The Light company Houston Lighting & Power P.O. Box 1700 Houston, Texas 77001 (713) 228-9211

March 11, 1986
ST-HL-AE-1621
File No.: G9.17

Mr. Vincent S. Noonan, Project Director
PWR Project Directorate #5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Auxiliary Feedwater Requirements

Reference: Letter ST-HL-AE-1609 dated 2/20/86; J. H. Goldberg to R. D. Martin

Dear Mr. Noonan:

Attached for your information and use is a copy of revised sections of the STP FSAR as transmitted to the NRC staff in Washington on March 6, 1986. These revised sections reflect the commitments as outlined in the referenced letter.

If you should have any questions on this matter, please contact Mr. M. E. Powell at (713) 993-1328.

Very truly yours,

M. R. Wisenburg

M. R. Wisenburg
Manager, Nuclear Licensing

LRS/yd

Attachments

B001 Limited Dist
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cc:

Hugh L. Thompson, Jr., Director
Division of PWR Licensing - A
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Robert D. Martin
Regional Administrator, Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

N. Prasad Kadambi, Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20814

Claude E. Johnson
Senior Resident Inspector/STP
c/o U.S. Nuclear Regulatory
Commission
P.O. Box 910
Bay City, TX 77414

M.D. Schwarz, Jr., Esquire
Baker & Botts
One Shell Plaza
Houston, TX 77002

J.R. Newman, Esquire
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Director, Office of Inspection
and Enforcement
U.S. Nuclear Regulatory Commission
Washington, DC 20555

T.V. Shockley/R.L. Range
Central Power & Light Company
P.O. Box 2121
Corpus Christi, TX 78403

H.L. Peterson/G. Pokorny
City of Austin
P.O. Box 1088
Austin, TX 78767

J.B. Poston/A. vonRosenberg
City Public Service Board
P.O. Box 1771
San Antonio, TX 78296

Brian E. Berwick, Esquire
Assistant Attorney General for
the State of Texas
P.O. Box 12548, Capitol Station
Austin, TX 78711

Lanny A. Sinkin
Christic Institute
1324 North Capitol Street
Washington, D.C. 20002

Oreste R. Pirfo, Esquire
Hearing Attorney
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Charles Bechhoefer, Esquire
Chairman, Atomic Safety &
Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. James C. Lamb, III
313 Woodhaven Road
Chapel Hill, NC 27514

Judge Frederick J. Shon
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Ray Goldstein, Esquire
1001 Vaughn Building
807 Brazos
Austin, TX 78701

Citizens for Equitable Utilities, Inc.
c/o Ms. Peggy Buchorn
Route 1, Box 1684
Brazoria, TX 77422

Docketing & Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, DC 20555
(3 Copies)

Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
1717 H Street
Washington, DC 20555

Revised 12/2/85

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TABLE 4.1-1

REACTOR DESIGN COMPARISON TABLE

THERMAL AND HYDRAULIC DESIGN PARAMETERS	W. B. McGuire UNITS 1 & 2	South Texas Project UNITS 1 & 2	
1. Reactor Core Heat Output, MWt	3,411	3,800	
2. Reactor Core Heat Output, 10^6 BTU/hr	11,641	12,969	
3. Heated Generated in Fuel, %	97.4	97.4	
4. System Pressure, Nominal, psia	2,250	2,250	
5. System Pressure, Minimum Steady State, psia	2,220	2,220	
6. Minimum Departure from Nucleate Boiling Ratio for Design Transients	>1.30	>1.30	18
7. DNB Correlation	"R" (W-3 with Modified Spacer Factor)	"R" (W-3 with Modified Spacer Factor)	
PEAKING FACTORS TO PREVENT DNB			
7.1 Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$	1.55	1.52	
7.2 Axial Power Shape (Chopped Cosine-Peak to Average)			
a) at Normal Full Power Operation	1.55	1.55	
b) at Overpower Conditions	1.55	1.61	
COOLANT FLOW			
8. Total Thermal Flow Rate, 10^6 lb_m/hr	140.3	~~139.8~~ 141.3	44
9. Effective Flow Rate for Heat Transfer, 10^6 lb_m/hr	134.0	~~133.2~~ 135.0	18
10. Effective Flow Area for Heat Transfer, ft^2	51.1	51.1	
11. Average Velocity Along Fuel Rods, ft/sec	16.7	16.7	18
12. Average Mass Velocity, 10^6 lb_m/hr-ft^2	2.62	~~2.61~~ 2.64	

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TABLE 4.1-1 (Continued)

REACTOR DESIGN COMPARISON TABLE

THERMAL AND HYDRAULIC DESIGN PARAMETERS	W. B. McGuire UNITS 1 & 2	South Texas Project UNITS 1 & 2	
COOLANT TEMPERATURE, °F			
13. Nominal Inlet	558.1	560.~~0~~ 4	
14. Average Rise in Vessel	60.2	~~66.1~~ 65.2	18
15. Average Rise in Core	62.7	~~68.8~~ 67.9	
16. Average in Core (Based on average enthalpy)	592.1	~~596.6~~ 596.5	44
17. Average in Vessel	588.2	593.0	18
HEAT TRANSFER			
18. Active Heat Transfer, Surface Area, ft²	59,700	69,700	
19. Average Heat Flux, Btu/hr-ft²	189,800	181,200	
20. Maximum Heat Flux for Normal Operation, Btu/hr-ft²	440,300	453,100	
21. Average Linear Power, kW/ft	5.44	5.20	
22. Peak Linear Power for Normal Operation, kW/ft	12.6	13.0	
23. Peak Linear Power Resulting from Overpower Transients/ Operator Errors (assuming a maximum overpower of 118%), kW/ft	18.0[a]	18.0[a]	
24. Heat Flux Hot Channel Factor, F_Q	2.32[b]	2.50[b]	
25. Peak Fuel Central Temperature at Peak Linear Power for Prevention of Centerline Melt, °F	4,700	4,700	

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	Sensor Drift (S)	Trip Setpoint	Allowable Value
1. Manual Reactor Trip	NA	NA	NA	NA	NA
2. Power Range, Neutron Flux,	7.5	4.56	0	≤109% of RTP**	≤111.1% of RTP**
a. High Setpoint					
b. Low Setpoint	8.3	4.56	0	≤25% of RTP**	≤27.3% of RTP**
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP** with a time constant ≥ 2 seconds	≤6.3% of RTP** with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP** with a time constant ≥ 2 seconds	≤6.3% of RTP** with a time constant ≥ 2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP**	≤30.9 of RTP**
6. Source Range, Neutron Flux	17.0	10.0	0	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7. Overtemperature ΔT	6.8	4.44	2.3+0.7	See note 1	See note 2
8. Overpower ΔT	5.5	1.4	0.2	See note 3	See note 4
9. Pressurizer Pressure - Low	3.1	0.71	1.5	≥1870 psig	≥1858.3 psig
10. Pressurizer Pressure - High	3.1	0.71	1.5	≤2380 psig	≤2391.7 psig
11. Pressurizer Water Level - High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span

* Loop DESIGN FLOW = 95.700 gpm Δ ~~94,100~~ 95,400

** RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	Sensor Drift (S)	Trip Setpoint	Allowable Value
12. Reactor Coolant Flow-Low	2.5	2.1	0.6 flow*	≥ 90% of loop design flow*	≥ 89.6 of loop design flow*
13. Steam Generator Water Level - Low-Low	15.0	12.18	1.5	≥ 33% of narrow range instrument span	≥ 31.3% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pump	10.6	0.3	0	≥ 10,300 volts	≥ 9815 volts
15. Underfrequency - Reactor Coolant Pumps	3.4	0.01	0	≥ 57.2 Hz	≥ 57.1 Hz
16. Turbine Trip					
A. Low Emergency Trip Fluid Pressure	Later	Later	Later	Later	Later
B. Turbine Stop Valve Closure	Later	Later	Later	Later	Later
17. Safety Injection Input from ESFAS	NA	NA	NA	NA	NA

95,400
~~94,100~~

* Loop DESIGN FLOW = 95,700 gpm
** RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	Sensor Error (S)	Trip Setpoint	Allowable Value
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	NA	NA	NA	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	NA	NA	NA	$\leq$ 10 % of RTP**	$\leq$ 12.2 % of RTP**
2) P-13 input	NA	NA	NA	$\leq$ 10 % RTP** Turbine Impulse Pressure Equivalent	$\leq$ 12.2 % RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8	NA	NA	NA	$\leq$ 48 % of RTP**	$\leq$ 50.2 % of RTP**
d. Power Range Neutron Flux, P-9	NA	NA	NA	$\leq$ 50 % of RTP**	$\leq$ 52.2 % or RTP**
e. Power Range Neutron Flux, P-10	NA	NA	NA	$\geq$ 10 % of RTP**	$\geq$ 7.8 % of RTP**
f. Turbine Impulse Chamber Pressure, P-13	NA	NA	NA	$\leq$ 10 % RTP** Turbine Impulse Pressure Equivalent	$\leq$ 12.2 % RTP** Turbine Impulse Pressure Equivalent

95,400
~~94,100~~

* Loop DESIGN FLOW = 95,700 gpm
** RTP = RATED THERMAL POWER

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

Functional Unit	Total Allowance (TA)	Z	Sensor Error (S)	Trip Setpoint	Allowable Value
19. Reactor Trip Breakers	NA	NA	NA	NA	NA
20. Automatic Trip and Interlock Logic	NA	NA	NA	NA	NA

95,400
~~94,400~~

* Loop DESIGN FLOW = 95,700 gpm
** RTP = RATED THERMAL POWER

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HEAT FLUX HOT CHANNEL FACTOR, RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a through d above are maintained. The combination of the RCS flow requirement (389600 gpm) and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

The $F^N_{\Delta H}$ requirement of 1.52 includes a 2% uncertainty in design and a 4% uncertainty on the measured value of $F^N_{\Delta H}$. Therefore, the measured value of $F^N_{\Delta H}$ should be increased by 4% before being compared with the required value of 1.52.

2.1%

The flow requirement 389600 gpm already includes a measurement uncertainty of ~~3.5%~~. Therefore no adjustment of the measured flow-value is necessary before comparing against the flow requirement.

Fuel rod bowing reduces the value of the DNB ratio, ~~credit~~ Credit ~~is available~~ in the generic margin is available to offset this reduction ~~the generic margin~~. The South Texas Project generic margins, totaling 3.3% DNBR, completely offset any rod bow penalties. This margin includes the following:

1. Design limit DNBR of 1.30 vs 1.28.
2. Grid Spacing Ks of 0.059 vs 0.066.
3. Thermal diffusion coefficient of ~~0.0059~~ 0.059 vs 0.061. (for use in modified spacer factor)

The applicable values of rod bow penalties are explained in FSAR Section ~~(Later)~~ 4.4.2.2.5.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the incore detector flux mapping system, and a 3% allowance is appropriate for manufacturing tolerance.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown in ~~Figure 3.2-3~~ Specification 3.2.3.

Question 410.18N

Provide a response to the staff's March 10, 1980 letter to near-term operating license applicants concerning your AFW system design (TMI-2 Task Action Plan, NUREG-0737, Item II.E.1.1). This response should include the following:

(a) A review of the AFW system design against Standard Review Plan Section 10.4.9, and Branch Technical Position ASB 10-1.

(b) A review of the AFW system design, Technical Specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter.

(c) The design basis for the AFW flow requirements and verification that the AFW system will meet these requirements (refer to Enclosure 2 of the March 10, 1980 letter).

Response

(a) Tables Q410.18N-1 and Q410.18N-2 summarize the STP conformance to SRP 10.4.9 and BTP ASB 10-1.

(b) The draft STP Technical Specifications were submitted on June 17, 1985 (reference letter ST-HL-AE-1271 to Mr. Hugh L. Thompson from J. H. Goldberg). ~~A review against the~~ Technical Specifications ~~is necessary to complete the response. It is anticipated the response will be provided by the fourth quarter of the year.~~

(c) ~~A response will be provided in the fourth quarter of 1985.~~
The response is provided in FSAR Section 7A, item II.E.1.1

for the AFWS have been prepared and were ~~will be~~ provided by ~~the end of the fourth quarter of 1985.~~
letter ST-HL-AE-1548 dated January 15, 1986.

Table 410.18N-1

Standard Review Plan, Section 10.4.9

Item	Acceptance Criteria	Related to	STP Position	Reference FSAR Section
1.	II.2	GDC 2	Conforms	10.4.9.2 & 10.4.9.3
2.	II.2	GDC 4	Conforms	3.5, Table 3.5-1, 3.6 & 10.4.9.2
3.	II.3	GDC 5	Conforms	10.4.9.2(1)
4.	II.4	GDC 19	Conforms	7.4.1, 7.4.1.1 & 10.4.9
5.	II.5 (a)	GDC 34 & 44	Conforms	10.4.9.1
	II.5 (b)	GDC 34 & 44	Conforms	10.4.9.1, 10.4.9.2, 10.4.9.3 & Table 10.4-3
	II.5 (c)	GDC 34 & 44	Conforms	10.4.9.2 (paragraph 7 & 11), 6.2.4, 10.4.9.3 and Appendix 10A (later).
6.	II.6	GDC 45	Conforms	6.6
7.	II.7	GDC 46	Conforms	10.4.9.4, 14.2 & STP Tech Specs (later)

Note:

(1) Each unit has an entirely independent Auxiliary Feedwater System.

Table 410.18N-2

Branch Technical Position ASB 10-1

Item	ASB 10-1 Position	Related to	STP Position	Reference FSAR Sections
1.	B.1	Independency & Diversity	Conforms(1)	10.4.9.2
2.	B.2	Diverse & Separate Motive Power	Conforms	10.4.9.2, 7.4.1.1, Table 10.1-1
3.	B.3	Train Separation & Crossconnect	Meets the intent.(2)	10.4.9.2, Fig 10.4.9.-1, 10.4.9.1.4, 10.4.9.3
4.	B.4	Redundancy	Conforms	10.4.9.1.4, 10.4.9.3
5.	B.5	AFW Flow Following HELB	Conforms(1)	10.4.9.1.4

1. Operator action is required to open the safety grade steam generator power operated relief valves for specific events.

Note

2 ~~1~~. The STP AFW system with its four independent trains is designed to function (provide the required AFW flow) following a postulated piping failure with or without off site power available considering, at the same time, any single failure.

Additionally the AFW trains are any or provided with a cross-connect for use during nonsafety-actuated AFW system operation. This allows one, two, ~~or~~ three or four operating pumps to feed all four SGs. In addition, the cross-connect valves are provided with manual actuators (locally operated) which would allow any operable AFW pump to be aligned with any effective SG during an extreme accident and failure combination.

II.E.1.1 AUXILIARY FEEDWATER SYSTEM EVALUATION

Position

The Office of Nuclear Reactor Regulation is requiring reevaluation of the auxiliary feedwater (AFW) systems for all PWR operating plant licensees and operating license applications. This action includes:

(1) Perform a simplified AFW system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFW system failure under various loss-of-main-feedwater-transient conditions. Particular emphasis is given to determining potential failures that could result from human errors, common causes, single-point vulnerabilities, and test and maintenance outages;

(2) Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance; and

(3) Reevaluate the AFW system flowrate design bases and criteria.

Clarification

Operating Plant Licenses--Items 1 and 2 above have been completed for Westinghouse (W), Combustion Engineering (C-E), and two Babcock and Wilcox (B&W) operating plants (Rancho Seco, short-term only, and TMI-1). As a result of staff review of items 1 and 2, letters were issued to these plants that required the implementation of certain short- and long-term AFW system upgrade requirements. Included in these letters was a request for additional information regarding item 3 above. The staff is now in the process of evaluating licensees' responses and commitments to these letters.

The remaining B&W operating plants (Oconee 1-3, Crystal River 3, ANO-1, and Davis-Besse 1) have submitted the analysis described in item 1 above. The analysis is presently undergoing staff review. When the results of the staff reviews are complete, each of the remaining B&W plants will receive a letter specifying the short- and long-term AFW system upgrade requirements based on item 1 above. Included in these letters will be a request for additional information regarding items 2 and 3 above.

Operating License Applicants--Operating license applicants have been requested to respond to staff letters of March 10, 1980 (W and C-E) and April 24, 1980 (B&W). These responses will be reviewed during the normal review process for these applications.

STP Response

The following information responds to the NRC letter of March 10, 1980, enclosure 2, relating to the Auxiliary Feedwater System Design Bases.

