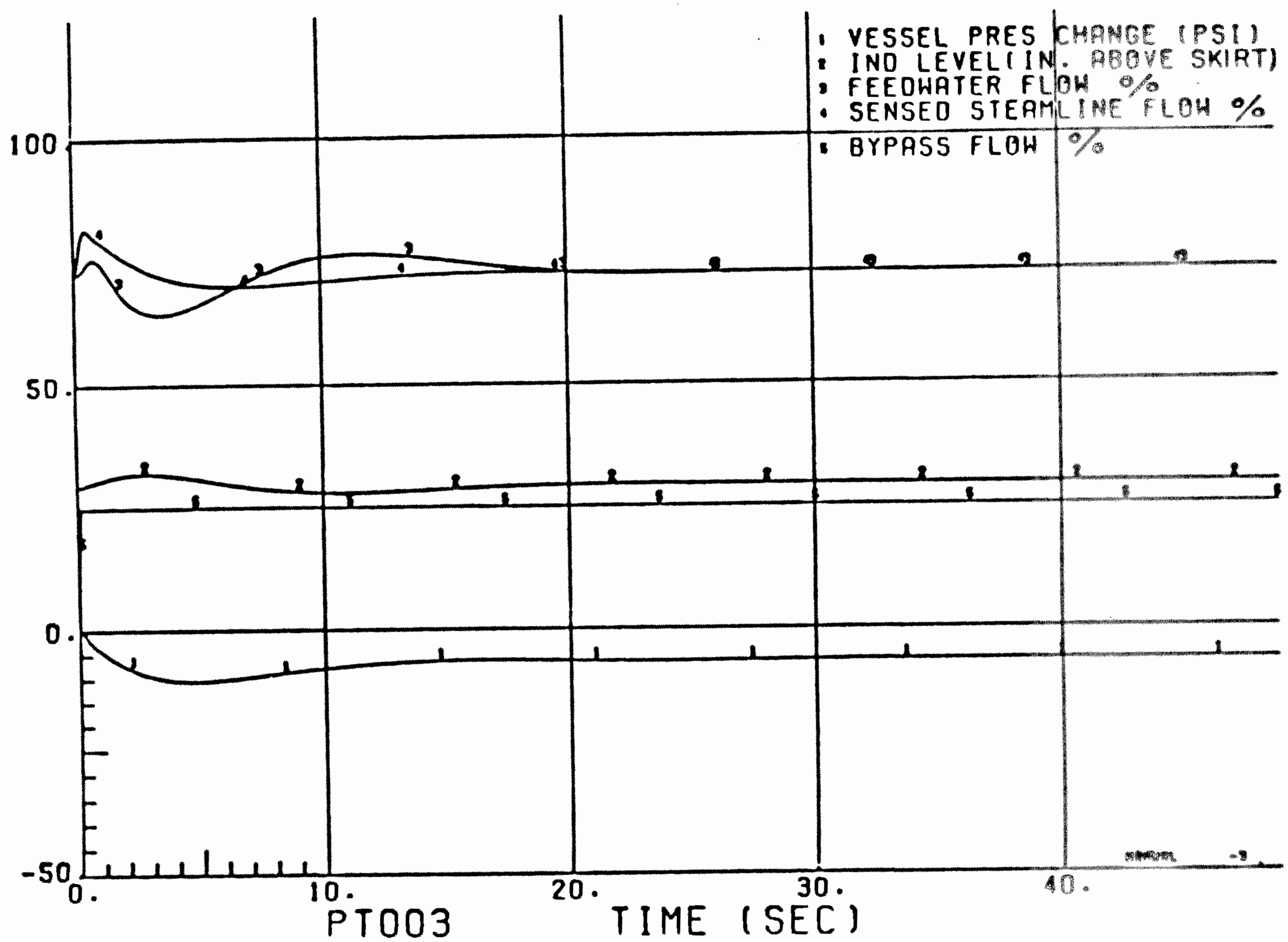


Figure R.2-16. Initial Core Inadvertent Full Bypass Opening: 75% Power, 100% Flow, (Continued)

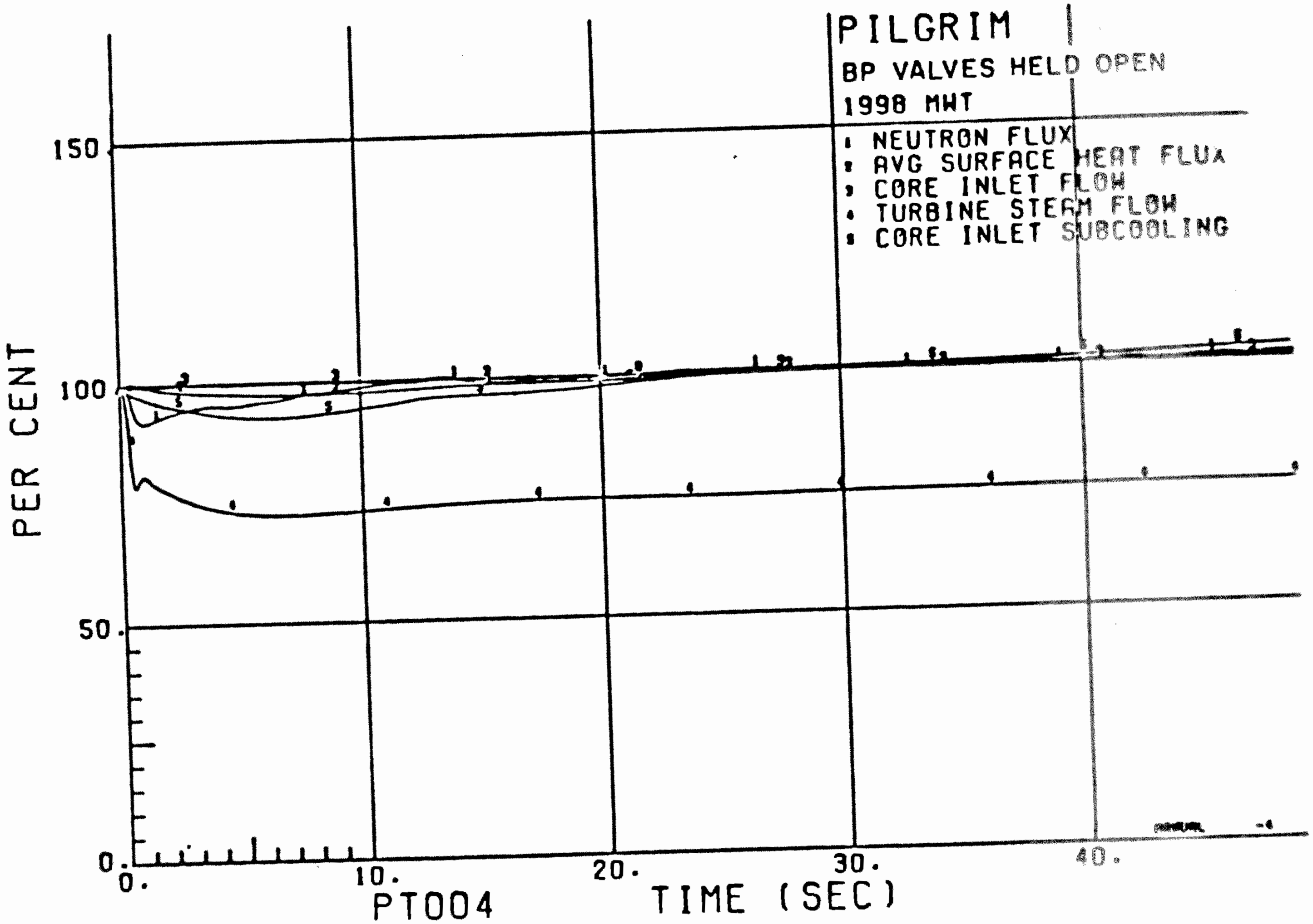


Figure R.2-17. Initial Core Inadvertent Full Bypass Opening: 1998 Mwe,

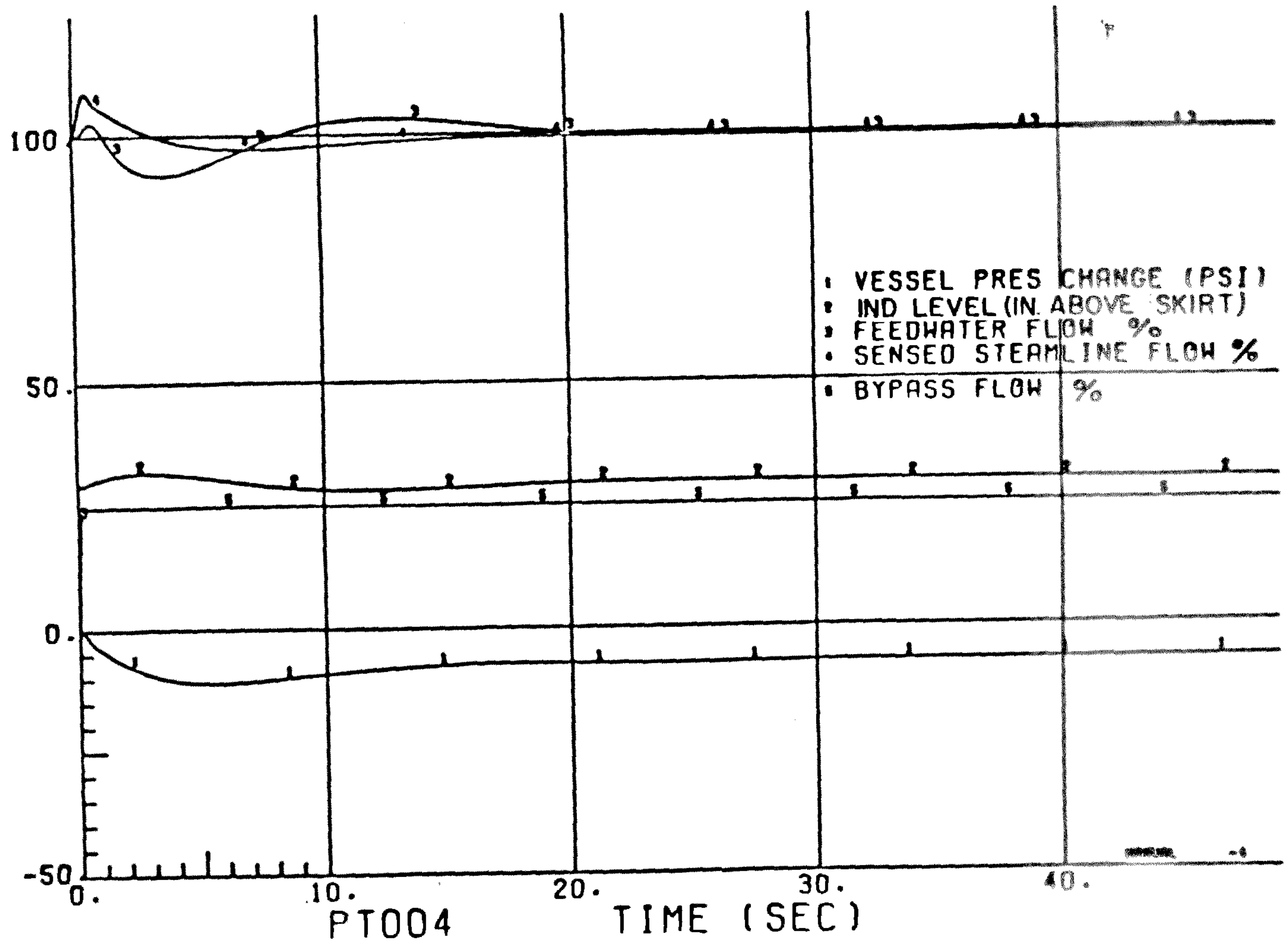
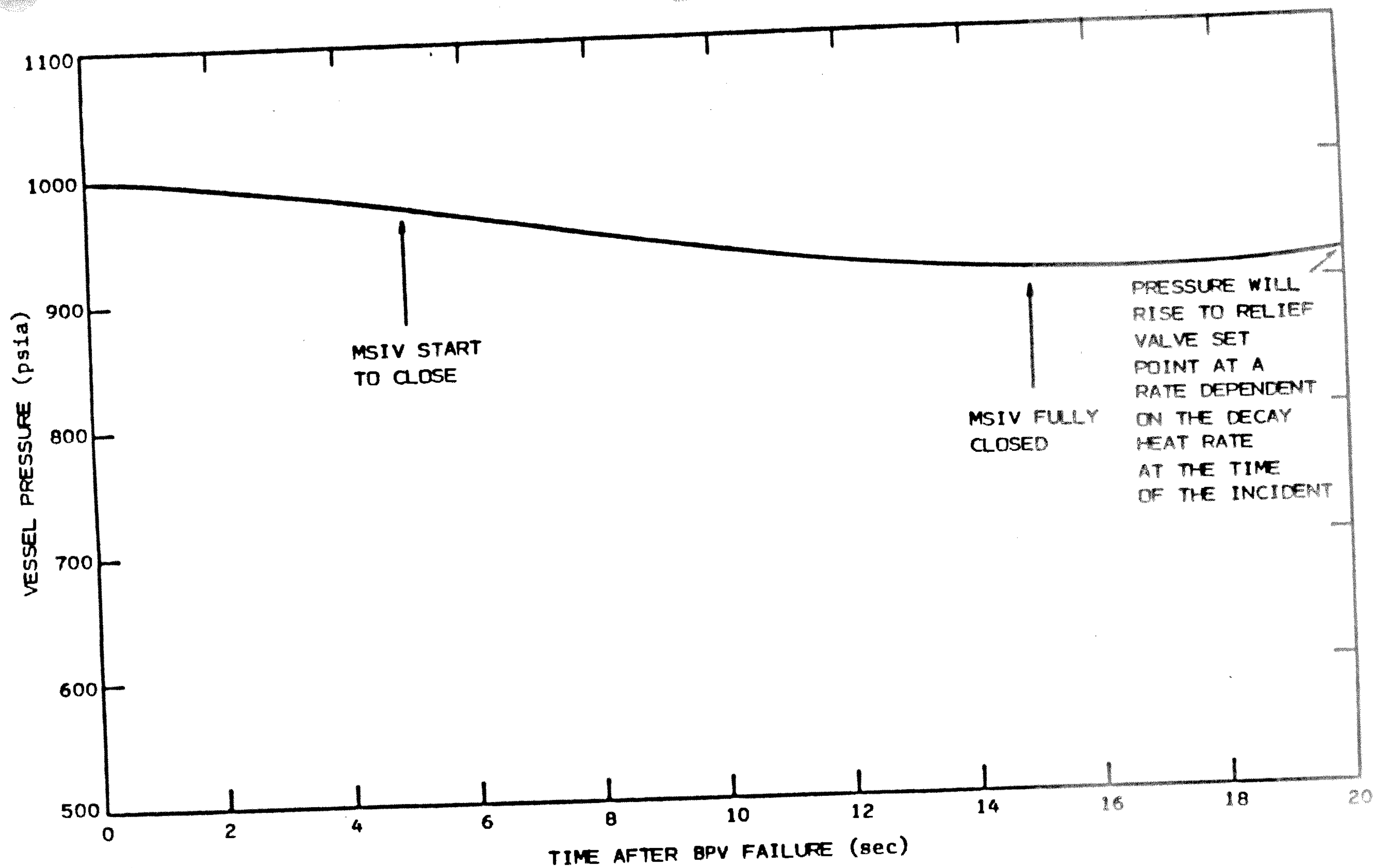


Figure R.2-17. Initial Core Inadvertent Full Bypass Opening: 1998 MWe, Pilgrim (Continued)



NOTE: FOLLOWING AN INADVERTENT FULL OPENING OF THE BYPASS SYSTEM WHEN THE REACTOR IS IN THE HOT STANDBY CONDITION

Figure R.2-18. Initial Core Vessel Pressure Transient

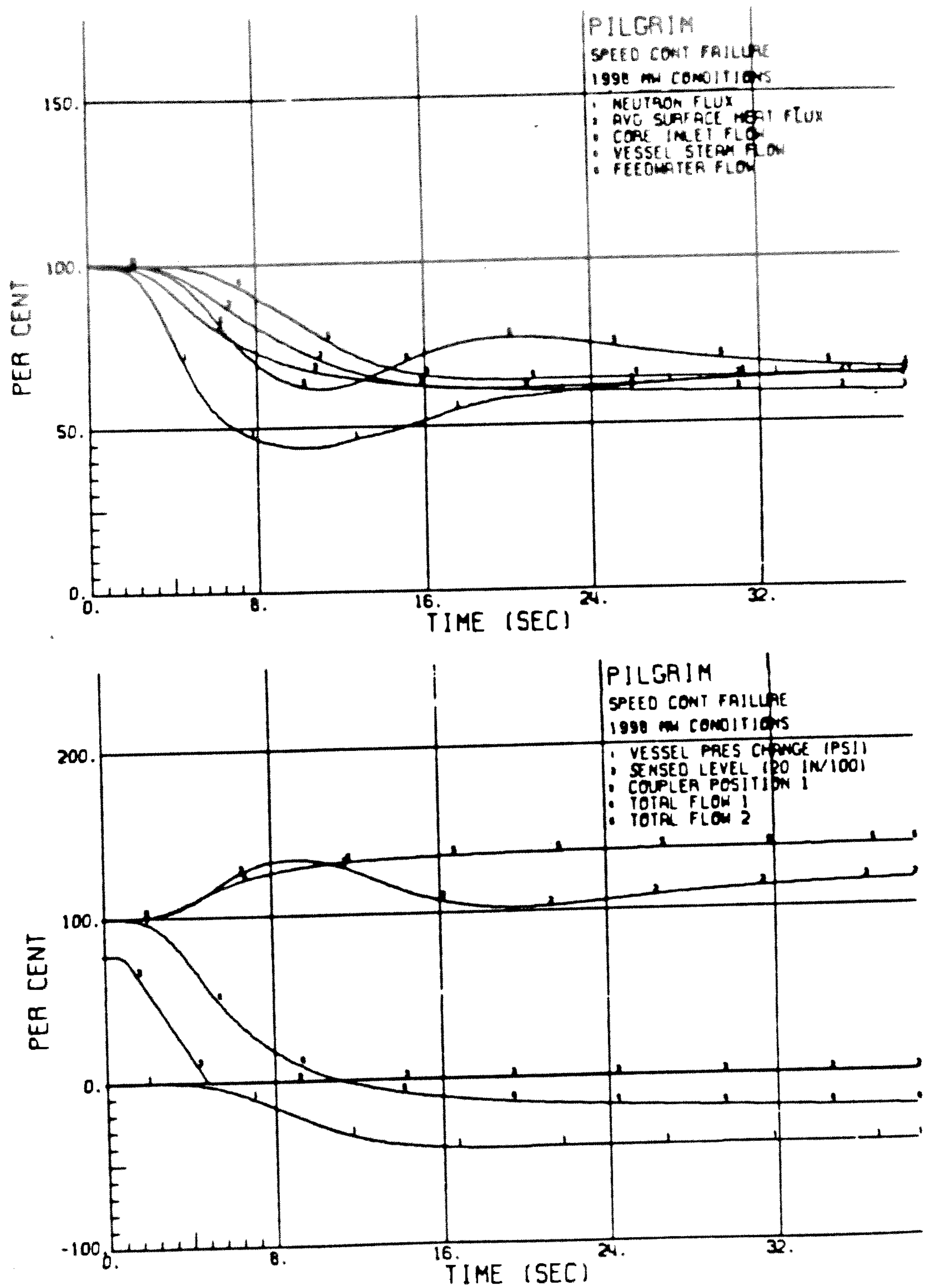
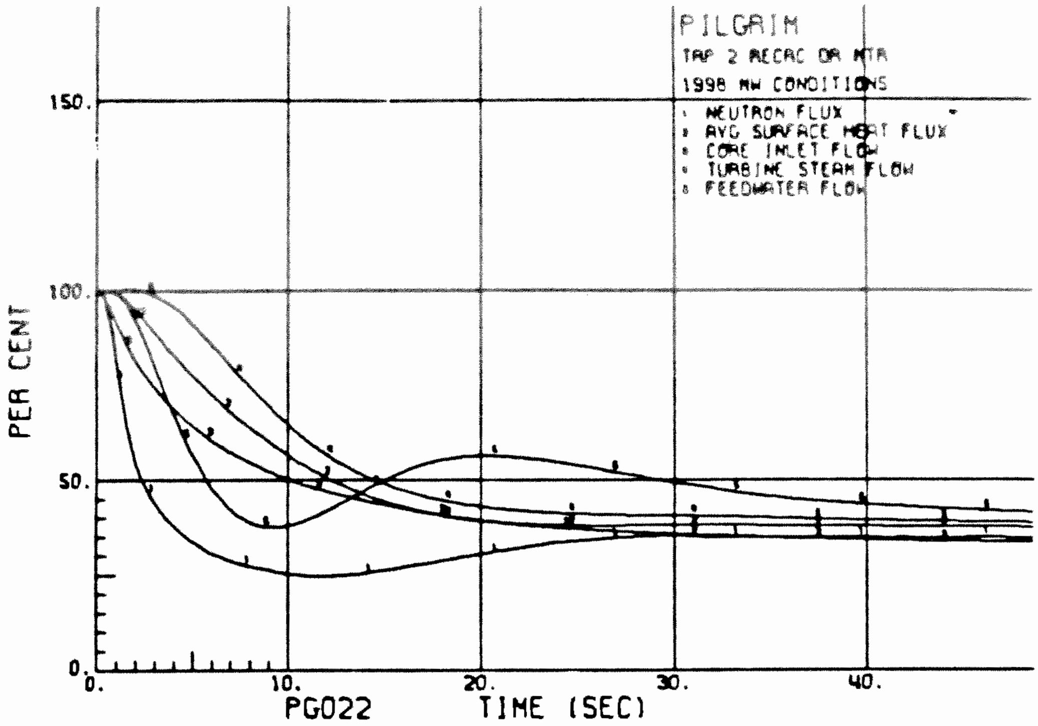


Figure R.2-19. Initial Core Speed Controller Failure Zero Speed Demand



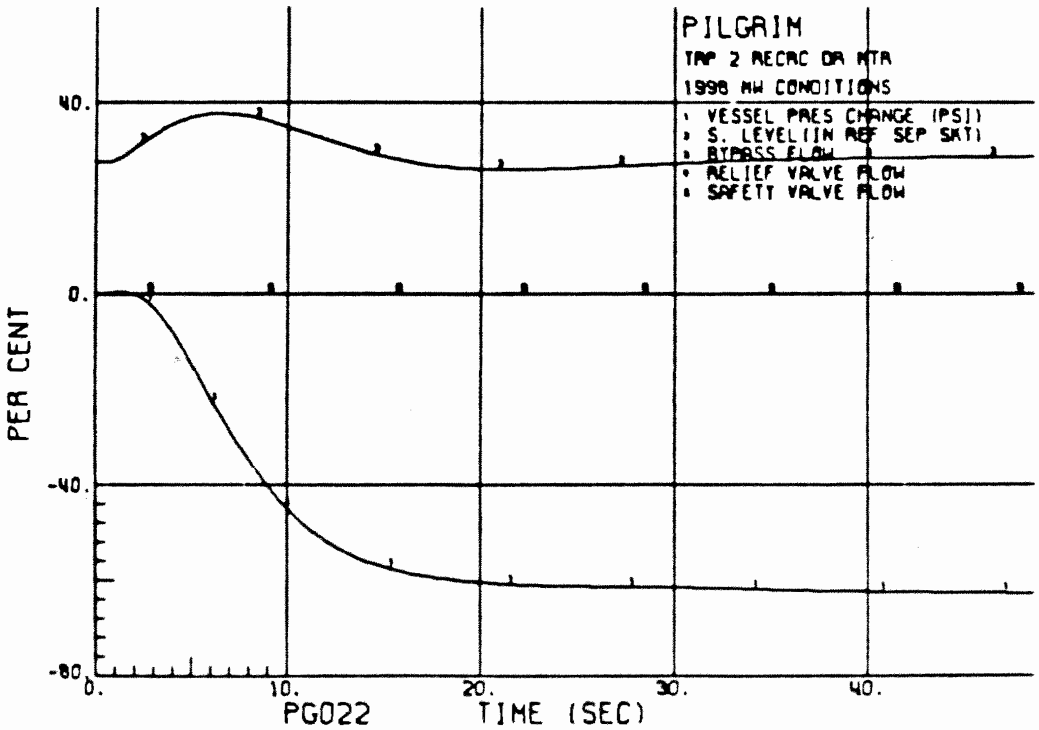


Figure R.2-20. Initial Core Trip of Both Recirculation Drive Motors

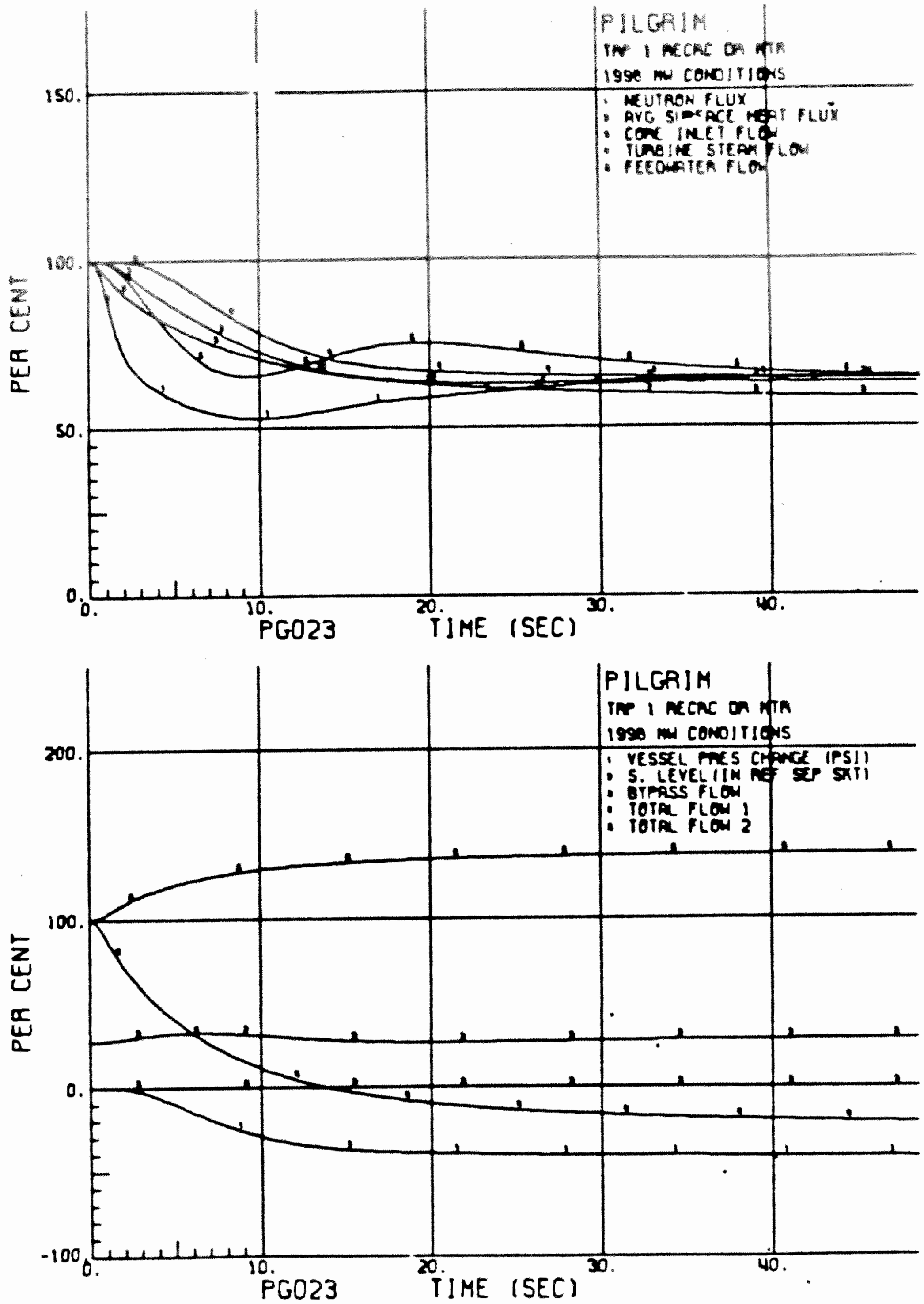


Figure R.2-21. Initial Core Trip of 1 Recirculation Drive Motor

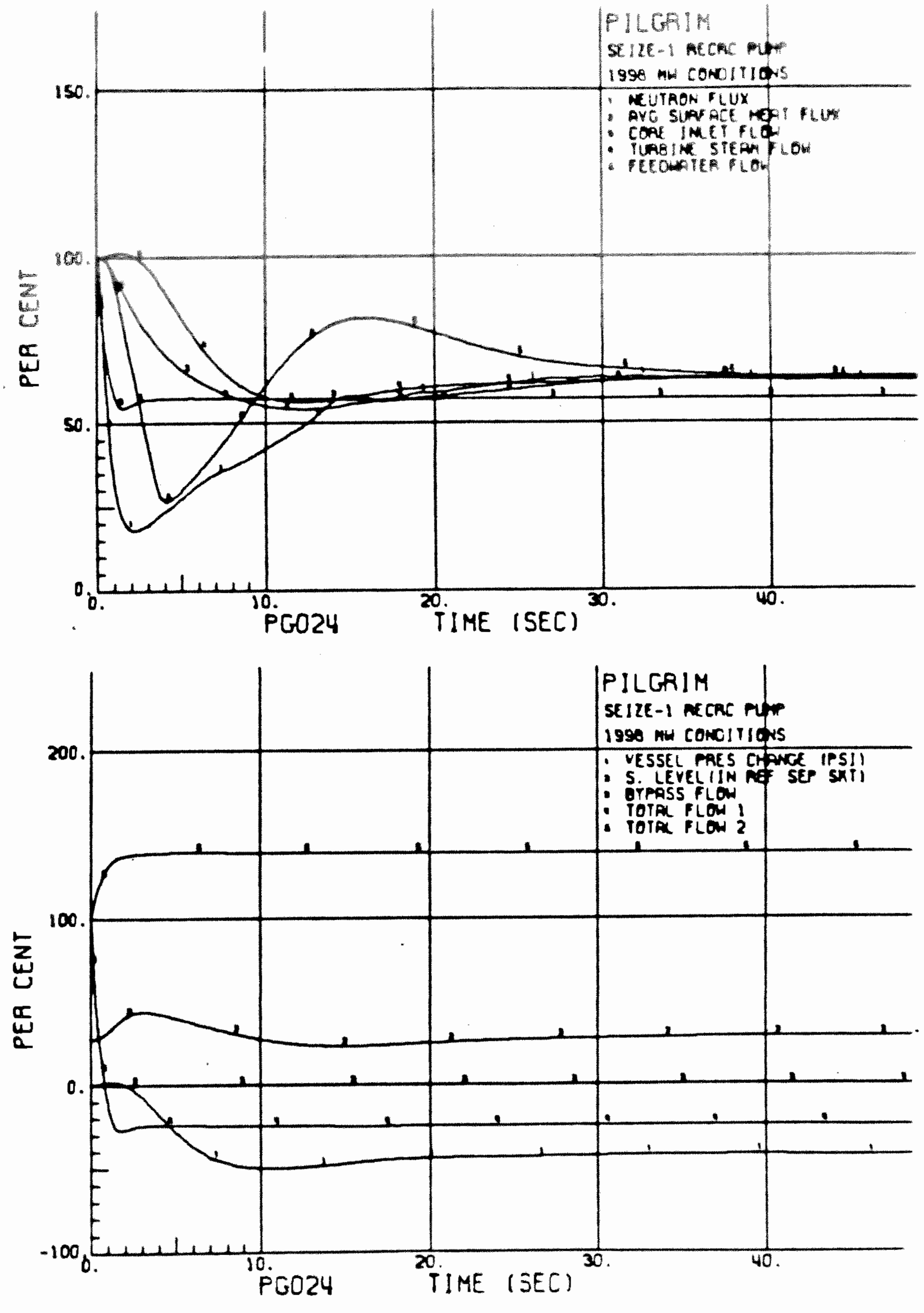
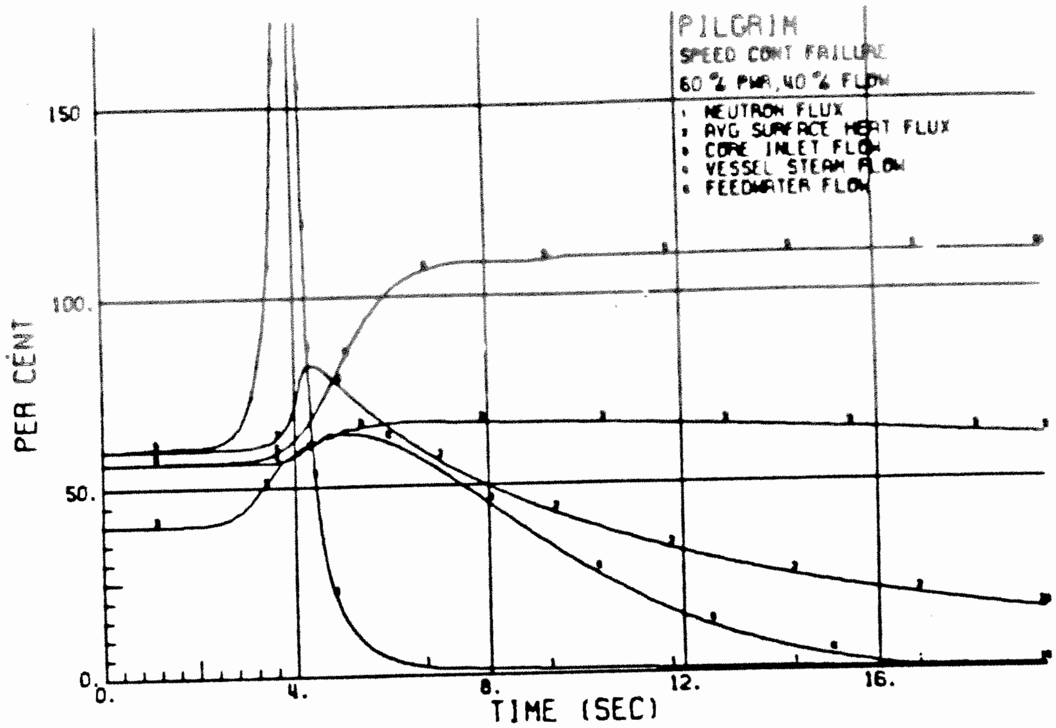