Question 1

a. Identify the plant transient and accident conditions considered in establishing AFWS flow requirements, including the following events:

1) Loss of Main Feedwater (LMFW)
2) LMFW w/loss of offsite AC power
3) LMFW w/loss of onsite and offsite AC power
4) Plant cooldown
5) Turbine trip with and without bypass
6) Main steam isolation valve closure
7) Main feedline break
8) Main steamline break
9) Small break LOCA
10) Other transient or accident conditions not listed above.

b. Describe the plant protection acceptance criteria and corresponding technical bases used for each initiating event identified above. The acceptance criteria should address plant limits such as:

1) Maximum RCS pressure (PORV or safety valve actuation)
2) Fuel temperature or damage limits (CNB, PCT, maximum fuel central temperature)
3) RCS cooling rate limit to avoid excessive coolant shrinkage
4) Minimum steam generator level to assure sufficient steam generator heat transfer surface to remove decay heat and/or cool down the primary system.

Response to 1.a

The Auxiliary Feedwater System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. As an Engineered Safeguards System, the Auxiliary Feedwater System is directly relied upon to prevent core damage and system overpressurization in the event of transients such as a loss of normal feedwater or a secondary system pipe rupture, and to provide a means for plant cooldown following any plant transient.

Following a reactor trip, decay heat is dissipated by evaporating water in the steam generators and venting the generated steam either to the condensers through the steam dump or to the atmosphere through the steam generator safety valves or the power-operated relief valves. Steam generator water inventory must be maintained at a level sufficient to ensure adequate heat transfer and continuation of the decay heat removal process. The water level is maintained under these circumstances by the Auxiliary Feedwater System which delivers an emergency water supply to the steam generators. The Auxiliary Feedwater System must be capable of functioning for extended periods, allowing time either to restore normal feedwater flow or to proceeed with an orderly cooldown of the plant to the reactor coolant conditions where the Residual Heat Removal System can assume the burden of decay heat removal. The Auxiliary Feedwater System flow and the emergency water supply capacity must be sufficient to remove core decay heat, reactor coolant pump heat, and sensible heat during the plant cooldown. The Auxiliary Feedwater System can also be used to maintain the steam generator water levels above the tubes following a LOCA. In the latter function, the water head in the steam generators serves as a barrier to prevent leakage of fission products from the Reactor Coolant System into the secondary plant.

DESIGN CONDITIONS

The reactor plant conditions which impose safety-related performance requirements on the design of the Auxiliary Feedwater System are as follows for the South Texas Units 1 & 2.

- Loss of Main Feedwater Transient
  - Loss of main feedwater with offsite power available
  - Loss of Offsite Power - LOOP (i.e., loss of main feedwater without offsite power available)

- Secondary System Pipe Ruptures
  - Feedline rupture
  - Steamline rupture

- Loss of all AC Power

- Loss of Coolant Accident (LOCA)

- Cooldown

Loss of Main Feedwater Transients

The design loss of main feedwater transients are those caused by:

- Interruptions of the Main Feedwater System flow due to a malfunction in the feedwater or condensate system

- Loss of offsite power or LOOP with the consequential shutdown of the system pumps, auxiliaries, and controls

These transients are discussed in Sections 15.2.6 and 15.2.7.

Loss of main feedwater transients are characterized by a reduction in steam generator water levels which results in a reactor trip, a turbine trip, and auxiliary feedwater actuation by the protection system logic. Following reactor trip from a high initial power level, the power quickly falls to decay heat levels. The water levels continue to decrease, progressively uncovering the steam generator tubes as decay heat is transferred and discharged in the form of steam either through the steam dump valves to the condenser or through the steam generator safety or power-operated relief valves to the atmosphere. The reactor coolant temperature increases as the residual heat in excess of that dissipated through the steam generators is absorbed. With increased temperature, the volume of reactor coolant expands and begins filling the pressurizer. Without the addition of sufficient auxiliary feedwater, further expansion will result in water being discharged through the pressurizer safety and/or relief valves. If the temperature rise and the resulting volumetric expansion of the primary coolant are permitted to continue, then (1) pressurizer safety valve capacities may be exceeded causing overpressurization of the Reactor Coolant System and/or (2) the continuing loss of fluid from the primary coolant system may result in bulk boiling in the Reactor Coolant System and eventually in core uncovering, loss of natural circulation, and core damage. If such a situation were ever to occur, the Emergency Core Cooling System would be ineffectual because the primary coolant system pressure exceeds the shutoff head of the safety injection pumps, the nitrogen over-pressure in the accumulator tanks, and the design pressure of the Residual Heat Removal Loop. Hence, the timely introduction of sufficient auxiliary feedwater is necessary to arrest the decrease in the steam generator water levels, to reverse the rise in reactor coolant temperature, to prevent the pressurizer from filling to a water solid condition, and eventually to establish stable hot standby conditions. Subsequently, a decision may be made to proceed with plant cooldown if the problem cannot be satisfactorily corrected.

The LOOP transient differs from a simple loss of main feedwater in that emergency power sources must be relied upon to operate vital equipment. The loss of power to the electric driven condenser circulating water pumps results in a loss of condenser vacuum and condenser dump valves. Hence, steam formed by decay heat is relieved through the steam generator safety valves or the power-operated relief valves. The calculated transient is similar for both the loss of main feedwater and the LOOP, except that reactor coolant pump heat input is not a consideration in the LOOP transient following loss of power to the reactor coolant pump bus.

Secondary System Pipe Ruptures

The feedwater line rupture accident not only results in the loss of feedwater flow to the steam generators but also results in the complete blowdown of one steam generator within a short time if the rupture should occur downstream of the last nonreturn valve in the main or auxiliary feedwater piping to an individual steam generator. Another significant result of a feedline rupture may be the spilling of auxiliary feedwater to the faulted steam generator. With a "typical" headered AFS arrangement, such situations can result in the injection of a disproportionately large fraction of the total auxiliary feedwater flow (the system preferentially pumps water to the lowest pressure region) to the faulted loop rather than to the effective steam generators which are at relatively high pressure. However, the South Texas units have four auxiliary feedwater pumps, with associated independent piping trains. Each auxiliary feedwater train delivers flow to a different steam generator. This arrangement allows the flow from only one auxiliary feedwater pump to spill through a break and ensures that sufficient flow will be delivered to the remaining effective steam generators. The concerns are similar for the main feedwater line rupture as those explained for the loss of main feedwater transients.

limit

Main steamline rupture accident conditions are characterized initially by plant cooldown and, for breaks inside containment, by increasing containment pressure and temperature. Auxiliary feedwater is not needed during the early phase of the transient but flow to the faulted loop will contribute to an excessive release of mass and energy to containment. Thus, steamline rupture conditions establish the upper limt on auxiliary feedwater flow delivered to a faulted loop. Eventually, however, the Reactor Coolant System will heat up again and auxiliary feedwater flow will be required to be delivered to the nonfaulted loops, but at somewhat lower rates than for the loss of feedwater transients described previously. Provisions must be made in the design of the Auxiliary Feedwater System to limit, control, or terminate the auxiliary feedwater flow to the faulted loop as necessary in order to prevent containment overpressurization following a steamline break inside containment, and to ensure the minimum flow to the remaining unfaulted loops.

X

Loss of All AC Power

Although the AFS must be designed to cope with a complete loss of ac power, i.e., the loss of both offsite and onsite ac power sources, this event is not considered to be a design basis event for overall plant design by current industry standards and government regulations.

The South Texas AFS provided three motor-driven pumps and one turbine-driven pump. Each pump is capable of delivering a minimum of 550 gpm at a pressure equivalent to the accumulation pressure of the lowest setpoint of the steam generator safety valves. The AFS is designed with diversity in pump motive power sources and essential instrumentation and control power sources. The AFS is capable of delivering the required flow of 550 gpm to at least one steam generator, assuming the loss of both onsite and offsite ac power.

Loss-of-Coolant Accident (LOCA)

The loss of coolant accidents discussed in Section 15.6.5 do not impose on the auxiliary feedwater system any flow requirements in addition to those required by the other accidents addressed in this response. The following description of the small LOCA is provided here for the sake of completeness to explain the role of the auxiliary feedwater system in this transient.

Small LOCAs are characterized by relatively slow rates of decrease in reactor coolant system pressure and liquid volume. The principal contribution from the Auxiliary Feedwater System following such small LOCAs is basically the same as the system's function during hot shutdown or following spurious safety injection signal which trips the reactor. Maintaining a water level inventory in the secondary side of the steam generators provides a heat sink for removing decay heat and establishes the capability for providing a buoyancy head for natural circulation. The auxiliary feedwater system may be utilized to assist in a system cooldown and depressurization following a small LOCA while bringing the reactor to a safe shutdown condition.

Cooldown

The cooldown function performed by the Auxiliary Feedwater System is a partial one since the reactor coolant system is reduced from normal zero load temperatures to a hot leg RCS temperature of approximately 350°F. The latter is the maximum temperature recommended for placing the Residual Heat Removal System (RHRS) into service. The RHR system completes the cooldown to cold shutdown conditions.

Cooldown may be required following expected transients, following an accident such as a main feedline break, or during a normal cooldown prior to refueling or performing reactor plant maintenance. If the reactor is tripped following extended operation at rated power level, the AFWS is capable of delivering sufficient AFW to remove decay heat and reactor coolant pump (RCP) heat following reactor trip while maintaining the steam generator (SG) water level. Following transients or accidents, the recommended cooldown rate is consistent with expected needs and at the same time does not impose additional requirements on the capacities of the auxiliary feedwater pumps, considering a single failure. The Auxiliary Feedwater System is provided with a seismic Category I Auxiliary Feedwater Storage tank which is sized with sufficient capacity for 4 hours of standby, followed by a 10 hour natural circulation cooldown, with an additonal 8 hour soak period.

Response to 1.b

Table II.E.1.1-1 summarizes the criteria which are the general design bases for each event, discussed in the response to Question 1.a. Specific assumptions used in the analyses to verify that the design bases are met are discussed in response to Question 2. (See also the response to NRC Question 410.18N.)

The primary function of the Auxiliary Feedwater System is to provide sufficient heat removal capability following reactor trip and to remove the decay heat generated by the core and prevent system overpressurization. Other plant protection systems are designed to meet short-term or pre-trip fuel failure criteria. The effects of excessive coolant shrinkage are evaluated by the analysis of the rupture of a main steam pipe transient. The maximum flow requirements determined by other bases are incorporated into this analysis, resulting in no additional flow requirements.

Question 2

Describe the analyses and assumptions and corresponding technical justification used with plant condition considered in 1.a above including:

a. Maximum reactor power (including instrument errors allowance) at the time of the initiating transient or accident.

b. Time delay from initiating event to reactor trip.

c. Plant parameter(s) which initiates AFWS flow and time delay between initiating event and actuation of AFWS flow.

d. Minimum steam generator water level when initiating event occurs.

e. Initial steam generator water inventory and depletion rate before and after AFWS flow commences -- identify reactor decay heat rate used.

f. Maximum pressure at which steam is released from steam generator(s) and against which the AFW pump must develop sufficient head.

g. Minimum number of steam generators that must receive AFW flow; e.g., 1 out of 2? 2 out 4?

h. RC flow condition -- continued operation of RC pumps or natural circulation.

i. Maximum AFW inlet temperature.

j. Following a postulated steam or feed line break, time delay assumed to isolate break and direct AFW flow to intact steam generator(s). AFW pump flow capacity allowance to accommodate the time delay and maintain minimum steam generator water level. Also identify credit taken for primary system heat removal due to blowdown.

k. Volume and maximum temperature of water in main feed lines between steam generator(s) and AFWS connection to main feed line.

l. Operating condition of steam generator normal blowdown following initiating event.

m. Primary and secondary system water and metal sensible heat used for cooldown and AFW flow sizing.

n. Time at hot standby and time to cooldown RCS to RHR system cut in temperature to size AFW water source inventory.

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Response to 2

Analyses have been performed for the limiting transients which define the AFWS performance requirements. These analyses have been provided for review in the FSAR. Specifically, they include:

- Loss of Main Feedwater/Loss of Offsite Power (LOOP)
- Rupture of a Main Feedwater Pipe
- Rupture of a Main Steam Pipe Inside Containment

In addition to the above analyses, calculations have been performed specifically for the South Texas Units to determine the plant cooldown flow (storage capacity) requirements. The Loss of All AC Power is evaluated via a comparison to the transient results of a LOOP, assuming an available auxiliary pump having a diverse (non-ac) power supply. The LOCA analysis, as discussed in response to Question 1.b, incorporates the system flow requirements as defined by other transients, and therefore is not performed for the purpose of specifying AFWS flow requirements. Each of the analyses listed above are explained in further detail in the following sections of this response.

Loss of Main Feedwater/Loss of Offsite Power (LOOP)

The Loss of Main Feedwater/ LOOP events were analyzed for the South Texas Units and are presented in FSAR Section 15.2.7. The difference between the two events is that, for the LOOP case, power to the Reactor Coolant Pumps is assumed to be lost following reactor trip. The acceptance criteria for these ANS Condition II events, as listed in Table II.E.1.1-1, are all met. The following assumptions, concerning the AFWS, have been made. Sixty seconds following generation of the low-low steam generator water level signal, auxiliary feedwater is initiated. Two ~~One~~ auxiliary feedwater pumps ~~is~~ are assumed to provide auxiliary feedwater to each of two ~~one~~ steam generators ~~at a rate of~~ @ 540 ~~575 gpm~~. It takes approximately 78 seconds to eliminate the 90 cubic foot purge volume before the relatively cold auxiliary feedwater (120 ~~100~~°F) reaches the steam generator. Table II.E.1.1-2 summarizes the assumptions used in these analyses. In addition, FSAR Section 15.2.7 provides more detail concerning the Loss of Main Feedwater/ LOOP analysis.

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Rupture of a Main Feedwater Pipe

The Main Feedwater Pipe Rupture event was analyzed for the South Texas Units and is presented in FSAR Section 15.2.8. Cases were analyzed both with and without offsite power available. The acceptance criteria for this ANS Condition IV event, as listed in Table II.E.1.1-1, are all met. The following assumptions, concerning the AFWS, have been made. Sixty seconds following generation of the low-low steam generator water level signal, auxiliary feedwater is initiated. One auxiliary feedwater pump is assumed to provide auxiliary feedwater to one nonfaulted steam generator at a rate of 540 gpm. It takes approximately 83 seconds to eliminate the 100 cubic foot purge volume before the relatively cold auxiliary feedwater (160°F) reaches the steam generator. Table II.E.1.1-2 summarizes the assumptions used in this analysis. In addition, FSAR Section 15.2.8 provides more detail concerning the Main Feedwater Pipe Rupture analysis.

OK

≥ 120

Rupture of a Main Steam Pipe Inside Containment

Because the steamline break transient is a cooldown, the AFWS is not needed to remove heat in the short term. Furthermore, addition of excessive auxiliary feedwater to the faulted steam generator will affect the peak containment pressure following a steamline break inside containment. This transient is performed at four power levels for several break sizes. Auxiliary feedwater is assumed to be initiated at the time of the break, independent of system actuation signals. The maximum flow is used for this analysis. Table II.E.1.1-2 summarizes the assumptions used in this analysis. At 30 minutes after the break, it is assumed that the operator has isolated the AFWS from the faulted steam generator which subsequently blows down to ambient pressure. The criteria stated in Table II.E.1.1-1 are met.

This transient establishes the maximum allowable auxiliary feedwater flow rate to a single faulted steam generator assuming all pumps operating, establishes the basis for runout protection, if needed, and establishes layout requirements so that the flow requirements may be met considering the worst single failure.

Plant Cooldown

Maximum and minimum flow requirements from the prevously discussed transients meet the flow requirements of plant cooldown. This operation, however, defines the basis for tank size, based on the required cooldown duration, maximum decay heat input and maximum stored heat in the system. As previously discussed in the response to Question 1.a, the Auxiliary Feedwater System (AFWS) partially cools the system to the point where the RHRS may complete the cooldown, i.e., 350°F in the RCS. Table II.E.1.1-2 shows the assumptions used to determine the cooldown heat capacity of the Auxiliary Feedwater System.

The cooldown is assumed to commence at the maximum rated power, and maximum trip delays and decay heat source terms are assumed when the reactor is tripped. Primary metal, primary water, secondary system metal and secondary system water are all included in the stored heat to be removed by the AFWS. See Table II.E.1.1-3 for the items constituting the sensible heat stored in the NSSS.

This operation is analyzed to establish minimum tank size requirements for auxiliary feedwater fluid source which are normally aligned.

Question 3

Verify that the AFW pumps in your plant will supply the necessary flow to the steam generator(s) as determined by items 1 and 2 above considering a single failure. Identify the margin in sizing the pump flow to allow for pump recirculation flow, seal leakage and pump wear.

Response to 3

The South Texas Auxiliary Feedwater System flow design capabilities, considering various single failures, are documented in the Failure Mode Analysis for the AFW System provided in Table 10.4-3 of the South Texas FSAR.

The South Texas AFW pump sizing is based on delivering the required flow at the lowest steam generator safety relief valve set pressure plus accumulation (1339 psia). The required flow does not include a continuous recirculation flow because of the system use of Automatic Recirculation Control (ARC) valves which provide 100% forward flow when the flowrate is above the pump minimum flow requirements. Likewise, the seal leakage is not considered in the pump design flow since the pumps are provided with mechanical seals. The AFW pump design wear margin is based on head rather than flow, and when converted to flow this wear margin is approximately 4%.

TABLE II.E.1.1-1

CRITERIA FOR AUXILIARY FEEDWATER SYSTEM DESIGN BASIS CONDITIONS

Condition or Transient	Classification*	Criteria	Additional Design Criteria
Loss of Main Feedwater	Condition II	Peak RCS pressure not to exceed design pressure +10% No consequential fuel failures	Pressurizer does not fill
Loss of Offsite Power	Condition II	(same as LMFW)	Pressurizer does not fill
Feedline Rupture	Condition IV	10CFR100 dose limits Containment design pressure not exceeded	Core does not uncover
Loss of all A/C Power*	N/A	Note 1 Containment design pressure not exceeded	Same as ~~blackout~~ LOOP assuming turbine driven pump
Loss of Coolant	Condition III	10 CFR 100 dose limits 10 CFR 100 PCT limits	
	Condition IV	10 CFR 100 dose limits 10 CFR 100 PCT limits	
Cooldown	N/A		100°F/hr 567°F to 350°F

Okay

*ANSI N18.2 (This information provided for those transients performed in the FSAR.)

Note 1: Although this transient establishes the basis for AFW pump and instrumentation/controls powered by a diverse power source, this is not evaluated relative to typical criteria since multiple failures must be assumed to postulate this transient.

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TABLE II.E.1.1-2

SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

Transient	Loss of Feedwater/ ~~Station Blackout~~ LOOP	Cooldown	Main Feedline Break	Main Steamline Break (Containment)
a. Max NSSS power	102% of nominal rating (102% of 3817 MWt)	4100 MWt	102% of nominal rating (102% of 3817 MWt)	0, 30, 70, 100% (percent of 3817 MWt)
b. Time delay from event to Rx trip	~~24~~ sec 55.2 sec 54.9 sec LNFW/~~SB~~ LOOP	2 sec	~~17.3~~ sec 22.0 sec	variable
c. AFWS actuation signal/time delay for AFWS flow	low-low SG level/ 1 minute	N/A	Low-low SG level/ 1 minute	Assumed immediately @ 0 sec (no delay)
d. SG water level at time of reactor trip	(low-low SG level) 239 ~~27.0%~~ NR span ~~91,344~~ lbm 84,663	N/A	(low-low SG level) 182 ~~22.9%~~ NR span ~~86,625~~ lbm 80077	N/A
e. Initial SG inventory	~~170,020~~ lbm/SG 148,635	98,100 lbm/SG at 556.3°F	Broken Loop - ~~120,074~~ 128,565 lbm Intact Loop - ~~118,830~~ 119,433 lbm	consistent with power
Rate of change before & after AFWS actuation	See FSAR Figure 15.2-10	N/A	See Figure II.E.1.1-1	N/A
Decay heat	ANS-5.1-1979 + 2σ	N/A	ANS-5.1-1979 + 2σ	ANS + 20%
f. SG pressure used for AFW pump design ~~pressure~~	1339 psia	1339 psia	1339 psia	N/A
g. Minimum # of SGs which must receive AFW flow	~~X~~ of 4 2	N/A	1 of 4	N/A

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TABLE II.E.1.1-2 (Continued)

SUMMARY OF ASSUMPTIONS USED IN AFWS DESIGN VERIFICATION ANALYSES

Transient	Loss of Feedwater/ ~~Station Blackout~~[2] LOOP	Cooldown	Main Feedline Break	Main Steamline Break (Containment)
h. RC pump status	*Tripped at reactor trip (2 second delay)	Tripped	*Tripped @ reactor trip (2 second delay)	All operating
i. Maximum AFW temperature	~~160°F**~~ 120°F	120°F	160°F** 120 F**	Same temperature as main feedwater at initial operating power
j. Operator action	None	N/A	None	Aux. feed flow terminated after 30 minutes
k. MFW purge volume/ S/G and temperature	90 ft^3/440°F	0 ft^3/440°F	100 ft^3/440°F	450 ft^3/loop (for dryout time)
l. Normal blowdown	none assumed	none assumed	none assumed	none assumed
m. Sensible heat	see cooldown	Table II.E.1.1-3	see cooldown	N/A
n. Time at standby/time to cooldown to RHR	2 hr/10 hrs*	2 hr/5 hrs	2 hr/5 hrs	N/A
o. AFW flowrate	~~550~~ 1010 ~~915~~ gpm*** constant	variable	540 gpm*** constant	1210 gpm (constant) to broken SG

* with offsite power not available
** 120°F is maximum temperature
*** system design is 550 gpm per train

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9318Q:1D/121085

TABLE II.E.1.1-3

SUMMARY OF SENSIBLE HEAT SOURCES

Primary Water Sources (initially at rated power temperature and inventory)

- RCS fluid
- Pressurizer fluid (liquid and vapor)

Primary Metal Sources (initially at rated power temperature)

- Reactor coolant piping, pumps and reactor vessel
- Pressurizer
- Steam generator tube metal and tube sheet
- Steam generator metal below tube sheet
- Reactor vessel internals

Secondary Water Sources (initially at rated power temperature and inventory)

- Steam generator fluid (liquid and vapor)
- Main feedwater purge fluid between steam generator and AFWS piping

Secondary Metal Sources (initially at rated power temperature)

- All steam generator metal above tube sheet, excluding tubes

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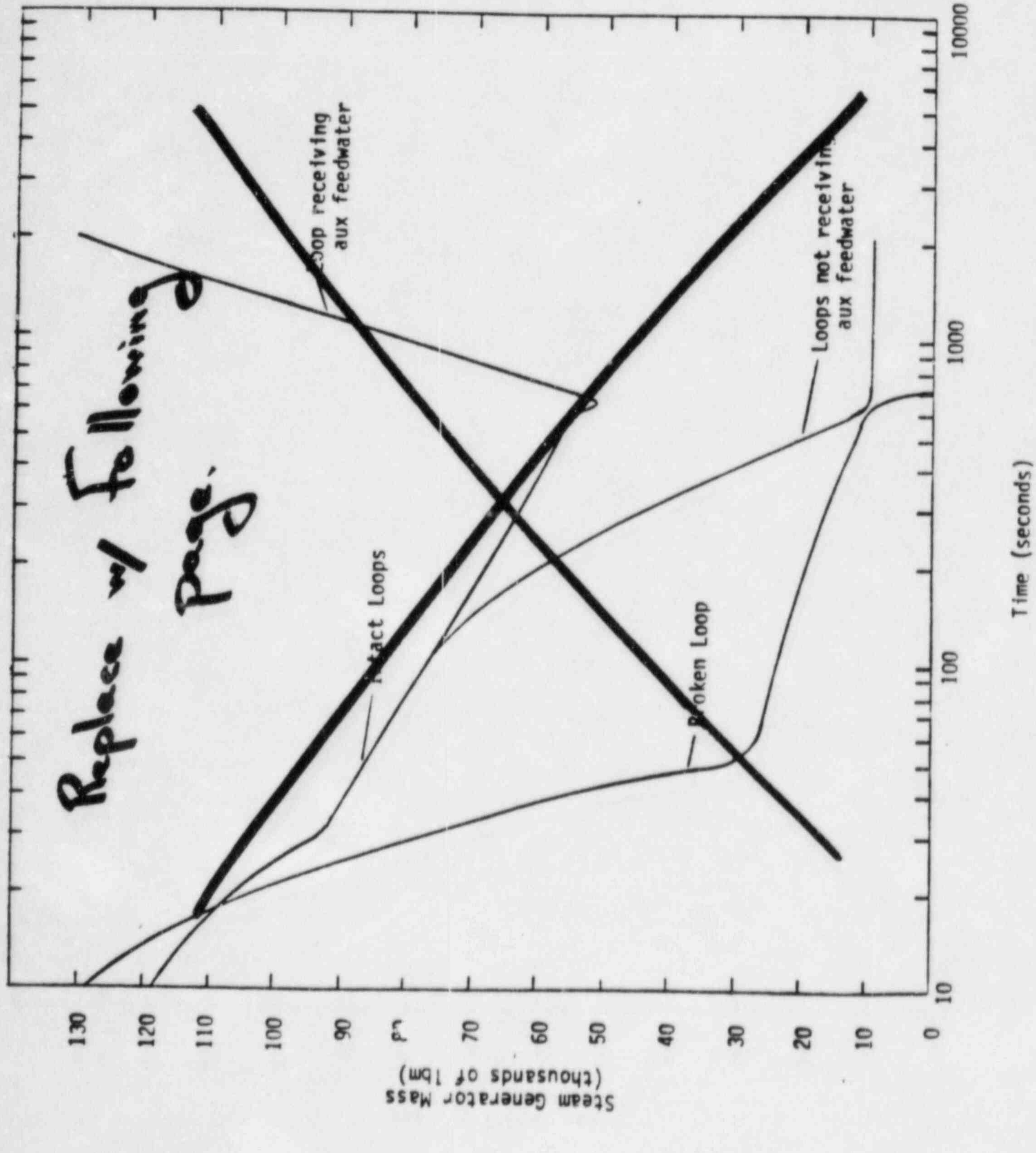
Figure II.E.1.1-1 Feedline Break



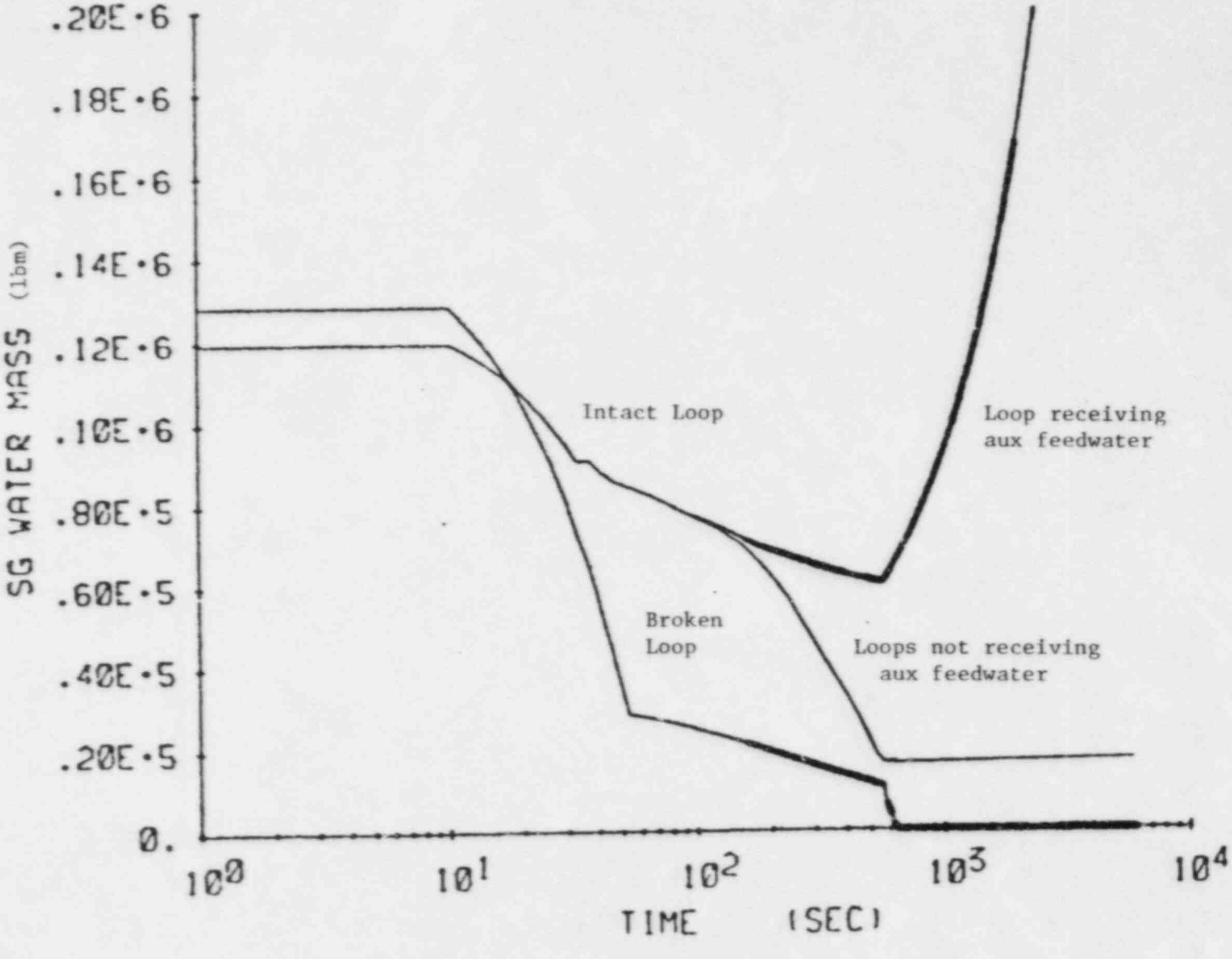


Figure II.E.1.1-1
Feedline Break

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2. Average Reactor Coolant System temperature	$\pm$ 4°F allowance for controller deadband and measurement error (unless otherwise specified in the text)
3. Pressurizer pressure	$\pm$ 30 pounds per square inch allowance for steady state fluctuations and measurement error (unless otherwise specified in the text)

Initial values for core power, average RCS temperature and pressurizer pressure are selected to minimize the initial departure from nucleate boiling ratio (DNBR) unless otherwise stated in the sections describing specific accidents. Table 15.0-2 summarizes the initial conditions and computer codes used in the accident analyses.

15.0.3.3 Power Distribution. The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of control rods and operating restrictions. Power distribution may be characterized by the radial factor ($F\Delta_H$) and the total peaking factor (F_q). The peaking factor limits will be given in the Technical Specifications. 27

For transients which may be departure from nucleate boiling (DNB) limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F\Delta_H$ is included in the core limits illustrated on Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F\Delta_H$ consistent with the initial power level defined in the Technical Specifications.

The axial power shape used in the DNB calculation is discussed in Section 4.4.

The radial and axial power distributions described above are input to the THINC Code as described in Section 4.4.

For transients which may be overpower limited the total peaking factor (F_q) is of importance. All transients that may be overpower limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, the fuel rod thermal evaluations are performed as discussed in Section 4.4. Examples are the CVCS malfunction that results in a decrease in the boron concentration in the reactor coolant inventory which lasts many minutes, and the excessive increase in secondary steam flow incident which may reach equilibrium without causing a reactor trip. For overpower transients which are fast with respect to the fuel rod thermal time constant, a detailed fuel heat transfer calculation must be performed. Examples are the uncontrolled RCCA bank withdrawal from subcritical or low power startup and RCCA ejection incidents which result in a large power rise over a few seconds. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and rod power, a typical value at beginning-of-life for high power rods is approximately five seconds.

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TABLE 15.0-3

NOMINAL VALUES OF PERTINENT PLANT PARAMETERS

UTILIZED IN THE ACCIDENT ANALYSES[a]

Thermal output of NSSS (MWt)	See Table 15.0-2	
Core inlet temperature (°F)	560.0	
Vessel average temperature (°F)	593.0	18
Reactor Coolant System pressure (psia)	2250	
Reactor coolant flow per loop (gpm)	94,100*	18
Steam flow from NSSS (lb/hr)	16,960,000	18
Steam pressure at steam generator outlet (psia)	1100	
Maximum steam moisture content (%)	0.25	
Assumed feedwater temperature at steam generator inlet (°F)	440	
Average core heat flux (Btu/hr-ft^2)	181200	

[a] Steady state errors discussed in Section 15.0.3 are added to these values to obtain initial conditions for transient analyses.

* A loop flow of 95,400 gpm was used in the Locked Rotor rods-in-DW^ analysis.

Reference 15.2-4 presents additional results of analysis for a complete loss of heat sink including loss of main feedwater. This analysis shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.3.3 Radiological Consequences. There are only minimal radiological consequences associated with this event, therefore, this event is not limiting. The radiological consequences resulting from atmospheric steam dump are less severe than the steam line break event discussed in Section 15.1.5.