Figure R.2-22. Initial Core Seizure of One Recirculation Pump



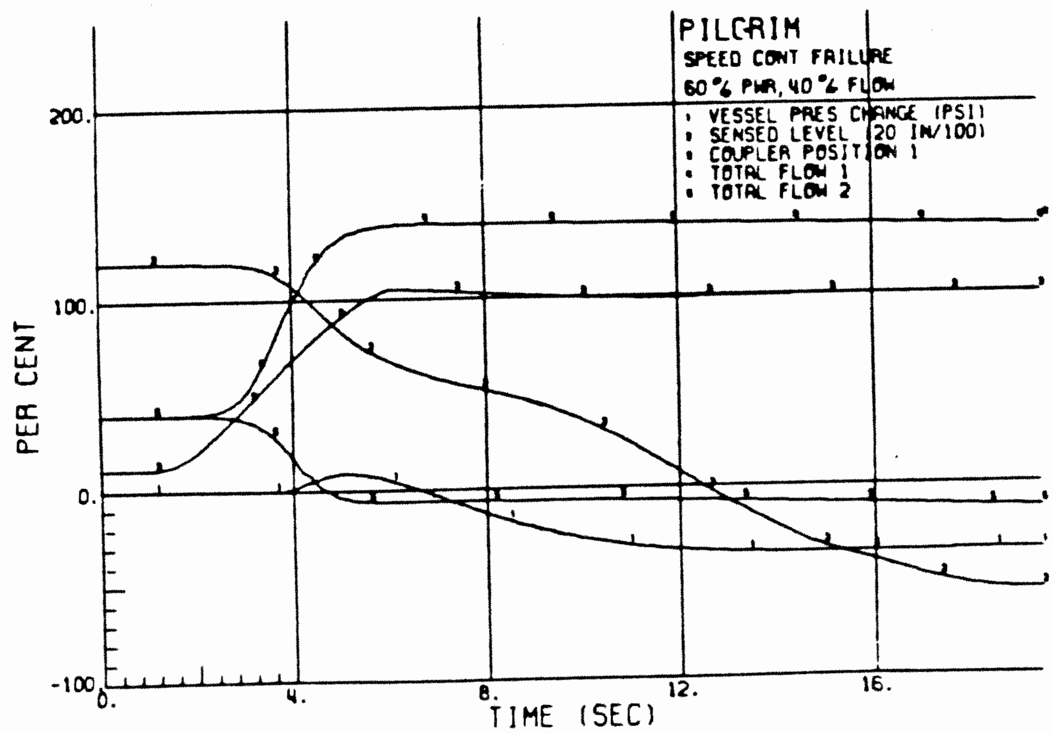


Figure R.2-23. Initial Core Speed Controller Failure Maximum Speed Demand

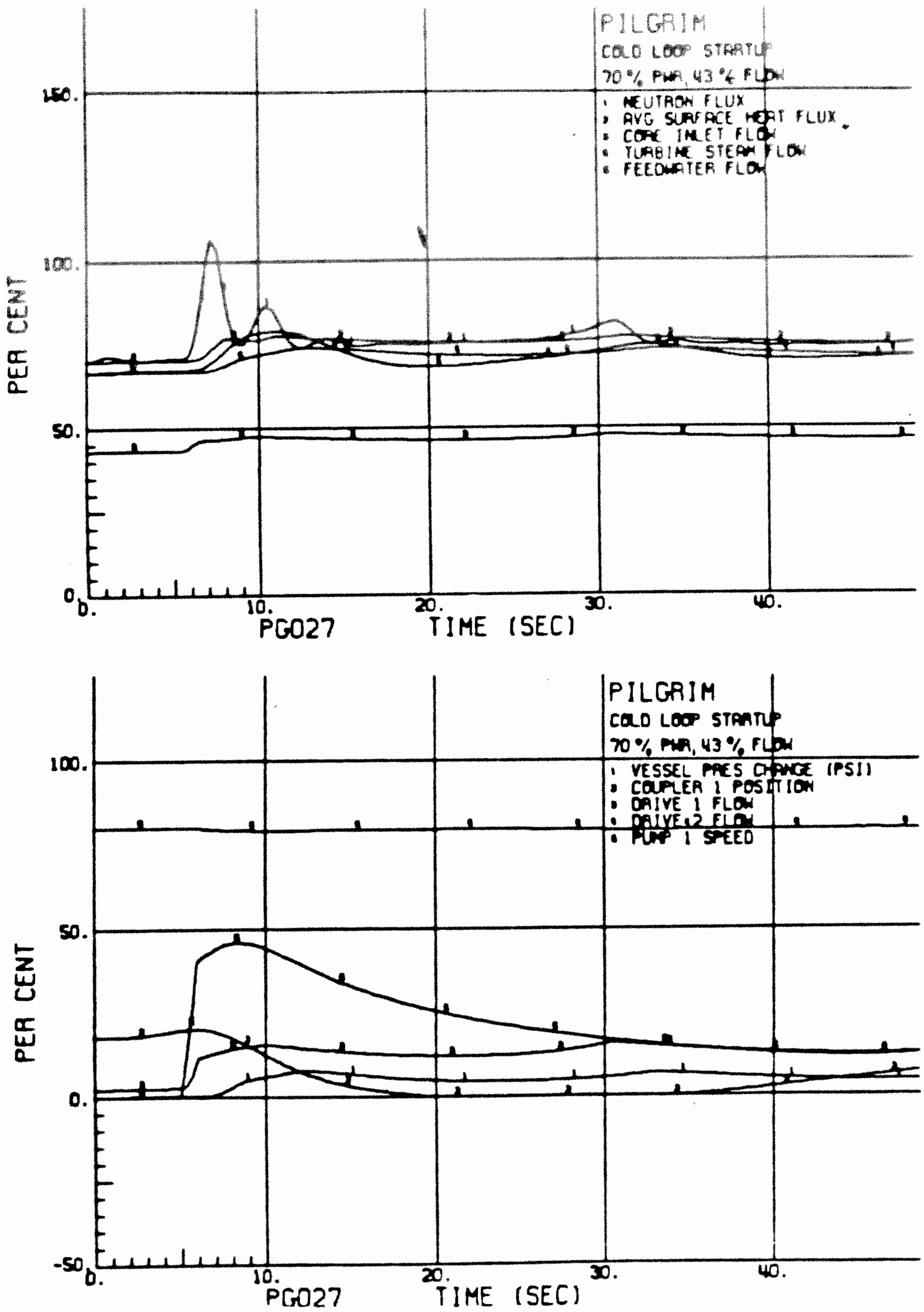
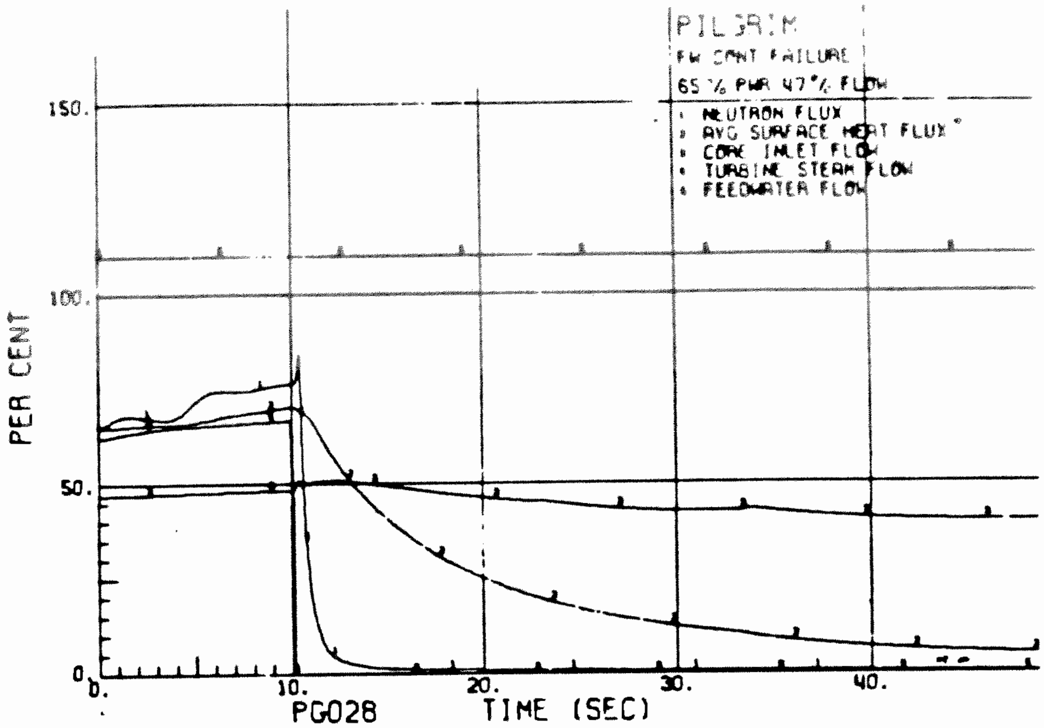


Figure R.2-24. Initial Core Startup of One 100°F Recirculation Loop



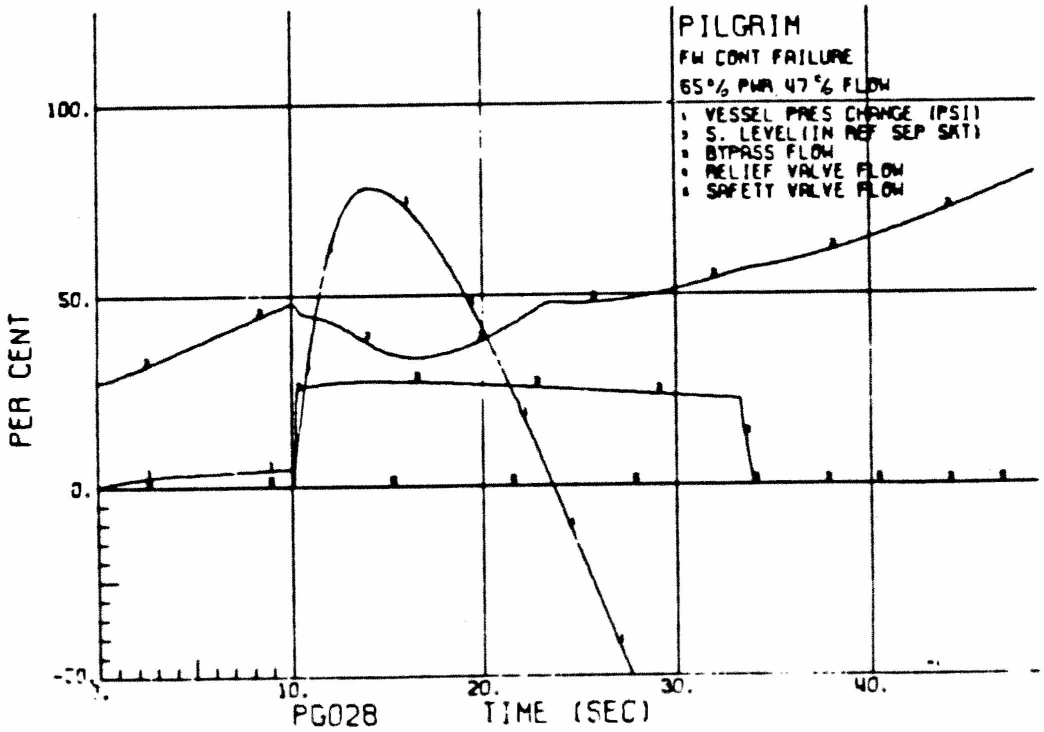


Figure R.2-25. Initial Core Feedwater Controller Failure Maximum Demand

R.3 ANALYSIS OF DESIGN BASIS ACCIDENTS (INITIAL CORE)

R.3.1 Introduction

The methods for identifying and evaluating accidents have resulted in the establishment of design basis accidents for the various accident categories as follows:

	Accident Category	Design Basis Accident
1.	Accidents that result in radioactive material release from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact	Rod drop accident (single control rod)
2.	Accidents that result in radioactive material release directly to the primary containment	Loss of coolant accident (rupture of one recirculation loop)
3.	Accidents that result in radioactive material release directly to the secondary containment with the primary containment initially intact	Accidents in this category are less severe than those in categories 4 and 5 below
4.	Accidents that result in radioactive material releases directly to the secondary containment with the primary containment not intact	Refueling accident (fuel assembly drops on core during refueling)
5.	Accidents that result in radioactive material releases outside the secondary containment	Steam line break accident (main steam line breaks outside of secondary containment). Radwaste System accidents (failures in systems containing significant amounts of radioactivity)

An investigation of accident possibilities reveals that accidents in Category 3 are less than those in Categories 4 and 5. There are two varieties of accidents in Category 3: Failures of the nuclear system process barrier inside the secondary containment and failures involving fuel that is located outside the primary containment but inside the secondary containment. Under the accident selection rules, a main steam line break inside the Reactor Building is the most severe accident of the first variety, but this accident results in a radioactivity release to the environs no greater than that resulting from the main

steam line break outside the secondary containment. Similarly, the most severe accident of the second variety is the dropping of a fuel assembly in the spent fuel pool, but this results in a smaller radioactivity release to the environs than that resulting from dropping a fuel assembly on the fuel in the reactor vessel during refueling. Because the consequences of accidents in Category 3 are less severe than those resulting from similar accidents in other categories, the accidents in Category 3 are not described.

R.3.2 Control Rod Drop Accident

Accidents that result in releases of radioactive material from the fuel with the nuclear system process barrier, primary containment, and secondary containment initially intact can result from various failures of the Control Rod Drive System. Examples of such failure are: Collet finger failures in one control rod drive mechanism, a Control Rod Drive System pressure regulator malfunction, and a control rod drive mechanism ball check valve failure. None of the single failures associated with the control rods or the Control Rod System result in a greater release of radioactive material from the fuel than the release that results when a single control rod drops out of the core after being disconnected from its drive and after the drive has been retracted to the fully withdrawn position. Thus, this control rod drop accident is established as the design basis accident for the category of accidents resulting in radioactive material release from the fuel with all other barriers initially intact. A highly improbable combination of events would be required for the design basis control rod drop accident to occur.

The following events are required for the design basis control rod drop accident to occur.

1. Failure of the rod to drive coupling. The design of the coupling itself reduces the probability of separation. Tests conducted under both simulated reactor conditions and the conditions more extreme than those expected in reactor service have shown that the coupling does not separate, even after thousands of scram cycles. Tests also show that the coupling does not separate when subjected to forces 30 times greater than that which can be achieved with a control rod drive.

2. Sticking of the control rod in its fully inserted position as the drive is withdrawn. The control rods are designed to minimize the probability of sticking in the core. The control rod blades, which are equipped with rollers that make contact with the channel walls, travel in gaps between the fuel assemblies with approximately 1/2 in total clearance. Control rods of similar design, now in use in operating reactors, have exhibited no tendency to stick in the core due to distortion or swelling of the blade.

3. Full withdrawal of the control rod drive.

4. Failure of the operator to notice the lack of response of neutron monitoring channels as the rod drive is withdrawn.

5. Failure of the operator to verify rod coupling. The control rod bottoms on a seal, preventing the control rod drive from being withdrawn to the overtravel position. Attempting to withdraw a control rod drive to the overtravel position provides a method for verifying rod coupling. This verification is required whenever neutron monitoring equipment response does not indicate that the rod is following the drive.

6. Control rod later becomes loose and falls freely to the fully withdrawn position.

The accident is analyzed over the full spectrum of power conditions. Nuclear excursion results are presented for three points in this range: The cold (68°F) critical condition for moderator and fuel, a hot (547°F) critical condition, and the 10 percent of rated power condition. The results of the rod drop accident initiated from higher than 10 percent power are less severe than the 10 percent power case because of the faster Doppler response. The radiological results of the most severe case are presented.

R.3.2.1 Initial Conditions

The following initial conditions are assumed for the three cases presented:

Case A (cold):	Reactor critical Moderator and fuel at 68°F Power level 10^{-8} x rated Rod worth (for dropped rod) 0.025 Vk
Case B (hot):	Reactor critical Moderator and fuel at 547°F Power level 10^{-6} x rated Rod worth (for dropped rod) 0.025 Vk
Case C (power):	Reactor critical Moderator and fuel at 547°F Power level 10^{-1} x rated Rod worth (for dropped rod) 0.038 Vk

In considering the possibilities of a control rod drop accident, only the rod worths of the lower curve of Figure R.3-1, are pertinent at less than 10 percent power. these are the rods which are normally allowed to be moved by operating procedures and the rod worth minimizer. The non-scheduled rods, those described by the central envelope, do not have a withdrawal permissive during the time their worths are greater than the lower curve, so they are held full in by the control rod drive and cannot drop from the core. If a non-scheduled rod were selected, the rod worth minimizer blocks rod movement. Therefore, the worth of the strongest rod which could be

stuck is limited to about 0.01 k, and the 0.025 Vk worth assumed for Cases A and B is considerably above the rod worth values available for stuck rods under the assumed reactor conditions. In the greater than 10 percent power range, the maximum rod worth is determined by the FLARE[1] and WANDA[2] computer codes and is shown on Figure R.3-2. thus, in Case C the rod worth is assumed to be 0.038 Vk.

R.3.2.2 Excursion Analysis Assumptions

The following assumptions are used in the analysis of the nuclear excursion for each case:

1. The velocity at which the control rod falls out of the core is assumed to be 5 ft/sec. The control rod velocity limiter[3], an engineered safeguard, limits the rod drop velocity to less than this value

2. Control rod scram motion is assumed to start at about 200 milliseconds after the neutron flux has attained 120 percent of rated flux. This assumption allows the power transient to be terminated initially by the Doppler reactivity effect of the fuel. This assumption is particularly conservative for Cases A and B because a high neutron flux scram would be initiated earlier by the intermediate range neutron (IRM) monitoring channels

3. No credit is taken for the negative reactivity effect resulting from the increased temperature of, or void formation in the moderator because the time constant for heat transfer between the fuel and the moderator is long compared with the time required for control rod motion

4. No credit is taken for the prompt negative effect of heating in the moderator due to gamma heating and neutron thermalization

R.3.2.3 Fuel Damage

Fuel rod damage estimates are based upon the UO_2 vapor pressure data of Ackerman[4] and interpretation of all the available SPERT, TREAT, KIWI, and PULSTAR test results which show that the immediate fuel rod rupture threshold is about 425 cal/g. Two especially applicable sets of data, as far as fuel failure thresholds are concerned, come from the PULSTAR[5] and ANL-TREAT[6],[7] tests.

The PULSTAR tests, which used UO_2 pellets of 6 percent enrichment with Zr-2 cladding, achieved maximum fuel enthalpies of about 200 cal/g with a minimum period of 2.83 msec. The coolant flow was by natural convection. Film boiling occurred and there were local clad bulges; however, fuel pin integrity was maintained and there were no abnormal pressure rises.

The two ANL-TREAT tests used Zircaloy clad UO_2 pins with energy inputs of 280 and 450 cal/g.

	Test 1	Test 2
Input Energy (cal/g)	280	450
Final Mean Particle Diameter (mils)	60	30
Pressure Rise Rate (psi/sec)	30	600

The ultimate degree of fuel fragmentation and dispersal of the two cases is not significantly different; however, the pressure rise rate in the higher energy test is increased by a factor of 20. This strongly implies that the dispersion rate in the higher energy test was significantly higher than that of the lower energy test. This leads to the logical conclusion that although a high degree of fragmentation occurs for fuel in the 200 to 300 cal/g range, the breakup and dispersal into the water is gradual and pressure rise rates are very modest. On the other hand, for fuel above the 400 cal/g range, the breakup and dispersal is prompt and much larger pressure rise rates are probable.

Based on the analysis of the above referenced data, it is estimated that 170 cal/g is the threshold for eventual fuel cladding perforation. Fuel melting is estimated to occur in the 220 to 280 cal/g range and a minimum of 425 cal/g is required to cause immediate rupture of the fuel rods due to UO_2 vapor pressures.

A parametric analysis was made of the rod drop accident for various starting conditions and rod worths. The results are shown on Figures R.3-3 and R.3-4, and the reduction in final peak fuel enthalpy with increasing initial power level is clearly shown. The cold critical case (Case A) is shown as point A on Figure R.3-3, and the hot standby critical case (Case B) is shown on point B on Figure R.3-4. Figure R.3-5 is a conservative description of the consequences when the core is at rated temperature and the coolant is boiling. Here the 10 percent of power case (Case C) is represented by point C. In these cases, the maximum initial enthalpy generally is not in the fuel which experiences the greatest enthalpy addition during the excursion. If a rod were dropped from a high initial enthalpy region, the results would not be as great as with one dropped from a lower enthalpy region. However, for conservatism, it is assumed that the peak enthalpy increment is added to the maximum fuel enthalpy that existed in the vicinity of the excursion center prior to the accident.

In the worst hot standby critical case (Case B), the power transient is calculated to have a total energy generation of 4,000 MW-sec. The excursion energy is calculated to be distributed in the fuel such that about 330 fuel rods have enthalpies greater than 170 cal/g. The maximum UO_2 enthalpy is calculated to be 220 cal/g. Essentially no fuel will melt because fuel melting occurs in the range from 220 to 280 cal/g.

The power transient in the 10 percent of power rod drop accident (Case C) is less severe than the one at hot standby (Case B). The peak enthalpy is about 200 cal/g and only about 50 fuel rods have enthalpies exceeding 170 cal/g.

The power transient in the cold condition rod drop accident (Case A) is calculated to be distributed in the fuel such that about 200 fuel rods have enthalpies greater than 170 cal/g. The maximum UO_2 enthalpy is calculated to be 250 cal/g. Approximately 50 lb of UO_2 have enthalpies in excess of 220 cal/g. Because fewer fuel rods are perforated and because the Shutdown Cooling System would be operating, allowing no radioactivity release to the main condenser, the radiological results of the cold rod drop accident are insignificant when compared to the worst hot standby case.

All of these peak enthalpies are far below 425 cal/g, which is estimated to be the threshold for immediate rupture of fuel rods due to UO_2 vapor pressure. Furthermore, the above peak enthalpies are well below the design limit of 280 cal/g. Thus, there are no damaging pressure pulses as a result of the rod drop accident, and the only damage expected would be the failed fuel rods.

R.3.2.4 Fission Product Release From Fuel

The following assumptions are used in the calculation of fission product activity release from the fuel.

1. In Case B, 330 fuel rods fail; this is the largest number of failed fuel rods resulting from the analysis of the rod drop accident over the full spectrum of power conditions.

2. The reactor has been operating at design power until 30 min before accident initiation. When translated into actual station operations, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30 min time represents a conservative estimate of the shortest time in which the required station changes could be accomplished, and defines the minimum decay time to be applied to the fission product inventory for the calculations.

3. The reactor has been operating at design power for 1,000 days prior to the accident. This assumption results in equilibrium concentration of fission products in the fuel. Longer operating histories do not significantly increase the concentration of longer lived fission products.

4. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity contained in a fuel rod is in the plenums and available for release if the cladding is perforated. These release percentages are consistent with actual measurements made on defective fuel experiments. The

basis for these values is presented in APED-5756.[8]

5. The following fission product inventories are applicable for the total core at the time the accident occurs:

Noble gases	2.62×10^8Ci
Halogens	4.76×10^8Ci

These inventories are the result of a nuclear analysis of the fuel assuming operation at design power for 1,000 days followed by a 30 min decay period.

6. None of the solid fission products are released from the fuel. Because the fraction of solid fission product activity available for release from the fuel is negligible, this assumption is reasonable.

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

Using the above assumptions the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:

Noble gases (Xe, Kr)	5.48×10^4Ci
Halogens (Br, I)	1.77×10^4Ci

R.3.2.5 Fission Product Transport

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

1. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and 5 percent steam flow are the maximum flow rates expected under the conditions assumed for the accident. The recirculation flow rate is used in determining the volume of coolant in which the activity released from the fuel is deposited. The 5 percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would actually be expected. Because of the relatively long fuel to coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within the time required for the main steam line isolation valves to achieve full closure.

2. The main steam line isolation valves are assumed to receive an automatic closure signal 0.5 sec after the start of the rod drop accident, and to be fully closed at 10 sec from the receipt of the closure signal. The automatic closure signal originates from the main steam line radiation monitors. The 10 sec closure time of the main steam line isolation valves is the maximum closing time permitted by valve setting. The total time required to isolate the main steam lines (10.5 sec), combined with the assumptions in 1., allow calculation of the total amount of fission product activity transported to the condenser before the steam lines are isolated.

3. All of the noble gas activity is assumed released to the steam space of the reactor vessel. None is retained in the liquid reactor coolant.

4. The ratio of the halogen concentration in steam to that of water is assumed to be 3×10^{-5}. Measurements, taken under applicable chemical and physical conditions of reactor coolant water and condensate at Dresden Nuclear Power Station Unit No. 1, indicate that the steam to water halogen concentration ratio is in the range of 1×10^{-5} to 3×10^{-5}.

5. Water carryover in the main steam lines is assumed to be 0.1 percent of the total mass of steam transferred to the condenser. Measurements of the steam separation effected by the same types of separators used in this reactor vessel show that water carryover is less than 0.1 percent even at rated steam flow. The carryover fraction permits computation of the halogn activity carried to the main condenser in the water entrained in the steam.

6. None of the fission products released from the fuel are assumed to plate out.

The main steam line radiation monitors initiate closure of all main steam line isolation valves when a preestablished radiation level is exceeded. This action prevents further transport of the fission products to the condenser.

Using the listed assumptions, the following amounts of fission product activities are carried to the condenser:

Noble gases	3.72×10^{3}Ci
Iodine 131	1.95×10^{0}Ci
Iodine 132	2.99×10^{-1}Ci
Iodine 133	1.03×10^{0}Ci
Iodine 134	2.38×10^{-1}Ci
Iodine 135	5.72×10^{-1}Ci

R.3.2.6 Fission Product Release to Environs

The following assumptions and initial conditions are used in the calculation of fission product activity release to the environs:

1. All of the noble gas activity transferred to the condenser during the assumed 10.5 sec isolation valve closure time is assumed to be airborne in the condenser. The halogen activity transferred to the condenser experiences the removal effects of the condensate and forms an equilibrium condition between the condensate and vapor volume.

2. The mechanical vacuum pump discharge and the air ejector offgas line are assumed to be isolated upon high radiation signals from the main steam line radiation monitors and the air ejector radiation monitors.

3. The condenser was assumed to be at atmospheric pressure so that outleakage of the activity could occur into the Turbine Building. A constant 0.5 percent/day condenser leak rate was assumed. This outleakage was conservatively assumed to escape from the Turbine Building through the Turbine Building roof exhausters. Based upon these conditions, the fission product release rate to the environment is shown on Table R.3-1.

R.3.2.7 Radiological Effects

The radiological exposure resulting from the activity discharged to the environment have been determined for the following meteorological conditions. These conditions range from very stable to unstable and consider a wind speed of 3 m/sec with turbulent building wake effects. Table R.3-2 shows the calculated offsite exposures beyond the nearest site boundary, which is approximately 0.2 mi from the release point. The maximum 24 hr whole body and thyroid doses are 1.6 x 10^{-5} rem and 8.2 x 10^{-3} rem, respectively. These doses are well within the guideline doses of 10CFR100, namely 25 rem whole body and 300 rem thyroid.

R.3.3 Loss of Coolant Accident (Pipe Break Inside Primary Containment)

Accidents that could result in release of radioactive material directly into the primary containment are the results of postulated nuclear system pipe breaks inside the drywell. All possibilities for pipe break sizes and locations have been investigated including the severance of small pipe lines, the main steam lines upstream and downstream of the flow restrictors, and the recirculation loop pipelines. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop pipelines. This accident is established as the design basis loss of coolant accident.

R.3.3.1 Initial Conditions and Assumptions

The analysis of this accident is performed using the following assumptions:

1. The reactor is operating at the most severe condition at the time the recirculation pipe breaks, which maximizes the parameter of interest; primary containment response, fission product release, or Core Standby Cooling System (CSCS) requirements.

2. A complete loss of normal ac power occurs simultaneously with the pipe break. This additional condition results in the longest delay time for the CSCS to become operational.

3. The recirculation loop pipeline is considered to be instantly severed. This results in the most rapid coolant loop and depressurization with coolant discharged from both ends of the break.

R.3.3.2 Nuclear System Depressurization and Core Heatup

In Section 6, core Standby Cooling Systems, the initial phases of the loss of coolant accident are described and evaluated. Included in that description are the rapid depressurization of the nuclear system, the operating sequences of the CSCS, the heatup of the fuel, and the perforation of fuel rods. Analysis of the initial core showed that a maximum of 9.0 percent of the fuel rods reach the pressure and temperature conditions necessary for perforation.

R.3.3.3 Primary Containment Response

See Section 14.5.3.1.

R.3.3.4 Fission Products Released to Primary Containment

The following assumptions and initial conditions were used in calculating the amounts of fission products released from the nuclear system to the drywell.

1. The reactor has been operating at design power of 1,998 MWt for 1,000 days prior to the recirculation pipe break. This assumption results in equilibrium concentrations of fission products in the fuel. Longer operating histories would not increase the concentrations of the longer lived fission products of significance.

2. Twenty-five percent of the fuel rods in the core are conservatively assumed to be perforated. This is about a factor of three above the predicted extent of fuel damage and accounts for any uncertainties in the fuel damage prediction.

3. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity contained in a fuel rod is in the plenums and available for release if the cladding is perforated.(8) These percentages are consistent with measurements made on defective fuel experiments.

4. Due to negligible particulate activity available for release from nonmolten fuel, none of the volatile or nonvolatile radioactive solids are assumed to be carried out of the reactor vessel during the accident.

5. All of the noble gases and halogens released from the perforated fuel rods are assumed to be transported to the drywell.

6. Fission product decay during depressurization is neglected.

7. The equilibrium fission product activity is:

Noble gases	5.26×10^8Ci
Halogens	5.77×10^8Ci

The fission product inventory reflects an assumed 1,000 days at design power followed by a decay period of 1 min. The 1 min assumption results in the decay of the very short lived fission products which contribute significantly to the fission products in the fuel but are insignificant as far as plenum activity is concerned.

8. The only mechanisms which will reduce the noble gas concentration will be radioactive decay and leakage to the secondary containment. This is because the noble gases released from the damaged fuel elements will not chemically combine with the liquid in the suppression pool, nor will they be removed by plateout or fallout.

9. The halogen activity released to the primary containment will experience such removal effects as washout, fallout, plateout, and removal by the pressure suppression pool. The affects of washout and plateout have been shown by various investigators to vary between 10 and 1,000.(8) A value of 2 has been conservatively chosen to represent these removal effects for this analysis.

10. An iodine partition factor has been conservatively assumed to be 102. Numerous experiments have also been conducted to investigate the iodine partition factor between water and air. The results of these experiments(8) show that a partition factor of 10^3 to 10^8 is appropriate for conditions existing in the primary containment as a result of a loss of coolant accident.

11. As a consequence of releasing elemental iodine into the primary containment the possibility exists that some of this activity may be converted to an organic iodine which is generally removed to a lesser extent by the various removal mechanisms previously discussed than is elemental iodine. Various experiments[8] have been conducted which show that for the conditions representative of a loss of coolant accident the conversion ratio of elemental iodine to methyl iodine varies between 0.001 percent and 1.0 percent. The conversion ratio chosen for this analysis is 1.0 percent. In addition to this high conversion ratio it is also conservatively assumed that the only removal mechanism affecting the organic iodines within the primary containment is radioactive decay.

Using the previous assumptions, the calculations result in the following amounts of fission products released to the drywell through the pipe break:

Noble gases	1.4×10^6Ci
Iodine 131	8.85×10^4Ci
Iodine 132	1.36×10^4Ci
Iodine 133	4.72×10^4Ci
Iodine 134	1.25×10^4Ci
Iodine 135	2.72×10^4Ci

The noble gas activity, and the sum of the elemental and organic iodines airborne in the primary containment is shown on Table R.3-3.

R.3.3.5 Fission Product Release to Secondary Containment

The fission product activity in the secondary containment at any time is a function of the leakage rate from the primary containment and the volumetric discharge rate from the secondary containment. During normal power operation the Reactor Building ventilation rate is one air change per hour, however, the normal ventilation system is isolated and Standby Gas Treatment System operation initiated (one air change per day) as a result of low reactor water level, high drywell pressure, or high radiation in the refueling ventilation exhaust ductwork. For ease of analysis, the primary containment leak rate is assumed to be constant at 0.5 percent/day. Any fission product removal effects in the secondary containment such as plateout, fallout, etc. are neglected. However, the effects of decay are considered. Based upon these values, the Reactor Building fission product inventory is calculated to reach concentrations as noted on Table R.3-4. Maximum values are attained in approximately 2 days.

R.3.3.6 Fission Product Release to Environs

The fission product activity released to the environs is dependent upon the fission product inventory airborne in the secondary containment, the volumetric flow from the secondary containment, and

the efficiency of the various components of the Standby Gas Treatment System.

The following assumptions and initial conditions were used in calculating the fission product release to the environs:

1. The activity airborne in the secondary containment is as presented on Table R.3-4.

2. A filter efficiency of 99 percent is used for iodine and 0 percent for the noble gases. The Standby Gas Treatment System also contains a demister (use of the demister is optional) for the removal of entrained water droplets and electric heaters for heating the incoming air to reduce the relative humidity of the incoming mixture to at least 70 percent.

3. There is one secondary containment air change per day through the Standby Gas Treatment System

Based upon these conditions, the fission product activity being released to the environs is shown on Table R.3-5.

R.3.3.7 Radiological Effects

The radiological exposures resulting from the activity released to the environment as a consequence of the loss of coolant accident have been determined for six meteorological conditions. These conditions range from very stable to unstable and consider wind speeds of 1 m/s and 5 m/s.

Table R.3-6 shows the calculated exposures beyond the nearest site boundary (approximately 0.2 mile from the release point). The values on Table R.3-6 assume a flat site; however, if terrain is considered, the doses will increase somewhat.

The most significant terrain feature is Manomet Hill located about 2,200 m SSW of the main stack. Assuming that the elevated plume intersects the ground at this location, a 1 m/s wind speed, and very stable atmospheric conditions, the 24 hour whole body and thyroid inhalation doses are 4.5×10^{-2} rem and 2.0×10^{-3} rem respectively. These doses, as well as those shown on Table R.3-7 are below the guideline values of 10 CFR 100, 25 rem whole body and 300 rem thyroid.

If consideration is given to a 30 day dose, the 24 hour cloud gamma dose values presented on Table R.3-6 would increase by a factor of 2.7 and the 24 hour inhalation doses would increase by a factor of 2.8. It is concluded that this accident will not result in any radiological exposures which endanger the health and safety of the public.

R.3.4 Refueling Accident

Accidents that result in the release of radioactive materials directly to the secondary containment are events that can occur when the primary containment is open. A survey of the various station conditions that could exist when the primary containment is open reveals that the greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the Reactor Building.

Various mechanisms for fuel failure under this condition have been investigated. The refueling interlocks, which impose restrictions on the movements of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. The RPS is capable of initiating a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. The possibility of mechanically damaging the fuel has been investigated.