15.2.3.4 Conclusions. Results of the analyses, including those in Reference 15.2-4, show that the plant design is such that a turbine trip without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The DNBR remains above 1.30 for all cases analyzed; thus, the DNB design basis as described in Section 4.4 is met. The above analysis demonstrates the ability of the NSSS to safely withstand a full load rejection. 18

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of main steam isolation valves would cause a turbine trip and other consequences as described in Section 15.2.5 below.

15.2.5 Loss of Condenser Vacuum and Other Events Causing a Turbine Trip

Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 10.2. A loss of condenser vacuum would preclude the use of turbine bypass to the condenser; however, since turbine bypass is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum. 43 In addition, analyses for the other possible causes of a turbine trip, as listed in Section 10.2, are covered by Section 15.2.3. Possible overfrequency 43 effects due to a turbine overspeed condition are discussed in Section 15.2.2.1 and are not a concern for this type of event.

15.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)

15.2.6.1 Identification of Causes and Accident Description. A complete loss of nonemergency ac power may result in the loss of all power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the plant or by a loss of the onsite ac distribution system.

This transient is more severe than the turbine trip event analyzed in Section 15.2.3 because, for this case, the decrease in heat removal by the secondary system is accompanied by a flow coastdown which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip due to (1) turbine trip; (2) ~~upon~~ reaching one of the trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal; or (3) ~~due to~~ loss of power to the control rod drive mechanisms (CRDMs) as a result of the loss of power to the plant.

Following a loss of ac power with turbine and reactor trips, the sequence described below will occur.

1. Plant vital instruments are supplied from emergency dc power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves may be automatically opened to the atmosphere. The condenser is assumed not to be available for turbine bypass. If the steam relief through the power-operated ~~a~~ relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor. 43

3. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power-operated relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition. 43

4. The standby diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

Three motor-driven and one turbine-driven auxiliary feedwater trains deliver water to their respective steam generators on any of the following: 43

1. Low-low water level in any steam generator

2. Safety injection signal

3. Manual actuation

turbine-driven auxiliary

The motor-driven auxiliary feedwater pumps are supplied power by the Diesel generators. The turbine-auxiliary driven feedwater pump utilizes steam from the secondary system. Both types of pumps are designed to start within one minute of the actuating signal. The turbine-auxiliary driven feedwater pump exhausts the secondary steam to the atmosphere. The auxiliary feedwater pumps take suction from the auxiliary feedwater storage tank for delivery to the steam generators. 43

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

A loss of nonemergency ac power event, as described above, is a more limiting event than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power, which was analyzed in Section 15.2.3. However, a loss of ac power to the plant auxiliaries as postulated above also results in a loss of normal feedwater since the feedwater booster pumps lose their power supply. A loss of normal feedwater caused by a loss of ac power is the most 43

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limiting Condition II event in the decrease in secondary heat removal category, and is analyzed in Section 15.2.7. Therefore, detailed analytical results for a loss of ac power transient will not be presented here. The results of the analysis in Section 15.2.7 are applicable to the loss of ac power event.

Following the reactor coolant pump coastdown caused by the loss of ac power, the natural circulation capability of the RCS will remove residual and decay heat from the core, aided by auxiliary feedwater in the secondary system. An analysis is presented to show that the natural circulation flow in the RCS following a loss of ac power event is sufficient to remove residual heat from the core.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-11. 2 Q211 .6

The plant systems and equipment available to mitigate the consequences of a loss of ac power event are discussed in Section 15.0.8 and listed in Table 15.0-6.

15.2.6.2 Analysis of Effects and Consequences.

Method of Analysis

A detailed analysis using the LOFTRAN ~~code~~ Code (Reference 15.2-3) is performed to obtain the natural circulation flow following a loss of offsite power. The simulation describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system. The digital program computes pertinent variables including the steam generator water level, pressurizer water level, and reactor coolant average temperature. 43

The assumptions used in the analysis are as follows:

1. The plant is initially operating at 102 percent of the nominal NSSS design rating;
2. A conservative core residual heat generation is based upon long-term operation at the initial power level preceding the trip;
3. A heat transfer coefficient in the steam generator is associated with RCS natural circulation.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

~~Steady-state cases are run at a number of power levels consistent with the decay heat generation rates expected. Equilibrium conditions are established, and the natural circulation flow through the core is recorded for each power level.~~

Results

The transient response of the RCS following a loss of ac power is less severe than for the loss of normal feedwater event analyzed in Section 15.2.7, and the results are not reproduced here.

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The first few seconds of the transient will closely resemble the complete loss of flow incident (see Section 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. |43

~~Natural circulation flow as a function of residual reactor power is presented in Table 15.2-2. The LOFTRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.~~

15.2.6.3 Radiological Consequences. A loss of nonessential ac power to plant auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power-operated relief valves or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences are less severe than the steam line break accident. |43

15.2.6.4 Conclusions. Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

|43

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description. A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The first postulated loss of normal feedwater event is one initiated by a loss of offsite power as described in Section 15.2.6. This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the reactor coolant pump coastdown.

As stated in Section 15.2.6.1, the following occur upon loss of ac power:

1. Plant vital instruments are supplied from essential dc power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Turbine bypass to the condenser is assumed not to be available. If the steam flow through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor. |43

3. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power-operated

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The first few seconds of the transient will closely resemble the complete loss of flow incident (see Section 15.3.2), i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual decay heat must be removed to prevent damage to either the RCS or the core. | 43

Natural circulation flow as a function of residual reactor power is presented in Table 15.2-2. The LOFTRAN results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and reactor coolant pump coastdown.

15.2.6.3 Radiological Consequences. A loss of nonessential ac power to plant auxiliaries would result in a turbine and reactor trip and loss of condenser vacuum. Heat removal from the secondary system would occur through the steam generator power-operated relief valves or safety valves. Since no fuel damage is postulated to occur from this transient, the radiological consequences are less severe than the steam line break accident. | 43

15.2.6.4 Conclusions. Analysis of the natural circulation capability of the RCS has demonstrated that sufficient heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage. | 43

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Accident Description. A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If an alternative supply of feedwater were not supplied to the plant, core residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur, resulting in a substantial loss of water from the RCS. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The worst postulated loss of normal feedwater event is one initiated by a loss of offsite power as described in Section 15.2.6. This is due to the decreased capability of the reactor coolant to remove residual core heat as a result of the reactor coolant pump coastdown.

As stated in Section 15.2.6.1, the following occur upon loss of ac power:

1. Plant vital instruments are supplied from essential dc power sources.

2. As the steam system pressure rises following the trip, the steam generator power-operated relief valves are automatically opened to the atmosphere. Turbine bypass to the condenser is assumed not to be available. If the steam flow through the power-operated relief valves is not available, the steam generator safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual decay heat produced in the reactor. | 43

3. As the no-load temperature is approached, the steam generator power-operated relief valves (or the safety valves, if the power-operated

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relief valves are not available) are used to dissipate the residual decay heat and to maintain the plant at the hot standby condition. 43

4. The standby diesel generators, started on loss of voltage on the plant emergency buses, begin to supply plant vital loads.

A loss of normal feedwater is classified as an ANS Condition II event, fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

Reactor trip on low-low water level in any steam generator provides protection for ~~a~~ the loss of normal feedwater.

The AFWS is started automatically as discussed in Section 15.2.6.1. The steam-driven auxiliary feedwater pump utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven auxiliary feedwater pumps are supplied power from the standby diesel generators. The pumps take suction directly from the auxiliary feedwater storage tank for delivery to the steam generators. 4

Upon loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and removal of residual heat is maintained by natural circulation in the reactor coolant loops. The analysis presented in Section 15.2.6 demonstrates the natural circulation capability of the RCS.

A loss of normal feedwater with a subsequent loss of power to the reactor coolant pumps ~~caused by a loss of offsite power~~ is the most limiting Condition II event in the decrease in secondary heat removal category. ~~Therefore,~~ a full analysis of the system transient is presented below to show that following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat, thus preventing either overpressurization of the RCS or loss of water from the reactor core, and returning the plant to a safe condition.

A

15.2.7.2 Analysis of Effects and Consequences.

Method of Analysis

A detailed analysis using the LOFTRAN Code (Reference 15.2.3) is performed in order to obtain the plant transient following a loss of normal feedwater. The simulation describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generator and feedwater system. The digital program computes pertinent variables including the steam generator water level, pressurizer water level, and reactor coolant average temperature. 4

Assumptions made in the analysis are:

1. The plant is initially operating at 102 percent of the nominal NSSS design rating.

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2. A conservative core residual heat generation is based upon long-term operation at the initial power level preceding the trip.

3. A heat transfer coefficient in the steam generator is associated with RCS natural circulation.

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2. Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 15.2-6). ANSI/ANS 5.1-1979 is a conservative representation of the decay energy release rates.

(A)

with and without reactor coolant pumps in operation has been done ~~for the transient~~. for the Loss of Normal Feedwater event. The limiting of the two transients, without reactor coolant pumps in operation,

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4. Reactor trip occurs on steam generator low-low water level. ~~No credit is~~ 43
leave - taken for immediate release of the control rod drive mechanisms caused by a loss of offsite power.

(a single train of auxiliary feedwater

5. The worst single failure in the AFW occurs (~~auxiliary feedwater pump fails to start~~). ~~actuation logic fails causing failure of two auxiliary feed~~ pump 43

6 ~~5~~. Auxiliary feedwater is delivered by ~~two~~ ~~one~~ two auxiliary feed water pumps to ~~two~~ ~~one~~ two steam generators. ~~This assumption is more severe than the single failure assumption of loss of one AFW pump.~~

7 ~~6~~. Secondary system steam relief is achieved through the steam generator safety valves.

8 ~~7~~. The initial reactor coolant average temperature is ~~4.0~~ ~~4.6~~ 4.7°F lower than the nominal value since this assumption results in a greater expansion of the RCS water during the transient and, thus, in a higher water level in the pressurizer at the time of maximum insurge. The initial pressurizer pressure uncertainty is 34 psi.

The loss of normal feedwater analysis is performed to demonstrate the adequacy of the RTS and ESF (e.g., the AFW) in removing long-term decay heat and preventing excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS water.

As such, the assumptions used in this analysis are designed to minimize the energy removal capability of the system and to maximize the possibility of water relief from the coolant system by maximizing the coolant system expansion, as noted in the assumptions listed above. 43

One such assumption is the loss of offsite power. This assumption results in coolant flow decay down to natural circulation conditions and a corresponding reduction in the steam generator heat transfer coefficient. Following a loss of offsite power, the first few seconds of a loss of normal feedwater transient will be virtually identical to the transient response (including DNBR and neutron flux versus time) presented in Section 15.3.2 for the complete loss of forced reactor coolant flow.

If offsite power were not lost ~~for this~~ ~~at the initiation of this~~ Start incident, the reactor coolant flow would remain at its normal value, and the reactor would trip via the low-low steam generator water level trip. The DNBR never falls below the value at the start of the transient. ~~The reactor coolant pumps may be manually tripped at some later time to reduce heat addition to the RCS and prevent filling the pressurizer.~~ 43

An additional assumption made for the loss of normal feedwater evaluation is that the pressurizer power-operated relief valves (PORV's) are assumed to function normally. Operation of the ~~valves~~ PORV's maintains peak RCS pressure ~~at or~~ below the actuation setpoint (~~2350~~ 2500 psia) of the pressurizer safety valves throughout the transient.

If these valves were assumed not to function, the coolant system pressure during the transient would rise to the actuation point of the pressurizer safety valves ~~(2500 psia)~~. The increased RCS pressure, however, results in less expansion of the coolant and, hence, more margin to the point where water relief from the pressurizer would occur. Plant characteristics and initial conditions are further discussed in Section 15.0.3.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-12.

Plant systems and equipment which are available to mitigate the effects of a loss of normal feedwater accident are discussed in Section 15.0.8 and listed in Table 15.0-6. Normal reactor control systems are not required to function. Pressurizer power-operated relief valves are assumed to function in order to provide a more limiting transient, as described above. The RTS is required to function following a loss of normal feedwater as analyzed here. The AFW system is required to deliver a minimum auxiliary feedwater flow rate. In the long term after automatic actuation of the AFW system, feedwater addition is manually controlled to maintain proper steam generator water level. No single active failure will prevent operation of any system required to function. A discussion of ATWT considerations is presented in Reference 15.2-2.

Results

15.2-9A through 15.2-9D

Figures ~~15.2-9~~ and 15.2-10 show the significant plant parameters following a loss of normal feedwater.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of the steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. Within one minute following the low-low water level signal, at least ~~two~~ auxiliary feedwater trains ~~are~~ delivering flow automatically, reducing the rate of water level decrease. 43

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The capacity of the auxiliary feedwater pumps is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves. From Figures ~~15.2-9 and~~ 15.2-10, it can be seen that at no time is the tubesheet uncovered in the steam generators receiving auxiliary feedwater flow, ~~and that at no time is there water relief from the pressurizer.~~

stet INSERT II

15.2-9A through 15.2-9D

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown on Figures ~~15.2-9~~ and 15.2-10, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.7.3 Radiological Consequences. The steam release and resulting radiological consequences from this transient would be the same as that for the loss of offsite power, and, similarly, radiological consequences resulting from this transient are less severe than the steam line break accident.

15.2.7.4 Conclusions. Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the auxiliary feedwater capacity is such that reactor coolant water is not relieved from the pressurizer relief or safety valves, and the water level in ~~all~~ steam generators receiving auxiliary feedwater is maintained above the tubesheets. The radiological consequences of this event are not limiting.

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The analysis also indicates that at no time is there water relief from the pressurizer; Figure 15.2-9D shows that the peak water volume in the pressurizer is less than 2100 ft^3 which is the filled pressurizer volume.

TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
	~~Rods begin to drop~~	~~6.4~~	
	~~Minimum DNBR occurs~~	~~(1)~~	
	~~Initiation of steam release from steam generator safety valves~~	~~8.0~~	18
	~~Peak pressurizer pressure occurs~~	~~8.0~~	
Loss of Normal Feedwater Flow	Main feedwater flow stops	10.0 ~~[illegible]~~	
	Low-low steam generator water level trip	63.2 ~~[illegible]~~ ~~22.9~~	43
	Rods begin to drop	65.2 ~~[illegible]~~ ~~24.9~~	
	Reactor coolant pumps begin to ~~coast~~down(2)	67.2 ~~24.8~~ ~~24.9~~	
	Ⓐ ~~One Two steam generators begins to receive auxiliary feed from one~~ ~~feedwater from its associated~~ ~~auxiliary feedwater pumps~~	~~[illegible]~~ ~~[illegible]~~	43
	Core decay heat decreases to auxiliary feedwater heat removal capacity	~~2000~~ ~~~2150~~ 2600	
	Two auxiliary feedwater pumps start and supply 2 steam generators	123.2	

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TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
move to (A)	Peak water ~~level~~ volume in pressurizer occurs	~~2811.0~~ ~~2196.0~~ 71.0	
~~Feedwater System Pipe Break~~			
~~1. With Offsite Power Available~~	~~Main feedline rupture occurs~~	~~10~~	
	~~Low-low steam generator water level reactor trip setpoint reached in affected steam generator~~	~~27~~	4
	~~Rods begin to drop~~	~~29~~	
	~~Auxiliary feedwater is delivered to intact steam generators~~	~~87~~	
	~~Low steamline pressure setpoint reached in affected steam generator~~	~~652~~	32 Q211.
	~~All main steam isolation valves close~~	~~660~~	43
	~~Pressurizer power-operated relief valve setpoint reached~~	~~788~~	43
	~~Steam generator safety valve setpoint reached in intact steam generator receiving auxiliary feedwater~~	~~960~~	
	~~Pressurizer water relief begins~~	~~2544~~	

TABLE 15.2-2

NATURAL CIRCULATION FLOW

Power (percent)	Natural Circulation Flow (percent)
4.0	6.45
3.5	6.16
3.0	5.86
2.5	5.53
2.0	5.15