The design basis accident for this case is one in which one fuel assembly is assumed to fall onto the top of the reactor core.

R.3.4.1 Assumptions

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

2. The entire amount of potential energy, referenced to the top of the reactor core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple, or the grapple cable breaks.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (UO_2).

R.3.4.2 Fuel Damage

Dropping a fuel assembly onto the reactor core from the maximum height allowed by the refueling equipment, less than 30 ft, results in an impact velocity of 40 ft/sec. The kinetic energy acquired by the falling fuel assembly is less than 17,000 ft lb and is dissipated in one or more impacts.

The first impact is expected to dissipate most of the energy and cause the largest number of cladding failures. To estimate the

expected number of failed fuel rods in each impact, an energy approach has been used.

The fuel assembly is expected to impact on the reactor core at a small angle from the vertical possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. Fuel rods are expected to absorb little energy prior to failure due to bending. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Actual bending tests with concentrated point loads show that each fuel rod absorbs about 1 ft lb prior to cladding failure. For rods which fail due to gross compression distortion, each rod is expected to absorb about 250 ft lbs before cladding failure (this is based on 1 percent uniform plastic deformation of the rods). The energy of the dropped assembly is absorbed by the fuel, cladding, and other core structure. A fuel assembly consists of about 72 percent fuel, 11 percent cladding, and 17 percent other structural material by weight. Thus, the assumption that no energy is absorbed by the fuel materials inserts considerable conservatism into the mass energy calculations that follow.

The energy absorption on successive impacts is estimated by consideration of a plastic impact. Conservation of momentum under a plastic impact shows that the fractional kinetic energy absorbed during impact is where M_1 is the impacting mass and M_2 is the struck

$$1 - \frac{M_1}{M_1 + M_2}$$

mass. Based on the fuel geometry within the reactor core, four fuel assemblies are struck by the impacting assembly. The fractional energy loss on the first impact is about 80 percent.

The second impact is expected to be less direct. The broad side of the dropped assembly impacts approximately 24 more fuel assemblies, so that after the second impact only 136 ft lb (about 1 percent of the original kinetic energy) is available for a third impact. Because a single fuel rod is capable of absorbing 250 ft lb in compression before cladding failure, it is unlikely that any fuel rods fail on a third impact.

If the dropped fuel assembly strikes only one or two fuel assemblies on the first impact, the energy absorption by the core support structure results in about the same energy dissipation on the first impact as in the case where four fuel assemblies are struck. The energy relations on the second and third impacts remain about the same as in the original case. Thus, the calculated energy dissipation is as follows:

First impact	80 percent
Second impact	19 percent
Third impact	1 percent (no cladding failures)

The first impact dissipates 0.80 x 17,000 or 13,600 ft lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure, and because 1 ft lb of energy is sufficient to cause cladding failure due to bending, all 49 rods of the dropped fuel assembly are assumed to fail. Because the 8 tie rods of each struck fuel assembly are more susceptible to bending failure than the other 41 rods, it is assumed the they fail upon the first impact. Thus 4 x 8 = 32 tie rods (total in four assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod due to compression, 250 ft lbs of energy is required. To cause failure of all the remaining rods of the four struck assemblies, 250 x 41 x 4 or 41,000 ft lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures due to compression is computed as follows:

$$\frac{0.5 \times 13,600 \times \left(\frac{11}{11 + 17}\right)}{250} = 11$$

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

Dropped assembly	49 rods (bending)
Struck assemblies	32 tie rods (bending)
Struck assemblies	11 rods (compression)
Total	92 failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie rods subjected to bending failure. Thus, 2 x 8 = 16 tie rods are assumed to fail. The number of fuel rod failures due to compression on the second impact is computed as follows:

$$\frac{\left(\frac{19}{2}\right) \times 17,000 \times \left(\frac{11}{11 + 17}\right)}{250} = 3$$

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

Struck assemblies	16 tie rods (bending)
Struck assemblies	3 rods (compression)
Total	19 failed rods

The total number of failed rods resulting from the accident is as follows:

First impact	92 rods
Second impact	19 rods
Third impact	0 rods
Total	111 failed rods

R.3.4.3 Fission Product Release from Fuel

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

1. The reactor fuel has an average irradiation time of 1,000 days at design power up to 24 hr prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern. The 24 hr decay time allows time for the reactor to be shut down, the nuclear system depressurized, the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected the these operations could be accomplished in less than 24 hrs.

2. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plenums and available for release. This assumption is based on fission product release data from defective fuel experiments.[8]

3. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products are assumed to be released from the fuel.

4. Failure is assumed in 111 fuel rods. This was the conclusion of the analysis of mechanical damage to the fuel.

5. The fission product activity contents of the fuel at the time of the accident are as follows:

Noble gases	1.45×10^8 Ci
Halogens	1.81×10^8 Ci

These activity contents are the result of an analysis of the fission product inventories in the core assuming equilibrium conditions at design power followed by a 24 hr decay period.

Using these assumptions, the following amounts of fission product activity are released from the fuel to the water in the reactor vessel as a result of the dropped fuel assembly:

Noble gases	1.21×10^4Ci
Iodine 131	2.13×10^3Ci
Iodine 132	2.92×10^2Ci
Iodine 133	5.69×10^2Ci
Iodine 134	6.2×10^{-6}Ci
Iodine 135	6.83×10^1Ci

R.3.4.4 Fission Product Release to the Secondary Containment

The following assumptions and initial conditions are used in calculating the fission product release to the secondary containment.

1. The fission product activity released to the refueling pool will be released to the secondary containment in proportion to the removal efficiency of the water in the refueling pool. Since water has a poor affinity for the noble gases they are assumed to be instantaneously released from the pool to the secondary containment.

2. As noted in Section R.3.3.5, the removal efficiency of the water for halogens can be defined in terms of the partition factor, for which values between 10^3 and 10^5 have been experimentally determined to be applicable for the conditions under investigation. A partition factor of 10^2 for the halogens has been conservatively assumed for this accident. Thus the computed inhalation exposures will be overestimated by a factor of 10 to 10^3.

3. The conservative assumption is also made that instantaneous equilibrium is attained between the refueling pool and secondary containment. In reality a true equilibrium will never by achieved. However, by assuming an instantaneous equilibrium condition, the resultant radiological exposures will be maximized.

4. The effects of plateout and fallout are neglected. Fission product plateout and/or fallout will occur in the secondary containment; but for the assumption that a true equilibrium is maintained, the effects of plateout or fallout would be compensated for by the evolutions of activity from the refueling pool.

5. The refueling cavity liquid volume is 3.6×10^4 ft^3 and the effective air volume in the secondary containment is 7.0×10^5 ft^3 above the refueling floor.

6. The Standby Gas Treatment System removes one secondary containment air volume per day.

Based upon these assumptions, the airborne activity is as shown on Table R.3-7.

R.3.4.5 Fission Product Release to Environs

The following assumptions and initial conditions are used in calculating the fission product release to the environs.

1. High radiation levels in the Reactor Building refueling ventilation exhaust will isolate the normal ventilation system and actuate the Standby Gas Treatment System.

2. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident, except for the addition of the released fission products. The relative humidity in the secondary containment will therefore be considerably below any levels which may be detrimental to the filter media in the Standby Gas Treatment System. However, as mentioned previously, the air flowing through the filter system is heated to reduce the relative humidity to 70 percent or less.

3. The filter efficiency is assumed to be 99 percent for iodines and 0 percent for the noble gases.

4. There is one secondary containment air change/day through the Standby Gas Treatment System.

5. The activity airborne in the secondary containment is shown on Table R.3-7.

Based upon these conditions, the fission product activity release rate to the environs is as shown on Table R.3-8.

R.3.4.6 Radiological Effects

The radiological exposures to the general population have been evaluated for six meteorological conditions ranging from very stable to unstable meteorology occurring with 1 m/s and 5 m/s winds. Two exposure periods have been evaluated, a 2 hr. exposure period and a 24 hr. exposure period, commonly referred to as the total dose. It should be emphasized that the radiological exposures presented on Table R.3-9 are based upon assumption that the stated meteorological conditions exist for the duration under consideration and that the wind blows in one direction during the entire release period.

Table R.3-9 shows the calculated offsite exposure beyond the nearest site boundary (approximately 0.2 mi from the release point). The values on Table R.3-9 assume a flat site; however, if terrain is considered, the doses will increase somewhat. The most significant terrain feature is Manomet Hill located about 2,200 m SSW of the main stack. Assuming that the elevated plume intersects the ground at

this location, a 1 m/s wind speed, and very stable atmospheric conditions, the 24 hr whole body and thyroid inhalation doses are 2.1 x 10^{-2} rem and 1.0 x 10^{-1} rem respectively. These doses, as well as those shown on Table R.3-9, are below the guideline whole body and lifetime thyroid doses of 25 rem and 300 rem (10CFR100), respectively.

If consideration is given to a 30 day dose, the 24 hr cloud gamma values would be increased by a factor of 1.08 while the thyroid inhalation dose would be increased by a factor of 1.22.

If is concluded that this accident will not result in any radiological exposures which endanger the health and safety of the public.

R.3.5 Main Steam Line Break Accident (Pipe Break Outside of Primary Containment

Accidents that result in the release of radoiactive materials outside the secondary containment are the results of postulated breaches in the nuclear system process barrier. The design basis accident (worst radiological effects) is a complete severance of one main steam line outside the secondary containment. Figure R.3-6 shows the break location. The analysis of the accident is described in three parts.

1. Nuclear System Transient Effects

   This includes analysis of the changes in nuclear system parameters pertinent to fuel performance and the determination of fuel damage.

2. Radioactive Material Release

   This includes determination of the quantity and type of radioactive material released through the pipe break and to the environs.

3. Radiological Effects

   This portion determines the dose effects of the accident to offsite persons.

R.3.5.1 Nuclear System Transient Effects

R.3.5.1.1 Assumptions

The following assumptions are used in evaluating response of nuclear system parameters to the steam line break accident outside the secondary containment.

1. The reactor is operating at design power.

2. Reactor vessel water level is normal for initial power level assumed at the time the break occurs.

3. Nuclear system pressure is normal for the initial power level.

4. The steam pipeline is assumed to be instantly severed by a circumferential break. The break is physically arranged so that the coolant discharge through the break is unobstructed. These assumptions result in the most severe depressurization of the nuclear system.

5. The main steam line isolation valves are assumed to be closed 10.5 seconds after the break. This assumption is based on the 0.5 second time required for the development of the automatic isolation signal (high differential pressure across the main steam line flow restrictor) and the 10-second closure time for the valves, which is the maximum setting for valve closure time.

   Faster main steam line isolation valve closure could reduce the mass loss until finally some other process line break would become controlling. However, the resulting radiological dose for this break would be less than the main steam line break with a 10-second valve closure. Thus, the postulated main steam line break outside the primary containment with a 10-second isolation valve closure results in maximum calculated radiological dose and is therefore the design basis accident.

6. The mass flow rate through the upstream side of the break is assumed not to be affected by isolation valve closure until the isolation valves are closed far enough to establish limiting critical flow at the valve location. After limiting critical flow is established at the isolation valve, the mass flow is assumed to decrease linearly as the valve is closed. This assumption results in an almost constant mass flow out of the break until the last 3 to 4 seconds of a 10 second valve closure.

7. The mass flow rate through the downstream side of the break is assumed not to be affected by the closure of the isolation valves in the unbroken steam lines until those valves are far enough closed to establish limiting critical flow at the valves. After limiting critical flow is established at the isolation valve positions, the mass flow is assumed to decrease linearly as the valves close. This assumption results in an almost constant mass flow through the break until the last 3 to 4 seconds of a 10 second valve closure.

8. In calculating the rate of water level rise inside the vessel, it is assumed that the steam bubbles formed during depressurization rise at an average velocity of about 1 foot/second relative to the liquid. This assumption is predicted by analysis [11,12] and confirmed experimentally.[13]

9. After the level of the mixture inside the reactor vessel rises to the top of the steam dryers, the break flow is assumed to change from steam to a two phase mixture.

10. Feedwater flow is assumed to decrease linearly to zero over the first 4 sec to account for the reduction in feedwater flow in response to the rise in reactor vessel water level.

11. A loss of offsite auxiliary ac power is assumed to occur simultaneously with the break. This results in the immediate loss of power to the recirculation pumps. Recirculation flow is assumed to coast down according to momentum computations for the Recirculation System.

12. Recirculation System drive pump head is assumed to be zero when the coolant in the reactor vessel downcomer reaches 1 percent quality. This assumption accounts for the effects of cavitation on recirculation drive pump performance as the pumps coast down.

R.3.5.1.2 Sequence of Events

The sequence of events following the postulated main steam line break is as follows.

The steam flow through both ends of the break increases to the value limited by critical flow considerations. The flow from the upstream side of the break is limited initially by the main steam line flow restrictor. The flow from the downstream side of the break is limited by the total area of the flow restrictors in the three unbroken lines. Main steam line isolation valve closure signals are generated as the differential pressure across the main steam line flow restrictors increase about isolation setpoints. The instruments sensing flow restrictor differential pressure generate isolation signals within about 500 milliseconds after the break occurs.

A reactor scram is initiated as the main steam line isolation valves begin to close. See Section 7.2, Reactor Protection System. In addition to the scram initiated from main steam line isolation valve closure, voids generated in the moderator during depressurization contribute significant negative reactivity to the core even before the scram is complete. Because the main steam line flow restrictors are sized for the main steam line break accident, reactor vessel water level remains above the top of the fuel throughout the transient.

R.3.5.1.3 Coolant Loss and Reactor Vessel Water Level

See Section 14.5.4.1 for coolant loss analysis.

Analysis of fuel conditions reveals that no fuel rod perforations due to high temperature occur during the depressurization, even with the conservative assumptions regarding the operation of the Recirculation and Feedwater Systems. The minimum critical heat flux ratio (MCHFR) remains above 1.0 at all times during the transient. See Section 6. No fuel rod failures due to mechanical loading during the depressurization occur because the differential pressures resulting from the transient do not exceed the designed mechanical strength of the core assembly. After the main steam line isolation valves close (10.5 seconds) depressurization stops and natural convection is established through the reactor core. No fuel cladding perforation occurs even if the stored thermal energy in the fuel were simply redistributed while natural convection is being established; cladding temperature would be about 1,000°F, well below the temperatures at which cladding can fail. Thus, it is concluded that even for a 10.5 second main steam line isolation valve closure, fuel rod perforations due to high temperature do not occur. For shorter valve closure times, the accident is less severe. After the main steam isolation valves are closed, the reactor can be cooled by operation of Normal or Standby Cooling System. The core flow and MCHFR during the first 10.5 seconds of the accident are shown on Figures R.3-27 and R.3-28. Since the MCHFR never drops below 1.0, the core is always cooled by very effective nucleate boiling.

R.3.5.2 Radioactive Material Release

R.3.5.2.1 Assumptions

The following assumptions are used in the calculation of the quantity and types of radioactive material released from the nuclear system process barrier outside the secondary containment.

1. The amounts of steam and liquid discharged are as calculated from the analysis of the nuclear system transient.

2. The concentrations of biologically significant radionuclides contained in the coolant discharged as liquid are as follows (zero decay):

Iodine 131	6.1×10^{-2} m Ci/cc
Iodine 132	3.0×10^{-1} m Ci/cc
Iodine 133	3.6×10^{-1} m Ci/cc
Iodine 134	4.3×10^{-1} m Ci/cc
Iodine 135	4.4×10^{-1} m Ci/cc
Other halogens	2.9×10^{-1} m Ci/cc
Molybdenum-Technetium 99m	6.4×10^{-1} m Ci/cc

Because of the steam to water halogen concentration ratio is on the order of 3×10^{-5} (8) only the halogens carried out

of the reactor vessel by the liquid phase during the discharge of the steam water mixture are significant. The coolant activity contents are based on data derived from reactor operation with an unusually large number of cladding failures, and an unusually high normal stack gas discharge rate which increases the activities attributed to each of the listed radioisotopes. Thus considerable conservation is inserted into the analysis.

3. The noble gas discharge rate, after 30 minute holdup, is assumed to be 0.1 Ci/second, an unusually high normal discharge rate. This assumption permits direct computation of the amount of noble gas activity leaving the reactor vessel at the time of the accident. The result is that 0.54 Ci of noble gas activity leaves the reactor vessel each second that the isolation valve is open.

4. It is assumed that the main steam line isolation valves are fully closed at 10.5 seconds after the pipe break occurs. This allows 500 milliseconds for the generation of the automatic isolation signal and 10 seconds for the valves to close. The valves and valve control circuitry are designed to provide main steam line isolation in no more than 10.5 sec. The actual closure time setting for the isolation valves is less than 10 seconds.

5. Due to the short half-life of nitrogen 16, the radiological effects from this isotope are of no major concern and are not considered in the analysis.

R.3.5.2.2 Fission Product Release from Break

Using the previously stated assumptions, the following amounts of radioactive materials are released from the nuclear system process barrier:

Noble gases	5.7×10^0Ci
Iodine 131	1.6×10^0Ci
Iodine 132	8.3×10^0Ci
Iodine 133	1.0×10^1Ci
Iodine 134	1.2×10^1Ci
Iodine 135	1.3×10^1Ci

These releases take into account the total amount of liquid released as well as the liquid converted to steam during the accident.

R.3.5.2.3 Fission Product Release to Environs

The following initial conditions and assumptions are used in calculating the fission product release to the environs:

1. Additional flashing to steam of the liquid exiting from the steam line break will occur due to its superheated condition in the atmosphere.

2. The steam cloud rises and is released from the turbine building roof exhausters in a matter of seconds. While the effect of steam cloud rise is a physical reality, this effect has been neglected for this accident and the assumption is made that the steam cloud remains at the elevation of the turbine building.

3. All of the activity released from the reactor vessel to the turbine building is conservatively assumed to escape to the environs.

R.3.5.3 Radiological Effects

The rising terrain around the site has been considered in evaluating the radiological consequences. In fact, for many inland directions the steam cloud release from the top of the turbine building could be at ground level prior to leaving the site. Table R.3-11 shows doses for the ground case for various distances and meteorological conditions. The most restrictive whole body and thyroid doses occur at the site boundary about 400 mSW of the Turbine Building. They are 2.4×10^{-3} rem and 2.8 rem, respectively. For other over water directions between about WNW through SE, these doses would be about 10 to 20 times too high because the steam cloud would not intersect rising terrain as the cloud travels downwind.

Since all of the activity is released to the environs in the form of a puff, the doses indicated are maximum values regardless of what dose period is being evaluated.

It is concluded that the health and safety of the public is not endangered as a consequence of this postulated accident.

R.3.6 Radwaste System Accidents

See Section 14.5.6.

R.3.7 References

1. Delp, D. L., et al. FLARE-A Three Dimensional Boiling Water Reactor Simulator. GEAP-4598, July 1964.

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4. Ackerman, R.J.; Gilles, W.P.; and Thorn, R.J. High Temperature Vapor Pressure of UO^2. Journal of Chemical Physic, Volume 25, No. 6, December 1956.

5. Macphee, J., and Lumb, R.F. Summary Report, PULSTAR Pulse Test-II. WNY-020, February 1965.

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8. Horton, N.R., Williams, W.A., Holtzclaw, J.W. Analytical Methods for Evaluating the Radiological Aspects of the General Electric Boiling Water Reactor. APED-5756, March 1969.

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10. Wilson, J.F., et al. The Velocity of Rising Steam in a Bubbling Two-Phase Mixture. ANS Transaction, Volume 5, No. 1, Page 151, 1962.

11. Ianni, P.W., et al. Design and Operating Experience of the ESADA Vallecitos Experimental Superheat Reactor (EVESR). APED-4784, February 1965.

Table R.3-1

INITIAL CORE CONTROL ROD DROP ACCIDENT FISSION PRODUCT RELEASE RATE TO ENVIRONS

Time after Accident	Fission Product Activity Being Released to Environs: Noble Gases (curies/sec)	Iodines (curies/sec)
1 min	2.2E-5*	1.0E-9
30 min	2.0E-4	9.3E-9
1 hr	2.0E-4	9.3E-9
2 hr	2.0E-4	8.8E-9
8 hr	1.7E-4	7.3E-9
1 day	1.4E-4	5.7E-9
2 days	1.2E-4	4.6E-9
4 days	9.2E-5	3.5E-9
30 days	1.0E-5	3.6E-10

NOTE:
*E-n = 10^{-n}

Table R.3-2

INITIAL CORE CONTROL ROD DROP ACCIDENT[1] RADIOLOGICAL EFFECTS

Dis-tance (mi)	2 Hr Dose VS-6	2 Hr Dose MS-6	2 Hr Dose N-6	2 Hr Dose U-6	24 Hr Dose VS-6	24 Hr Dose MS-6	24 Hr Dose N-6	24 Hr Dose U-6
Passing Cloud Whole Body Dose (rem)								
1/5[2]	4.8E-6[3]	4.3E-6	4.1E-6	3.1E-6	1.6E-5	1.4E-5	1.3E-5	1.0E-5
1/2	3.2E-6	2.7E-6	2.1E-6	1.3E-6	1.1E-5	9.0E-6	6.8E-6	4.3E-6
1	2.1E-6	1.7E-6	9.9E-7	5.4E-7	7.1E-6	5.5E-6	3.3E-6	1.8E-6
5	6.7E-7	4.5E-7	1.2E-7	4.8E-8	2.2E-6	1.5E-6	3.9E-7	1.6E-7
10	3.8E-7	2.4E-7	4.0E-8	1.5E-8	1.3E-6	7.9E-7	1.3E-7	5.1E-8
Lifetime Thyroid Dose (rem)								
1/5[2]	1.2E-4	8.4E-5	5.0E-5	3.0E-5	8.2E-3	5.9E-3	4.2E-3	2.1E-3
1/2	6.9E-5	3.9E-5	1.5E-5	6.2E-6	4.8E-3	2.8E-3	1.0E-3	4.4E-4
1	4.1E-5	1.9E-5	4.5E-6	1.8E-6	2.9E-3	1.4E-3	3.2E-4	1.3E-4
5	1.1E-5	3.5E-6	3.5E-7	1.4E-7	7.4E-4	2.4E-4	2.5E-5	9.7E-6
10	5.6E-6	1.7E-6	1.3E-7	5.0E-8	4.0E-4	1.2E-4	9.2E-6	3.5E-6

	Meteorology	Wind Speed (mph)
VS-6	Very stable	6
MS-6	Moderately stable	6
N-6	Neutral	6
U-6	Unstable	6

NOTES:
1. Analysis with building dilution effect
2. Site boundary
3. 4.8E-6 = 4.8×10^{-6}

Table R.3-3

INITIAL CORE LOSS OF COOLANT ACCIDENT
PRIMARY CONTAINMENT AIRBORNE FISSION
PRODUCT INVENTORY

Time after Accident	Fission Product Activity Airborne Primary Containment	
	Noble Gases (curies)	Halogens (curies)
30 min	1.3E6[1]	8.9E3
1 hr	1.3E6	8.6E3
2 hr	1.3E6	7.1E3
8 hr	1.1E6	6.7E3
1 day	9.9E5	5.2E3
2 days	8.8E5	4.2E3
4 days	7.4E5	3.2E3
30 days	3.0E5	3.2E2

NOTE:

1. $En = 10^n$

Table R.3-4

INITIAL CORE LOSS OF COOLANT ACCIDENT
SECONDARY CONTAINMENT AIRBORNE FISSION
PRODUCT INVENTORY

Time after Accident	Fission Product Activity Airborne Secondary Containment	
	Noble Gases (curies)	Halogens (curies)
30 min	1.4E2(1)	9.2E-1(2)
1 hr	2.7E2	1.8E0
2 hr	5.1E2	3.2E0
8 hr	1.6E3	9.5E0
1 day	3.1E3	1.6E1
2 days	3.8E3	1.8E1
4 days	3.7E3	1.6E1
30 days	1.5E3	1.5E0

NOTE:

1. $En = 10^{n}$
2. $E - n = 10^{-n}$

Table R.3-5

INITIAL CORE LOSS OF COOLANT ACCIDENT
FISSION PRODUCT RELEASE RATE TO ENVIRONS

Time after Accident	Fission Product Activity Being Released to Environs	
	Noble Gases (curies/sec)	Halogens (curies/sec)
30 min	1.6E-3[1]	1.1E-7
1 hr	3.1E-3	2.0E-7
2 hr	5.9E-3	3.7E-7
8 hr	2.5E-2	1.4E-6
1 day	3.6E-2	1.9E-6
2 days	4.4E-2	2.0E-6
4 days	4.3E-2	1.8E-6
30 days	1.7E-2	1.8E-7

NOTE:

1. $E\text{-}n = 10^{-n}$

TABLE R.3-6

INITIAL CORE LOSS OF COOLANT ACCIDENT RADIOLOGICAL EFFECTS

Distance	2-Hr Dose						24 Hr Dose					
(mi)	VS-2	MS-2	N-2	N-10	U-2	U-10	VS-2	MS-2	N-2	N-10	U-2	U-10
Passing Cloud Whole Body Dose (rem)												
1/5*	4.0E-4**	4.0E-4	4.0E-4	7.1E-5	5.0E-4	8.0E-5	2.3E-2	2.3E-2	2.4E-2	4.1E-3	2.9E-2	4.7E-3
1/2	3.3E-4	3.2E-4	3.7E-4	5.3E-5	4.6E-4	6.6E-5	1.9E-2	1.9E-2	2.1E-2	3.1E-3	2.7E-2	3.9E-3
1	2.3E-4	2.3E-4	2.9E-4	4.1E-5	2.4E-4	3.7E-5	1.3E-2	1.3E-2	1.7E-2	2.4E-3	1.4E-2	2.2E-3
5	0.6E-5	7.2E-5	4.4E-5	1.1E-5	2.0E-5	4.6E-6	3.9E-3	3.9E-3	2.5E-3	6.1E-4	1.1E-3	2.6E-4
10	3.2E-5	3.5E-5	1.2E-5	4.2E-6	5.1E-6	1.6E-6	2.0E-3	2.0E-3	7.2E-4	2.4E-4	2.9E-4	9.4E-5
Lifetime Thyroid Dose (rem)												
1/5	a	a	a	a	1.9E-6	1.5E-7	a	a	a	a	9.6E-5	7.1E-6
1/2	a	a	6.7E-7	2.2E-8	5.6E-6	1.1E-6	a	2.5E-9	1.0E-6	1.0E-6	2.7E-4	5.0E-5
1	a	7.1E-9	2.5E-6	3.6E-7	2.5E-6	5.7E-7	a	3.3E-7	1.6E-5	1.6E-5	1.1E-4	2.5E-5
5	a	4.6E-7	4.8E-7	1.3E-7	2.0E-7	5.3E-8	a	1.7E-5	4.4E-6	4.4E-6	7.3E-6	1.8E-6
10	a	5.1E-7	1.8E-7	5.1E-8	7.2E-8	1.9E-8	1.5E-9	1.6E-5	1.5E-6	1.5E-6	2.2E-6	5.5E-7

Symbols:

"a" means 1×10^{-10}

Notes

*Site Boundary = 0.2 mi = 320m

**E-n = 10^{-n}

Meteorology		Wind Speed (mph)
VS-2	Very Stable	2
MS-2	Moderately stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

Table R.3-7

INITIAL CORE REFUELING ACCIDENT
SECONDARY CONTAINMENT AIRBORNE
FISSION PRODUCT INVENTORY

Time after Accident	Fission Product Activity Airborne Secondary Containment Noble Gasses (curies)	Halogens (curies)
1 min	1.2E4[1]	5.0E2
30 min	1.2E4	4.9E2
1 hr	1.1E4	4.8E2
2 hr	1.0E4	4.6E2
8 hr	7.5E3	4.0E2
1 day	3.2E3	3.1E2
2 days	1.0E3	2.6E2
4 days	1.0E2	1.3E2
30 days	6.8E-11	1.9E-1[2]

NOTES:

1. En = 10^{n}
2. E-n = 10^{-n}

Table R.3-8

INITIAL CORE REFUELING ACCIDENT
STATION MAIN STACK FISSION
PRODUCT RELEASE RATE

Time after Accident	Fission Product Activity Being Released to Environs: Noble Gasses (curies/sec)	Halogens (curies/sec)
1 min	1.4E-1[1]	5.8E-5
30 min	1.4E-1	5.7E-5
1 hr	1.3E-1	5.6E-5
2 hr	1.2E-1	5.4E-5
8 hr	8.7E-2	4.6E-5
1 day	3.8E-2	3.6E-5
2 days	1.2E-2	2.6E-5
4 days	1.2E-3	1.5E-5
30 days	7.9E-16	2.2E-8

NOTE:

1. E-n = 10^{-n}

Table R.3-9

INITIAL CORE REFUELING ACCIDENT RADIOLOGICAL EFFECTS

Distance (mi)	2 Hr Dose VS-2	MS-2	N-2	N-10	U-2	U-10	24 Hr Dose VS-2	MS-2	N-2	N10	U-2	U-10
Passing Cloud Whole Body Dose (rem)												
1/5(1)	1.7E-3(2)	1.7E-3	1.7E-3	2.2E-3	3.1E-4	3.5E-4	9.9E-3	9.9E-3	1.0E-2	1.8E-3	1.2E-2	2.0E-3
1/2	1.4E-3	1.4E-3	1.6E-3	2.3E-4	2.0E-3	2.0E-4	8.0E-3	8.0E-3	9.1E-3	1.3E-3	1.1E-2	1.6E-3
1	9.9E-4	9.9E-4	1.3E-3	1.7E-3	1.0E-4	1.6E-4	5.7E-3	5.7E-3	7.2E-3	1.0E-3	6.0E-3	9.2E-4
5	2.8E-4	3.1E-4	1.9E-4	4.5E-5	8.4E-5	2.0E-5	1.6E-3	1.6E-3	1.1E-3	2.6E-4	4.8E-4	1.1E-4
10	1.4E-4	1.5E-4	5.3E-5	1.8E-5	2.2E-5	6.9E-6	8.0E-4	8.0E-4	3.1E-5	1.0E-5	1.2E-5	4.0E-5
Lifetime Thyroide Dose (rem)												
1/5	a	a	5.3E-10	a	7.0E-4	5.2E-5	a	a	3.5E-9	a	4.6E-3	2.4E-4
1/2	a	1.8E-8	2.4E-4	8.0E-6	2.1E-3	3.8E-4	a	1.2E-7	1.6E-3	5.1E-5	1.3E-2	2.5E-3
1	a	2.6E-6	9.0E-4	1.3E-4	9.0E-4	2.1E-4	a	1.6E-5	5.7E-7	8.0E-4	5.7E-3	1.3E-3
5	a	1.7E-4	1.75E-4	4.7E-5	7.4E-5	1.9E-5	a	9.2E-5	9.6E-4	2.5E-4	4.0E-4	1.0E-4
10	1.8E-8	1.8E-4	6.5E-5	1.8E-5	2.6E-4	7.0E-6	9.3E-8	9.4E-4	3.3E-4	9.1E-5	1.3E-4	3.4E-5

Symbols:

"a" means 1×10^{-10}

Meteorology		Wind Speed (mph)
VS-2	Very Stable	2
MS-2	Moderately stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

NOTES

1. Site Boundary = 0.2 mi = 320m
2. E-n = $^{10\text{-}n}$

Table R.3-10

INITIAL CORE MAIN STEAM LINE BREAK ACCIDENT
RADIOLOGICAL EFFECTS

2 Hr Dose

Distance (mi)

Passing Cloud Whole Body Dose (rem)

	VS-2	MS-2	N-2	N-10	U-2	U-10
1/5[1]	2.7E-3[2]	1.8E-3	1.4E-3	3.5E-4	6.8E-4	1.6E-4
1/2	1.6E-3	1.0E-3	6.1E-4	1.6E-4	2.5E-4	6.1E-5
1	9.7E-4	6.0E-4	2.9E-4	8.3E-5	9.3E-5	2.5E-5
5	1.9E-4	1.1E-4	2.4E-5	9.8E-6	4.8E-6	1.7E-6
10	7.0E-5	3.8E-5	5.3E-6	2.9E-6	9.8E-7	4.5E-7

Lifetime Thyroid Dose (rem)

1/5	7.9E-0	3.7E-0	2.0E-0	6.8E-1	4.6E-1	1.2E-1
1/2	2.8E-0	1.2E-0	3.9E-1	1.3E-1	8.4E-2	2.2E-2
1	1.1E-0	4.6E-1	1.1E-2	3.6E-2	2.3E-2	6.2E-3
5	6.8E-2	3.9E-2	6.0E-3	1.9E-3	1.2E-3	3.2E-4
10	1.3E-2	1.2E-2	1.7E-3	5.4E-4	3.4E-4	8.8E-5

Meteorology		Wind Speed (mph)
VS-2	Very stable	2
MS-2	Moderately stable	2
N-2	Neutral	2
N-10	Neutral	10
U-2	Unstable	2
U-10	Unstable	10

NOTES:

These dose values apply to the inland directions (about W through SSE) from the Turbine Building. For all other directions, the doses would be less.