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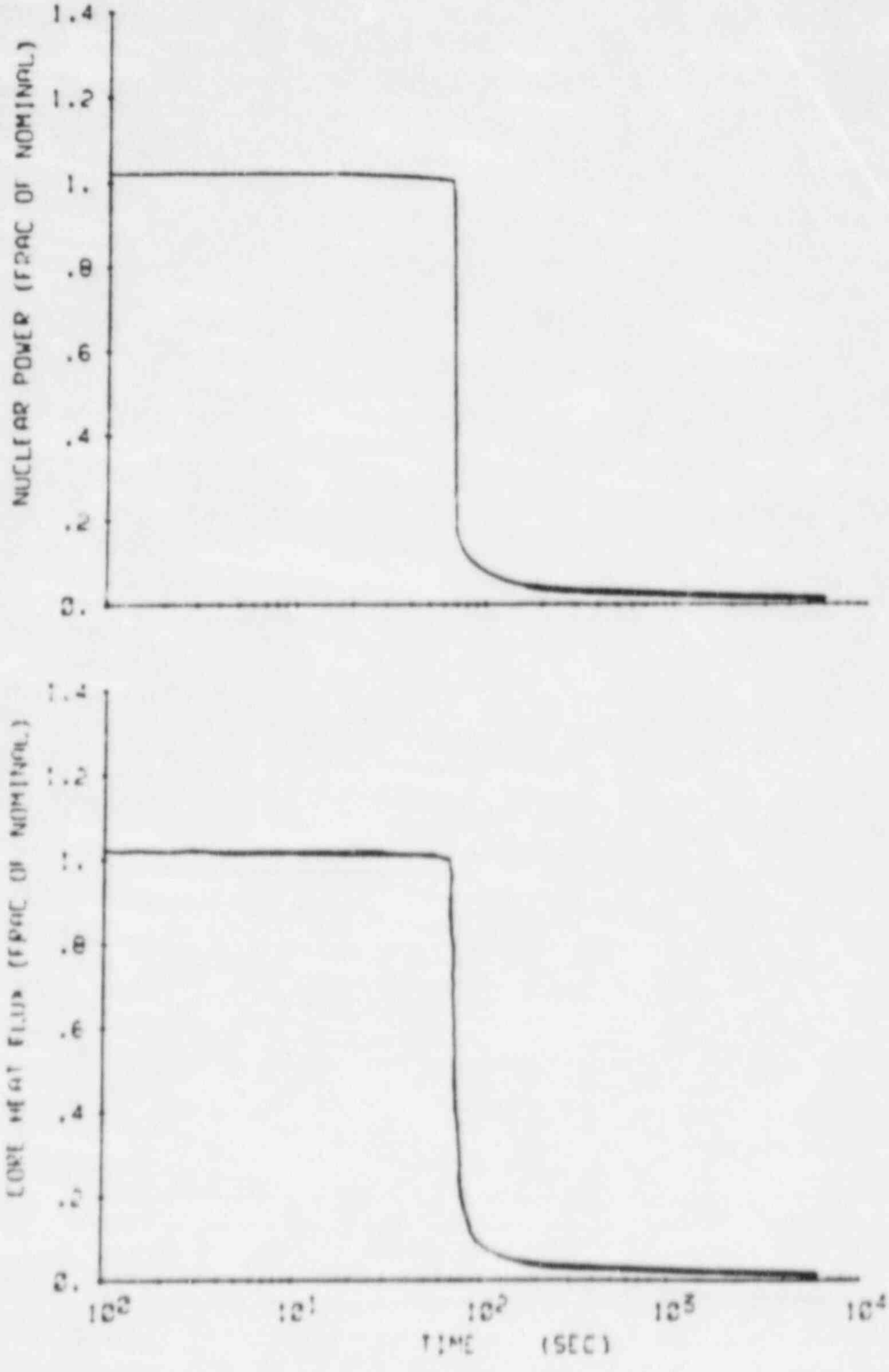

Figure 15.2-9A NUCLEAR POWER AND CORE HEAT FLUX TRANSIENTS

LOSS OF NORMAL FEEDWATER

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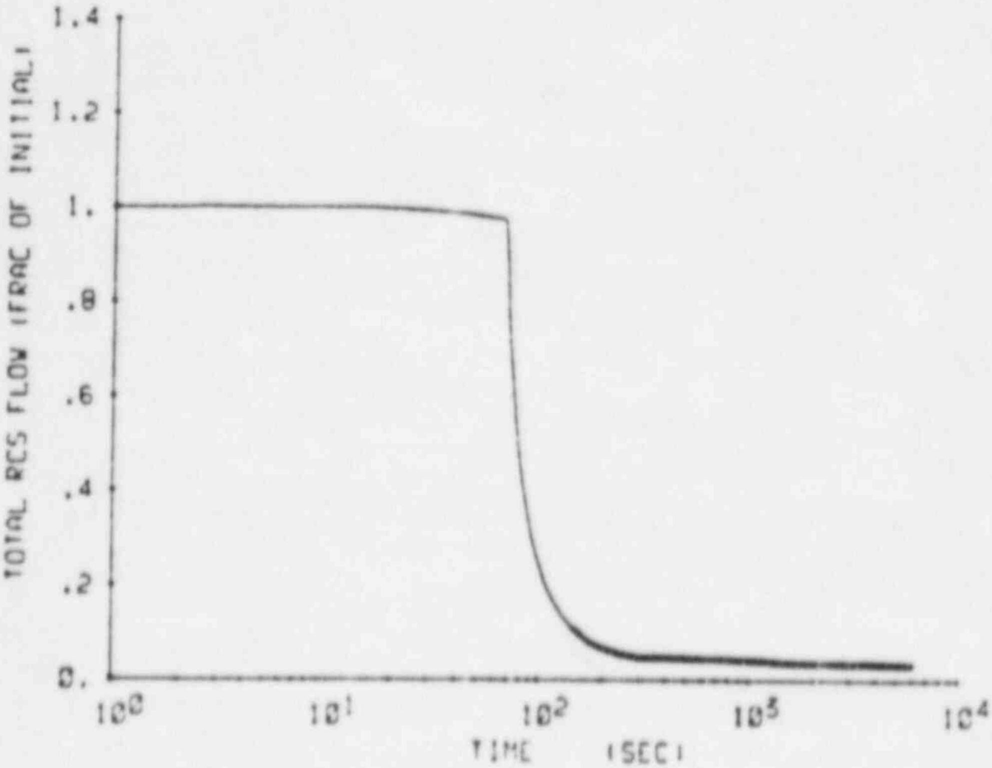

Figure 15.2-9B TOTAL REACTOR COOLANT SYSTEM FLOW TRANSIENT

LOSS OF NORMAL FEEDWATER

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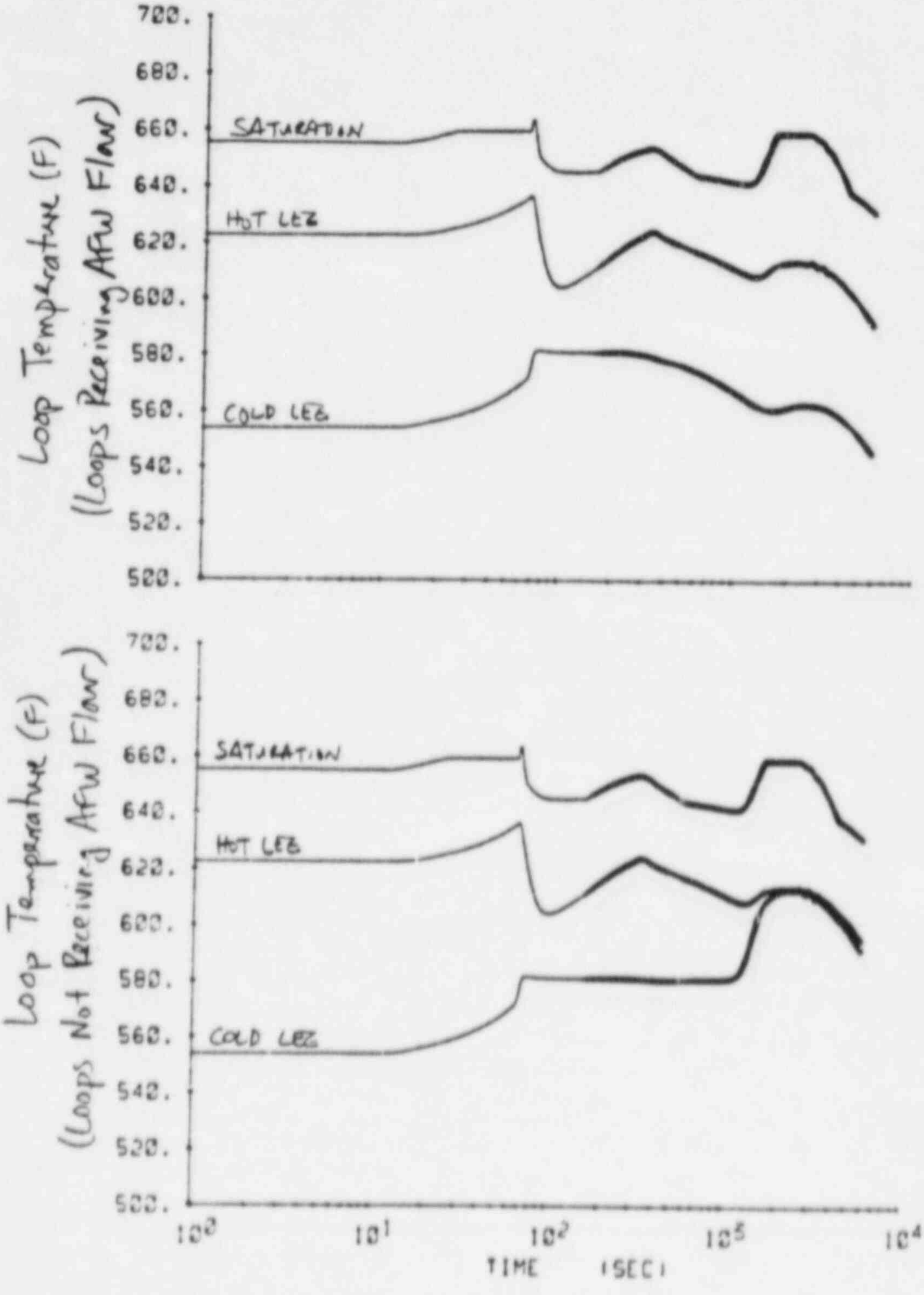

Figure 15.2-9C LOOP TEMPERATURE TRANSIENTS
LOSS OF NORMAL FEEDWATER

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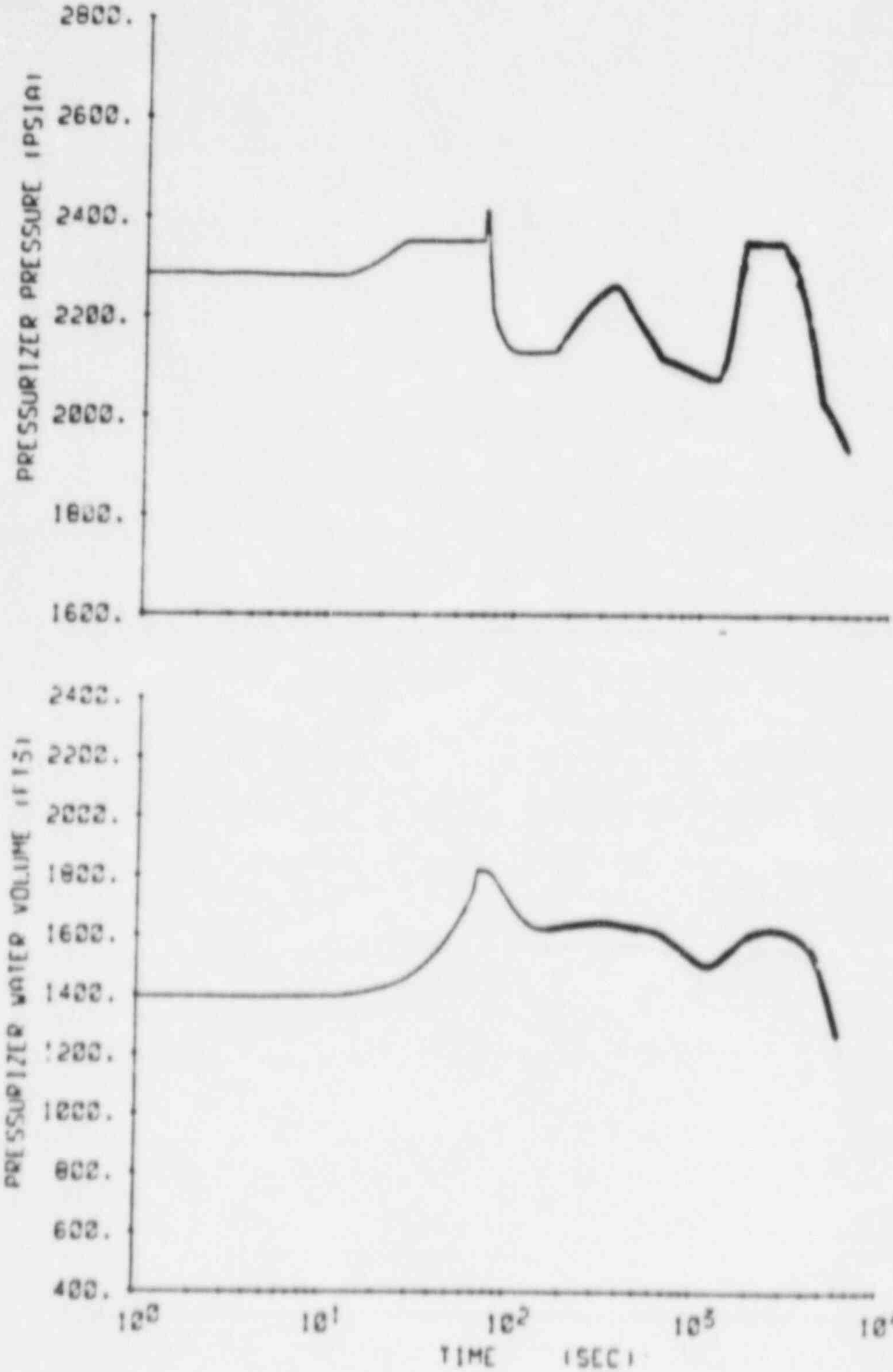

Figure 15.2-9D PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS
LOSS OF NORMAL FEEDWATER

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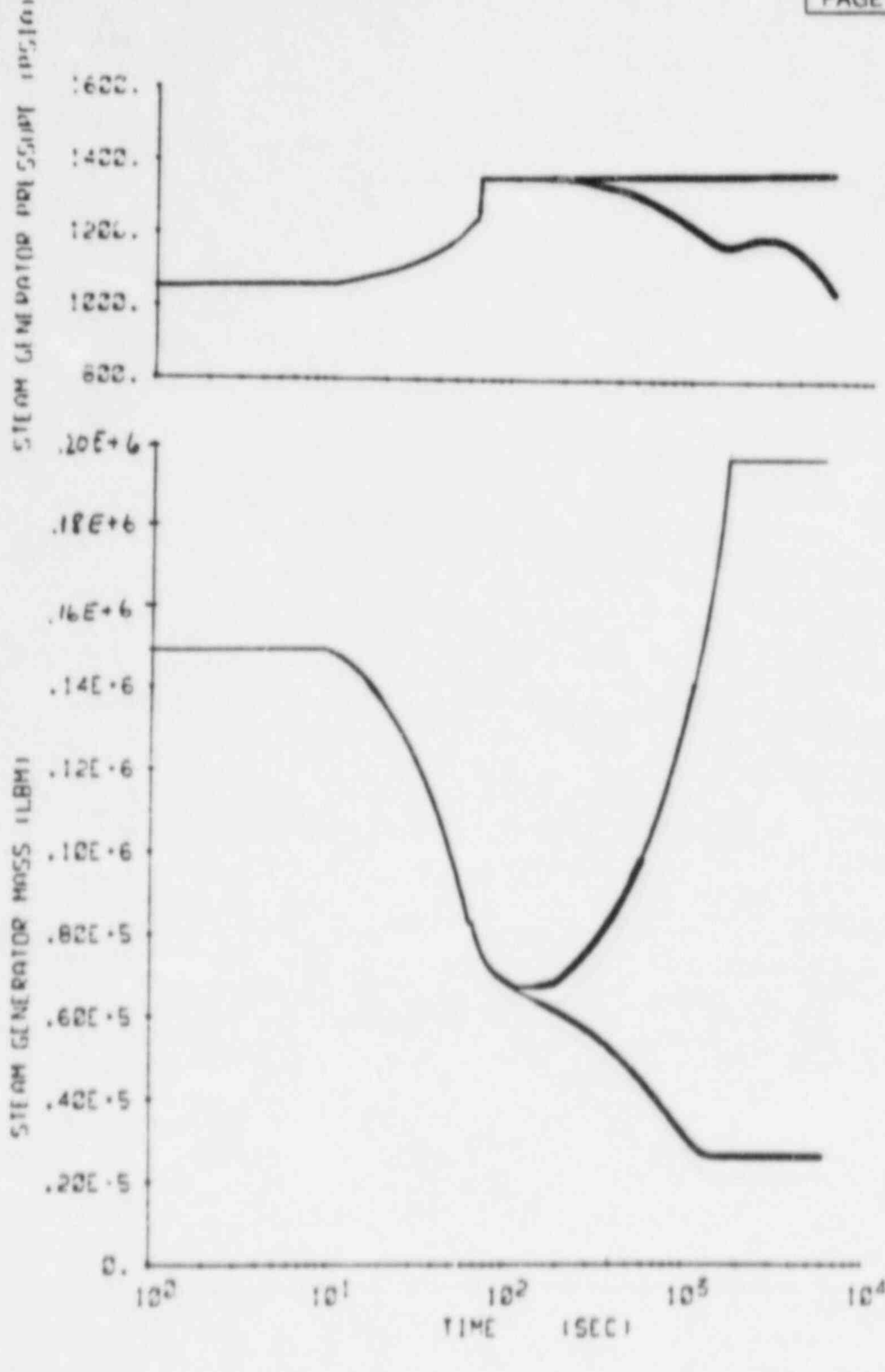

Figure 15.2-10 STEAM GENERATOR PRESSURE AND MASS TRANSIENTS

LOSS OF NORMAL FEEDWATER

SOUTH TEXAS PROJECT UNITS 1 & 2

15.2.8 Feedwater System Pipe Break

15.2.8.1 Identification of Causes and Accident Description. A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. (A break upstream of the feedwater line check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Section ~~15.2.7~~ 15.2.7.)