1. Site Boundary = 0.2 mi = 320m
2. E-n = $\times 10^{-n}$

TABLE R.3-11

INITIAL CORE OFFSITE THYROID DOSES RESULTING FROM FAILURES OF CLASS II COMPONENTS CONTAINING LIQUID WITH SIGNIFICANT AMOUNTS OF RADIOACTIVE MATERIALS NOT ENCLOSED IN CLASS I STRUCTURES

Vessel Name	Maximum Iodine Release Rate through Building Exhaust Vent (microcuries/sec)	Total Iodine Released (curies)	2 Hour Thyroid Dose at 320 Meters (Mrem)	30 Day Thyroid Dose at 6,800 Meters (Mrem)
Condenser Hotwell	3.3×10^{-4}	4.0×10^{-5}	5.2×10^{-4}	6.5×10^{-5}
Chemical Waste Receiver Tank	2.6×10^{-2}	1.1×10^{-2}	1.2×10^{-1}	1.6×10^{-5}
Monitor Tank	2.6×10^{-2}	1.1×10^{-2}	1.2×10^{-1}	1.6×10^{-2}
Treated Water Holdup Tank	0.7×10^{-5}	2.6×10^{-5}	2.9×10	4.0×10^{-5}
Clean Waste Receiver Tank	4.1×10^{-1}	1.6×10^{-1}	1.75	2.4×10^{-1}
Total Tankage Failure	3.0×10^{-2}	1.8×10^{-2}	1.3×10^{-1}	1.9×10^{-2}

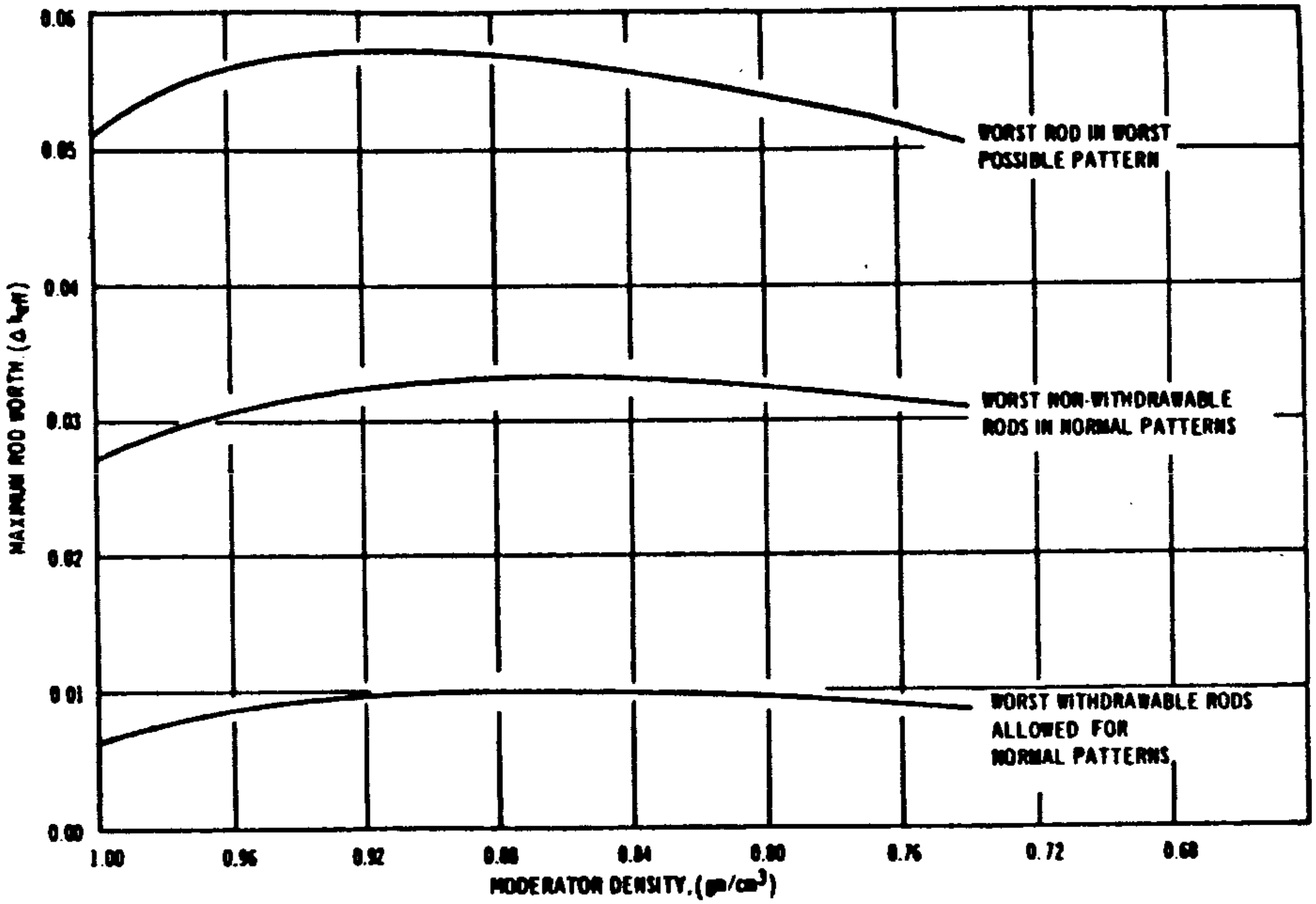


Figure R.3-1. Initial Core Maximum Rod Worth Versus Moderator Density

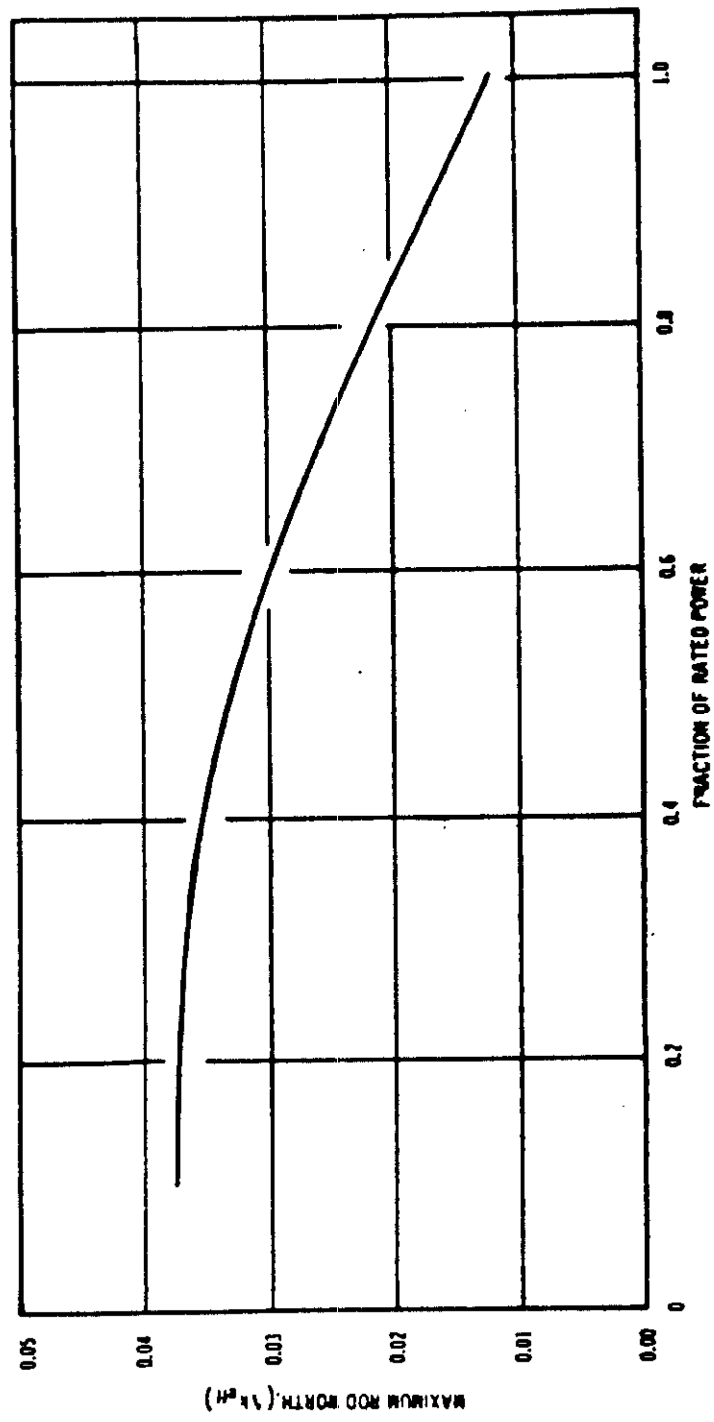


Figure R.3-2. Initial Core Maximum Rod Worth Versus Power Level

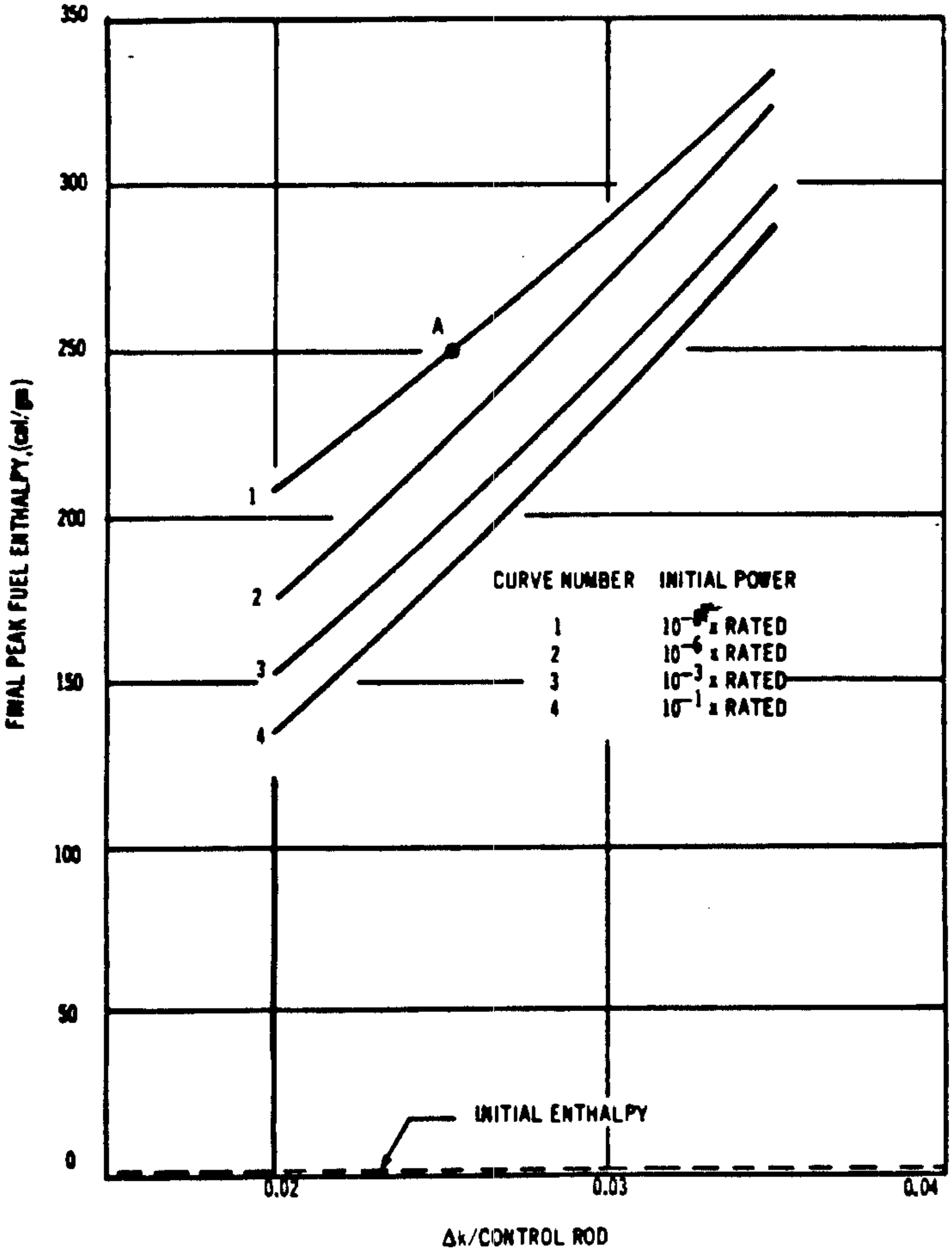


Figure R.3-3. Initial Core Rod Drop Accident (Cold, Critical) Peak Fuel Enthalpy

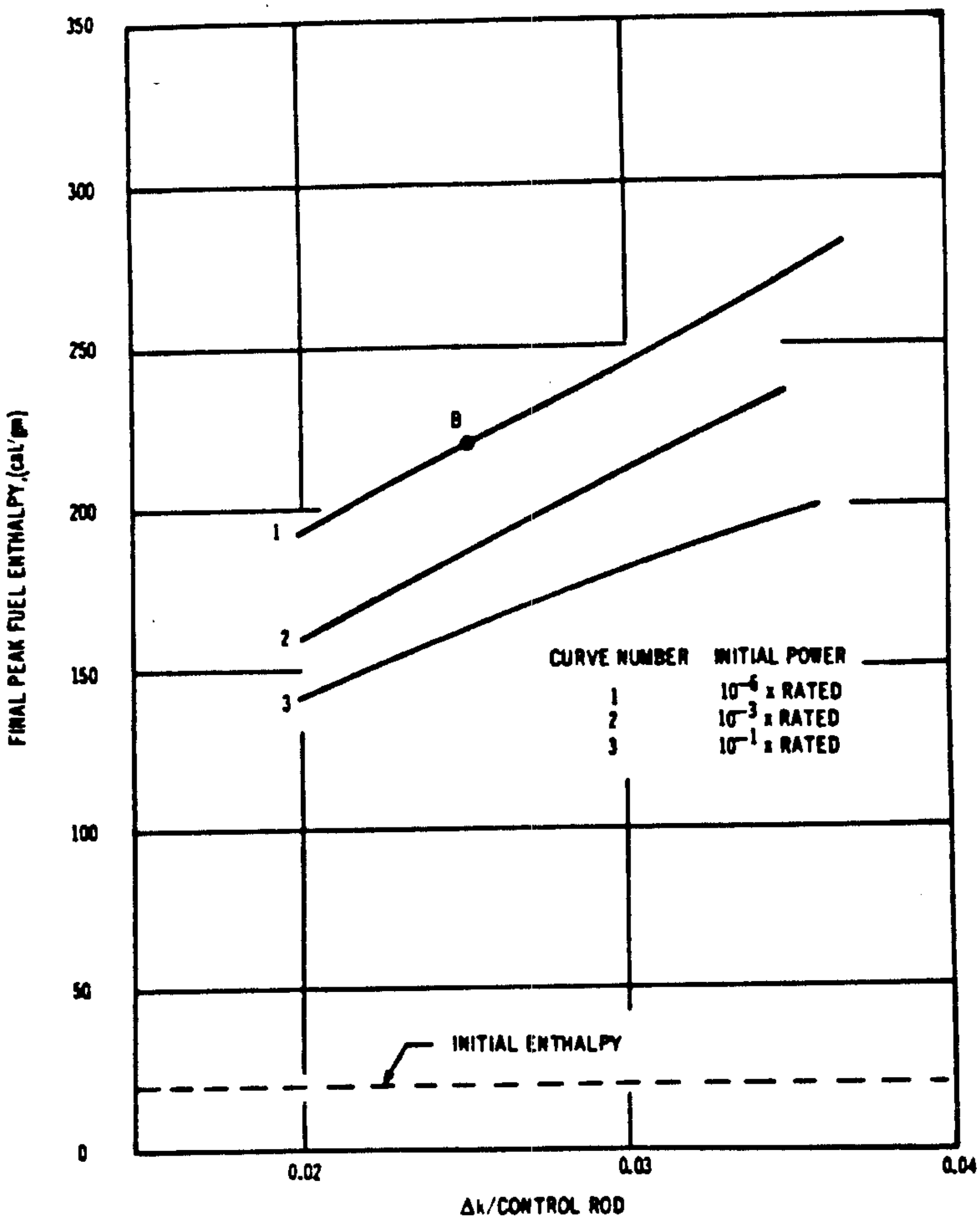


Figure R.3-4. Initial Core Rod Drop Accident (Hot, Critical) Peak Fuel Enthalpy

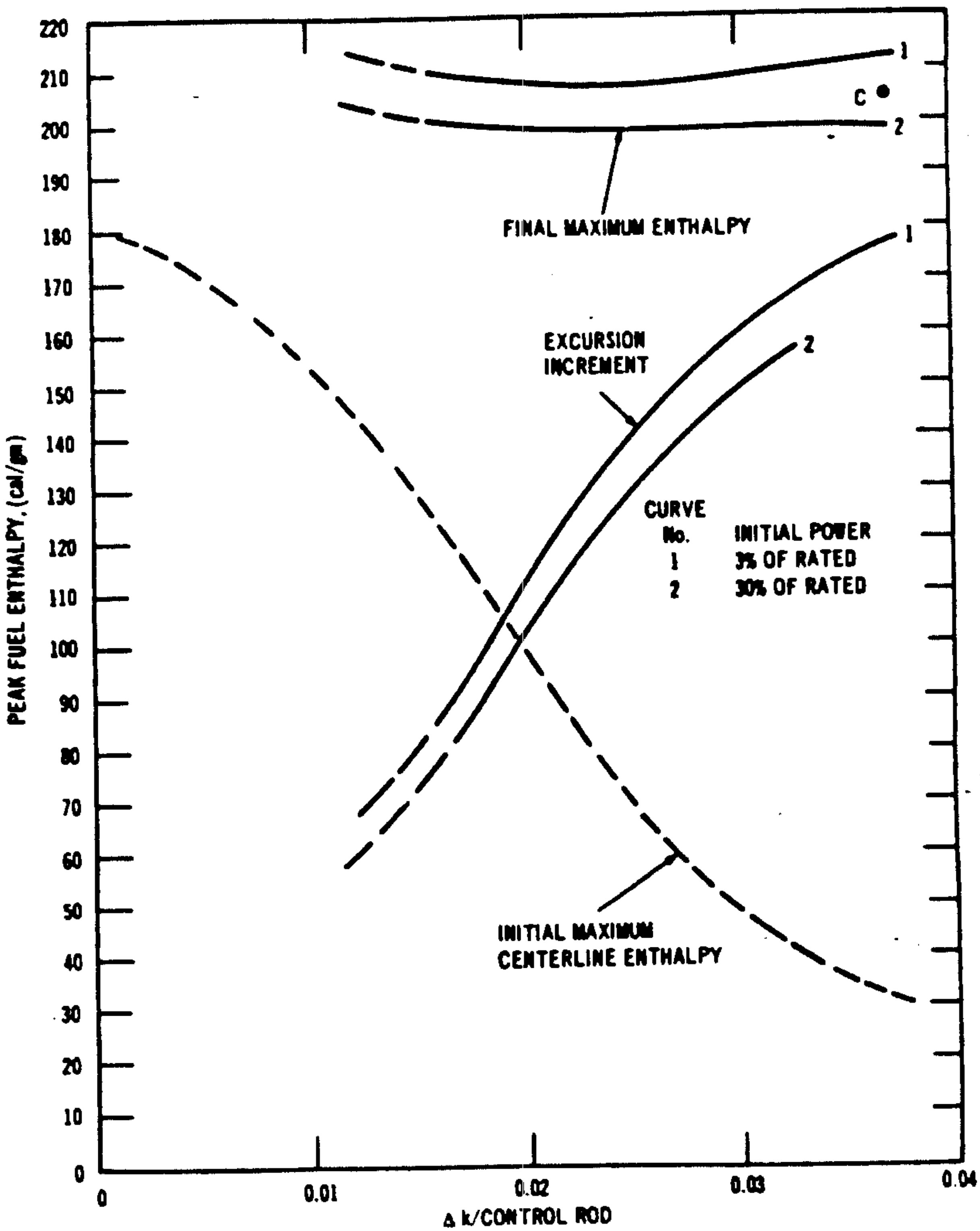


Figure R.3-5. Initial Core Rod Drop Accident (Power Range) Peak Fuel Enthalpy

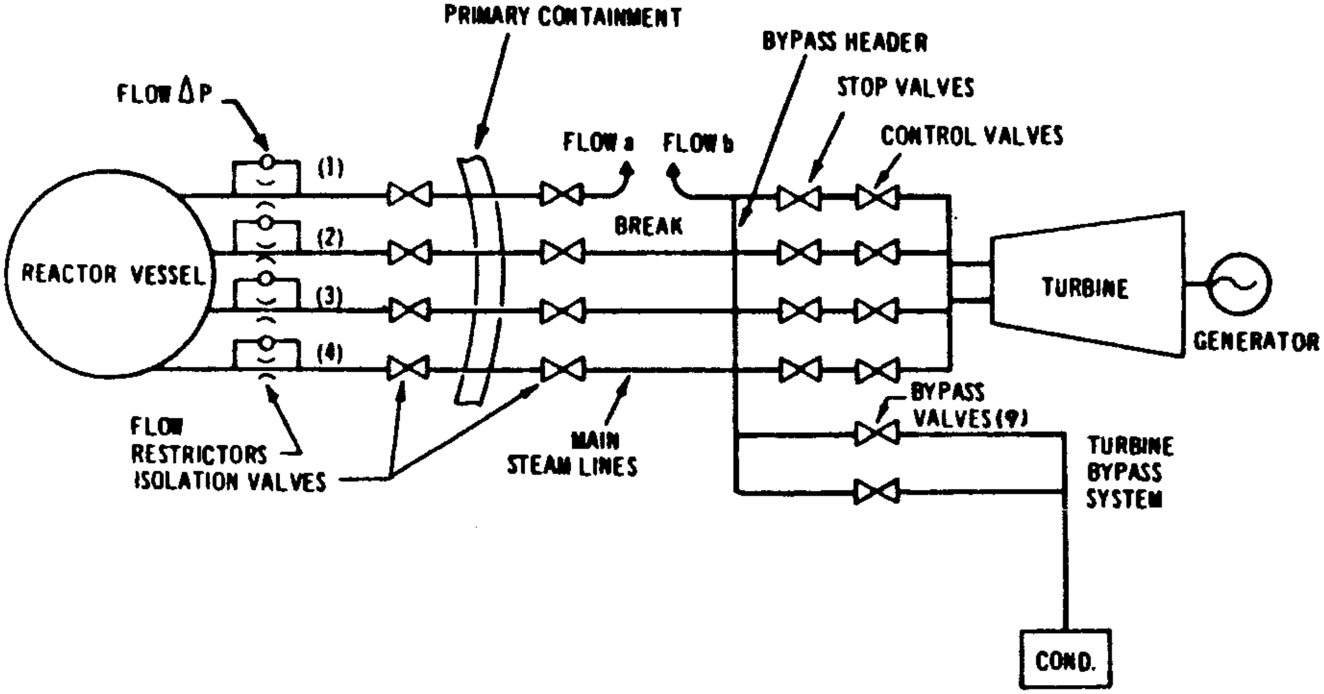


Figure R.3-6. Main Steam Line Break Accident, Break Location

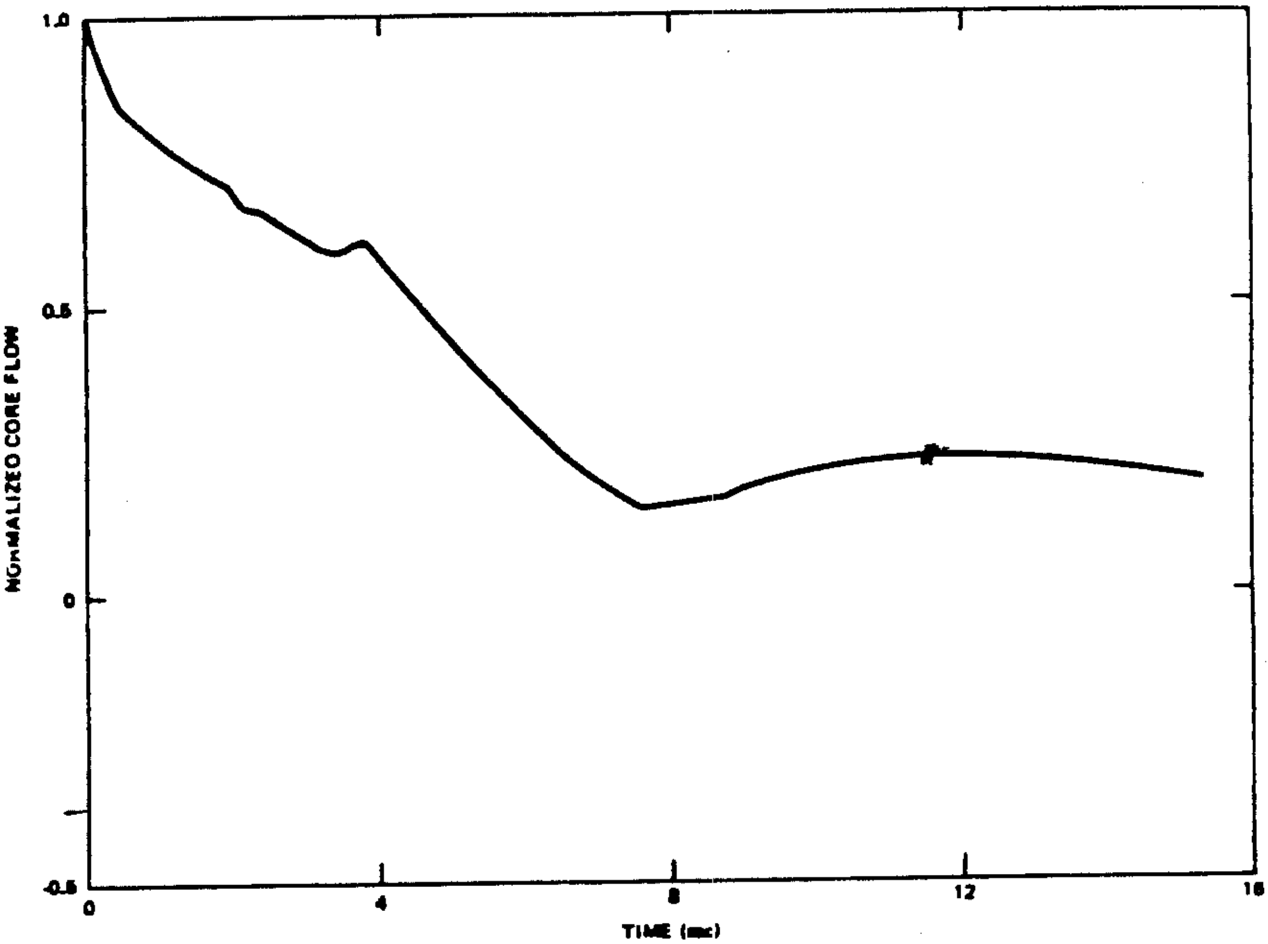


Figure R.3-7. Main Steam Line Break Accident Normalized Core Inlet Flow

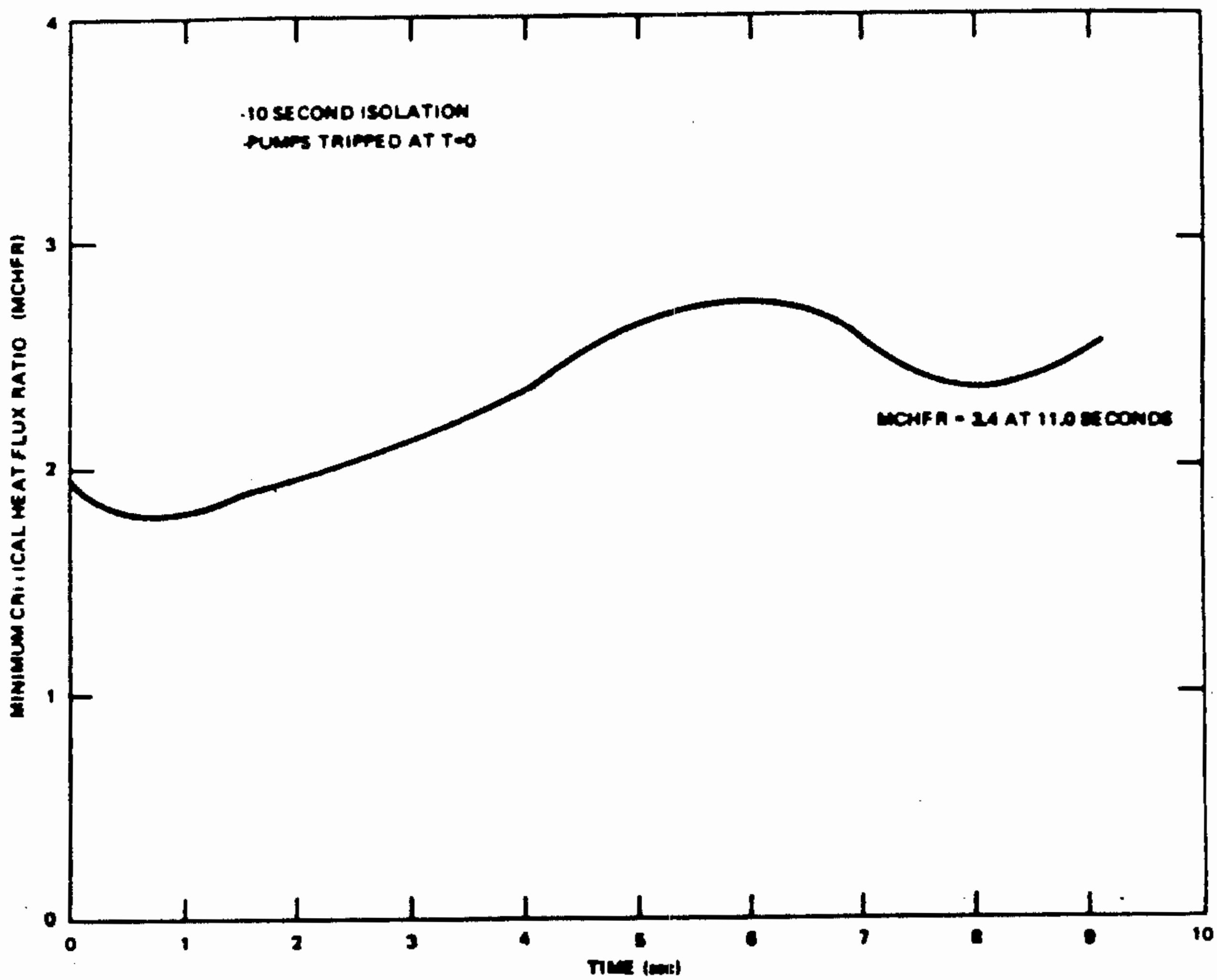


Figure R.3-8. Initial Core Main Steam Line Break Accident Minimum Critical Heat Flux Ratio

R.4 SPECIAL EVENTS (INITIAL CORE)

See Subsections 14.2.3 and 14.6.

R.5 ANALYTICAL METHODS (INITIAL CORE)

R.5.1 Nuclear Excursion Analysis

R.5.1.1 Introduction

Although extensive preventative measures in the forms of equipment design and procedural controls are taken to avoid nuclear excursions, such an event is assumed as a design basis accident. A continued effort is made in the area of analytical methods to assure that nuclear excursion calculations reflect the state of the art in the field. This section outlines only the broader aspects of the subject. Greater detail is available in technical literature.(1)

R.5.1.2 Description

There are many ways of inserting reactivity into a large core BWR. However, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. The one category of reactivity additions that must be considered in evaluating large nuclear excursions is that associated with the Control Rod System. It appears, at this time, that the rapid removal of a high-worth control rod is the only way to obtaining a high enough rate of reactivity insertion to result in a potentially significant excursion.

The rapid removal of a high-worth rod results in a high local K_{eff} in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion; therefore, the method of analysis must be capable of properly accounting for any possible effects of the power distribution shifts. This is an effect which is not significant in small cores.

With this background in mind, it is now possible to categorize nuclear excursions in water-moderated oxide cores. This categorization criterion that seems most definitive is one based on the principal shutdown mechanisms that come into play. This method is particularly useful because for fuel of this design the principal shutdown mechanisms have a direct relationship to both the consequences of the excursion and the applicable method of analysis. With respect to the energy densities presented, the following reference points are used.

Enthalpy = 0 cal/g ambient temperature
Enthalpy = 220 cal/g at incipient melting of UO_2
Enthalpy = 280 cal/g at fully molten UO_2
Enthalpy = 425 cal/g when UO_2 vapor pressure is 1,000 psi

Table R.5-1 describes the three categories of nuclear excursions, assuming a very low initial power level. As shown in the table, there is some overlap in the three ranges of excursions. The indicated numbers for reactivity insertion rate, minimum period, and

peak energy density are nominal values and will vary somewhat from one reactor to another.

In the low reactivity insertion rate range, the reactor is barely prompt critical, and the energy that is stored in the fuel as a result of the nuclear burst is built up at a relatively slow rate. As a result, there may be a significant amount of heat transfer out of the fuel during the burst, and the negative moderator coefficient as well as the U-238 Doppler effect contributes to the shutdown mechanisms. In the medium range, the period is much shorter, and there is very little heat transfer out of the fuel during the burst. In this case, the principal shutdown mechanism is the Doppler effect. Finally, in the high range, there exists the possibility of core disassembly during the burst, due to high internal pressure causing prompt failure of fuel rods. This results in a significant contribution toward shutdown of the excursion.

In terms of consequences, the low range is limited to no fuel cladding damage, or at worst, a small amount of burnout. This poses no threat to nuclear system integrity; therefore, from a safety viewpoint, only the medium and high ranges are considered. The design basis rod drop accident is in the medium range, well below the range where core disassembly is possible.

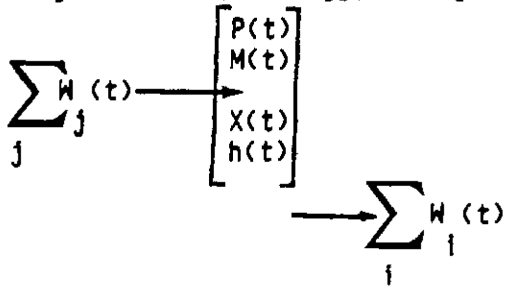
R.5.2 Reactor Vessel Depressurization Analysis

R.5.2.1 Introduction

The analytical methods used to calculate the energy and mass release rates issuing from a reactor vessel during rapid depressurization are described in this section. Conservation of mass and energy equations are written for a constant volume system containing saturated steam and liquid in thermodynamic equilibrium to determine the thermodynamic state in the vessel. Mass flow rates into and out of the vessel are then used to find the rate of 7 change of system pressure and mass inventory.

R.5.2.2 Theoretical Development

The mathematical formulation for the depressurization of the reactor vessel can be derived by considering the conservation of mass and energy in the constant volume system during rapid depressurization as shown in the control volume sketch below. If the mass flow rates are known, it is possible to develop expressions of the rate of change of mass, energy, and pressure within the system.



(R.6)

R.5.2.2.1 Mass Balance

The volume of the Control System is comprised of saturated liquid and saturated vapor in equilibrium:

$$V = M_f\ v_f + M_g\ v_g = \text{constant} \tag{R.2}$$

where:

V = Total volume of the system (i.e., the reactor vessel)
v = Specific volume
M = Mass

The subscripts f and g refer to the liquid and vapor phases, respectively.

Since the total mass in the system is simply

$$M = M_f + M_g. \tag{R.3}$$

then the steam quality by weight is given as,

$$x = \frac{M_g}{M}. \tag{R.4}$$

R.5.2.2.2 Mass Rate of Change in Vessel

From continuity the rate of change of vapor mass in the system is equal to the net inflow of vapor plus the rate at which liquid is flashed to vapor due to depressurization. Hence,

$$\frac{dM_g}{dt} = \sum_j w_{g_j} \sum_i w_{g_i} + W_{fg}' \tag{R.5}$$

where

w = mass flow rate
W_{fg} = net flashing rate

The subscript j corresponds to inflow while i refers to the outflow from the vessel evaluated at the thermodynamic conditions within the system. Similarly, the rate of change of liquid mass in the vessel is

$$\frac{dM_f}{dt} = \sum_j w_{f_j} \sum_i w_{f_i} + W_{fg} \tag{R.6}$$

R.5.2.2.3 Rate of Change of Energy in Vessel

The rate of change of energy in the system can be expressed from the First Law of Thermodynamics:

(Net energy inflow) - (net energy outflow) = (rate of change of internal energy)

$$\left(q + \sum_{j} w_f h_f + \sum_{j} w_g h_g\right) - \left(\sum_{i} w_f h_f + \sum_{i} w_g h_g\right)$$

$$= \frac{d}{dt}\left(M_f h_f + M_g h_g - VP\right) \qquad (R.7)$$

where:

h = Enthalpy
P = Saturated pressure in the system
q = Heat transfer rate to the fluid from the surroundings (solids)

The right hand side of Equation R.7 can be expanded, using the chain rule, to yield

(Rate of change of internal energy)

$$= \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP}\right] \frac{dP}{dt} + h_g \frac{dM_g}{dt} + h_f \frac{dM_f}{dt} - V \frac{dP}{dt} \qquad (R.7a)$$

R.5.2.2.4 Flashing Rate in Vessel

After substituting Equations R.5, R.6, and R.7a into Equation R.7, the expression for the net flashing rate is:

$$W_{f_g} = \frac{1}{h_{fg}} \left\{ \dot{q} + \sum_j w_g h_g - \left(\sum_j w_g \quad h_g \right) \right.$$

$$+ \sum_j w_f h_f - \left(\sum_j w_f \right) h_f$$

$$\left. - \left[M_g \frac{dh_g}{dP} + M_f \frac{dh_f}{dP} - V \right] \frac{dP}{dt} \right\} \qquad \text{(R.8)}$$

R.5.2.2.5 Vessel Depressurization Rate

In order to arrive at an expression for depressurization rate, start by differentiating Equation R.2, realizing that for a fixed total system volume = 0; then,

$$M_g \frac{dv_g}{dt} + v_g \frac{dM_g}{dt} + M_f \frac{dv_f}{dt} + f_v \frac{dM_f}{dt} = 0 \qquad \text{(R.9)}$$

Now expanding this by means of the chain rule, obtain:

$$v_g \frac{dM_g}{dt} + v_f \frac{dM_f}{dt} + \left(M_f \frac{dv_f}{dP} + M_g \frac{dv_g}{dP} \right) \frac{dP}{dt} = 0 \qquad \text{(R.10)}$$

With expressions for dM_g/dt and dM_f/dt as given in Equation R.5 and R.6, Equation R.10 can be written:

$$v_g \left[\sum_j w_g - \sum_i w_g + W_{fg} \right] + v_f \left[\sum_j w_f - \sum_i w_f - W_{fg} \right]$$

$$+ \left[M_f \frac{dv_f}{dP} + M_g \frac{dv_g}{dP} \frac{dP}{dt} \right] = 0. \qquad \text{(R.11)}$$

After substituting Equation R.8 into Equation R.11 and rearranging, the following expression for depressurization rate is obtained:

$$\frac{dP}{dt} = - \left[\frac{f_1(P) + f_2(P)}{f_3(P)} \right] \qquad \text{(R.12)}$$

where

$$f_1(P) = v_f\left(\sum_j w_f - \sum_i w_f\right) + v_g\left(\sum_j w_g - \sum_i w_g\right)$$

$$f_2(P) = \frac{v_{fg}}{h_{fg}}\left[q + \left(\sum_j w_f h_f\right) - \left(\sum_j w_f\right) h_f + \left(\sum_j w_g h_g\right) - \left(\sum_j w_g\right) h_g\right]$$

$$f_3(P) = M_g\left[\frac{dv_g}{\partial P} - \left(\frac{V_{fg}}{h_{fg}}\right)\frac{dh_g}{\partial P}\right]$$

$$+M_f\left[\frac{dv_f}{dP} - \left(\frac{v_{fg}}{h_{fg}}\right)\frac{dh_f}{dP}\right] + \left(\frac{v_{fg}}{h_{fg}}\right)\frac{\underline{V}}{J}$$

R.5.2.2.6 Mass Flow Rates

The mass flow rates entering the reactor vessel during the blowdown are treated as functions of time and are independent of the internal thermodynamic conditions in the vessel. These flow rates can be liquid or vapor or some combination of the two. The outlet flow rate can be calculated from one or two flow models: critical flow as a function of the control volume stagnation properties P_o and h_o, or supercritical flow as a function of the pressure difference P_o - P_{sink} (sink refers to the pressure outside the vessel).

Critical flow is flow which is "choked" at some point where the Mach number is unity in the line through which depressurization is taking place. Critical or maximum flow (both single-phase and two-phase) persists when the ratio of driving pressure (vessel pressure) to sink

pressure (drywell) is greater than approximately two. The critical flow analysis of F. J. Moody[2] is used to determine the flow rate for critical flow conditions.

For the instantaneous values of pressure, P, enthalpy, h, and friction coefficient, fL/d, a three variable interpolation is performed using Moody's results to find the critical mass velocity:

$$G_c = G\ (P, h\ \bar{f}L/d). \qquad \text{(R.13)}$$

The mass flow rate is now calculated from

$$w_c = AG_c \qquad \text{(R.14)}$$

where

A = minimum flow area in the line

Supercritical flow will exist prior to the formation of bubbles in liquid flow and establishment of two-phase critical flow, or when the source pressure is low so that the ratio of P_o/P_{sink} 2.

Supercritical mass velocity is calculated from:

$$G_{sc} = \frac{2g\ (P_0 - P_{sink})}{v_f\ (1.4 + f\ L/d)\ \varnothing^2} \qquad \text{(R.15)}$$

where:

$\varnothing$ = Martinelli-Nelson two-phase multiplier.[3]

The mass flow rate is:

$$w_{sc} = A\ G_{sc} \qquad \text{(R.16)}$$

R.5.2.3 Numerical Solution

If a function of time and its time derivatives are known at time t_1, the value of the function at time t_1 + V t can be obtained from a Taylor series expansion. The first three terms of the series are:

$$f(t_1 + \Delta t) = f(t_1) + \frac{\Delta t}{1!} f'(t_1) + \frac{\Delta t^2}{2!} f''(t_1) + \ldots \qquad \text{(R.17)}$$

where:

$$f'(t_1) = \frac{df}{dt} \text{ at } t = t_1$$

$$f''(t_1) = \frac{d^2 f}{dt^2} \text{ at } t = t_1$$

Δt = Size of time step

Integration

If the term involving the second derivative is negligible, the Euler forward integration method is obtained

$$f(f_1 + \Delta t) = f(t_1) + \Delta t\, f'(f_1) \qquad (R.18)$$

Time Step

A variable time step based on an accuracy criterion has been used in the integration method. The error made in one extrapolation of the Euler method can be approximated by the third term of Taylor's series given by Equation R.17; i.e.,

$$e = \frac{\Delta t^2}{2!} f''(t_1) \qquad (R.19)$$

An exact equation for the second time derivative can be approximated by the rate of change of the first derivative, i.e.,

$$f''(t_1) = \frac{f'(t_1 + \Delta t) - f'(t_1)}{\Delta t} \qquad (R.20)$$

After substituting Equation R.11 and R.19, an approximation of the error made in one time step can be calculated:

$$e = \frac{\Delta t}{2} f'(t_1 + \Delta t) - f'(t_1) \qquad (R.21)$$

If the magnitude of this error is within the error criterion, , then the time step is doubled for the next calculation. If >|e|≥E, then the time step is halved and the previous calculations are repeated.

Calculations

Equations R.5, R.6, and R.12 are programmed for machine calculation using the numerical methods described above.

R.5.3 Reactor Core Heatup Analysis

R.5.3.1 Introduction

The analytical method used to calculate reactor core thermal transient following a loss of coolant accident is described in this section. The fuel temperature, cladding temperature, channel temperature, and amount of metal water reaction are calculated as functions of time from the start of the accident. In this analysis the power of decaying fission products, the chemical energy released by metal water reactions, and the stored heat in the fuel, cladding, and other metal in the core are included as heat sources.

The fuel rods are classified such that those with similar power levels and fuel bundle locations are analyzed as a group. A one dimensional heat balance is then written for each type of fuel rod. Heat is transferred from the surface of the fuel rods by convection to the water, steam, or hydrogen formed in the metal water reaction. In addition, thermal radiation between fuel rods and from the rods to the channel is accounted for in the overall heat balance.

R.5.3.2 Theoretical Development

A typical fuel rod consists of UO_2 fuel with a Zircaloy cladding. A fuel bundle consists of 49 fuel rods, grouped together to form a square array which is surrounded by a metal channel. The fuel rods are divided into three radial temperature zones for the numerical calculations as shown on Figure R.5-1. The cladding, on the other hand, is described by the average cladding temperature, with an outer surface temperature computed from the average temperature. The channel (Figure R.5-1) is considered to be at a uniform temperature radially. The fuel rods within the channel are divided into four representative zones to describe the spatial variation of power generation. The entire reactor core is made up of several hundred fuel bundles and channels. To describe the radial variations of

power generation, the core is divided into five radial zones. The fuel rods and channels are divided into five axial regions. Axial conduction between regions is neglected. Each channel is considered to be isolated from the rest of the core so that interactions between adjacent channels are neglected.

R.5.3.2.1 Heat Sources

The energy generated by delayed neutrons and decaying fission products is assumed to be uniform within a fuel rod and to have the same radial and axial variation within the core as the steady state power distribution. The chemical energy released by the metal water reaction is described by the parabolic rate law given by Baker,[4] where the rate of change of the metal oxide thickness is written as:

$$\frac{d\delta}{dt} = \frac{K}{\delta} \exp(D/T_C) \qquad \text{(R.22)}$$

where:

K = Rate coefficient
T_c = Cladding temperature
D = Activation coefficient
δ = Oxide thickness

The heat generation rate and hydrogen release rate are proportional to the rate of change of oxide generated. The chemical heat liberated is given as follows:

$$\frac{dQ_c}{dt} = \frac{d\delta}{dt} \Delta H \rho_c A_s \qquad \text{(R.23)}$$

where:

H = Heat of reaction
ρ_c = Density of metal
A_s = Exposed surface area of oxide

The mass rate of hydrogen generated is:

$$\frac{dW_H}{dt} = 2\frac{d}{dt} \rho_c A_s \left(\frac{N_{H_2}}{N_{METAL}}\right)$$

where:

W_H = Mass of hydrogen generated
N = Molecular weight

The above reaction rate considers that there is an unlimited source of saturated steam available for the reaction. The empirical reaction constants, K and D, are based upon experimental data obtained under conditions where the metal and water are at the same temperature. Therefore, for Equation R.22 to be correct the water must be heated to the cladding temperature. The energy required to heat this water is deducted from the total chemical energy added to the system.

R.5.3.2.2 Conduction Heat Transfer

The heatup analysis considers only radial conduction of heat from the fuel to the cladding surface. Axial conduction along the fuel rods or to support structures is neglected. Resistance to heat flow through the fuel cladding gap is taken into account.

R.5.3.2.3 Convection Heat Transfer

Heat is transferred from the cladding and channel to the surrounding fluid by thermal radiation and convection. During the blowdown a convection heat transfer coefficient must be calculated. The water level is calculated from the mass inventory in the reactor vessel during the blowdown. If an axial node is covered with water or steam water mixtures, the heat transfer coefficient for that node is obtained from the Jens-Lottes correlation for boiling heat transfer:

$$h_B = \frac{e^{P/900}}{1.9}\left[(Q_s)^{0.75}\right] \qquad (R.25)$$

where:

P = Reactor pressure
Q_s Surface heat flux

Equation R.25 is used to describe the heat transfer coefficient if the calculated water level is above the center of the node. When water level drops below the center of the node, it is treated as being completely uncovered and the convective heat transfer rate diminishes to zero.

R.5.3.2.4 Radiation

Thermal radiation between fuel rods and the fuel channel box is permitted if they are not covered with water. To simplify

calculations, the fuel rods are grouped into four groups. Figure R.5-1 shows the channel configuration. Group 1 rods exchange radiation with Groups 2, 3, and 4 rods and the channel. Group 2 rods exchange radiation with Groups 1, 3, and 4 rods and the channel. Group 3 rods exchange radiation with Groups 1, 2 and 4 rods and the channels. Finally, Group 4 rods exchange radiation only with Groups 1, 2, and 3 rods. Radiation view factors are also calculated for each group of rods. The view factors together with the emissivity and relative areas are converted to radiation coefficients used in the Stephan-Boltzman equation for obtaining the radiant heat transfer.

R.5.3.3 Method of Solution

The fuel, cladding, and channel temperature are calculated at each time step by considering the aforementioned energy consideration. All temperatures are integrated using a simple Euler forward difference method:

$$(t + \Delta t) = \phi(t) + \frac{d\phi(t)}{dt} \Delta t \tag{R.26}$$

All physical properties are considered constant with temperature and time. The model utilizes the calculated histories of pressure, water level, and heat transfer coefficients. The sink temperature for all convective heat transfer calculations is determined by the saturation temperature at the given pressure.

R.5.4 Containment Response Analysis

R.5.4.1 Short Term Containment Response

R.5.4.1.1 Introduction

This section describes the analytical model used to study the pressure suppression part of the LOCA. The system consists of a drywell which has mass and energy flowing into it from the reactor vessel, and a suppression chamber which receives flow from the drywell via the drywell vents.

The chain of events occuring during the pressure-suppression phase of an accident is as follows:

1. Mass and energy enter the drywell from the reactor vessel

2. Drywell temperature and pressure increase

3. The increased pressure acts to clear water from the vents

4. When the vents are cleared; air, steam, and water flow through the vents and into the suppression pool. The steam is condensed by the pool increasing the pool temperature. The air passes through the pool and enters the suppression

chamber air space, causing the pressure in the suppression chamber to rise

5. As the transient proceeds, energy inflow from the reactor vessel decreases, because of reduced reactor vessel pressure, while vent flow continues. The result is reduced drywell pressure

6. Vent flow decreases with decreasing drywell pressure, and the drywell and suppression chamber come to an equilibrium pressure

R.5.4.1.2 Theoretical Development

The total energy, E_D, air mass, M_{a_D}, and water mass, M_{w_D} (liquid vapor), in the drywell at any time are known by numerical integration of the appropriate rate equations, which are developed by taking mass and energy balances on the drywell.

An independent expression for total drywell energy is:

$$E_D = M_{f_D} u_f + M_{g_D} u_g + M_{a_D} C_v T_D \qquad \text{(R.27)}$$

where:

$$M_{f_D} = (1 - x_D) M_{w_D}$$

$$M_{g_D} = x_D M_{w_D}$$

$$x_D = \frac{v - v_f}{v_g - v_f}$$

$$v = \frac{V_D}{M_{w_D}}$$

and all thermodynamic properties are evaluated at T_D, the equilibrium saturation temperature of the air-stream-liquid mixture within the drywell. Numerical solution of Equation R.27 gives T_D.

With T_D known, all thermodynamic properties are known, and the total drywell pressure is

$$P_D = P_{g_D} + P_{a_D}$$

where:

$$P_{a_D} = \frac{M_{a_D} R_a T_D}{V_{a_D}}$$

and

$$V_{a_D} = V_D - v_f M_{f_D}$$

Flow Rate through Vents

The flow through vents is treated as compressible flow of an ideal gas with friction except modifications are made to account for the effects of the liquid flowing in the vents. The total mass flow rate is then given as a function of friction factor, vent area, drywell/suppression chamber pressure ratio, and drywell fluid properties. Specifically:

$$\dot{w} = f\left(\frac{fL}{D}, A_V, \frac{P_D}{P_B}, v_M, x\right) \qquad \text{(R.28)}$$

where:

$\frac{fL}{D}$ = vent friction factor

A_V = vent flow area

$\frac{P_D}{P_B}$ = pressure ratio (P_B = back pressure at vent exit)

v_M = specific volume of gaseous phase (air and steam)

x = quality in drywell

The specific volume of the gaseous phase is given by

$$v_M = \frac{V_{a_D}}{M_{a_D} + M_{g_D}} \qquad \text{(R.29)}$$

The following expressions are used to determine the vent flow rate of each constituent:

$$\dot{w}_{g_2} = \frac{M_{g_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad \text{(R.30)}$$

and

$$\dot{w}_{a_2} = \frac{M_{a_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad \text{(R.31)}$$

$$\dot{w}_{f_2} = \frac{M_{f_D}}{M_{a_D} + M_{g_D} + M_{f_D}} \dot{w}_T \qquad \text{(R.31)}$$

The subscript 2 represents flow from drywell to suppression chamber.

Drywell Time Derivatives

To compute the rate of change of mass in the drywell, a mass balance gives

$$\dot{M}_{w_D} = \dot{w}_{g_1} + \dot{w}_{f_1} - \dot{w}_{g_2} - \dot{w}_{f_2} \qquad \text{(R.33)}$$

and

$$\dot{M}_{a_D} = \dot{w}_{a_2}$$

where the subscript 1 represents the source flow into the drywell, the rate of change of energy in the drywell, assuming no heat loss, is

$$\dot{E}_D = h_{g_1} \dot{w}_{g_1} + h_{f_1} \dot{w}_{f_1} - h_{g_2} \dot{w}_{g_2} - h_{f_2} \dot{w}_{f_2} - C_p T_D \dot{w}_{a_2} \qquad \text{(R.35)}$$

An energy balance on the lumped suppression pool, assuming complete condensation of the steam entering the pool and neglecting heat losses and evaporation, gives

$$h_{g_2}\dot{w}_{g_2} + C_p(T_D - T'_a)\dot{w}_{a_2} = \frac{d}{dt}\left[M_{w_s}h\right] \qquad (R.36)$$

where M_w is the pool mass and h is the enthalpy of the pool. After expanding the right hand side of Equation R.36 becomes

$$\frac{d}{dt}\left[M_{w_s}h\right] = h\dot{M}_{w_s} + M_{w_s}\left(\frac{h}{T}\right)_p \dot{T}_s + M_{w_s}\left(\frac{h}{P}\right)_T \dot{P}_s \qquad (R.37)$$

For liquid water, $(h/T)_P = 1.0$, $(h/P)_T = 0.0$, thus,

$$\dot{T}_s = \frac{h_{f_2}w_{f_2} + h_{g_2}w_{g_2} + C_P(T_D - T'_a)w_{a_2} - h\dot{M}_{w_s}}{M_{w_s}} \qquad (R.38)$$

A mass balance gives

$$\dot{M}_{w_s} = \dot{w}_{f_2} + \dot{w}_{g_2}$$

It is assumed that in bubbling through the pool the air reaches the pool temperature, thus

$$T'_a = t_s \qquad (R.40)$$

Also knowing T_s from the integral of Equation R.38, P_{g_s}, the partial pressure of vapor in the suppression chamber can be determined

$$P = P_{g_s} + \frac{M_{a_s} R_a T_{a_s}}{V_{a_s}}, \qquad (R.41)$$

where:

$$V_{a_s} = V_s - vM_{w_s}$$

and

v = Specific volume of the pool water

R.5.4.1.3 Solution

The equations described above have been programmed for computer solution. Time derivatives are integrated using numerical methods. Thermodynamic properties in the drywell are found using tabular interpolations.

R.5.4.2 Long Term Containment Pressure Response

The previously developed equations are used to calculate the containment transient during the reactor vessel depressurization and during the containment depressurization which follows the vessel transient. Once the depressurization transient is over, a considerably simplified model can be used. The key assumptions employed in the simplified model are:

1. Drywell and suppression chamber both saturated and at the same total pressure.

2. An energy balance is performed to determine the temperature of the emergency core cooling flow as it drains by gravity back into the suppression chamber. The drywell is conservatively assumed to be 5°F hotter than the water draining back into the suppression pool.

3. The suppression chamber air temperature is taken equal to the pool temperature which is determined from an energy balance on the pool mass.

4. No credit is taken for heat losses from the primary containment.

Since no mass is being added to the suppression pool, the pool temperature can be calculated based on the following energy balance:

$$\dot{T}_s = \frac{h_D \dot{m}_{D_o} - h_s \dot{m}_{s_o} - \dot{q}_{H_x}}{M_{W_s}}$$

h_D = enthalpy of water leaving drywell

$\dot{m}_{D_o}$ = flow rate out of drywell

h_s = enthalpy of water in suppression chamber

$\dot{m}_{s_o}$ = flow rate out of suppression chamber

q_{H_x} = heat removal rate of heat exchanger

M_{W_s} = mass of water in suppression chamber

Assuming no storage in drywell

$$\dot{m}_{D_o} = \dot{m}_{s_o} = \dot{m}_{CSCS}$$

And since the only heat source is the core decay heat, we have:

$$(h_D - h_s)\,\dot{m}_{CSCS} = \dot{q}_D \qquad (R.43)$$

Therefore,

$$\dot{T}_s = \frac{\dot{q}_D - \dot{q}_{H_x}}{M_{W_s}} \qquad (R.44)$$

which can be integrated to give T_s as a function of time. At any point in time the drywell temperature is given by:

$$T_D = T_s + \frac{\dot{q}_D}{\dot{m}_{CSCS}} + 5°F.$$

With the suppression chamber and drywell temperatures known and their total pressure assumed equal, it is now possible to solve for the total pressure

$$P_D = P_S$$

$$P_{a_D} + P_{v_D} = P_{a_s} + P_{v_s}$$

$$\frac{M_{a_D} R T_D}{V_D} + P_{v_D} = \frac{M_{a_s} R T_s}{V_{a_s}} + P_{v_S} \qquad (R.46)$$

The total mass M_T, can be determined from a mass balance on the primary containment:

$$\dot{M}_T = \dot{M}_{a_D} + \dot{M}_{a_s} = \dot{m}_i - \dot{m}_{LEAK} \qquad (R.47)$$

where:

$\dot{m}_i$ = all non-condensible flow into containment, e.g., hydrogen from metal water reaction

m_{LEAK} = leakage from primary containment

Therefore, at any time, M_T is known, and

$$M_T = M_{a_s} + M_{a_D} \qquad (R.48)$$

The two Equations, R.46 and R.47, can be solved for the two unknown, M_{a_s} and M_{a_D}, and the pressure determined.

The leakage rate from the primary containment is determined from the following relationship:

$$\dot{m}_{LEAK} = L_T \left[\frac{1 - \left(\frac{1}{P}\right)^2}{1 - \left(\frac{1}{P_T}\right)^2} \right]^{1/2} \qquad (R.49)$$

where:

L_T = Leak rate at reference pressure

P_T = Reference pressure in absolute atmospheres

P = Containment pressure in absolute atmospheres

The above equations are solved simultaneously on a step-by-step basis to obtain the long term pressure transient and leakage rate of the primary containment.

R.5.5 Analytical Methods for Evaluating Radiological Effects

R.5.5.1 Introduction

This section describes the analytical techniques used to calculate the radiological exposure for each of the design basis accidents. The external exposures are from airborne fission products and the internal exposures are from inhalation of airborne radioactive materials.

The first portion of the analysis concerns the meteorological considerations that describe the dissemination of the radioactive material as it emanates from the source and spreads through the atmosphere. The second portion of the analysis describes the radiological effects on man as a result of the dispersed radioactive materials.

The radiological effects of the design basis accidents are evaluated at various discrete distances from the station. The nearest distance is approximately the site boundary with other distances given to illustrate the decrease of the radiological effects with distance.

Since airborne materials are released via an elevated release point, the effects at short distances for any diffusion condition are much

less for all modes of exposure except from the passing cloud. At these short distances, the plume has not yet reached ground level so that exposure from inhalation is small. The passing cloud effect, however, remains nearly constant due to essentially line-source geometry of the elevated plume.

R.5.5.2 Meteorological Diffusion Evaluation Methods

R.5.5.2.1 General

Six points in the atmospheric diffusion spectrum are used to evaluate the radiological effects of secondary containment leakage via the elevated release point. These represent the meteorological conditions which could exist at the site.

The atmospheric diffusion methods are the same as those reported in the Journal of Applied Meteorology.(5)

R.5.5.2.2 Height of Release

Discharge from the secondary containment to the atmosphere emanates from the elevated release point. The effective height of release is the sum of the release point height plus any effluent rise due to momentum or buoyancy. For most of the design basis accidents, the additional effects of momentum and buoyancy are negligible, so that the release height is equal to the top of the main stack.

While buoyancy effects are significant for the steam line break accident, the conservative assumption is made that the release height is equal to the top of the Turbine Building.

Since the terrain rises inland, the assumption was made that the effective height of the stack plume or steam cloud is reduced by such terrain. This is considered to be a conservative assumption since it is expected that plumes would travel in the free air "streamlining" over and around obstacles, not "intersecting" the ground. The closest "high" terrain point is toward Manomet Hill about 2,200 m SSW of the stack. This location is about the same elevation as the top of the stack. Therefore, it was selected as the offsite location resulting in the highest dosages during stack releases. All other directions would obviously give lower dosages. The steam line break accident steam cloud is assumed to "intersect" the nearest "high" ground about 400 m SW of the Turbine Building.

The frequency of wind blowing onshore towards these higher terrain locations is in the range of 4.5 percent/yr. For these same directions the frequency of 1 m/s wind speeds is in the range of 0.05 to 0.15 percent/yr. However, the so-called worst case meteorological condition for these hillside cases is with a very stable atmosphere. The frequency then is reduced to the range of about 0.009 to 0.05 percent/yr (8 to 44 hr/yr). Certainly not all of these hours occur consecutively as was assumed for the accident analysis. On a probability basis the chances of having the meteorological conditions assumed in the direction of rising terrain are quite small and when

coupling this probability with the probability of the postulated design basis accident itself, the dosages calculated would be for a very unlikely event.

R.5.5.2.3 Diffusion Conditions

An important parameter used in the atmosphere diffusion calculation is the measure of wind direction persistence and variability of direction. This parameter is the product of the standard deviation of the horizontal wind direction fluctuations WR, and the average wind velocity u. Combined with the assumed stability condition, specification of WR_u permits calculation of air concentrations at various distances from the source.

A conservative value of 0.1 rad-m/s is used for this parameter to describe the horizontal spreading of the plume for 1 m/s wind speed conditions. A value of 1.0 rad-m/s is typical for a 5 m/s condition. These values are typical for a 1 hr period. A choice of wind direction persistence (number of continuous hours) of 24 hr is used for poor diffusion conditions. This period is conservative when used with WR_u of 0.1 rad-m/s and 1 m/s wind speed, based upon the U.S. Weather Bureau Data shown on Table R.5-2.[6]

This data shows that wind persistency of periods as long as 24 hr occurs only about 0.1 percent of the time or less at the site listed. The sites include flat terrain, coastal and lake shore sites, and some valley locations. For a wind speed of 5 m/s, a value of 1.0 rad-m/s corresponds to a WR of 0.20 rad which is similar to the value of 0.1 rad for the 1 m/s case. Thus, about the same amount of wind variability is considered and the conservative 24 hr persistence assumption is applicable to both cases.

R.5.5.2.4 Applied Meteorology

The diffusion and wind direction persistence conditions and breathing rates used for the design basis accident calculations are given on Table R.5-3.

R.5.5.2.5 Cloud Dispersion Calculations

The dispersion of the released effluent is described by the Gaussian Diffusion Equation given below.

$$\frac{X}{Q_o} = \frac{f_d}{2\pi \sigma_y \sigma_z \bar{u}} e^{-1/2 \left(\frac{y^2}{\sigma_y^2} + \frac{z^2}{\sigma_z^2} \right)} \qquad \text{(R.50)}$$

where:

X/Q_o = Integrated air concentration (X) per unit activity release (Q_o)

y = Distance from centerline crosswind (since plume centerline is used, y = o)

z = Height of plume above ground

f_d = Cloud depletion factor (halogens only) See Section R5.5.2.6

σ_y = Horizontal diffusion coefficient

σ_z = Vertical diffusion coefficient

$\bar{u}$ = Average wind speed

σ_y and σ_z are defined as follows:

$$\sigma_y^2 = AT - A\alpha + A\alpha e^{-t/\alpha} \quad (R.51)$$[5]

where:

$$A = 13 + 232.5\,(\sigma \quad \bar{u})$$

$$\frac{A}{2(r \quad u)2}$$

t = time after release and = $x/\bar{u}$ where x is downwind distance

The vertical cloud growth, as defined by the standard deviation of width σ_z is given by

$$\sigma_z^2 = a\left[1 - e^{-k^2 t^2}\right] + bt \text{ stable case} \quad (5) \qquad \text{(R.52)}$$

$$\sigma^2 = \frac{C_z^2 x^{2-n}}{2} \text{ neutral and unstable case} \quad (7) \qquad \text{(R.53)}$$

The values of the constants in Equations R.52 and R.53 are given as shown on Table R.5-4.

The conventional "reflection" factor of 2 usually applied for releases for ground level is not included. For the passing cloud dose, which is primarily a gamma dose, the entire cloud volume is integrated as an "infinite" number of point sources to plus and minus infinity in the z-direction ignoring interception by the ground, so that the entire cloud volume is included. Inhalation doses are a function of concentration at the ground and subject to "reflection" effects if they exist. Since the materials of interest in inhalation effects deposit on the ground, "perfect" reflection will not occur, but rather the cloud will expand, distorting the Gaussian mass distribution resulting in at most a small increase in concentration. In addition, no account is taken of the better diffusion near the ground compared to the stack exit elevation used. In any event, an increase by a factor of less than two but perhaps more than one may be a result of this "reflection" effect. A factor of 1.0 is used in this analysis.

No distinction in the choice of the diffusion parameter is made between the first 2 hour period and the total accident dose calculations. This is inconsistent because larger values of this parameter are obviously appropriate for the longer time period. That is the values used, as discussed in Section R.5.5.2.3, are for 1 hour periods, and thus are somewhat conservative when applied to the 2 hour period dose calculation and are markedly conservative for the total accident calculation.

R.5.5.2.6 Cloud Depletion and Ground Deposition

The fallout concentrations of radioactive materials are determined on the basis of particle setting by eddy diffusion only, since settling by gravity is expected to be negligible in this case.

The extent of halogen and solid fission product deposition on the ground is a function of the apparent deposition velocity, which, in turn, is considered to be a function of the diffusion condition and wind speed. Deposition velocities used in this evaluation are given on Table R.4-5.[7]

These values of deposition velocity are used in the calculation of the cloud depletion term, f_d.

$$f_d = \exp\left[\frac{-V_g}{u_o}\sqrt{\frac{2}{\pi}}\,\frac{u_o}{u_h}\int_o^t \frac{u_h \exp\left(\frac{-z^2}{2\sigma_z^2}\right)}{\sigma_z}\,dt\right] \qquad (R.54)$$

where:

f_d = Cloud depletion factor due to fallout

V_g = Deposition velocity of isotope in question (cm/sec)

u_o = Wind speed at ground level (cm/sec)

u_h = Wind speed at height of release (cm/sec)

σ_z = Vertical diffusion coefficient (cm)

R.5.5.2.7 Air Concentration Calculation

Using the equations developed above, the integration air concentration from a release of 1 curie of activity is calculated in curie-sec/m3. This data is given on Tables R.5-6 and R.5-7 for the specified release heights and meteorological conditions.

R.5.5.3 Radiological Effects Calculation

The radiological doses of primary consideration are inhalation and cloud gamma. While the deposition gamma dose may be important from a decontamination viewpoint, it is of minor importance in evaluating the radiological consequences of a design basis accident and is, therefore, insignificant in this analysis.

The downwind radiological effects, such as cloud gamma and inhalation exposure, are a function principally of the integrated air concentration at any point. Calculation of this integrated concentration has been described in the preceding sections. This section describes the conversion of air concentration to radiation dose.

R.5.5.3.1 Passing Cloud Dose

The ground level whole body cloud gamma dose which is received from airborne radioactive materials is determined by summing the dose contribution from each incremental volume of air containing fission product activity. The dose from a point in space to a receptor located at coordinates x, y, and z is determined as follows:

$$D_g = \sum_{K=1}^{m} C_1 C_k f_k \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} \chi G_K \, dx\,dy\,dz \qquad (R.55)$$

D_g = Gamma dose at the receptor point (rem)

C_1 = Conversion factor (3.7×10^{10} disintegration/sec-curie)

χ = Integrated air concentration (curie-sec/m^3)

f_k = Number of photons of the K isotope released per disintegration (photons/dis)

C_k = Flux to dose conversion factor

$$\left(\frac{\text{rem} - m^2 - \text{sec}}{\text{Sec} -}\right)$$

G_k = Dose attenuation kernel which is defined as follows

$$G_K = \frac{B\,e^{-ut}}{4\pi^2}$$

where:

$\bar{B}$ = Buildup factor = $1 + kuT$

$k = \frac{u-u_a}{u_a}$ where u is the total absorption coefficient and u_a is the energy absorption coefficient (M^{-1})

T = Distance from the source to the detector position and is equal to

$$\sqrt{x_1^2 + y_1^2 + z_1^2}$$

R.5.5.3.2 Inhalation Dose

The inhalation dose is an internal exposure which is received as a consequence of inhaling airborne radioactive fission products. Depending upon the isotopes inhaled there may be one or more organs which are effected.

The total activity inhaled during the inhalation period is

$$Q_{dep.} = X_i B_r \qquad \text{(R.56)}$$

Where:

X = Time integral of the air concentration previously defined in Section R.5.5.2.5 (curie/sec/m^3)

B_r = Breathing rate (m^3/sec)

When the above equation is multiplied by an appropriate conversion factor C_i (rem/sec-curie inhaled) a dose rate in the organ is obtained. The total dose resulting from inhalation of a mixture of fission products is

$$D_I = \sum_{i=1}^{N} \int_{o}^{t} \chi_i B C_{ri} e^{-\lambda_i t} dt \qquad (R.57)$$

Where:

D_I = Total inhalation dose (rad)

λ_i = Effective decay constant of the i^{th} isotope in the organ of reference (sec^{-1})

The summation sign indicates that all isotopes contributing to the organ dose are added together to obtain the total inhalation dose.

The conversion factor C_{ri}, which is applicable to the isotope of the organ of interest, is calculated by the use of the following mathematical mode[8]

$$C_i = \frac{C_1 f_a E C_3}{M C_2} \qquad (R.58)$$

Where:

c_i = Activity to dose conversion factor (rad/sec-curie inhaled)

C_1 = 1.6×10^{-6} ergs/mV

f_a = Fraction of inhaled material reaching the organ of reference

E = effective energy absorbed per disintegration (MV/dis)

C_3 = 3.7×10^{10} dis/sec-curie

M = Mass of the organ (g)

C_2 = 100 ergs/g-rad

Therefore:

$$C_i = \frac{(1.6 \times 10^{-6})(f_a)(E)(3.7 \times 10^{10})}{(M)(100)}$$

$$= \frac{5.92 \times 10^{2} f_a E}{M} \text{ (rad/sec curie inhaled)} \qquad \text{(R.59)}$$

Upon integration of Equation R.57 the total inhalation dose is

$$D_T = \sum_{i=1}^{N} \frac{x_i B_r C_i (1 - e^{-\lambda_i t})}{\lambda_i} \qquad \text{(R.60)}$$

If T is large compared to λ Equation R.59 can be simplified to

$$D_T = \sum_{i=1}^{N} \frac{x_i B_r C_i}{g_i} = \sum_{i=1}^{N} \frac{x_i B_r C_i T_i}{0.693} \qquad \text{(R.61)}$$

Where:

T = Effective half life of the i^{th} isotope and is equal to

$$T_i = \frac{T_b T_r}{T_b + T_r}$$

T_b = Biological half life (sec)

T_r = Radioactive half life (sec)

If the effective half life is defined in terms of days and is combined with the conversion factor C_i Equation R.61 can be expressed as follows

$$D_T = \sum_{i=1}^{N} \chi_i B_r C_i \qquad \text{(R.62)}$$

Where:

$$C_i' = \frac{8.64 \times 10^{4}}{6.93 \times 10^{-1}} T_i C_i = 1.25 \times 10^{-5} C_i T_i$$

For the thyroid gland the dose conversion factor is

$$C_i' = \frac{(1.25 \times 10^{5})(5.92 \times 10^{2})(f_a)ET_i}{M}$$

$$= \frac{7.40 \times 10^{7} f_a ET_i}{20} = 3.7 \times 10^{6} f_a ET_i$$

The numerical values which are used in Equation R.58 as well as the dose conversion factor, C_i are given o Table R.5-8.