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown (by excessive energy discharge through the break) or a RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

A feedwater line rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

1. Feedwater flow to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip;

2. Fluid in the steam generator may be discharged through the break and would not be available for decay heat removal after trip;

3. The break may be large enough to prevent the addition of any main feedwater after trip.

The AFWS (Section 10.4.9) is provided to assure that adequate feedwater will be available such that:

1. No substantial overpressurization of the RCS shall occur; and

2. Sufficient liquid in the RCS shall be maintained in order to provide adequate decay heat removal.

A major feedwater line rupture is classified as an ANS Condition IV event. See Section 15.0.1 for a discussion of Condition IV events.

The severity of the feedwater line rupture transient depends on a number of system parameters including break size, initial reactor power, and credit taken for the functioning of various control and safety systems. A number of cases of feedwater line break have been analyzed. Based on these analyses, it has been shown that the most limiting feedwater line rupture is a double-ended rupture of the largest feedwater line. Analyses have been performed at full power with and without loss of offsite power. These cases are analyzed below.

The following provide protection for a main feedwater rupture:

1. A reactor trip on any of the following conditions:

   a. High pressurizer pressure,

   b. Overtemperature ΔT,

   c. Low-low steam generator water level in any steam generator,

   d. Safety injection signals from any of the following:

      1) 2/3 Low steam line pressure in any loop

      2) 2/3 High containment pressure (HI-1)

   (Refer to Chapter 7 for a description of the actuation ~~system~~ s systems.)

2. An AFW system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10.4.9).

15.2.8.2 Analysis of Effects and Consequences.

Method of Analysis

A detailed analysis using the LOFTRAN code (Reference 15.2-3) is performed in order to determine the plant transient following a feedwater line rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators and feedwater system and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature. Q32

The cases analyzed assume a double-ended rupture of the largest feedwater pipe at full power. Major assumptions made in the analyses are as follows: Q211.74

1. The plant is initially operating at 102 percent of the nominal NSSS design rating.

2. Initial reactor coolant average temperature is ~~4.0°F~~ 4.1°F above the nominal value, and the initial pressurizer pressure is ~~30~~ 34 psi above its nominal value.

3. No credit is taken for the pressurizer spray control system.

4. Initial pressurizer level is at the nominal programmed value +~~2~~ 5 percent (error); initial steam generator water level is at the nominal value +5 percent in the faulted steam generator and at the nominal value -5 percent in the intact steam generators.

5. No credit is taken for the high pressurizer pressure reactor trip.

6. Main feedwater to all steam generators is assumed to stop at the time the break occurs (all main feedwater spills out through the break).

7. The worst possible break area is assumed. This maximizes the blowdown discharge rate following the time of trip which maximizes the resultant heatup of the reactor coolant.

8. A conservative feedwater line break discharge quality is assumed prior to the time the reactor trip occurs, thereby maximizing the time the trip setpoint is reached. After the trip occurs, a saturated liquid discharge is assumed until all the water inventory is discharged from the affected steam generator. This minimizes the heat removal capability of the affected steam generator.

9. Reactor trip is assumed to be actuated when the low-low steam generator water level trip setpoint minus 10 percent of narrow range span in the affected steam generator is reached. 43

540 ~~[illegible]~~

10. The AFW is actuated by the low-low steam generator water level signal. The AFW is assumed to supply a total of [illegible] gal/min to one intact steam generator. A 60-second delay was assumed following the low-low water level signal to allow time for startup of the standby diesel generators and the auxiliary feedwater pumps. An additional ~~[illegible]~~ 83 seconds was assumed before the feedwater ~~lines were~~ line was purged and the relatively cold (~~[illegible]~~ 120°F) auxiliary feedwater entered the intact steam generator. 32 Q211.74 43

11. No credit is taken for heat energy deposited in RCS metal during the RCS heatup.

12. No credit is taken for charging or letdown.

13. Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

14. Conservative core residual heat generation is assumed based upon long-term operation at the initial power level preceding the trip. INSERT I

15. No credit is taken for the following potential protection logic signals to mitigate the consequences of the accident:

    a. High pressurizer pressure

    b. Overtemperature ΔT

    c. High pressurizer level

    d. High Containment pressure

Receipt of a low-low steam generator water level signal in at least one steam generator starts the motor-driven and turbine-driven auxiliary feedwater pumps, which in turn initiate auxiliary feedwater flow to the steam generators. Similarly, receipt of a low steamline pressure signal in at least one steamline initiates a safety injection signal which closes all main steam isolation valves, ~~and initiates flow of borated water into the RCS. The amount of safety injection flow is a function of RCS pressure.~~ 43

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14. Core residual heat generation is based on the 1979 version of ANS 5.1 (Reference 15.2-6). ANSI/ANS-5.1-1979 is a ~~more realistic~~ conservative representation of the decay energy release rates. ~~The previous model used, per [illegible] Appendix K of 10CFR50, assumes 1.2 times the values for infinite operating time in the ANS Standard. ANSI/ANS-5.1-1979 is based on current research which more accurately quantifies decay heat and its uncertainty.~~ ~~[illegible]~~

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Emergency operating procedures following a main feedwater line rupture require the operator to isolate feedwater flow spilling from the ruptured feedwater line (B)

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-13.

Plant characteristics and initial conditions are further discussed in Section 15.0.3.

No reactor control systems, with the exception of the pressurizer PORVs, are assumed to function. (A) The RTS is required to function following a feedwater line rupture as analyzed here. No single active failure will prevent operation of this system.

Only one auxiliary feedwater pump is assumed to function following receipt of an initiating signal. Following initiation, the auxiliary pump is assumed to deliver ~~[illegible]~~ 540 gal/min of auxiliary feedwater to one intact steam generator. 32

Results

Calculated plant parameters following a major feedwater line rupture are shown on Figures 15.2-11 through 15.2-24. Results for the case with offsite power available are presented on Figures 15.2-11 through 15.2-17. Results for the case where offsite power is lost are presented on Figures 15.2-18 through 15.2-24. The calculated sequence of events for both cases analyzed is listed in Table 15.2-1.

The system response following the feedwater line rupture is similar for both cases analyzed. Results presented on Figures 15.2-12 and 15.2-15 (with offsite power available) and Figures 15.2-19 and 15.2-22 (without offsite power) show that pressures in the RCS and main steam system remain below 110 percent of the respective design pressures. Pressurizer pressure increases until reactor trip on low-low steam generator water level. Pressure then decreases, due to the loss of heat input. Coolant expansion occurs due to reduced heat transfer capability in the steam generators; the pressurizer power-operated relief valves open to maintain RCS pressure at an acceptable value. 43

DNBR remains above 1.30 at all times during the transients, as shown on Figures 15.2-17 and 15.2-24; thus, the DNB design basis as described in Section 4.4 is met. 18

The reactor core remains covered with water throughout the transient, as water relief due to thermal expansion is limited by the heat removal capability of the AFW system.

The major difference between the two cases analyzed can be seen in the plots of hot and cold leg temperatures, Figures 15.2-13 and 15.2-14 (with offsite power available) and Figures 15.2-20 and 15.2-21 (without offsite power). It is apparent that for the duration of the ~~initial~~ transient ~~(300 seconds)~~, the case without offsite power results in higher temperatures in the hot leg. ~~For longer times, however, the case with offsite power results in a slightly more severe rise in temperature until the coolant pumps are turned off.~~

(A)

The operation of the PORVs serves to worsen the transient via minimizing the saturation temperature and therefore minimizing the margin to subcooling. It also allows a greater discharge of mass from the primary system, thus maximizing the liquid volume in the pressurizer.

(B)

and to control the RCS temperature which also prevents the pressurizer from filling.

15.2.8.3 <u>Radiological Consequences</u>. The feedwater line break with the most significant consequences would be one that occurred inside the Containment between a steam generator and the feedwater check valve. In this case, the contents of the steam generator would be released to the Containment. Since no fuel failures are postulated, the radioactivity released is less than that for the steam line break. Furthermore, automatic isolation of the Containment would further reduce any radiological consequences from this postulated event. 43 Q211 .41

15.2.8.4 <u>Conclusions</u>. Results of the analyses show that for the postulated feedwater line rupture, the assumed AFW capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core. Radiological doses from the postulated feedwater line rupture are less than those previously presented for the postulated steam line break. All applicable acceptance criteria are thus met. The radiological consequences of this event are not limiting.

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~~In addition, when the Effect of the uncertainty of ANSI/ANS-5.1-1979 (Reference 15.2-6) decay heat release rates are assumed, calculations show the pressurizer does not become water-solid as a result of a feedwater system line break accident.~~

delete!

REFERENCES

SECTION 15.2:

15.2-1 Mangan, M. A., "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, October, 1971.

15.2-2 "Westinghouse Anticipated Transients Without Trip Analysis," WCAP-8330, August, 1974.

15.2-3 Burnett, T. W. T., et.al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary Class 2), WCAP-7907-A (Proprietary Class 3), April 1984. 43

15.2-4 Balwin, M. S., Merrian, M. M., Schenkel, H. S. and VandeWalle, D. J., "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWR's," WCAP-8424, Revision 1, June, 1975.

15.2-5 Hunin, C., "FACTRAN a Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972.

15.2-6 "American National Standard for Decay Heat Power in Light Water Reactors," ANSI/ANS-5.1-1979, August, 1979.

TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
Feedwater System Pipe Break			
1. With Offsite Power Available	Main feedline rupture occurs	10	
	Low-low steam generator water level reactor trip setpoint reached in affected steam generator	30	43
	Rods begin to drop	32	
	Low steamline pressure setpoint reached in affected steam generator	566	32 Q211.7
	All main steam isolation valves close	574	43
	Pressurizer power-operated relief valve setpoint reached	1250	43
	Steam generator safety valve setpoint reached in intact steam generator receiving auxiliary feedwater	1300	
	~~Pressurizer water relief begins~~		
	Peak water volume in pressurizer occurs	36	
	One auxiliary feedwater pump starts and supplies 1 intact steam generator	90	

TABLE 15.2-1 (Cont'd)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

Accident	Event	Time (sec)	
2. Without Offsite Power	Main feedline rupture occurs	10	
	Low-low steam generator water level reactor trip setpoint reached in affected steam generator	30	43
	Rods begin to drop (2) ~~power lost to the reactor coolant pumps (2)~~	32	
	Power lost to the reactor coolant pumps	34	
	One auxiliary feedwater pump starts and ~~delivers~~ supplies 1 intact steam generator	90	
	Low steamline pressure setpoint reached in affected steam generator	553	32
	All main steam isolation valves close	561	43
	Pressurizer power-operated relief valve setpoint reached	1268	43
	Steam generator safety valve setpoint reached in intact steam generator receiving auxiliary feedwater	2616	
	Pressurizer water relief begins (3)	3352	

Notes:

(1) DNBR does not decrease below its initial value

(2) Loss of offsite power assumed to occur at reactor trip 43
a two second delay is assumed prior to RCP trip

(3) operator action precludes the filling of the pressurizer

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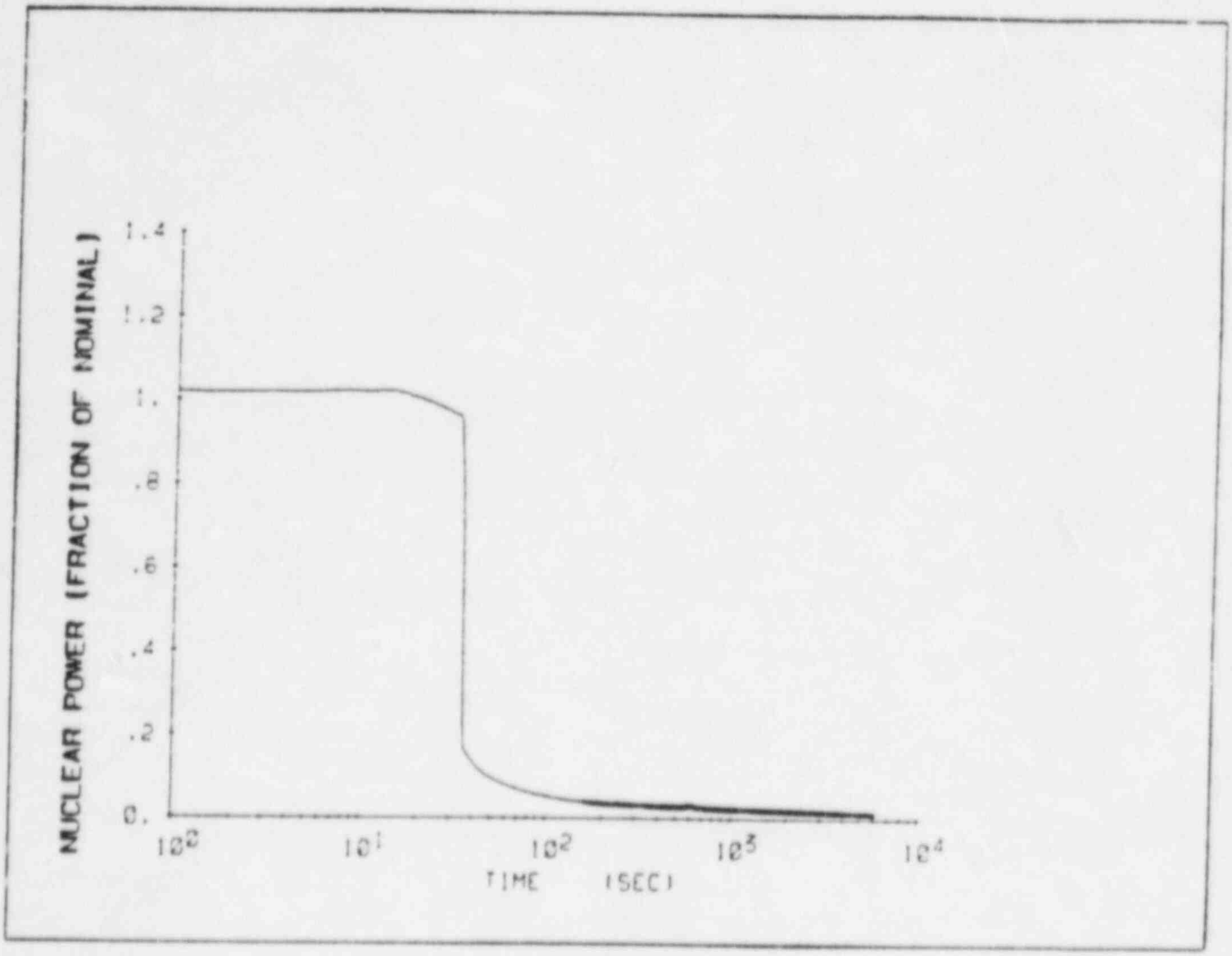

Figure 15.2-11. Nuclear Power Transient, Total Core Reactivity Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available (1 of 3)