R.5.5.3.3 Pseudo-Fumigation from Sea Breeze Flow

The Pilgrim Station site, due to its proximity to water, experiences sea breeze circulation during the time that a high pressure area is situated over New England. It only occurs when a weak overall pressure gradient exists. This is most pronounced in spring and early summer. Sea Breeze studies performed at the site during the summer of 1969, when correlated to Van der Hoven's prediction technique(9) for determining the location where pseudo-fumigation could occur, were in good agreement. See Appendix I for details of the study.

Basically, pseudo-fumigation can occur as follows: Following a postulated accident, the stack emission plume would travel inland in stable air. Once this plume intersects the turbulent boundary layer which increases in height with distance inland, the lower part of the plume could be subjected to vertical mixing within the turbulent layer. The result would be an increase in ground level air concentration beyond the point of plume intersection with the elevated boundary (inversion). According to the study, plumes blowing towards Plymouth (E to ESE wind) frequently touch the ground over Rocky Point or come down to the surface over Plymouth Harbor_under sea breeze conditions. However, the plume (smoke release) used in studies was released at 300 feet msl, whereas the actual stack release would be at 400 feet msl.

The only other direction which fumigation may possibly cause higher ground level air concentrations is toward Manomet Hill (NNE wind). A wind in this direction with a sea breeze condition is usually transient in nature and results in plume downwash quite close to the emission point. Since Rocky Hill Road traverses the site, the pseudo-fumigation was assumed to occur at this road location.

In evaluating the case where stable air initially blows over water, then over land for awhile, then over water again towards Plymouth Harbor, the distance from the stack that pseudo-fumigation could occur must be found. This model rather crudely assumes that the elevated plume is uniformly distributed vertically over (h) once this pseudo-fumigation starts. Material released may be trapped within the unstable layer between the top of the sloping turbulent boundary layer and the ground. The worst 1 hour activity release from a design basis accident was used to calculate the doses at the site boundary.

The pseudo-fumigation is only postulated to occur for short time periods (~1 hour). Therefore, the highest 1 hour activity release is from the refueling accident. A whole body dose of 0.003 rem and a thyroid dose of 0.0001 rem result. The following input values to the above model were used: r_y = 62m, 1 = 2m/s, and h = 100 m based upon the sea breeze study. The same 1 hour release occurring during the time pseudo-fumigation could occur towards Rocky Hill Road results in a thyroid dose increase to 0.011 rem. The following input valves were used: ry = 17m, 1 = 2 m/sec, h = 104 m.

Using Van der Hoven's model modified by terrain (see Appendix I), $u\Delta\theta$ values of less than about 12 are required before plume downwash could start. Most of the test smoke runs in the sea breeze study gave - values of between 12.2 and 18.6. For the sake of conservatism, the assumption was made that fumigation occurred at the site boundary WNW of the main stack. In actual fact, it would probably occur farther away, such as at the shoreline beyond the site boundary. Certainly the site boundary dose result would be higher of the two.

Ground level air concentrations due to this pseudo-fumigation were calculated using the following model:[10]

$$\left(\frac{X}{Q}\right)_f = \frac{1}{2\pi \sigma_y \bar{u} h} \quad \frac{\mu C_i}{cc} \text{ per } \frac{C_i}{sec} \qquad \text{(R.63)}$$

R.5.6 References

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Table R.5-1

CHARACTERISTICS OF NUCLEAR EXCURSIONS
WATER-MODERATED OXIDE CORE

Range	Reactivity Insertion Rate ($/sec)	Minimum Period (ms)	Peak Energy Density (cal/g)	Principal Shutdown Mechanisms
Low	<2.5	>4	<120	Doppler effect moderator effects
Medium	2-25	7-2	100-424	Doppler effect
High	>20	<3	>380	Doppler effect core disassembly

Table R.5-2

DOSE COMPUTATIONAL METHODS WIND DIRECTION PERSISTENCE[1]

Station	Direction[2]	Frequency of Duration (hr) 50%	10%	1%	0.1%	Longest No. Hr.	Longest No. Hr[3] in Any Direction	
Augusta, GA	W	2	3	8	13	18	W	48
Birmingham, AL	S	2	4	9	16	16	SSE	20
Chicago, IL	SSW	2	5	12	21	22	NNE	25
Little Rock, AR	SSW	2	4	9	17	28	SSE	28
Phoenix, AZ	E	2	3	6	9	12	E	12
Rochester, NY	WSW	2	6	13	23	28	WSW	28
Salt Lake City, UT	SSE	2	4	7	13	15	S	17
San Diego, CA	NW	2	6	12	16	17	WNW	33
Tampa, FL	ENE	2	3	7	13	14	SSW	18
Yakima, WA	W	2	5	8	14	17	WNW	19

NOTES:

1. One sector- 22.5 deg.
2. Direction examined is the one showing greatest frequency of persistent winds.
3. Longest number of hours observed may not be same direction as direction showing most frequency of persistent winds.

Table R.5-3

METEOROLOGY APPLICABLE TO DESIGN BASIS ACCIDENTS

Time/ after Accident	Diffusion Conditions Investigated: Stability Category[1]	Diffusion Conditions Investigated: % Occurence	Wind Variance during Indicated Time Period	Breathing Rate m(3)/
0-8 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100	None (centerline concentration)	3.47×10^{-4}
8-24 hr	VS-1, MS-1, N-1, N-5, U-1, U-5	100	22.5 deg sector spread	1.75×10^{-4}
1-4 days	N-2, VS-2	25	22.5 deg sector spread	2.32×10^{-4}
>4 days	N-3, VS-2	16.5	22.5 deg sector spread	2.32×10^{-4}

NOTE:

1. VS denotes very stable, MS-moderately stable, N-neutral, and U-unstable atmospheric conditions. 1, 2, 3, and 5 denote wind speeds in m/s.

Table R.5-4

VALUES OF CONSTANTS USED IN CLOUD DISPERSION CALCULATIONS

Stability	Wind Speed (m/sec)	a (m)	b (m /sec)	k (sec^{-2})	c (m n/2)	n
VS, MS	1	3.4×10^{1}	2.5×20^{-2}	8.8×10^{-4}	-	-
U	1	-	-	-	3.0×10^{-1}	2.0×10^{-1}
U	5	-	-	-	2.6×10^{-1}	2.0×10^{-1}
N	1	-	-	-	1.5×10^{-1}	2.5×10^{-1}
N	5	-	-	-	1.2×10^{-1}	2.5×10^{-1}

Table R.5-5

DEPOSITION VELOCITIES OF HALOGENS AND NOBLE GASES

Meteorology	Wind Velocity (m/sec)	Deposition Velocity (cm/sec) Noble Gases	Deposition Velocity (cm/sec) Halogens
Very stable	1	0	0.24
Moderately stable	1	0	0.34
Unstable	1	0	0.80
Unstable	5	0	4.00
Neutral	1	0	0.46
Neutral	5	0	2.70

Table R.5-6

CALCULATED AIR CONCENTRATION FOR 100 METER RELEASE HEIGHT
(Curie-sec/m^3/curie released)

Distance (m)	Activity of Interest	Meteorological Conditions					
		VS-2	MS-1	N-1	U-5	V-1	U-5
320	Noble Gases	a(1)	6.9E-15	3.3E-12(2)	a	4.3E-6	1.7E-7
	Halogens	a	6.9E-15	3.3E-12	a	4.3E-6	1.7E-7
800	Noble Gases	a	1.1E-10	1.5E-6	2.7E-8	1.3E-5	1.3E-6
	Halogens	a	1.1E-10	1.5E-6	2.7E-8	1.2E-5	1.3E-6
1,600	Noble Gases	a	1.6E-8	5.6E-6	4.7E-7	5.8E-6	7.7E-7
	Halogens	a	1.6E-8	5.6E-6	4.7E-7	5.6E-6	7.5E-7
8,000	Noble Gases	1.3E-14	1.0E-6	1.1E-6	2.1E-7	4.8E-7	8.6E-8
	Halogens	1.3E-14	1.0E-6	1.1E-6	2.0E-7	4.4E-7	7.9E-8
16,000	Noble Gases	1.1E-10	1.2E-6	4.3E-7	8.7E-8	1.7E-7	3.2E-8
	Halogens	1.1E-10	1.1E-6	3.9E-7	7.9E-8	1.5E-7	2.9E-8

NOTES:

Symbols refer to stability and wind speed, i.e., VS, MS, N, U, means very stable, moderately stable, neutral, and_unstable respectively; and 1 and 5 means 1 m/s and 5 m/s respectively. The diffusion parameter u assumed is 0.1 rad-m/s for the 1 m/sec cases and 1.0 rad-m/s for the 5 m/s cases.

1. "a" means $<1 \times 10^{-15}$

2. E-n = $x10^{-15}$

Table R.5-7

CALCULATED AIR CONCENTRATION FOR 30 METER RELEASE HEIGHT
(Curie-sec/m^3/curie released)

Distance (m)	Activity of Interest	Meteorological Conditions					
		VS-2	MS-1	N-1	N-5	U-1	U-5
320	Noble Gases	4.3E-8	9.5E-5	1.1E-4	1.4E-5	8.0E-5	1.9E-5
	Halogens	4.3E-8	9.5E-5	1.1E-4	1.4E-5	7.8E-5	1.9E-5
800	Noble Gases	2.0E-7	9.0E-5	8.4E-5	2.2E-5	2.0E-5	5.1E-6
	Halogens	2.0E-7	8.9E-5	8.1E-5	2.1E-5	1.8E-5	4.8E-6
1,600	Noble Gases	9.2E-7	6.7E-5	3.1E-5	9.3E-6	5.9E-6	1.6E-6
	Halogens	9.2E-7	6.4E-5	3.0E-5	8.8E-6	5.4E-6	1.4E-6
8,000	Noble Gases	9.1E-6	1.5E-5	2.0E-6	6.4E-7	3.3E-7	8.8E-8
	Halogens	8.6E-6	1.2E-5	1.8E-6	5.5E-7	2.9E-7	7.5E-8
16,000	Noble Gases	9.2E-6	6.8E-6	6.1E-7	1.9E-7	9.5E-8	2.5E-8
	Halogens	7.8E-6	4.7E-6	5.2E-7	1.6E-7	8.0E-8	2.1E-8

NOTES:

Symbols refer to stability and wind speed, i.e., VS, MS, N, and U mean very stable, moderately stable, neutral, and_unstable respectively, and 1 and 5 mean 1 m/s and 5 m/s respectively. The diffusion parameter u assumed is 0.1 radian-m/s for the 1 m/s cases and 1.0 radian-m/s for the 5 m/s cases.

1. E-n = x10-n

Table R.5-8

THYROID DOSE CONVERSION FACTORS

Isotope	Effective Half Life (days)	F_a	E (meV/dis)	Ci rad/curie inhaled
I-131	7.6×10^{0}	2.3×10^{-1}	2.3×10^{-1}	1.48×10^{6}
I-132	9.7×10^{-2}	2.3×10^{-1}	6.5×10^{-1}	5.65×10^{4}
I-133	8.7×10^{-1}	2.3×10^{-1}	5.4×10^{-1}	4.21×10^{5}
I-134	3.6×10^{-2}	2.3×10^{-1}	8.2×10^{-1}	2.64×10^{4}
I-135	2.8×10^{-1}	2.3×10^{-1}	5.2×10^{-1}	1.30×10^{5}

FUEL ROD DETAIL

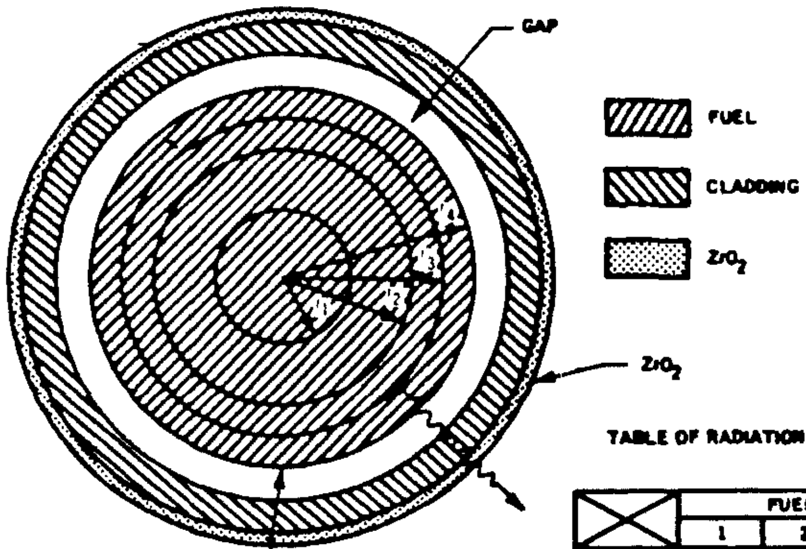
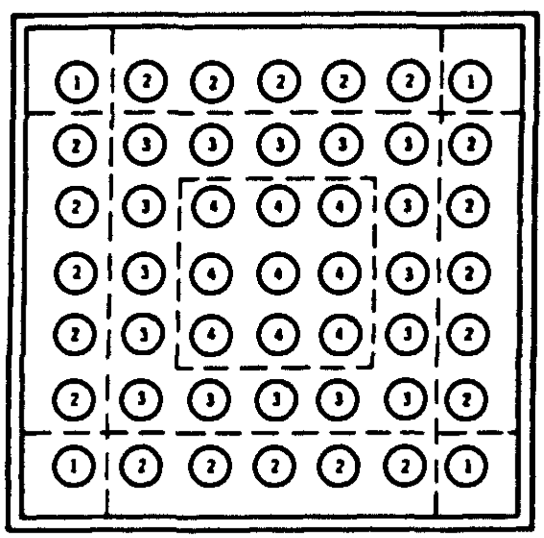


TABLE OF RADIATION HEAT TRANSFER COEFFICIENTS

RECEPTOR

SOURCE		FUEL ROD TYPE 1	FUEL ROD TYPE 2	FUEL ROD TYPE 3	FUEL ROD TYPE 4	CHANNEL SIDE WALL
FUEL ROD TYPE	1	-	0.206	0.0040	0.007	0.2914
FUEL ROD TYPE	2	0.04%	-	0.177	0.0287	0.1786
FUEL ROD TYPE	3	0.0212	0.221	-	0.150	0.0235
FUEL ROD TYPE	4	0.003	0.0537	0.264	-	NIL
CHANNEL SIDE WALL		0.0075	0.299	0.031	NIL	-

CHANNEL CROSS SECTION



FUEL BUNDLE DETAIL

CHANNEL		
ROD TYPE	No. OF RODS	ROD POWER FACTOR
1	4	1.24
2	20	1.00
3	16	0.91
4	9	0.85

Figure R.5-1. Initial Core Fuel Rod and Fuel Bundle Details

R.6 EVALUATION USING STANDARD NRC APPROACH (INITIAL CORE)

R.6.1 Evaluation of Station Systems Using TID-14844 Source Terms

An evaluation has been made of the performance of the Pilgrim Station containment and engineered safety features, using the assumptions of TID-14844 as a basis for fission product source terms. This TID-14844 source term evaluation is presented in response to recent requests by the Advisory Committee on Reactor Safeguards that such analyses be supplied by the Applicant. The Applicant does not believe that the fission product source terms postulated in TID-14844 are appropriate for the Pilgrim Nuclear Power Station, considering the design of the Core Standby Cooling Systems (CSCS) incorporated to protect against such gross fission product releases. This TID-14844 evaluation is presented to indicate the large margin of design conservatism available in the containment systems provided for the Pilgrim Station. The Applicant's engineering evaluation of the protection afforded to the health and safety of the public by the overall station design is reported in the previous portions of Appendix R.

It should be noted that an apparent inconsistency exists in the calculations of containment and equipment capability and those calculations used to evaluate the offsite radiological exposures. For equipment capabilities, it is assumed that all halogen and particulate activity released from the core is retained by the suppression pool. It is also assumed that 25 percent of the halogen activity and 100 percent of the noble gas and particulate activity are airborne in the primary containment. For the secondary containment and equipment capabilities, it was assumed that 100 percent mixing in the secondary containment and 100 percent filter efficiency were applicable. For the purpose of evaluating the offsite radiological exposures, it was assumed that the suppression pool had no reduction effects for either the particulate, noble gas, or halogen activities. It was also assumed that zero mixing in the secondary containment and 90 percent filter efficiency were applicable. It is therefore apparent that the above conditions cannot be applicable to equipment and containment capabilities and offsite radiological exposures. The above assumptions and the following analyses are in accordance with requirements set forth for the Hatch Nuclear Power Station, Docket No. 50-321.

R.6.1.1 Source Term Assumptions

For the purpose of calculating the worst dose, heat loading, airborne, or waterborne activity, the following assumptions were made:

1. Activity in suppression pool - 50 percent of the core halogen inventory and 1 percent of the core particulate inventory are instantaneously released to the suppression pool.

2. Activity airborne in primary containment - 100 percent of the core noble gas activity, 25 percent of the core halogen activity, and 1 percent of the core particulate activity are airborne in the primary containment.

3. Activity airborne in secondary containment - the airborne primary containment activity noted in item 2 is released at a constant leak rate of 0.5 percent/day to the secondary containment, uniformly mixed in the secondary containment, and released to the Standby Gas Treatment System at the rate of 1 air change/day.

4. Filter activity and heat loading - the airborne primary containment activity noted in item 2 is released at a constant leak rate of 0.5 percent/day directly to the Standby Gas Treatment System where the filter efficiency is assumed to be 100 percent.

Use of the above assumptions results in the activities and heat loadings presented on Table R.6-1. The effect of primary containment leak rate on these values can be determined by using the appropriate factor as given on Table R.6-2. The relative factor of increase shown on Table R.6-2 shows the effect of increasing the leak rate from 0.5 percent/day to 2 percent/day. As noted on Table R.6-2, the factor of increase is time dependent. It should be noted that the factor of increase is applicable to only the activity or heat loading in the secondary containment or on filters. A leak rate of greater than 0.5 percent/day would reduce the activity and heat load in the primary containment.

R.6.1.2 Standby Gas Treatment System

The data presented and the discussion that follows show that the Standby Gas Treatment System, as designed, will tolerate service conditions that arise from utilizing TID-14844 source terms for a core melt loss of coolant accident (LOCA), when considering the proposed Technical Specification primary containment leakage rate of 2 percent. Data is included for a 0.5 percent leak rate, although this case is less severe than the 2 percent case. The design conservatism of the Standby Gas Treatment System is shown on Table R.6-3.

The Standby Gas Treatment System has been designed to prevent a single active component failure from preventing operation of the system. It is designed to process up to 4,000 standard ft^3/min of air from the secondary containment as well as to maintain a minimum cooling air flow of 500 standard ft^3/min through the idle filter train. The temperatures shown on Table R.6-3 represent filter media (high efficiency particulate adsorber (HEPA) filter) or carbon (charcoal filter) temperatures at air flows of 250 standard ft^3/min and 4,000 standard ft^3/min; the higher of the two temperatures indicating the lower air flow case.

High Efficiency Particulate Air Filter

These filters are rated at 250°F temperature for continuous service. Using data on Table R.6-3 with 500 standard ft^3/min air flow (due to a failure resulting in a transfer to the "B" train), a filter efficiency of 100 percent and 1 percent of the core particulate activity available for release at a 2.0 percent primary containment leakage rate, after 30 days of continuous operation the temperature of the filter media would remain within acceptable limits.

The PNPS design as shown on Figure 5.3-2 incorporates two such filters, one upstream of the charcoal filter and one downstream. The analysis above applies to the upstream filter since all fission products are assumed to be contained on it. However, the downstream filter acts as a backup filter and would retain particulates getting past the upstream filter in the event that the upstream filter developed leaks.

Charcoal Filters

Each charcoal filter train contains a minimum of 504 lb of charcoal.

Using data on Table R.6-3, the maximum temperature of the charcoal filter would be reached at 500 standard ft^3/min air flow (due to a failure resulting in a transfer to the "B" train) and 25 percent core halogen inventory release from the primary containment of a leak rate of 2.0 percent/day and retained on the filter with a 100 percent filter efficiency. This peak temperature occurs within about 10 days, assuming TID-14844 source terms. The maximum charcoal temperature is calculated to be well below the ignition temperature of 625°F and the charcoal de-adsorbtion temperature of 330°F. Heaters are provided to maintain the relative humidity of the air stream below 70 percent.

R.6.1.3 Core Standby Cooling System Components

There are no CSCS components located within the primary containment that would be required to function following the postulated LOCA that could suffer significant radiation damage other than electrical penetrations that are discussed separately. Valves in the Residual Heat Removal (RHR) and Core Spray Systems are fluid actuated check valves and consequently not subject to loss of operability because of radiation damage.

Components within the secondary containment that could be affected by radiation doses are pump seals and motor operator valves as discussed below.

1. Pumps - the RHR and core spray pumps that would be expected to operate following a LOCA have seals consisting of Buna "N" O-rings and carbon seal rings. Irradiation tests on the Buna "N" material indicate that an integrated dose of 4 x 10^6 rad results in a 25 percent decrease in compression set

properties. Carbon materials have a dose threshold on the order of 3 x 10^{11} rad. Source terms presented on Table R.6-1 were converted to the equivalent integrated dose rates as shown on Table R.6-4. For the outside surface of a schedule 80, 24 in dia pipe, the integrated dose rate for 180 days is about 6.2 x 10^5 rad. It is expected that this value would represent the maximum dose received by the seal; therefore, failure due to radiation damage is not expected.

2. Valves - no valve operation is expected to be required after the accident is contained and the RHR System is placed in the shutdown cooling mode. This event could occur about 8 hrs after the accident so that the total integrated dose given on Table R.6-4 at 12 hrs of 5.9 x 10^4 rad adequately represents the maximum expected equipment dose. No loss of valve operability would be experienced at this low dose.

The electrical components associated with items 1 and 2 above are not affected by the radiation fields that exist in the corner room locations. Electrical equipment is these areas would receive lower doses than the pump seals (Table R.6-4) which is substantially less than the allowable dose for electrical components (10^7 rad). Design margins between maximum expected doses shown and equipment capabilities illustrate the degree of conservatism available.

R.6.1.4 Electrical Penetrations

As shown on Table R.6-4, the maximum total integrated dose due to TID-14844 source terms at the interior surface of the drywell would be 2.6 x 10^7 rad after 180 days. The electrical penetrations are sealed and insulated with a silicon ceramic inorganic compound capable of accepting a total integrated dose of 10^8 rad without damage. In addition, the penetration is sealed both inside and outside the drywell, with the outer seal receiving less than 10^5 rad total integrated dose. Thus, the seals would be fully capable of maintaining containment integrity.

R.6.1.5 Control Room

Radiation protection for operating personnel in the control room under accident conditions is provided by operation of either of two high efficiency air filtration trains in conjunction with the installed control room shielding. Two 1,000 ft^3/min high efficiency filter trains are provided in parallel with the normal outside air inlet duct. Each of the filter trains consists of inlet and outlet isolation dampers, a heating coil, prefilter, HEPA, charcoal filter, and final (HEPA). Should fission products leaving the main stack reach ground level during a brief atmospheric fumigation, the Control Room Ventilation System can be isolated by dampers from the outside air intake. The Ventilation System is designed to operate on a closed cycle, if necessary. See Section 10.17.

Shielding

Biological dose rates inside the control room have been reevaluated based on fission product source terms developed using the assumptions contained in TID-14844. The fission product activity in the secondary containment as a function of time after a LOCA may be determined by applying the appropriate relative factor of increase (as demonstrated on Table R.6-2) to the source terms given on Table R.6-1. This activity level results from a primary containment airborne activity level of 100 percent of the noble gases, 25 percent of the halogens, and 1.0 percent of the solids contained in the core at the time of the accident.

The gamma ray energy spectrum of the fission product activity in the secondary containment 1 hr and 1 day after the accident are shown on Tables R.6-5 and R.6-6. The average gamma ray activity decreases in energy as a function of time. The average energies may be grouped as a function of time after the accident as follows:

Time after Accident	Average Photon Energy
0-12 hr	0.6 MeV
12-48 hr	0.3 MeV
2-30 days	0.8 MeV
30 days - 6 months	0.02 MeV

Activity levels were based on a 0.5 percent leak rate from the primary containment. Variations in dose rate attributable to variations in leak rate were obtained by scaling the activity levels by the relative increase factors on Table R.6-2. Activity levels presented are the total activities contained in the secondary containment.

Dose rate calculations were performed using a point kernel type shielding computer code. The Reactor Building was represented by a cylindrical drywell of solid concrete; 1 ft thick concrete floors at the 23, 51, 74, and 117 ft levels, and the 1.5 ft thick concrete Reactor Building wall separating the Reactor Building and the Control Building. The control room was modeled to include the 2 ft control room wall, the 2 ft ceiling in the control room, the 1 ft control room floor, the 1 ft south wall and the building floor at 51 ft. The source was distributed throughout the Reactor Building based on total mixing. Each floor was treated as a separate source region. Six gamma ray energy groups were used. See Tables R.6-5 and R.6-6. The results of these calculations are shown on Table R.6-7.

Total whole body doses due to Reactor Building shine the first 30 days following a TID-14844 LOCA assuming control room occupancy for 8 hr/day for 22 of 30 days are given on Table R.6-7. Total thyroid dose received in the control room, based upon the above source term and occupancy factor, and a 95 percent efficiency for the Main Control Room Environmental Control System filters is shown on Table R.6-8. In addition, the thyroid and whole body radiation exposures

of control room personnel due to the periodic need for personnel to leave the main control room were evaluated.

The following radiation sources were considered when evaluating the whole body dose received by control room personnel while in the control room and while traveling to and from the control room across the site: gamma radiation from fission products in the reactor building, gamma radiation from fission products in the station main stack, and gamma radiation from the fission product cloud released from the main stack.

The thyroid dose received by control room personnel was evaluated considering exposure while in the control room and while traveling to and from the control room in a stack-released fission product cloud containing radioactive iodines. A fumigation type meteorology condition lasting 1/2 hour was assumed to occur once during the first 1/2 hour of the accident. The contribution to the whole body dose in the control room from main stack shine, treated as a time dependent line source for the computation, was found to be negligible. The stack shine contribution was also found to be negligible for personnel traveling across the site during shift changes.

The main control room environmental control system is described in detail above and in Section 10.17. It is not expected that the control room outside air additions will contain significant concentrations of fission products following a LOCA release of TID-14844 fission product source term except during the assumed fumigation condition considering the design of the secondary containment system.

In the event of an accident, the control room operator may manually initiate high efficiency filtration of the outside air supplied to the control room by activating one of two 1,000 ft^3/min high efficiency filter trains which are provided in parallel with the normal outside air inlet duct. The system is designed to effect isolation of the main control room from unfiltered outside air and to maintain a positive pressure with respect to other station ventilation zones. The ventilation system may be operated on a closed cycle (no outside air addition) during a fumigation condition if necessary.

A process radiation monitor detects high radiation in the outside air intake duct, and an area radiation monitor in the control room monitors control room area radiation levels. Those two monitors alarm in the main control room upon detection of high radiation conditions.

In evaluating the capability of the control room ventilation system, an estimate of the maximum concentration of activity prior to intake into the ventilation system was made. Since the plume is released from the main stack, the normal dispersion of the plume results in negligible concentration of activity at the elevation of the control room air intake duct on the lee side of the turbine building. The only meteorological condition that could possibly cause a significant concentration at the air intake from an elevated plume is a fumigation condition. Of the two types of fumigation, the sea breeze

type could not occur since the sloping boundary separating stable air above from unstable air below could not intersect the elevating plume near the plant. The plume would be much higher than this boundary layer. However, the upward dissipation of a surface based nocturnal radiation inversion could possibly reach the lower portion of the elevated plume and mix it downward. Such a condition lasts no longer than 1/2 hr/day. This latter case, assumed to occur once over the course of the accident, was evaluated to determine the maximum air concentrations prior to entering the ventilation intake to the control room.

The probable radiation exposure of personnel due to the periodic need for personnel to leave the control room and return was evaluated assuming normal rotation of operating shifts and allowing 5 min for the incoming or outgoing personnel to travel approximately 1.4 mi from the Turbine Building exit along the access road to the site boundary. Personnel were assumed to travel this route twice a day on each of 22 days/month. Exposure to the various radiation sources resulting from the release of the TID-14884 fission product source term in the primary containment was assumed to be negligible along the 170 ft route from the main control room to the Turbine Building exit due to the shielding provided by the surrounding concrete floors and walls and the short travel time in the structure. No credit was taken anywhere along the route to the site boundary for breathing apparatus or special whole body shielding. Figures R.6-1 and R.6-2 show the route traveled by the control room personnel.

The time dependent distribution of fission products in the primary containment, Reactor Building, and environment was determined assuming a constant primary containment leak rate of 1.5 percent/day, a flow rate of 100 percent/day of the Reactor Building volume through the Standby Gas Treatment System, and a charcoal filter efficiency of 95 percent in the Standby Gas Treatment System and Main Control Room Environmental Control System. Calculations were also performed for a primary containment leak rate of 0.635 percent/day. For Reactor Building shine calculations, fission products were assumed to be held up for decay and uniformly distributed in the free volume of the building. For fission product cloud shine and thyroid dose calculations, fission products were assumed to pass straight through the Reactor Building, Standby Gas Treatment System, and main stack without holdup.

Whole body dose calculations were performed using a point kernel type shielding computer code. The Reactor Building was represented by a cylindrical volumetric source having 6 in concrete walls. It was assumed that for the travel route taken by control room personnel, the Turbine Building was an effective shield for all Reactor Building levels below el 105 ft and that the distance of closest approach to the Reactor Building for personnel was approximately 200 ft. The whole body and thyroid doses from fumigation and from normal diffusion conditions were calculated assuming TID sources, typical AEC meteorology, and assuming submersion in a semi-infinite cloud having a uniform fission product distribution averaged over the first 30 days of the accident and also averaged over the distance from the Turbine Building to the site boundary.

Based on the above assumptions, the total whole body dose from ingress, egress, and occupancy to a control room operator over the first 30 days of the accident was calculated to be 1.3 rem for a primary containment leak rate of 1.5 percent/day. For a leak rate of 0.635 percent/day the calculated dose was 0.55 rem. The total thyroid dose from ingress, egress, and occupancy to a control room operator over the first 30 days of the accident was calculated to be 9.5 rem for a primary containment leak rate of 1.5 percent/day. For a leak rate of 0.635 percent/day the calculated thyroid dose was 4.0 rem.

R.6.1.6 Materials Within the Primary Containment

No material within the primary containment will fail by decomposition or corrosion and affect vital systems. The effect of hydrogen evolution by radiolysis and the General Electric program for determining its effect is discussed in Appendix J.

The interior surface of the suppression pool has been coated with Carboline 368 which will withstand the accident environment without failure. This coating in wet atmospheres will tolerate radiation doses of 109 Routins without damage other than becoming discolored. Since the integrated 180-day dose in the suppression pool would be less than that in the drywell (Table R.6-4) of 2.6 x 107 Routines, no failure will occur.

R.6.1.7 Summary

All of the necessary safety-related core cooling, containment, and/or design basis accident mitigating systems or components have sufficient design conservatism to operate and perform their intended function while subject to the ultraconservative TID-14844 (AEC) sources without exceeding 10CFR100 guideline limits at the site boundaries.

R.6.2 Evaluation of Accident Doses Using AEC/DRL Assumptions Incorporated with TID-14844 Source Terms

The following sections present an analysis of accident doses using the AEC/DRL assumptions regarding Containment System performance and meteorological conditions combined with TID-14844 source terms. The AEC/DRL assumptions are not publicly documented and the following sections represent a summary of the Applicant's interpretation of the normal AEC methods of analysis.

R.6.2.1 Loss of Coolant Accident Assumptions (Elevated Release)

See Section 14.5.3.2.2 [Historical information. See Revision 23 or previous for cross-reference.]

R.6.2.2 Refueling Accident Assumptions (Elevated Release)

1. Assumptions 1, 4, 5, 7, 8, and 9 of the LCCA are applicable.

2. Each damaged fuel rod contains 50 percent more activity than the average fuel rod in the core.

3. Twenty percent of the noble gases and 10 percent of the iodine contained within the damaged rods are released within 2 hours.

4. Ninety-five percent of the iodine released from the rods is retained by the refueling pool water.

5. One bundle is assumed to be damaged (49 rods).

6. Meteorology - Refueling Accident

   For the duration of the accident, the concentrations are those at the plume centerline with a 1 m/s wind speed. For the exclusion area calculations, the meteorological conditions are fumigation Pasquill F for the first 1/2 hour and extremely unstable Pasquill A for the following 1-1/2 hour.

   For the low population zone calculations, the meteorological conditions are moderately stable Pasquill F for the duration of the accident. It should be emphasized that these conditions are impossible to attain simultaneously and are evaluated for the express purpose of maximizing the offsite dose at the above respective locations.

R.6.2.3 Steam Line Break Accident Assumptions (Ground Level Release)

See Section 14.5.4.2.2. [Historical information. See Revision 23 or previous for cross-reference]

R.6.2.4 Control Rod Drop Accident Assumptions (Ground Level Release)

1. Assumptions 1, 7, 8, and 9 for the LOCA are applicable.

2. The damaged fuel rods are from the highest burnup (activity) regions of core.

3. One hundred percent of the noble gases and 50 percent of the iodines are released from the damage rods.

4. Ninety percent of the iodine released from the damaged fuel rods is retained by the reactor water.

5. The radionuclides released from the reactor water travel to the condenser where 50 percent of the iodine experiences plateout.

6. The mechanical vacuum pump and Offgas System is isolated and the leak rate from the condenser if 0.5 percent/day.

7. The accident duration is 24 hours.

8. Meteorology - for the duration of the accident the concentrations are those for Pasquill F, 1 m/s wind speed. For the exclusion area calculations, the concentrations are

those at the plume centerline. For the low population zone calculations, the concentrations are those at the plume centerline for the first 8 hrs. For the remaining 16 hrs the plume stays within a 22.5 deg sector.

9. Building dilution effects are considered for exclusion area with maximum dilutions factors of 1/3.

R.6.2.5 Radiological Dose Evaluation

Radiological consequences of the design basis accidents have been evaluated using all of the above assumptions of the AEC/DRL (see Sections R.6.2.1 through R.6.2.4) and the results are reported on Table R.6-9. As stated in Section R.6.1, these results do not represent the Applicant's engineering evaluation of the protection afforded the health and safety of the public by the overall station design.

To demonstrate the relative effects of the various factors involved in making radiological dose calculations, Tables R.6-10 through R.6-13 list many assumptions and the effect that changing these assumptions has upon the resulting calculated dose. Note that many of these factors are nonlinear in nature and therefore cannot be interpolated or extrapolated without performing sophisticated calculations.

R.6.2.6 Discussion of Assumptions

The LOCA has generally been interpreted as a complete core melt (10 CFR 100.11(a) Note 1) without consideration of the geometry aspects of molten fuel and its resultant consequences. Only the fission product release has been considered. Such a situation would only be evaluated in light of little or no core cooling protection. It states in 10 CFR 100.10 that "... the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

1. Characteristics of reactor design and proposed operation including:

   a. Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials

   b. The extent to which generally accepted engineering standards are applied to the design of the reactor

   c. The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability of consequences of accidental release of radioactive materials

d. The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive materials to the environment can occur.

This BWR is designed to keep the thermal response of the core below clad melt. Because of this, the use of TID-14844 assumptions of core melt fission product release do not apply for this BWR.

The previously referenced topical report (APED-5756) summarizes the technical basis for all assumptions and models used on current generation GE BWRs. The use of this topical report in evaluating the radiological aspects of Pilgrim's BWR is consistent with good engineering practice and actual design.

Table R.6-1

ACTIVITY, MASS LOADING, AND HEAT LOADING AT VARIOUS LOCATIONS
FOR TID-14844 SOURCE TERM ASSUMPTIONS

(Time and Parametric Value)

	1 Hr	10 Hr	1 Day	10 Days	30 Days	Peak Value and Time	
Activity in S.P. (curie)	2.5×10^8	1.1×10^8	7.2×10^7	2.1×10^7	1.0×10^7	T = 0.0 hr	3.3×10^8
Heat Load S.P. (kW)	2.9×10^3	7.4×10^2	3.5×10^2	9×10^1	5.2×10^1	T = 0.0 hr	4.2×10^3
Activity Airborne P.C. (curies)	3.7×10^8	2.1×10^8	1.6×10^8	5.2×10^8	1.4×10^7	T = 0.0 hr	5.6×10^8
Heat Load P.C. (kW)	2.5×10^3	6.8×10^2	3.9×10^2	9.3×10^1	4.2×10^1	T = 0.0 hr	5.7×10^3
Activity Airborne S.C. (curies)	6.8×10^4	3.0×10^5	4.7×10^5	2×10^5	3.3×10^4	48 hrs	5.2×10^5
Heat Load S.C. (W)	4.6×10^2	9.2×10^2	9.8×10^2	3.5×10^2	7.6×10^1	24 hrs	9.8×10^2
Activity on HEPA (curies)	4.4×10^3	2.9×10^4	5.5×10^4	3.0×10^5	6×10^5	30 days	6.0×10^5
Heat Load on HEPA (W)	1.4×10^1	7.0×10^1	1.2×10^2	5.2×10^2	9.8×10^2	30 days	9.8×10^2
Activity on C.F. (curies)	1.9×10^4	8.8×10^4	1.4×10^5	3.0×10^5	1.6×10^5	280 hrs	3.1×10^5
Head Load on C.F. (W)	8.4×10^1	3×10^2	3.8×10^2	5.8×10^2	3.2×10^2	280 hrs	6.0×10^2

KEY

S.P. - Suppression Pool
P.C. - Primary Containment
S.C. - Secondary Containment
HEPA - High Efficiency Particulate Absorber Filter
C.F. - Charcoal Filter

Table R.6-2

RELATIVE FACTOR OF INCREASE OVER A PRIMARY
CONTAINMENT LEAK RATE OF 0.5 PERCENT PER DAY

	Factor of Increase Over Leak Rate of 0.5%/Day	
Time	0.5%/Day	2.0%/Day
1 Hr	1	4
10 Hr	1	4
1 Day	1	4
10 Days	1	3.7
30 Days	1	3.3

Table R.6-3

STANDBY GAS TREATMENT SYSTEM PERFORMANCE

	Design Temperature Capability	Assuming TID-14844 Sources: 0.5%/Day Leakage	Assuming TID-14844 Sources: 2.0%/Day Leakage
Iodine Accumulated		600g	1975 g
HEPA Filter	250°F		
Watts		980(1)	3,230(1)
Temperature at 250 standard ft^3/min		127°F	164°F
Temperature at 4,000 standard ft^3/min		101°F	102°F
Charcoal Filter	625°F/330°F(3)		
Watts		600(2)	2,200(2)
Temperature at 250 standard $ft.^3$/min		135°F	198°F
Temperature at 4,000 standard $ft.^3$/min		105°F	120°F

NOTES:

1. Peak heat load evaluated at approximately 30 days
2. Peak heat load occurs in approximately 10 days
3. 625°F is combustion temperature, 330°F is de-adsorbtion temperature

Table R.6-4

DOSE RATES FOR VARIOUS EQUIPMENT OR LOCATIONS BASED ON TID-14844 FISSION PRODUCT RELEASE ASSUMPTIONS

Location or Equipment	Max Dose Rate (R/hr)	Integrated Dose (Rad) for 12 Hr	2 Days	30 Days	180 Days
Surface 24 in. Sched 80 pipe	$1.1x10^4$	$5.9x10^4$	$2.0x10^5$	$4.4x10^5$	$6.2x10^5$
Interior Surface Drywell	$7.8x10^5$	$3.8x10^6$	$9.4x10^6$	$1.8x10^7$	$2.6x10^7$
Floor or Corner Compartment Containing Core Spray Pump Seals	$2.6x10^1$	$1.0x10^2$	$1.0x10^3$	$3.0x10^3$	$7.1x10^3$
Pump Seals	$1.1x10^4$	$5.9x10^4$	$2.0x10^5$	$4.4x10^4$	$6.2x10^5$
Secondary Containment Ground Elevation	$1.0x10^2$	$4.2x10^2$	$3.8x10^3$	$1.1x10^4$	$2.6x10^4$
Refueling Floor	$4.2x10^2$	$1.7x10^3$	$1.6x10^4$	$4.5x10^4$	$1.1x10^5$

Table R.6-5

GAMMA RAY ENERGY SPECTRUM OF FISSION PRODUCTS IN THE SECONDARY CONTAINMENT
(1 Hr After Loss of Coolant Accident Total Activity 32.5 Curies/MWt)

Nuclide	Curies/ MWt	Gamma Ray Energy MeV/ sec-MWt x 10^{10}					
		0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5
Kr 83^m	0.042	0.0176					
85^m	1.1	0.8569					
85	0.061	0.0475					
87	1.4	2.0874	0.0835			0.637	.590
88	2.7	2.7405	5.5188		6.193	3.886	
Xe 131^m	0.037	0.0224					
133^m	0.38	0.3271					
133	11.00	33.000					
135	0.23	0.447					
135^m	1.0	1.757	0.011				
138	0.78	1.386	0.117				
I 131	1.5	3.205	0.081	3.3898			
132	1.7	0.394	11.271				
133	2.4	5.524	0.8112	0.2465			
134	1.4	0.194	9.310	2.2708	0.882		
135	2.4	0.256	0.840	9.9960	4.658		
Totals	29.4	52.2624	28.796	15.903	11.734	4.535	5.590

$$= \frac{\text{Total Energy}}{\text{Total Photons}} = 0.64 \text{ MeV}$$

MeV	MeV/sec-MWt
0-0.5	5.2262×10^{11}
0.5-1.0	2.8796×10^{11}
1.0-1.5	1.5903×10^{11}
1.5-2.0	1.1734×10^{11}
2.0-2.5	4.5230×10^{10}
>2.5	5.5900×10^{10}

Table R.6-6

GAMMA RAY ENERGY SPECTRUM OF FISSION PRODUCTS
IN THE SECONDARY CONTAINMENT
(24 Hr After Loss of Coolant Accident Total Activity = 225
Curies/MWt)

		Gamma Ray Energy MeV/ sec-MWt x 10^{-10}					
Nuclide	Curies/ MWt	0.0-0.5	0.5-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5
Kr 83^m	0.0012	0.001					
85^m	0.47	0.366					
85	0.95	0.740					
87	1.1-4					0.0001	00044
88	0.14	0.142	0.286		0.321	2.0150	
Xe 131^m	0.54	0.327					
133^m	4.4	0.3788					
133	150.0	45.000					
135^m	0.0						
135	5.0	4.625	0.030				
138	0.0						
I 131	22.0	46.970	1.188				
132	0.25	0.005	0.165	0.004			
133	17.0	39.134	5.746				
134	0.						
135	3.4	0.363	1.190	14.161	6.599		
Totals	203.9	141.465	8.605	14.165	6.9205	2.0151	0.0044

E = 0.197 MeV

MeV	MeV/sec-MWt
0.0-0.5	1.4146×10^{12}
0.5-1.0	8.6058×10^{10}
1.0-1.5	1.4165×10^{11}
1.5-2.0	6.9205×10^{10}
2.0-2.5	2.0150×10^{10}
>2.5	4.4000×10^{6}

Table R.6-7

TID-14844 EVALUATION WHOLE BODY DOSE IN THE CONTROL ROOM
(8 Hr/Day, 22 Days of the First Month

Primary Containment Leak Rate	Whole Body Dose
0.635%/day	0.30 rem
1.5%/day	0.71 rem

Table R.6-8

TID-14844 EVALUATION THYROID DOSE IN THE CONTROL ROOM
(8 Hr/Day, 22 Days of the First Month)

Primary Containment Leak Rate	Whole Body Dose
0.635%/day	2.2 rem
1.5%/day	5.1 rem

Table R.6-9

INITIAL CORE RADIOLOGICAL DOSES FOR DESIGN BASIS ACCIDENTS BASED ON TID-14844 ASSUMPTIONS

Accident Evaluated		Dose Location	Radiological Exposure (rem) Cloud	Inhalation
Loss of Coolant	See Table 14.5-2(3)			
Refueling(1)	(2 hour)	EA(2)	1.5×10^{-1}	1.4×10^{1}
	(30 day)	LPZ	4.7×10^{-2}	1.8×10^{0}
Steamline Break	See Table 14.5-2(3)			
Control Rod Drop	(2 hour)	EA(2)	3.2×10^{-2}	1.6×10^{1}
	(30 day)	LPZ	9.3×10^{-3}	1.5×10^{0}

NOTES:

EA - Exclusion Area (See Figure 1.6-3)
LPZ - Low Population Zone (6,840 m)

1. As a result of terrain variances, the effective stack height was used for the loss of coolant and refueling accidents. For both events, the effective stack height is 91 m at a distance of 320 m, and is 0 at a distance of 6,840 m. Also, the inhalation dose was calculated assuming 95 percent filter efficiency.

2. Highest 2 hour dose at any site boundary.

3. Historical information. See Revision 23 or previous for cross-reference.

Table R.6-10

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
LOSS OF COOLANT ACCIDENT

Assumptions	Design Base Evaluation	Assumed AEC Evaluation	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission products released to drywell	1.8% noble gases[1], 0.32% iodines released from 25% of the fuel rods which are assumed to be perforated. 1% of total iodines in organic form. Negligible solids	100% noble gases, 50% iodines, and 1% solids in total core inventory. 5% of total iodine in organic form	625	220
Iodine retained in water	Based on partition factor of 100 between the volumes of air and water in suppression chamber and drywell	None	12.5[2]	1
Elemental iodine plateout in drywell	50%	50%	1	1
Leakage rate from primary containment	Function of drywell pressure: peaks close to 0.5% volume/day	0.635% volume/day, constant throughout accident	1.3 (2 hr)[3] or 1 (30 day)	1.3 (2 hr) or 1 (30 day)
Uniform mixing in Reactor Building	Yes	No	22 (2 hr) 1.2 (30 day)	28 (2 hr) 1.1 (30 day)
Iodine filter efficiency	99%	90% (95% for solids)	10	1

Table R.6-10 (Continued)

Assumptions	Assumed AEC Design Base Evaluation	Factor Affecting Evaluation	Thyroid Dose	Whole Body Dose
Effectiveness of stack	Yes	Yes	1	1
Leakage rate from secondary containment	100%/day	100%/day	1	1

NOTES:

1. One percent of iodines released in organic form, which is not reduced by fallout or plateout in the drywell and reactor building. Elemental iodines are carried into suppression pool during blowdown and a fraction retained according to the assumed equilibrium partition factor of 100. Iodines become airborne in the suppression chamber and drywell before leaking out to the secondary containment.

2. Takes into account the organic iodine fraction.

3. Two hour dose is evaluated at site boundary of 320 m; 30 day dose is evaluated at low population zone of 4.800 m.

Table R.6-11

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
REFUELING ACCIDENT

Assumptions	Design Base Evaluation	Assumed AEC Evaluation	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission products release to reactor water[1]	1.8% noble gases, 0.32% iodines from 111 perforated rods, solids negligible	20% noble gases 10% iodines from 49 perforated fuel rods[2]	0.31	11.1
Iodines retained in water	Equilibrium partition factor[3] of 100 for iodines and water	90%[2]	0.4	1
Plateout of iodines in Reactor Building	None	50%	0.5	1
Uniform mixing in refueling chamber	Yes	No	14 (2 hr)[4] 1.3 (30 days)	18 (2 hr) 1.1 (30 days)
	Fission products exponentially released from water to Reactor Building till exhausted	Fission products exponentially released from water in 2 hrs		
Iodine filter efficiency	99%	90% (95% for solids)	10	1

Table R.6-11 (Continued)

Assumptions	Assumed AEC Design Base Evaluation	Evaluation	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Effectiveness of stack	Yes	Yes	1	1
Leakage rate from secondary containment	100%/day	100%/day	1	1

NOTES:

1. Accident occurs 24 hrs after shutdown.
2. Assumptions in Hatch (Docket No. 50.321) evaluation.
3. Amount of retention depends on the ration of air space to water space. In this case, the equivalent value of 75 percent is obtained.
4. Two hr dose is evaluated at site boundary of about 320 m. Thirty day dose is evaluated at low population zone of 4,800 m.

Table R.6-12

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
STEAM LINE BREAK ACCIDENT

Assumptions	Design Base Evaluation	Assumed AEC Evaluation	Factor Affecting	
			Thyroid Dose	Whole Body Dose
Steam and water mass lost in blow-down (10.5 sec closure)	85,000 lb (25,000 lb steam, 60,000 lb water)	85,000 lb	1	1
Total fission gases released	70.8 Ci iodines, and 5.7 Ci noble gases[2]	Proportional to operating limit, assumed 10 times the base case value	10	10[2]
Concentration in water and steam	Equilibrium separation	Equilibrium separation	1	1
Steam cloud rise	No	No	1	1

NOTES:

1. Based on fission product concentration in coolant such that the offgas release rate at stack reaches the maximum expected value of 100,000 Ci/sec.

2. In the steam line break accident, the noble gases contribution to the whole body dose is insignificant.

Table R.6-13

SENSITIVITY OF DOSES TO VARIATION OF ASSUMPTIONS
CONTROL ROD DROP ACCIDENT

Assumptions	Design Base Evaluation	Assumed AEC Evaluation	Factor Affecting Thyroid Dose	Factor Affecting Whole Body Dose
Fission products released to water	1.8% noble gases, 0.32% iodines from 330 perforated fuel rods. Solids negligible	100% noble gases, 50% iodines from 330 perforated fuel rods	155	55
Noble gases carry-over to condenser hotwell	Uniformly mixed with steam, carried over at 5.0% steam flow rate, isolation valve closure at 10.5 sec	100%	1	10
Iodine carryover to condenser hotwell	Retention in water.(1) uniform mixing in steam dome, carryover at 5.0% steam flow, and isolation at 10.5 sec	10%	2,700	1.0
Iodine plateout in condenser hotwell	None	50%	0.5	1

NOTES:

1. Amount of retention in condenser hotwell water depends on relative ratio of steam space to water space. The "base" case uses an equilibrium partition factor of 100 and a steam-water space ratio of about 12.

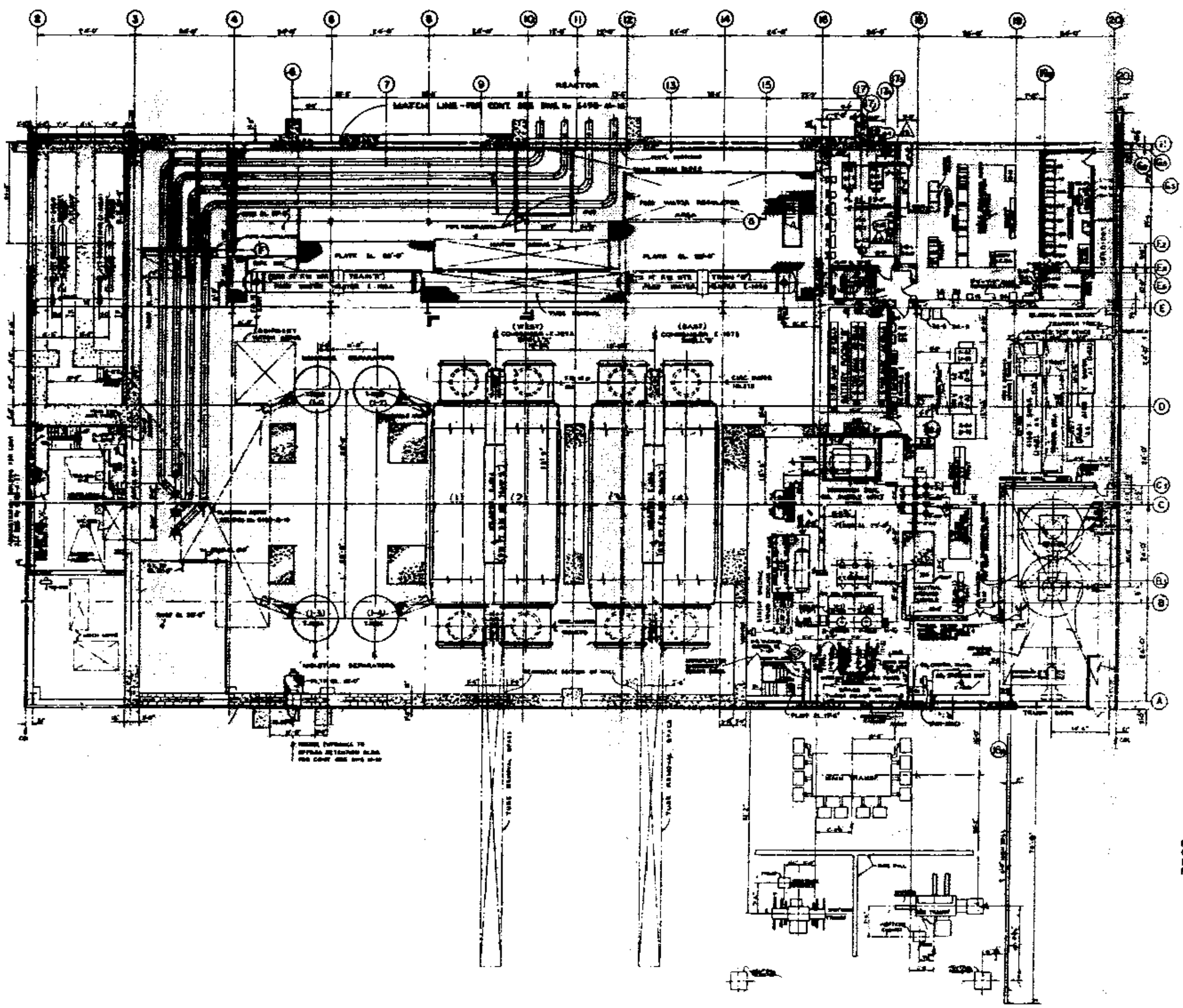


FIG. R.Q-1
ROUTE TRAVELED BY
CONTROL ROOM PERSONNEL
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

REV. 6 JULY 1986

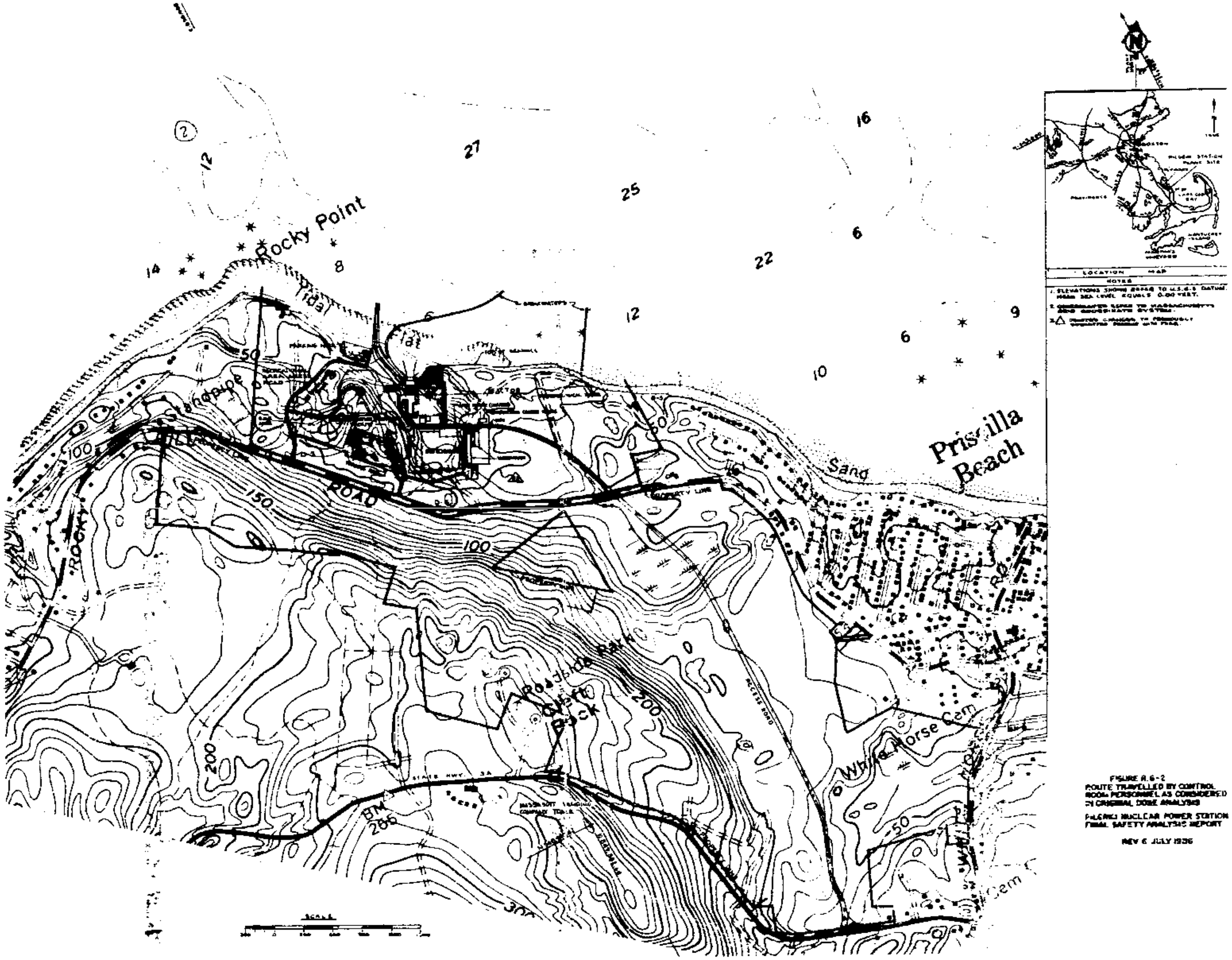


FIGURE A.6-2
ROUTE TRAVELLED BY CONTROL ROOM PERSONNEL AS CONSIDERED IN ORIGINAL DOSE ANALYSIS

PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

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APPENDIX S

LICENSE RENEWAL COMMITMENTS

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LICENSE RENEWAL COMMITMENTS

S.1 Supplement for Renewed Operating License

The Pilgrim Nuclear Power Station license renewal application (Reference S.4-1) and information in subsequent related correspondence provided sufficient basis for the NRC to make the findings required by 10 CFR 54.29 (Final Safety Evaluation Report) (Reference S.4-2). As required by 10 CFR 54.21(d), this UFSAR supplement contains a summary description of the programs and activities for managing the effects of aging (Section S.2) and a description of the evaluation of time-limited aging analyses for the period of extended operation (Section S.3). The period of extended operation is the 20 years after the expiration date of the original operating license.

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S.2 Aging Management Programs and Activities

The integrated plant assessment for license renewal identified aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the aging management programs and activities required during the period of extended operation. All aging management programs have been initiated prior to entering the period of extended operation.

PNPS quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The Entergy Quality Assurance Program applies to safety-related structures and components. Corrective actions and administrative (document) control for both safety-related and nonsafety-related structures and components are accomplished per the existing PNPS corrective action program and document control program and are applicable to all aging management programs and activities that will be required during the period of extended operation. The confirmation process is part of the corrective action program and includes reviews to assure that proposed actions are adequate, tracking and reporting of open corrective actions, and review of corrective action effectiveness. Any follow-up inspection required by the confirmation process is documented in accordance with the corrective action program.

The corrective action controls of the Entergy (10 CFR Part 50, Appendix B) Quality Assurance Program are applicable to all aging management programs and activities that will be required during the period of extended operation.

Operating experience is used at PNPS to enhance plant aging management programs. External nuclear industry operating experience, including operating experience related to the effects of aging, is screened, evaluated, and acted on to prevent or mitigate the consequences of similar events or conditions. External operating experience includes NRC generic communications (e.g., generic letters, bulletins, information notices, and regulatory information summaries) and other documents (e.g., 10

CFR 21 reports, licensee event reports, vendor bulletins, information for INPO and other industry groups). Internal operating experience includes information from event investigation reports, trending reports, lessons learned from in-house events, self-assessments, and the 10 CFR 50 Appendix B corrective action process. Operating experience is operating information pertinent to plant safety and reliability originating both within and outside the Entergy organization. Operating experience information describes events, issues, equipment failures, etc., including those resulting from the effects of aging, that represent opportunities to apply lessons learned to avoid negative consequences or to recreate positive experiences.

Entergy procedures provide direction for the evaluation of operating experience continuing through the period of extended operation. These procedures implement two programs that monitor, on an ongoing basis, industry and plant-specific operating experience that includes, but is not limited to, operating experience related to the effects of aging on in-scope structures and components. These programs are the Operating Experience Program and the Corrective Action Program. Procedures for these programs provide a method for evaluating and initiating action for operating experience information at all Entergy nuclear stations. The primary objective of assessing operating experience is to identify and transfer lessons learned into actions that enhance the safety and reliability of Entergy’s nuclear plants. Operating experience involving age-related degradation mechanisms or aspects of programs that manage the effects of aging is provided to the respective program point of contact. Operating experience involving age-related degradation mechanisms for which no program can be readily identified is reviewed as documented in a written evaluation by the aging management operating experience point of contact for the station. The evaluations completed under these two programs ensure that aging management programs remain effective in managing the effects of aging for which they are credited.

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S.2.1 Boraflex Monitoring Program

The Boraflex Monitoring Program assures that degradation of the Boraflex panels in the spent fuel racks does not compromise the criticality analysis in support of the design of the spent fuel storage racks. The program relies on (1) neutron attenuation testing, (2) determination of boron loss through correlation of silica levels in spent fuel pool water samples and periodic areal density measurements, and (3) analysis of criticality to assure that the required 5% subcriticality margin is maintained.

License renewal commitment 49 governs implementation of this program. This program is being tracked by administrative controls.

S.2.2 Buried Piping and Tanks Inspection Program

The Buried Piping and Tanks Inspection Program included in its scope the Firewater, CST inlet to HPCI/RCIC, SSW, SBO fuel and coolant, EDG fuel, and the SBGTS piping.

The Firewater piping is addressed through the adoption of NFPA25 flow testing. The CST SS piping has been addressed through the GW testing, and is subject to future surveillance. The SSW inlet piping was determined to be non-susceptible. The SSW outlet has been modified with a non-susceptible CIPP, and is subject to the NRC credited Service Water Integrity Program, GL 89-13, which include regular internal visual surveillance and ongoing flow rate testing. The two EDG fuel oil tanks have been NDE tested and both tanks are subject to future NDE testing. The SBO, EDG, and the SBGTS piping have been excavated and inspected and are subject to future inspections. The Cathodic Protection Systems are operable and subject to ongoing surveillance. Pilgrim has committed to the Corporate Buried Pipe and Tank Program for all underground piping.

These inspections will be conducted at least once every ten years during the PEO. If measured soil resistivity is <20,000 ohms or scores higher than 10 points using the American Water Works Association C10S, or if backfill is found to have damaged the coating, the length of SBGTS pipe inspected will be doubled during subsequent ten (10) year inspections.

The two buried carbon steel EDG fuel oil tanks were inspected prior to the PEO and will be reinspected on a ten (10) year interval following the PEO.

If trending within the corrective action program identifies susceptible locations, the areas with a history of corrosion problems are evaluated for the need for additional inspection, alternate coating, or replacement.

License renewal commitment 1 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.3 BWR CRD Return Line Nozzle Program

Under the BWR CRD Return Line Nozzle Program, PNPS has cut and capped the CRD return line nozzle to mitigate cracking and continues in-service inspection (ISI) examinations to monitor the effects of crack initiation and growth on the intended function of the control rod drive return line nozzle and cap. ISI examinations include ultrasonic inspection of the nozzle-to-vessel weld and ultrasonic inspection of the dissimilar metal weld overlay at the nozzle.

License renewal commitment 30 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.4 BWR Feedwater Nozzle Program

Under the BWR Feedwater Nozzle Program, PNPS has removed feed water blend radii flaws, removed feed water nozzle cladding, and installed a triple-sleeve-double-piston sparger to mitigate cracking. This program continues with enhanced in-service inspection (ISI) of the feed water nozzles in accordance with the requirements of ASME Section XI, Subsection IWB and the recommendation of General Electric (GE) NE-523-A71-0594 to monitor the effects of cracking on the intended function of the feed water nozzles.

This program is being tracked by administrative controls.

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S.2.5 BWR Penetrations Program

The BWR Penetrations Program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP) documents BWRVIP-27 and BWRVIP-49 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel penetrations and nozzles.

This program is being tracked by administrative controls.

S.2.6 BWR Stress Corrosion Cracking Program

The BWR Stress Corrosion Cracking Program includes (1) preventive measures to mitigate intergranular stress corrosion cracking (IGSCC), and (2) inspection and flaw evaluation to monitor IGSCC and its effects on reactor coolant pressure boundary components made of stainless steel or CASS.

PNPS has taken actions to prevent IGSCC and will continue to use materials resistant to IGSCC for component replacements and repairs following the recommendations delineated in NUREG-0313, Generic Letter 88-01, and the staff-approved BWRVIP-75 report. Inspection of piping identified in NRC Generic Letter 88-01 to detect and size cracks is performed in accordance with the staff positions on schedule, method, personnel qualification and sample expansion included in the generic letter and the staff-approved BWRVIP-75 report.

License renewal commitment 2 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.7 BWR Vessel ID Attachment Welds Program

The BWR Vessel ID Attachment Welds Program includes (1) inspection and flaw evaluation in accordance with the guidelines of staff-approved BWR Vessel and Internals Project (BWRVIP) BWRVIP-48, and (2) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity and safe operation of reactor vessel inside diameter (ID) attachment welds and support pads.

This program is being tracked by administrative controls.

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S.2.8 BWR Vessel Internals Program

The BWR Vessel Internals Program includes (a) inspection, flaw evaluation, and repair in conformance with the applicable, staff-approved BWR Vessel and Internals Project (BWRVIP) documents, and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-130 to ensure the long-term integrity of vessel internals components.

License renewal commitments 3, 33, 34, and 37 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.9 Containment Leak Rate Program

As described in 10 CFR 50, Appendix J, containment leak rate tests are required to assure that (a) leakage through primary reactor containment and systems and components penetrating primary containment shall not exceed allowable values specified in technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of containment, and systems and components penetrating primary containment. Corrective actions are taken if leakage rates exceed acceptance criteria.

This program is being tracked by administrative controls.

S.2.10 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program entails sampling to ensure that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of fuel systems. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic draining and cleaning of tanks and by verifying the quality of new oil before its introduction into the storage tanks.

License renewal commitments 4, 5, 6, and 38 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

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S.2.11 Environmental Qualification (EQ) of Electric Components Program

The PNPS EQ of Electric Components program manages the effects of thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are refurbished, replaced, or their qualification extended prior to reaching the aging limits established in the evaluations. Aging evaluations for EQ components are considered time-limited aging analyses (TLAAs) for license renewal.

This program is being tracked by administrative controls.

S.2.12 Fatigue Monitoring Program

In order not to exceed design limits on fatigue usage, the Fatigue Monitoring Program tracks the number of critical thermal and pressure transients for selected reactor coolant system components. The program ensures the validity of analyses that explicitly assumed a fixed number of thermal and pressure fatigue transients by assuring that the actual effective number of transients does not exceed the assumed limit.

The transient cycles tracked by this program are referenced in Section 4.2.6.

License renewal commitments 31 and 35 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

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S.2.13 Fire Protection Program

The Fire Protection Program includes a fire barrier inspection and a diesel-driven fire pump inspection. The fire barrier inspection requires periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The diesel-driven fire pump inspection requires that the pump be periodically tested and system components internally inspected to ensure that the fuel supply line can perform its intended function. The program also includes periodic inspection and testing of the Halon fire suppression system.

Corrective actions, confirmation process, and administrative controls in accordance with the requirements of 10 CFR 50 Appendix B are applied to the Fire Protection Program.

License renewal commitments 7 and 8 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.14 Fire Water System Program

The Fire Water System Program applies to water-based fire protection systems that consist of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, and aboveground and underground piping and components that are tested in accordance with applicable National Fire Protection Association (NFPA) codes and standards. Such testing assures functionality of systems. To determine if significant corrosion has occurred in water-based fire protection systems, periodic flushing, system performance testing, and inspections are conducted. Also, many of these systems are normally maintained at required operating pressure and monitored such that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated.