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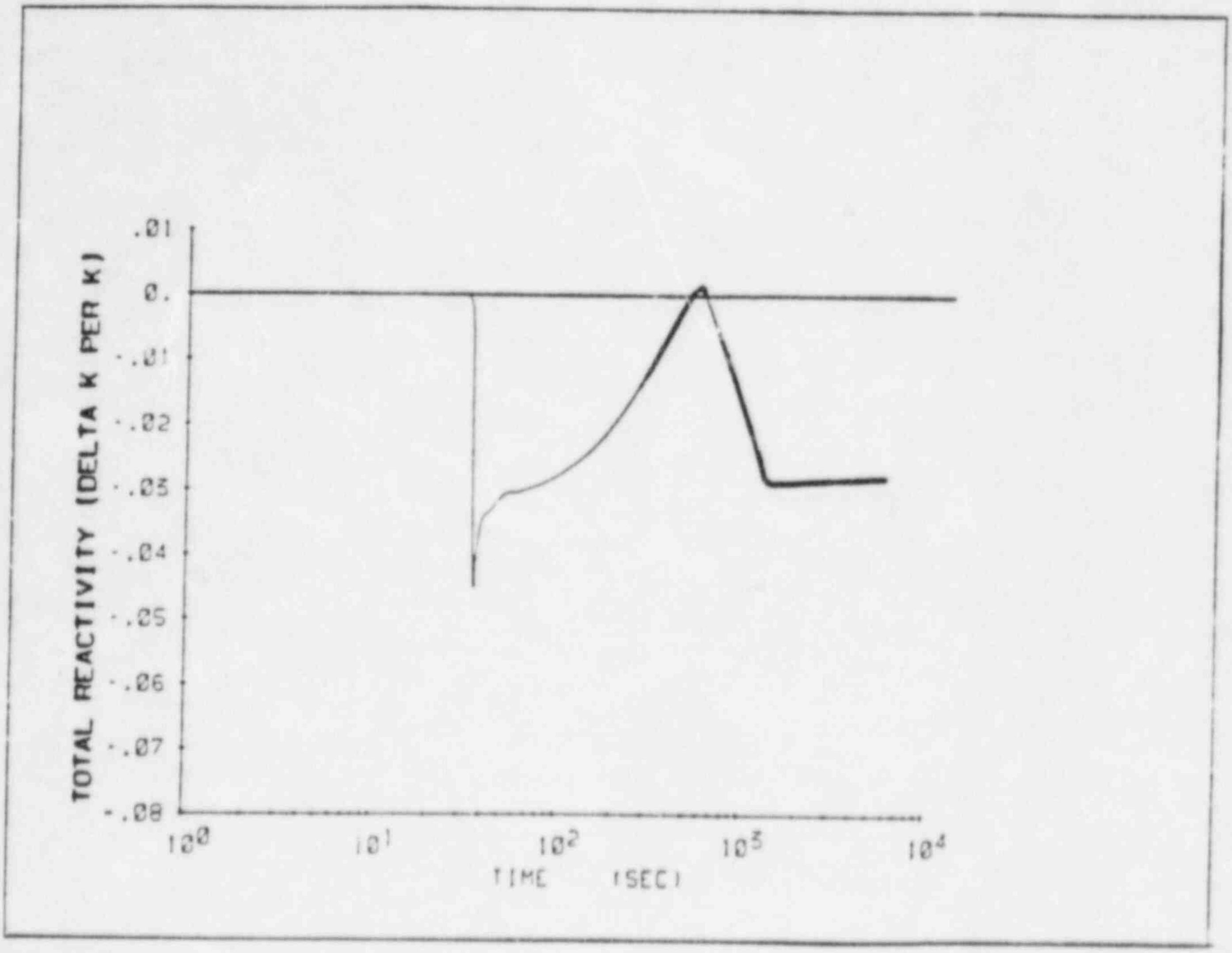

Figure 15.2-11. Nuclear Power Transient, Total Core Reactivity Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available (2 of 3)

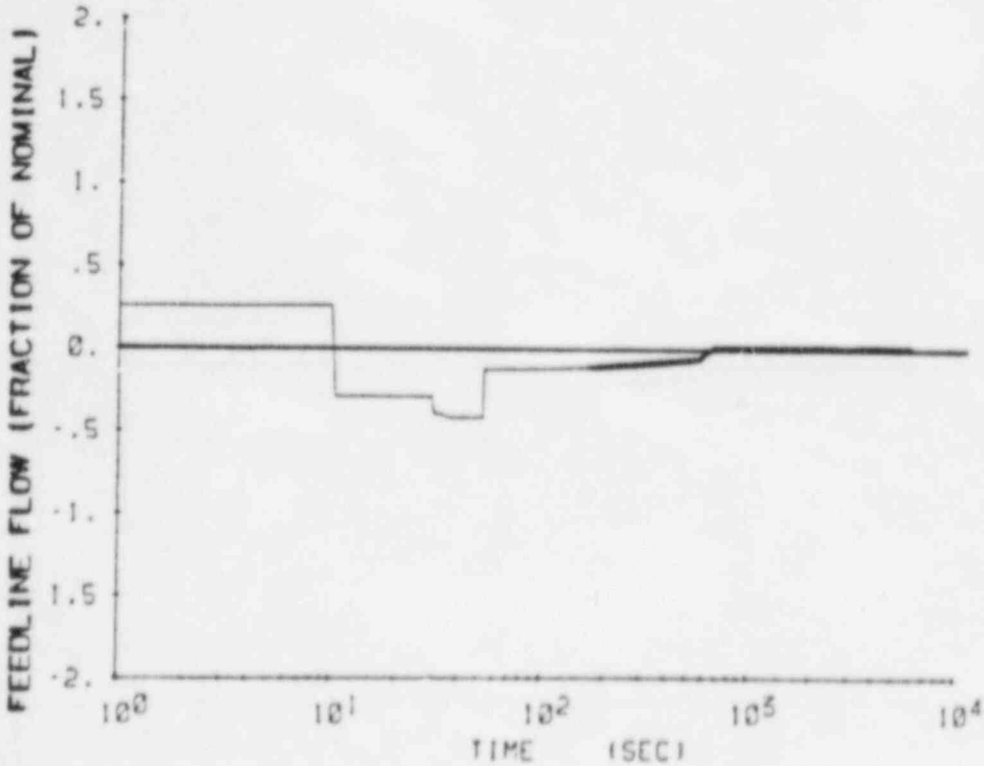

Figure 15.2-11. Nuclear Power Transient, Total Core Reactivity Transient, and Feedline Break Flow Transient for Main Feedline Rupture With Offsite Power Available (3 of 3)

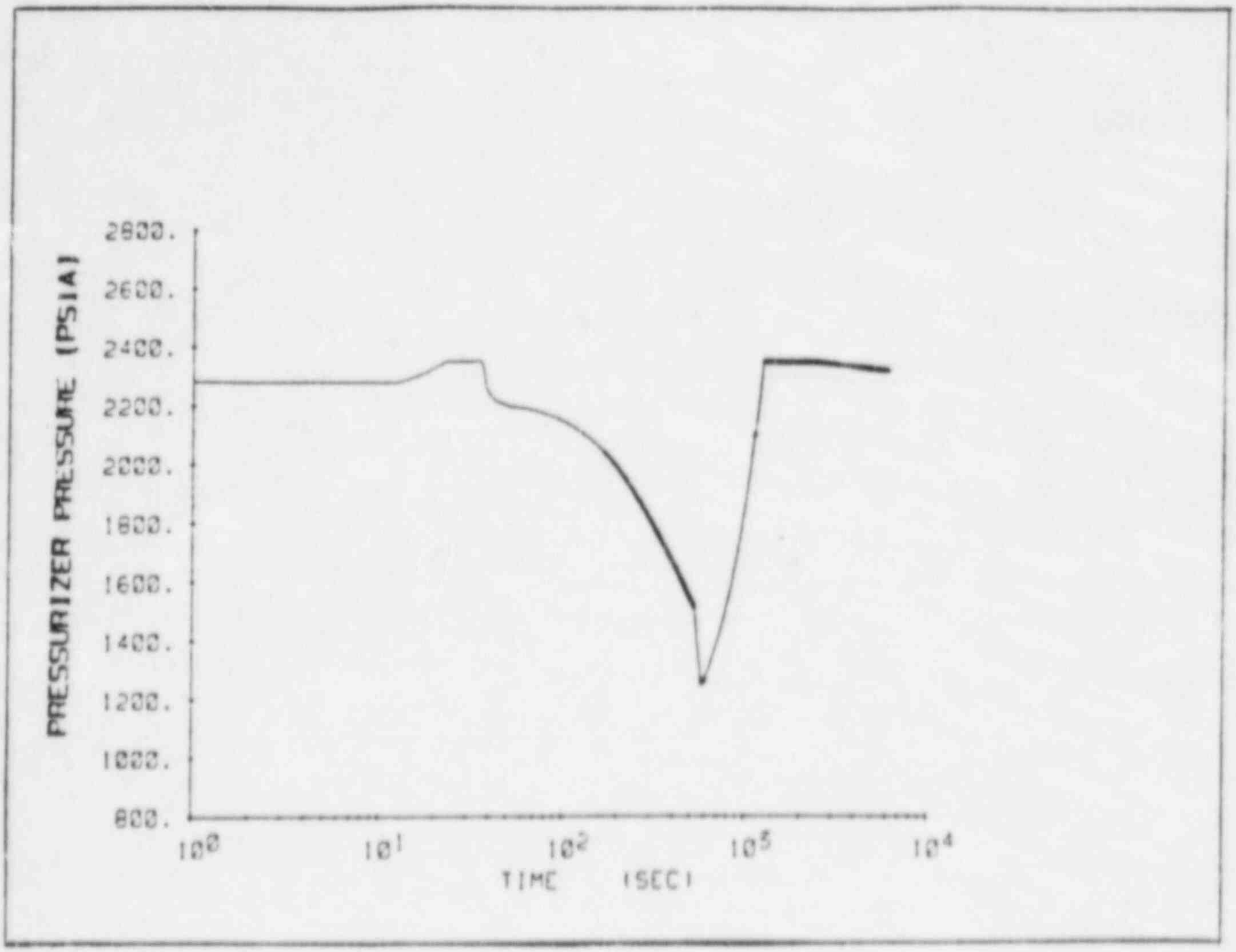

Figure 15.2-12. Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available (1 of 3)

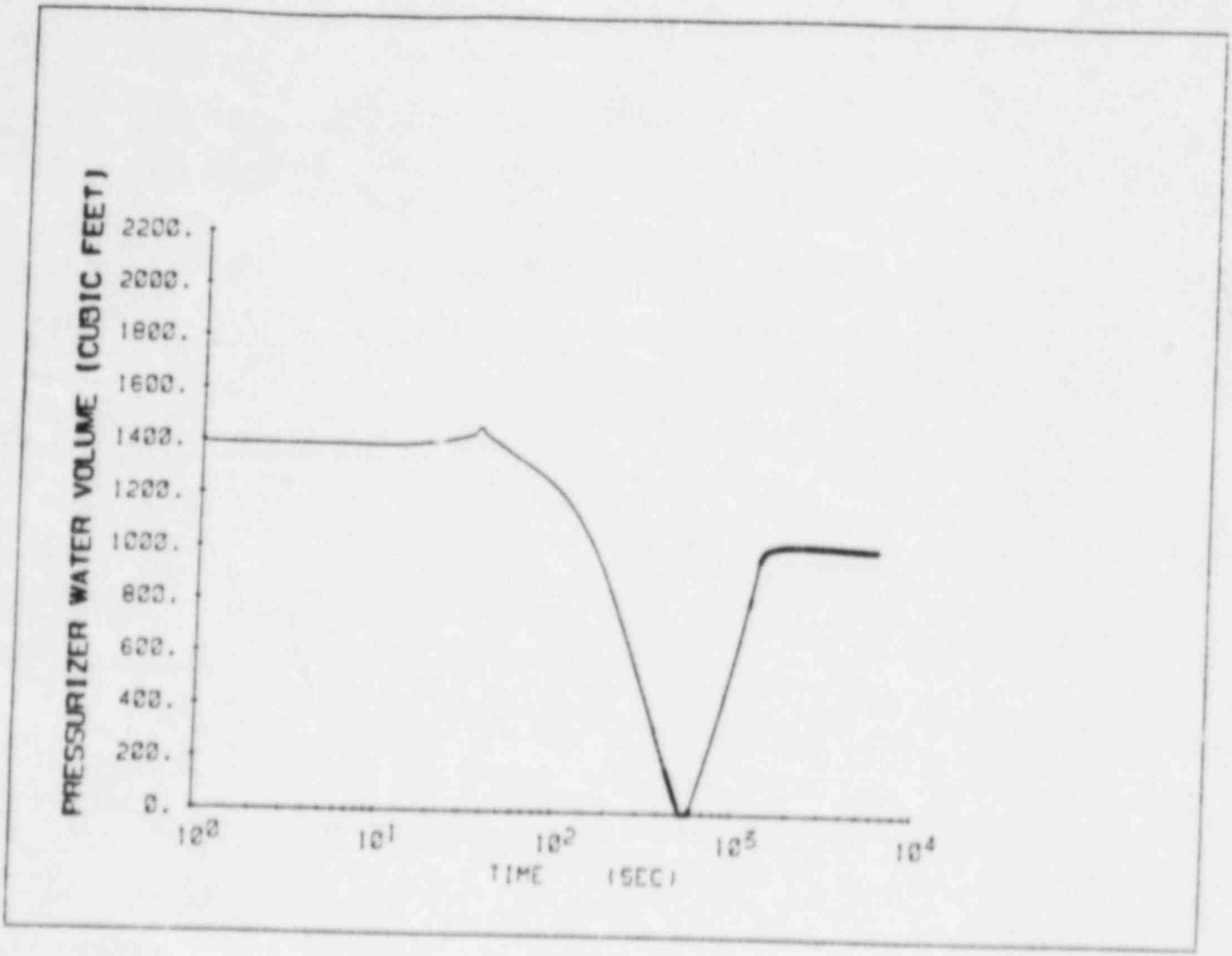

Figure 15.2-12. Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available (2 of 3)

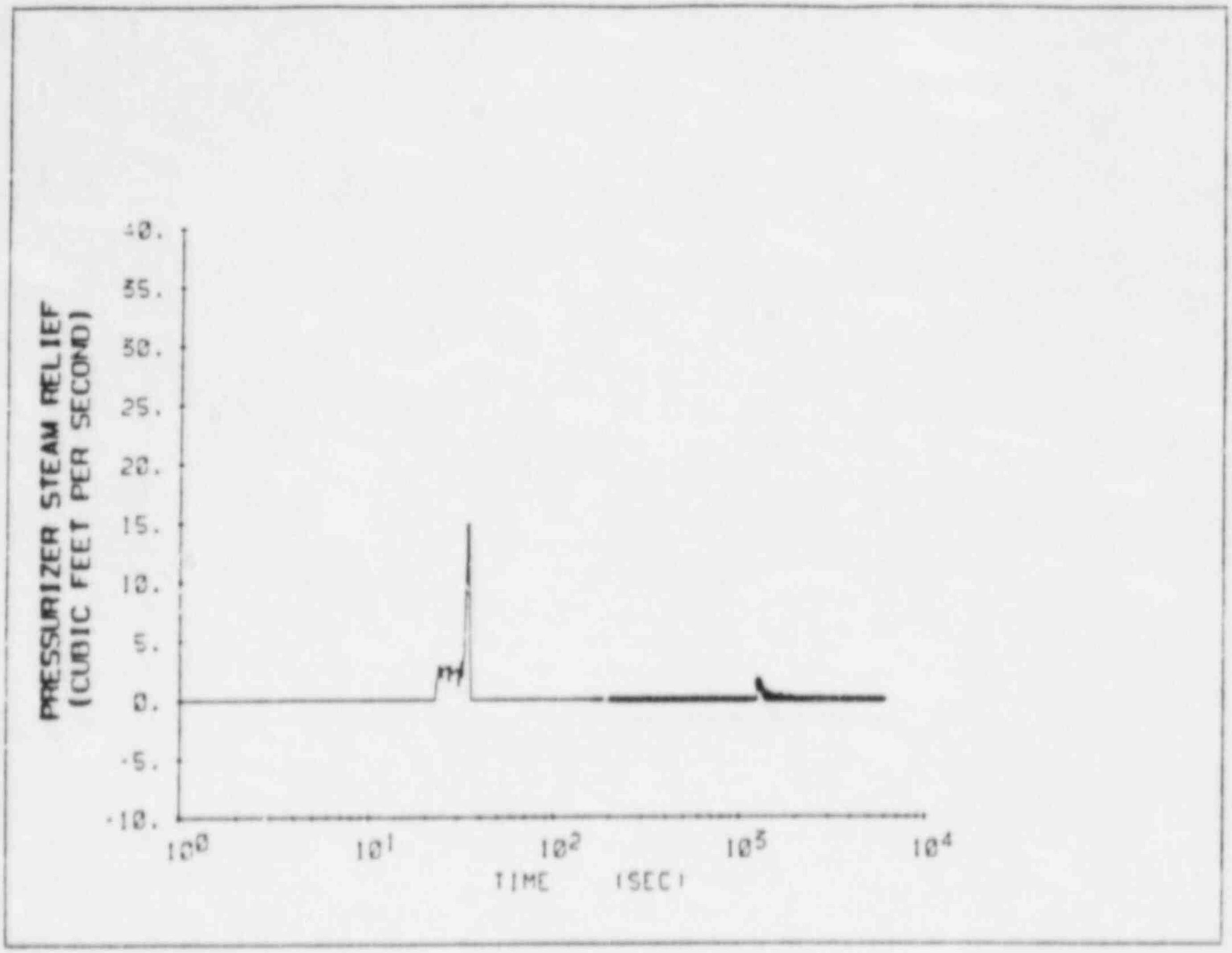

Figure 15.2-12. Pressurizer Pressure, Water Volume, and Relief Transients for Main Feedline Rupture With Offsite Power Available (3 of 3)