In addition, wall thickness evaluations of fire protection piping are periodically performed on system components using non-intrusive techniques (e.g., volumetric testing) to identify evidence of loss of material due to corrosion.

A sample of sprinkler heads will be inspected using the guidance of NFPA 25 (2002 Edition) Section 5.3.1.1.1, which states, "Where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing."

This sampling will be repeated every 10 years after initial field service testing.

License renewal commitments 9, 10, and 11 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.15 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion Program applies to safety-related and nonsafety-related carbon steel components in systems containing high-energy fluids carrying two-phase or single-phase high-energy fluid > 2% of plant operating time.

The program, based on EPRI recommendations for an effective flow-accelerated corrosion program, predicts, detects, and monitors FAC in plant piping and other pressure retaining components. This program includes (a) an evaluation to determine critical locations, (b) initial operational inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm predictions. The program specifies repair or replacement of components as necessary.

This program is being tracked by administrative controls.

S.2.16 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program inspects heat exchangers for degradation. If degradation is found, then an evaluation is performed to evaluate its effects on the heat exchanger's design functions including its ability to withstand a seismic event.

Representative tubes within the population of heat exchangers are eddy current tested at a frequency determined by internal and external operating experience and trended to ensure that effects of aging are identified prior to loss of intended function. Along with each eddy current test, visual inspections are performed on accessible heat exchanger heads, covers and tube sheets to monitor surface condition for indications of loss of material. The population of heat exchangers includes the RHR heat exchangers, RHR pump seal heat exchangers, core spray pump motor thrust bearing lube oil coolers, HPCI gland seal condenser, HPCI turbine lube oil cooler, RCIC lube oil cooler, recirculation pump motor generator set fluid coupling oil and bearing coolers, CRD pump oil coolers, recirculation pump motor lube oil coolers, clean up recirculation pump lube oil coolers and stuffing box cooler, fuel pool heat exchangers, CRD pump thrust bearing coolers, recirculation pump seal water coolers, clean up demineralizer

non-regeneration heat exchangers, and EDG lube oil coolers.

License renewal commitment 12 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.17 In-service Inspection - Containment In-service Inspection (CII) Program

The Containment In-service Inspection Program outlines the requirements for the inspection of Class MC pressure-retaining components (primary containment) and their integral attachments in accordance with the requirements of 10 CFR 50.55a(b)(2) and the 1998 Edition of ASME Section XI with 2000 Addenda, Inspection Program B.

The primary inspection method for the primary containment and its integral attachments is visual examination. Visual examinations are performed either directly or remotely with illumination and resolution suitable for the local environment to assess general conditions that may affect either the containment structural integrity or leak tightness of the pressure retaining component. The program includes augmented ultrasonic exams to measure wall thickness of the containment drywell structure.

License renewal commitment 41 specifies enhancement(s) to this program.

License renewal commitment 44 specified the performance of another set of UT measurements just above and adjacent to the sand cushion region prior to the period of extended operation and an additional set of measurements will be taken once within the first 10 years of the period of extended operation.

These programs are being tracked by administrative controls.

S.2.18 In-service Inspection – In-service Inspection (ISI) Program

The ISI Program is based on ASME Inspection Program B (Section XI, IWA-2432), which has 10-year inspection intervals. Every 10 years the program is updated to the latest ASME Section XI code edition and addendum approved in 10 CFR 50.55a.

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The program consists of periodic volumetric, surface, and visual examination of components and their supports for assessment, signs of degradation, flaw evaluation, and corrective actions.

This program is being tracked by administrative controls.

S.2.19 Instrument Air Quality Program

The Instrument Air Quality Program ensures that instrument air supplied to components is maintained free of water and significant contaminants, thereby preserving an environment that is not conducive to loss of material. Dewpoint, particulate contamination, and hydrocarbon concentration are periodically checked to verify the instrument air quality is maintained.

License renewal commitment 13 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.20 Metal-Enclosed Bus Inspection Program

Under the Metal-Enclosed Bus Inspection Program, internal portions of the non-segregated phase bus which connects the 4.16kV switchgear (A3 through A6) are inspected for cracks, corrosion, foreign debris, excessive dust buildup, and evidence of water intrusion. Bus insulation is inspected for signs of embrittlement, cracking, melting, swelling, or discoloration, which may indicate overheating or aging degradation. Internal bus supports have been and will be inspected for structural integrity and signs of cracks. Since bolted connections are covered with heat shrink tape or insulating boots per manufacturer's recommendations, a sample of accessible bolted connections is visually inspected for insulation material surface anomalies. Enclosure assemblies have been and will be visually inspected for evidence of loss of material and, where applicable, enclosure assembly elastomers are visually inspected and manually flexed to manage cracking and change in material properties.

These inspections are performed at least once every five years. License renewal commitment 14 governs implementation of this program.

This program is being tracked by administrative controls.

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S.2.21 Non-EQ Inaccessible Medium-Voltage Cable Program

The Non-EQ Inaccessible Medium Voltage Cable Program is based on, and consistent with NUREG-1801, Rev 2, section XI.E3. In scope, inaccessible medium-voltage and low-voltage (400V to 35KV) cables exposed to significant moisture are tested at least once every six years to provide an indication of the condition of the insulation. Significant moisture is defined as periodic exposures that last more than a few days. The specific test performed is a proven test for detecting deterioration of the insulation, such as power factor, partial discharge, polarization index, or other testing that is state-of-the-art at the time the test is performed. Evaluation of the test results are used to determine the need for increased test frequencies.

Inspections for water collection in cable manholes and conduit containing in scope medium and low voltage cables with a license renewal intended function (400V to 35KV) occur at least once every year. Additional condition-based inspections of these manholes are performed based on natural events for a coastal site. The results of the inspections are reviewed to determine if the inspection frequency, and/or testing frequency should be modified.

License renewal commitment 15 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.22 Non-EQ Instrumentation Circuits Test Review Program

Under the Non-EQ Instrumentation Circuits Test Review Program, calibration or surveillance results for non-EQ electrical cables in circuits with sensitive, high voltage, low-level signals; (i.e., neutron flux monitoring instrumentation); are reviewed. Most neutron flux monitoring system cables and connections are calibrated as part of the instrumentation loop calibration at the normal calibration frequency, which provides sufficient indication of the need for corrective actions based on acceptance criteria related to instrumentation loop performance. The review of calibration results is performed once every 10 years.

For neutron flux monitoring system cables that are disconnected during instrument calibrations, testing is performed at least once every 10 years using a proven method for detecting deterioration for the insulation system (such as insulation resistance tests or time domain reflectometry).

A review of the neutron monitoring system calibration and cable testing was completed before the period of extended operation and future tests will occur at least once every 10 years.

License renewal commitment 16 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.23 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program provides reasonable assurance that intended functions of insulated cables and connections exposed to adverse localized environments caused by heat, radiation and moisture can be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is significantly more severe than the specified service condition for the insulated cable or connection.

A site walkdown of accessible insulated cables and connections is visually inspected at least once every 10 years for cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking, or surface contamination.

License renewal commitment 17 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.24 Oil Analysis Program

The Oil Analysis Program maintains oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to loss of material, cracking, or fouling. Activities include sampling and analysis of lubricating oil for detrimental contaminants, water, and particulates.

Sampling frequencies are based on vendor recommendations, accessibility during plant operation, equipment importance to plant operation, and previous test results.

License renewal commitments 18, 19, and 40 specify enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.25 One-Time Inspection Program

A one-time inspection activity is used to verify the effectiveness of the water chemistry control programs by confirming that unacceptable cracking, loss of material, and fouling is not occurring on components within systems covered by water chemistry control programs [Sections S.2.36, S.2.37, and S.2.38].

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The elements of the One-Time Inspection Program include (a) determination of the sample size based on an assessment of materials of fabrication, environment, plausible aging effects, and operating experience, (b) identification of the inspection locations in the system or component based on the aging effect, (c) determination of the examination technique, including acceptance criteria that would be effective in managing the aging effect for which the component is examined, and (d) evaluation of the need for follow-up examinations to monitor the progression of any aging degradation.

One-time inspection activities included:

- bottom surface of the condensate storage tanks
- main stack foundation
- verify absence of fatigue cracking for miscellaneous items not covered by a fatigue TLAA
- internal surfaces of buried carbon steel pipe on the standby gas treatment system discharge to the stack,
- internal surfaces of compressed air and EDG system components containing untreated air,
- internal surfaces of stainless steel radioactive waste and sanitary soiled waste and vent system components containing untreated water,
- small bore piping in the reactor coolant system and associated systems that form the reactor coolant pressure boundary,
- reactor vessel flange leak-off line, and
- main steam flow restrictors

The results were used to confirm that loss of material, and cracking as applicable, are not occurring or are so insignificant that an aging management program is not warranted.

When evidence of an aging effect was revealed by a one-time inspection, the corrective action process was used.

License renewal commitments 20, 36, and 39 govern implementation of this program.

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S.2.26 Periodic Surveillance and Preventive Maintenance Program

The Periodic Surveillance and Preventive Maintenance Program include periodic inspections and tests that manage aging effects not managed by other aging management programs. The preventive maintenance and surveillance testing activities are generally implemented through repetitive tasks or routine monitoring of plant operations.

Temperatures are monitored during periodic emergency diesel generator (EDG), station blackout diesel, and security diesel surveillance tests to verify that associated heat exchangers are capable of removing the required amount of heat, thereby managing fouling of the heat exchanger tubes.

Periodic inspections using visual or other non-destructive examination techniques verify that the following components are capable of performing their intended function.

- reactor building crane, rails, and girders
- refueling platform carbon steel components
- main stack components
- standby liquid control system discharge accumulators
- carbon steel piping in the waterline region of the torus
- HPCI gland seal condenser blower and suction piping
- RCIC steam supply and exhaust piping downstream of the strainers and steam traps
- standby gas treatment system expansion joints, demister drain valves, and demister drain piping
- drain lines from each reactor building auxiliary bay passing into the water trough in the torus
- clean-up recirculation pump P-204B stuffing box cooler
- RBCCW copper alloy cooling coils
- EDG, station blackout diesel, and security diesel intake air, air start, and exhaust components

- EDG, station blackout diesel, and security diesel jacket water radiators
- security diesel oil cooler and aftercooler
- area coolers VAC-210A/B, VAC-202A/B, and VAC-204A/B/C/D
- VSF-103A/B, VAC-202A/B, VAC-204A/B/C/D, and EDG engine driven fan duct flexible connections
- condensate storage tanks
- circulating water, potable and sanitary water, radioactive waste, sanitary soiled waste and vent, plumbing and drains, and screen wash system components
- flex/expansion joints in the circulating water, HVAC/chilled water, and radioactive waste systems
- diesel fuel oil emergency transfer skid hoses, piping, pump casing, strainer, and valve bodies

License renewal commitment 21 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.27 Reactor Head Closure Studs Program

The Reactor Head Closure Studs Program includes in-service inspection (ISI) in conformance with the requirements of the ASME Code, Section XI, Subsection IWB, and preventive measures (e.g. rust inhibitors, stable lubricants, appropriate materials) to mitigate cracking and loss of material of reactor head closure studs, nuts, washers, and bushings.

This program is being tracked by administrative controls.

S.2.28 Reactor Vessel Surveillance Program

PNPS is a participant in the BWR vessel and internals project (BWRVIP) Integrated Surveillance Program (ISP) as incorporated into the plant Technical Specifications by License Amendment 209. The Reactor Vessel Surveillance Program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of PNPS, the actual shift in the reference temperature for nil-ductility

transition of the vessel material may be updated. In accordance with 10 CFR 50 Appendices G and H, PNPS reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10 CFR 50 Appendix H are met for the period of extended operation. The BWRVIP-116 report which was approved by the Staff will be implemented at PNPS with the conditions documented in Sections 3 and 4 of the Staff's final SE dated March 1, 2006, for the BWRVIP-116 report.

The Reactor Vessel Surveillance Program has been enhanced to proceduralize the data analysis, acceptance criteria, and corrective actions described in this program description.

If the PNPS standby capsule is removed from the reactor vessel without the intent to test it, the capsule will be stored in a manner which would permit its future use if necessary.

License renewal commitment 22 specifies enhancement(s) to this program.

S.2.29 Selective Leaching Program

The Selective Leaching Program ensures the integrity of components made of cast iron, aluminum, bronze, brass, and other alloys exposed to raw water, treated water, or groundwater that may lead to selective leaching. The program includes a one-time visual inspection and/or hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

License renewal commitment 23 governs implementation of this program.

This program is being tracked by administrative controls.

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S.2.30 Service Water Integrity Program

The Service Water Integrity Program relies on implementation of the recommendations of NRC GL 89-13 to ensure that the effects of aging on the salt service water (SSW) system are managed for the period of extended operation. The program includes component inspections for erosion, corrosion, and blockage and performance monitoring to verify the heat transfer capability of the safety-related heat exchangers cooled by SSW. Chemical treatment using biocides and chlorine and periodic cleaning and flushing of redundant or infrequently used loops are the methods used to control or prevent fouling within the heat exchangers and loss of material in SSW components.

License renewal commitment 24 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.31 Structures Monitoring - Masonry Wall Program

The objective of the Masonry Wall Program is to manage cracking so that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation.

The program includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4. Included components are the 10 CFR 50.48 required masonry walls, radiation shielding masonry walls, masonry walls with the potential to affect safety-related components, and the torus compartment water trough.

Masonry walls are visually examined at a frequency selected to ensure there is no loss of intended function between inspections.

This program is being tracked by administrative controls.

S.2.32 Structures Monitoring - Structures Monitoring Program

Structures monitoring is in accordance with 10 CFR 50.65 (Maintenance Rule) as addressed in Regulatory Guide 1.160 and NUMARC 93-01. Periodic inspections are used to monitor the condition of structures and structural components to ensure there is no loss of structure or structural component intended function.

License renewal commitments 25, 26, and 43 specify enhancement(s) to this program.

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License renewal commitment 45 specifies that if groundwater continues to collect on the Torus Room floor, obtain samples and test such water to determine its pH and verify the water is non-aggressive as defined in NUREG-1801 Section III.A1 item III.A.1-4 once prior to the period of extended operation and once every five years during the period of extended operation.

License renewal commitment 46 specifies inspection of the condition of a sample of torus hold-down bolts and associated grout and determine appropriate actions based on the findings prior to the period of extended operations.

This program is being tracked by administrative controls.

S.2.33 Structures Monitoring - Water Control Structures Monitoring Program

The Water Control Structures Monitoring Program includes visual inspections to manage loss of material and loss of form for water-control structures (breakwaters, jetties, and revetments). The water-control structures are of rubble mound construction with the outer layer protected by heavy capstone. Parameters monitored include settlement (vertical displacement) and rock displacement. These parameters are consistent with those described in RG 1.127.

License renewal commitment 27 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.34 System Walkdown Program

The System Walkdown Program entails inspections of external surfaces of components subject to aging management review. The program is also credited with managing loss of material from internal surfaces, for situations in which internal and external material and environment combinations are the same such that external surface condition is representative of internal surface condition.

Surfaces that are inaccessible during plant operations are inspected during refueling outages. Surfaces are inspected at frequencies to provide reasonable assurance that effect of aging will be managed such that applicable components will perform their intended function during the period of extended operation.

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System walkdown guidance includes visual inspection of high voltage insulators required for station blackout recovery.

License renewal commitment 28 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

S.2.35 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program

The purpose of the Thermal Aging and Neutron Irradiation Embrittlement of CASS Program is to assure that reduction of fracture toughness due to thermal aging and reduction of fracture toughness due to radiation embrittlement will not result in loss of intended function during the period of extended operation. This program evaluates CASS components in the reactor vessel internals and requires non-destructive examinations as appropriate.

License renewal commitment 29 governs implementation of this program.

This program is being tracked by administrative controls.

S.2.36 Water Chemistry Control - Auxiliary Systems Program

The purpose of the Water Chemistry Control - Auxiliary Systems Program is to manage loss of material for components exposed to treated water.

Program activities include sampling and analysis of the stator cooling water system to minimize component exposure to aggressive environments.

The One-Time Inspection Program confirmed the effectiveness of the program.

S.2.37 Water Chemistry Control - BWR Program

The objective of the Water Chemistry Control - BWR Program is to manage aging effects caused by corrosion and cracking mechanisms. The program relies on monitoring and control of water chemistry based on EPRI Report 1008192 (BWRVIP-130). BWRVIP-130 has three sets of guidelines: one for primary water, one for condensate and feed water, and one for control rod drive (CRD) mechanism cooling water. EPRI guidelines in BWRVIP-130 also include recommendations for controlling water chemistry in the torus, condensate storage tank, demineralized water storage tanks, and spent fuel pool.

The Water Chemistry Control - BWR Program optimizes the primary water chemistry to minimize the potential for loss of material and cracking. This is accomplished by limiting the levels of contaminants in the RCS that could cause loss of material and cracking. Additionally, PNPS has instituted hydrogen water chemistry (HWC) to limit the potential for intergranular SCC (IGSCC) through the reduction of dissolved oxygen in the treated water.

The One-Time Inspection Program confirmed the effectiveness of the program.

S.2.38 Water Chemistry Control - Closed Cooling Water Program

The Water Chemistry Control - Closed Cooling Water Program includes preventive measures that manage loss of material, cracking, and fouling for components in closed cooling water systems (reactor building closed cooling water, turbine building closed cooling water, emergency diesel generator cooling water, station blackout diesel cooling water, security diesel generator cooling water, and plant heating). These chemistry activities provide for monitoring and controlling closed cooling water chemistry using PNPS procedures and processes based on EPRI guidance for closed cooling water chemistry.

The One-Time Inspection Program confirmed the effectiveness of the program.

S.2.39 Bolting Integrity Program

The Bolting Integrity Program relies on recommendations for a comprehensive bolting integrity program, as delineated in NUREG-1339, and industry recommendations, as delineated in the Electric Power Research Institute (EPRI) NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting. The program relies on industry recommendations for comprehensive bolting maintenance, as delineated in EPRI TR-104213 for pressure retaining bolting and structural bolting.

License renewal commitment 32 specifies enhancement(s) to this program.

This program is being tracked by administrative controls.

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S.2.40 Bolted Cable Connections Program

The Bolted Cable Connections Program focused on the metallic parts of the cable connections. This sampling program provides a one-time inspection to verify that the loosening of bolted connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation was not an aging issue that requires a periodic aging management program. A representative sample of the electrical cable connection population subject to aging management review were inspected or tested. Connections covered under the EQ program, or connections inspected or tested as part of a preventive maintenance program were excluded from aging management review. The factors considered for sample selection were application (medium and low voltage), circuit loading (high loading), and location (high temperature, high humidity, vibration, etc.) The technical basis for the sample selection was documented.

This program was completed prior to the period of extended operation.

License renewal commitment 42 governed implementation of the program.

S.2.41 Neutron Absorber Monitoring Program

The Neutron Absorber Monitoring Program is a new program that manages loss of material and reduction of neutron absorption capacity of Boral and Metamic neutron absorption panels in the spent fuel racks. The program will rely on periodic inspection, testing, monitoring of coupons, and analysis of the criticality design to assure that the required five percent subcriticality margin is maintained during the period of extended operation.

The program was initiated prior to the period of extended operation. One test on each material was performed within the five years preceding the period of extended operation, with additional testing performed on each material at least once every ten years during the period of extended operation.

This program is being tracked by administrative controls.

S.2.42 Protective Containment Coatings

The Protective Coating Monitoring and Maintenance Program manage the effects of aging on Service Level 1 coatings inside containment by means of periodic visual inspections. The program also includes direction to select and review the suitability of

the coatings applied to surfaces inside containment (e.g., steel containment shell, structural steel, supports, penetrations, and concrete walls and floors). Inspection of coatings inside containment is performed of accessible areas in accordance with the IWE requirements of ASME Section XI during every other refueling outage (once per ASME Section XI IWE period) which is a maximum of four years.

This program is being tracked by administrative controls.

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S.3 Evaluation of Time-Limited Aging Analyses

In accordance with 10 CFR 54.21(c), an application for a renewed license requires an evaluation of time-limited aging analyses (TLAA) for the period of extended operation. The following TLAA have been identified and evaluated to meet this requirement.

S.3.1 Reactor Vessel Neutron Embrittlement

The reactor vessel neutron embrittlement TLAA will either remain valid for the period of extended operation (P-T limits) in accordance with 10 CFR 54.21(c)(1)(i) or have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii). Fifty-four EFPY would be the effective full power years at the end of the period of extended operation assuming an average capacity factor of 90% for 60 years.

S.3.1.1 Reactor Vessel Fluence

Calculated fluence is based on a time-limited assumption defined by the operating term. As such, fluence is the time-limited assumption for the time-limited aging analyses that evaluate reactor vessel embrittlement.

Fluence values were calculated using the RAMA fluence calculation method. The RAMA fluence method was developed for the Electric Power Research Institute, Inc. and the Boiling Water Reactor Vessel and Internals Project (BWRVIP) for the purpose of calculating neutron fluence in boiling water reactor components. This method has been approved by the NRC (Reference S.4-9) for application in accordance with Regulatory Guide 1.190 provided the fluence calculations for the reactor are appropriately benchmarked.

The benchmarking validation of the RAMA fluence calculation is ongoing for the PNPS reactor vessel. The RAMA calculated fluence is approximately 56% of the benchmark fluence calculated from the available surveillance capsule dosimetry. Uncertainties between the calculated and measured results from the dosimetry are still being examined to determine a possible cause for the discrepancy. An action plan to improve benchmarking data to support approval of new P-T curves will be developed and submitted for NRC review.

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An alternative analysis to determine the limiting fluence value has been performed (Reference S.4-12). This analysis assumes increasing fluence levels until an ASME Code or regulatory limit is reached based on the projected changes in material properties. Changes in the vessel (Ferritic) steel material properties are measured by an increase in adjusted reference temperature or a decrease in Charpy upper shelf energy. The effects of increasing fluence on the austenitic stainless steel core shroud and internals was also considered. By assuming increasing fluence levels, the analysis identifies the maximum fluence that can be experienced while meeting the Code and regulatory criteria.

The analysis determined that the limiting fluence value is set by the maximum mean RT_{NDT} value for the vessel axial welds of 114°F to remain below a calculated reactor vessel failure frequency of $5x10^{-6}$ per reactor-year. The corresponding maximum allowable ID fluence for the axial welds was determined to be 3.37E+18 n/cm^2. This fluence level is the limiting fluence value identified.

Entergy submitted to the NRC calculations consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

License renewal commitment 48 directed PNPS to submit fluence calculations consistent with RG 1.190 by June 8, 2010.

S.3.1.2 Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires that reactor vessel bolt up, hydro test, pressure tests, normal operation, and anticipated operational occurrences are accomplished within established pressure-temperature (P-T) limits. These limits are established by calculations that utilize the materials and fluence data obtained through the Reactor Vessel Surveillance Program.

Pilgrim received License Amendment 227 dated March 29, 2007 that extended the existing P-T limit curves for Pilgrim through Cycle 18.

The P-T limit curves will continue to be updated, as required by Appendix G of 10 CFR Part 50 or as operational needs dictate. This updating will assure that the operational limits remain valid through the period of extended operation. Maintaining the P-T limit curves in accordance with Appendix G of 10 CFR 50 assures that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation consistent with 10 CFR 54.21(c)(1)(iii).

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S.3.1.3 Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy … of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb…." The initial (unirradiated) values of upper-shelf energy (CvUSE) for PNPS beltline welds were provided to the NRC in correspondence responding to Generic Letter 92-01.

Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, provides two methods for determining Charpy upper-shelf energy (CvUSE). Position 1 applies for material that does not have surveillance data and Position 2 applies for material with surveillance data. Position 2 requires a minimum of two sets of credible material surveillance data. Since PNPS has data from only one material surveillance capsule, Position 2 does not apply. For Position 1, the percent drop in CvUSE for a stated copper content and neutron fluence is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial CvUSE to obtain the adjusted CvUSE.

The predictions for percent drop in CVUSE at 54 EFPY must be based on chemistry data, the maximum 1/4T fluence values, and unirradiated CvUSE data submitted to the NRC in the PNPS response to GL 92-01. The predicted CvUSE values for 54 EFPY will utilize Regulatory Guide 1.99 Position 1. The predictions will use Regulatory Guide 1.99, Position 1, Figure 2; specifically, the formula for the lines will be used to calculate the percent drop in CvUSE.

PNPS will use chemistry data from previous licensing submittals, the PNPS response to GL 92-01, and the 1/4T fluence values to be determined to perform linear interpolation on the CvUSE percent drop values in RG 1.99, Revision 2, Figure 2.

The license renewal SER for BWRVIP-74, Action Item #10, states that each license renewal applicant shall demonstrate that the percent reduction in Charpy USE for their beltline materials is less than that specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds given in BWRVIP-74. This action item is not applicable to PNPS if the PNPS projected CvUSE remains above the 50 ft-lb limit, even for the period of extended operation.

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An analysis determined that the limiting fluence value is set by the maximum mean RT_{NDT} value for the vessel axial welds of 114°F to remain below a calculated reactor vessel failure frequency of $5x10^{-6}$ per reactor-year. The corresponding maximum allowable ID fluence for the axial welds was determined to be 3.37E+18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel Charpy upper shelf energy TLAS. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy submitted to the NRC calculations consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

S.3.1.4 Adjusted Reference Temperature

Irradiation by high-energy neutrons raises the value of RT_{NDT} for the reactor vessel. RT_{NDT} is the reference temperature for nil-ductility transition as defined in Section NB-2320 of the ASME Code. The initial RT_{NDT} is determined through testing of unirradiated material specimens. The shift in reference temperature, ΔRT_{NDT}, is the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. The adjusted reference temperature (ART) is defined as initial RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in RG 1.99, Revision 2. The P-T curves are developed from the ART value for the vessel materials. RG 1.99 Revision 2 defines the calculation methods for RT_{NDT} and ART.

The PNPS reactor vessel was evaluated for an assumed exposure of less than 10^{19} nvt of neutrons with energies exceeding 1 MeV. After approximately 4.17 EFPY, the first surveillance capsule was withdrawn from the vessel and tested. The capsule test report concludes that the shift in RT_{NDT} and upper-shelf energy over 32 EFPY will be within 10 CFR 50 guidelines.

PNPS will project values for ΔRT_{NDT} and ART at 54 EFPY using the methodology of RG 1.99. These values will be calculated using the chemistry data, margin values, initial RT_{NDT} values, and chemistry factors (CFs) contained in the PNPS response to GL 92-01. Initial RT_{NDT} values are from report SIR-00-082, which was submitted in 2001 as part of the PNPS P-T limit change request. The 1/4T fluence values discussed in Section 4.2.1 will be used.

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New fluence factors (FFs) will be calculated using the expression in RG 1.99, Revision 2, Equation 2, where the fluence factor is given by

$$FF = f^{(0.28-0.10*\log f)}.$$

In this equation, f is the 1/4T fluence value. The new ΔRT_{NDT} values will be calculated by multiplying the CF and the FF for each plate and weld. Calculated margins and the initial RT_{NDT} will then be added to the calculated ΔRT_{NDT} in order to arrive at the new value of ART.

An analysis determined that the limiting fluence value is set by the maximum mean RT_{NDT} value for the vessel axial welds of 114°F to remain below a calculated reactor vessel failure frequency of $5x10^{-6}$ per reactor-year. The corresponding maximum allowable ID fluence for the axial welds was determined to be 3.37E+18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel adjusted reference temperature TLAS. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy submitted to the NRC consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

S.3.1.5 Reactor Vessel Circumferential Weld Inspection Relief

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on an analysis indicating acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

PNPS received NRC approval for this relief for the remainder of the original 40-year license term. The basis for this relief request is an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended operating period require additional analysis to extend this relief request.

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The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the reactor pressure vessel (RPV) shell weld failure probabilities. Three key inputs to the PFM analysis are (1) the estimated end-of-life mean neutron fluence, (2) mean chemistry values based on vessel types, and (3) the assumption of potential for beyond-design-basis events.

PNPS will compare the reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. The data will be from the NRC SER for PNPS Relief Request 28, and from the data in Table 2.6.4 of the NRC SER for BWRVIP-05. (For comparison, the EOL mean RT_{NDT} will be calculated without margin and hence will be lower than the Section 4.2.2 RT_{NDT} value.)

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when PNPS requested approval of the BWRVIP-05 technical alternative for the current license term.

An analysis determined that the limiting fluence value is set by the maximum mean RT_{NDT} value for the vessel axial welds of 114°F to remain below a calculated reactor vessel failure frequency of $5x10^{-6}$ per reactor-year. The corresponding maximum allowable ID fluence for the axial welds was determined to be 3.37E+18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel circumferential weld failure probability TLAS. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy submitted to the NRC calculations consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

S.3.1.6 Reactor Vessel Axial Weld Failure Probability

The BWRVIP recommendations for inspection of reactor vessel shell welds (BWRVIP-05) are based on generic analyses supporting an NRC SER conclusion that the generic-plant axial weld failure rate is no more than $5x10^{-6}$ per reactor year. BWRVIP-05 showed that this axial weld failure rate is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described above.

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PNPS received relief from the circumferential weld inspections for the remainder of the original 40-year operating term. The basis for this relief request was a plant-specific analysis that showed the limiting conditional failure probability for the PNPS circumferential welds at the end of the original operating term was less than the values calculated in the BWRVIP-05 SER. The BWRVIP-05 SER concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than $5x10^{-6}$ per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected. The PNPS relief request requires additional relief request if less than 90% coverage is achieved.

Applicant Action Item 12 from the NRC SER for BWRVIP-74 specified that applicants should monitor axial beltline weld embrittlement. One acceptable method was to determine that the mean RT_{NDT} of the limiting axial beltline weld at the end of the period of extended operation is less than the values specified in Table 1 of the FSER for BWRVIP-74. The limiting mean RT_{NDT} value of 114°F for the axial welds was determined to be equivalent to a failure frequency of less than $5x10^{-6}$ per reactor-year.

An analysis determined that the ID fluence value that yields a mean RT_{NDT} value for the vessel axial welds of 114°F is 3.37E+18 n/cm^2. This fluence is the limiting fluence value identified.

If fluence remains below this limiting value during the period of extended operation, the fluence will result in acceptable results for the reactor vessel axial weld failure probability TLAS. To confirm that this TLAA will be valid to the end of the period of extended operation, Entergy submitted to the NRC calculations consistent with Regulatory Guide 1.190 that demonstrated limiting fluence values will not be reached during the period of extended operation.

S.3.2 Metal Fatigue

S.3.2.1 Class 1 Metal Fatigue

Class 1 components evaluated for fatigue and flaw growth include the reactor pressure vessel (RPV) and appurtenances, certain reactor vessel internals, the reactor recirculation system (RRS), and the reactor coolant system (RCS) pressure boundary. The PNPS Class 1 systems include components within the ASME Section XI, IWB inspection boundary.

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The design of the reactor vessel internals is in accordance with the intent of ASME Section III. A review of design basis documents reveals that the only reactor vessel internals components for which there is a fatigue evaluation are the core shroud tie rods (stabilizer), the result of a repair to structurally replace circumferential shroud welds.

The PNPS fatigue monitoring program will assure that the allowed number of transient cycles is not exceeded. The program requires corrective action if transient cycle limits are approached. Consequently, the TLAA (fatigue analyses) based on those transients will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

License renewal commitment 35 addresses metal fatigue for reactor vessel components, including the feedwater nozzles.

S.3.2.2 Non-Class 1 Metal Fatigue

For non-Class 1 components identified as subject to cracking due to fatigue, a review of system operating characteristics was conducted to determine the approximate frequency of any significant thermal cycling. If the number of equivalent full temperature cycles is below the limit used for the original design (usually 7000 cycles), the component is suitable for extended operation. If the number of equivalent full temperature cycles exceeds the limit, evaluation of the individual stress calculations require evaluation. No components were identified with projected cycles exceeding 7000. Therefore, the TLAA for non-Class 1 piping and components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c) (i).

S.3.2.3 Environmental Effects on Fatigue

The effects of reactor water environment on fatigue were evaluated for license renewal. Projected cumulative usage factors (CUFs) were calculated for the limiting locations identified in NUREG/CR-6260. Several locations may exceed a CUF of 1.0 with consideration of environmental effects during the period of extended operation. For these locations, at least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the PNPS vintage, PNPS will refine the current fatigue analyses to include the effects of reactor water

environment and verify that the cumulative usage factors (CUFs) are less than 1. This includes applying the appropriate Fen factors to valid CUFs determined in accordance with one of the following:

1. For locations, including NUREG/CR-6260 locations, with existing fatigue analysis valid for the period of extended operation, use the existing CUF to determine the environmentally adjusted CUF.

2. More limiting PNPS-specific locations with a valid CUF may be added in addition to the NUREG/CR-6260 locations.

3. Representative CUF values from other plants, adjusted to or enveloping the PNPS plant specific external loads may be used if demonstrated applicable to PNPS.

4. An analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case) may be performed to determine a valid CUF.

During the period of extended operation, PNPS may also use one of the following options for fatigue management if ongoing monitoring indicates a potential for a condition outside the analysis bounds noted above.

1. Update and/or refine the affected analyses described above.

2. Implement an inspection program that has been reviewed and approved by the NRC (e.g., periodic nondestructive examination of the affected locations at inspection intervals to be determined by a method acceptable to the NRC).

3. Repair or replace the affected locations before exceeding a CUF of 1.0.

License renewal commitment 31 addresses environmental assisted fatigue for the locations identified in NUREG/CR-6260 for BWRs of the PNPS vintage.

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S.3.3 Environmental Qualification of Electrical Components

The PNPS EQ Program implements the requirements of 10 CFR 50.49 (as further defined by the Division of Operating Reactors Guidelines, NUREG-0588, and Reg. Guide 1.89). The program requires action before individual components exceed their qualified life. In accordance with 10 CFR 54.21(c)(1)(iii), implementation of the EQ Program provides reasonable assurance that the effects of aging on components associated with EQ TLAAs will be adequately managed such that the intended functions can be maintained for the period of extended operation.

S.3.4 Fatigue of Primary Containment, Attached Piping, and Components

In conjunction with the Mark I Containment Long-Term Program, the torus and attached piping systems were analyzed for fatigue due to mechanical loadings as well as thermal and anchor motion. This analysis was based on assumptions of the number of SRV actuations, operating basis earthquakes, and accident conditions during the life of the plant.

The fatigue usage calculated for PNPS is zero. However, the analysis considered all BWR plants which utilize the Mark I containment design. The analysis concluded that for all plants and piping systems considered the fatigue usage factor for an assumed 40-year plant life was less than 0.5. Extending plant life by an additional 20 years would produce a usage factor below 0.75. Since this is less than 1.0, the fatigue criteria are satisfied. This TLAA has been projected through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

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