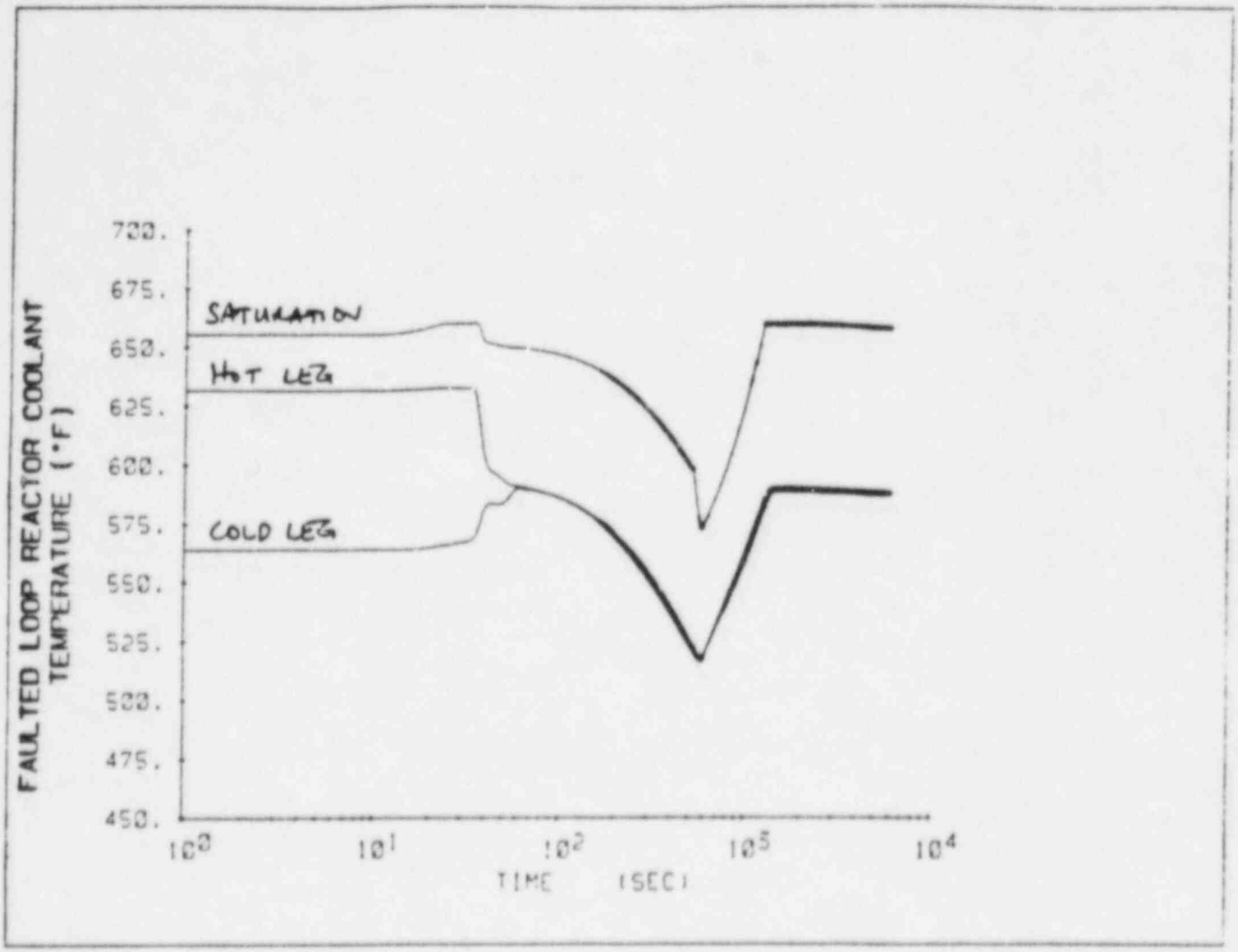

Figure 15.2-13. Reactor Coolant Temperature Transient for the Faulted Loop for Main Feedline Rupture With Offsite Power Available (1 of 1)

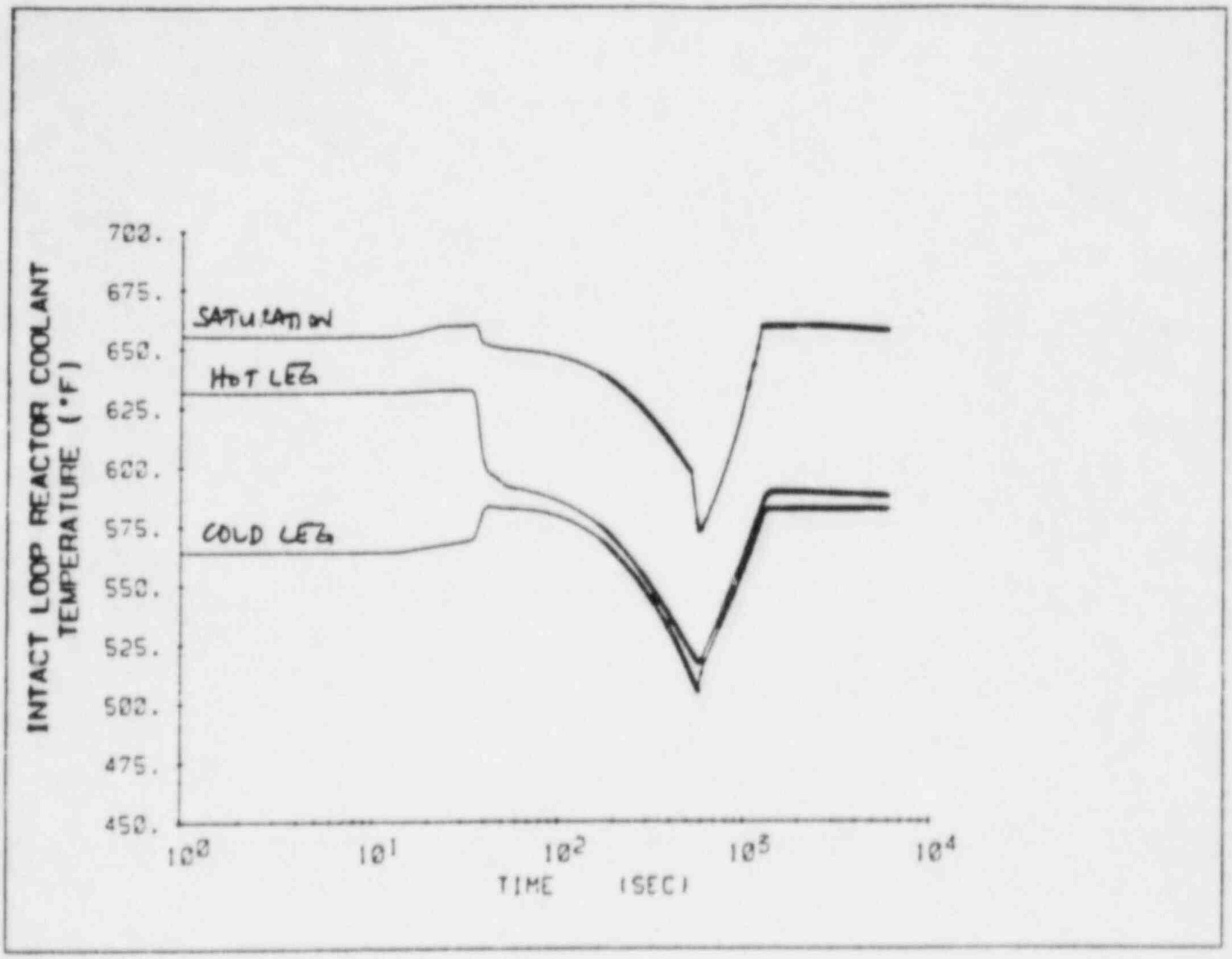

Figure 15.2-14. Reactor Coolant Temperature Transients for an Intact Loop for Main Feedline Rupture With Offsite Power Available (1 of 1)

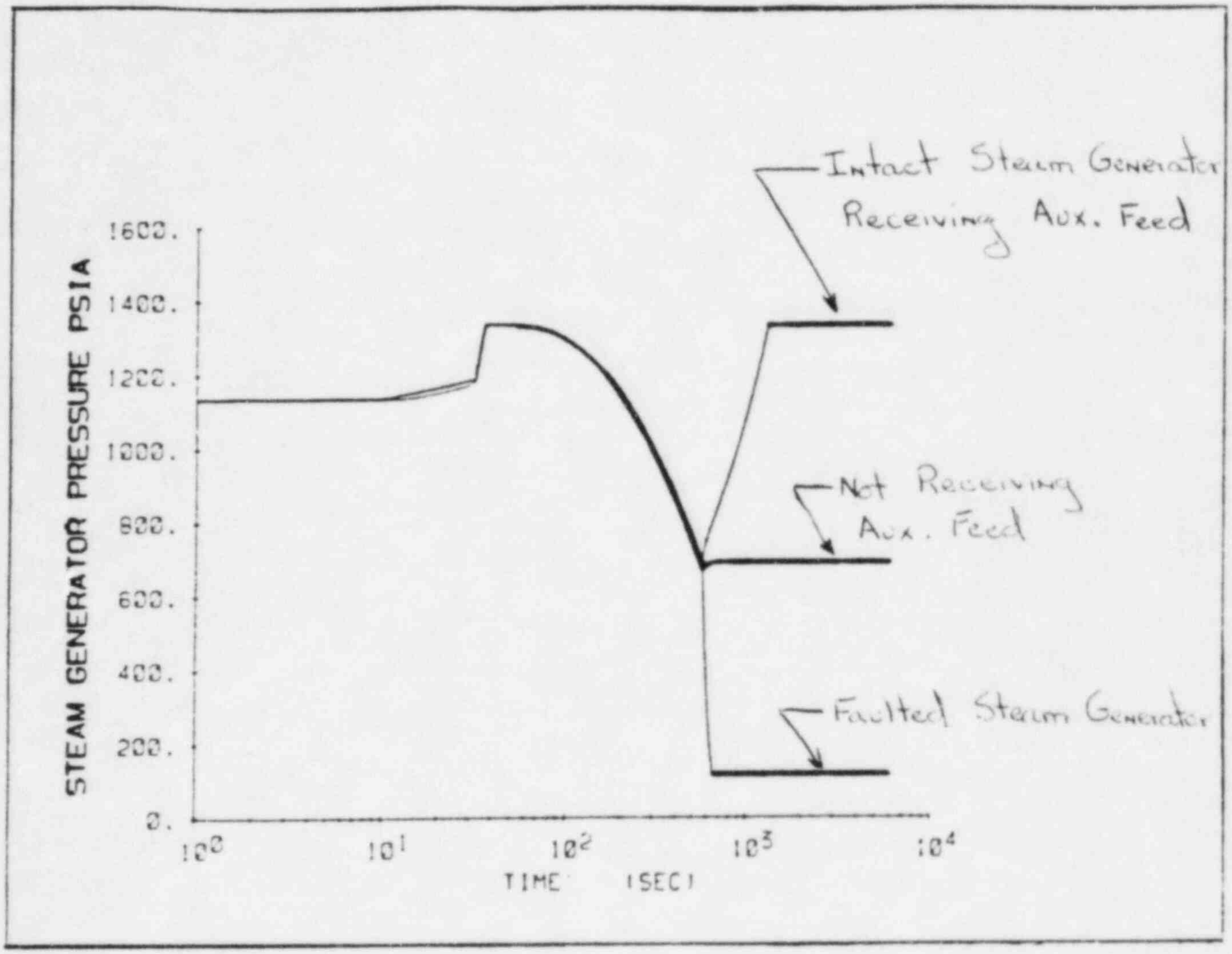

Figure 15.2-15. Steam Generator Pressure Transients for Main Feedline Rupture With Offsite Power Available (1 of 1)

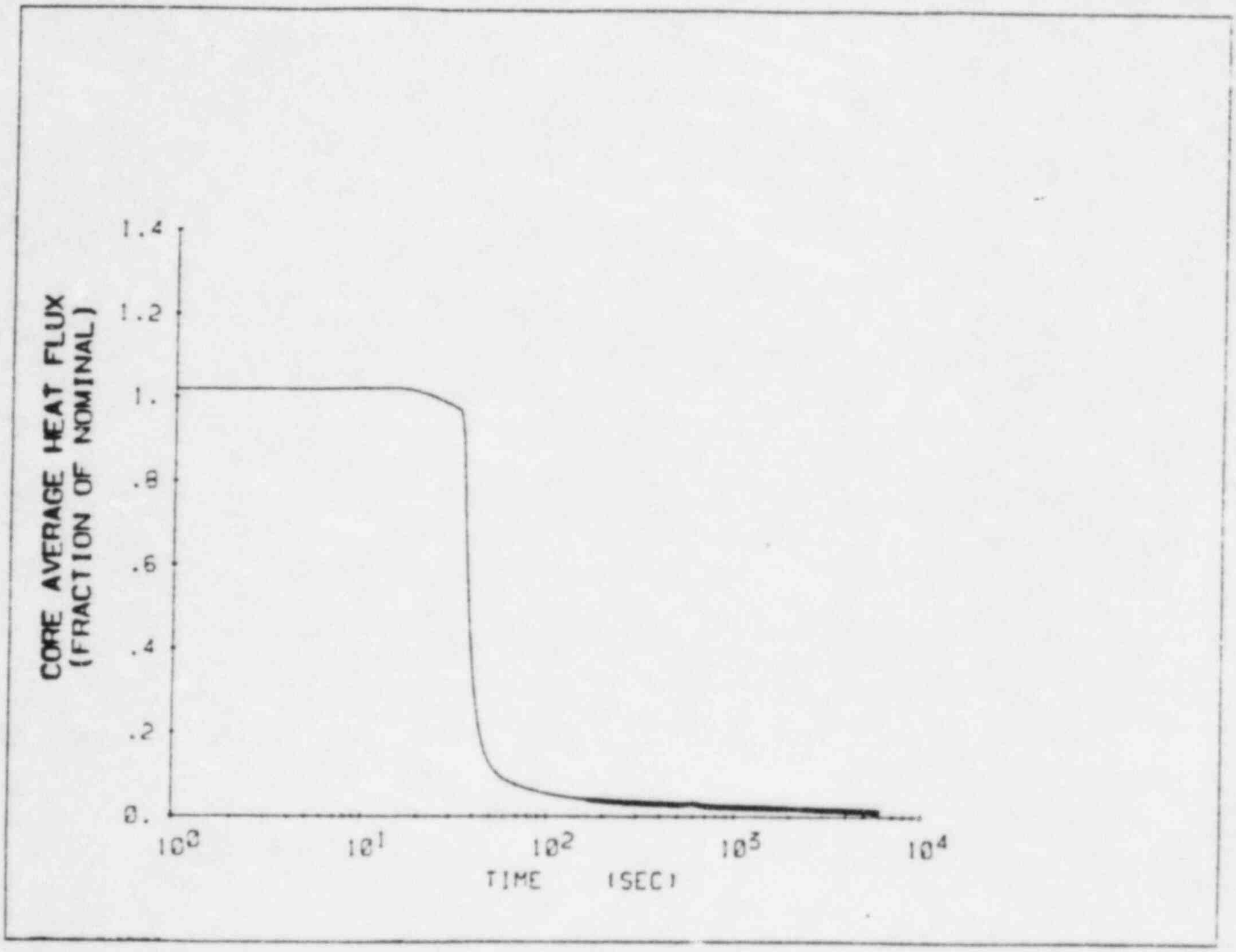

Figure 15.2-16. Core Average Heat Flux Transient for Main Feedline Rupture With Offsite Power Available (1 of 1)

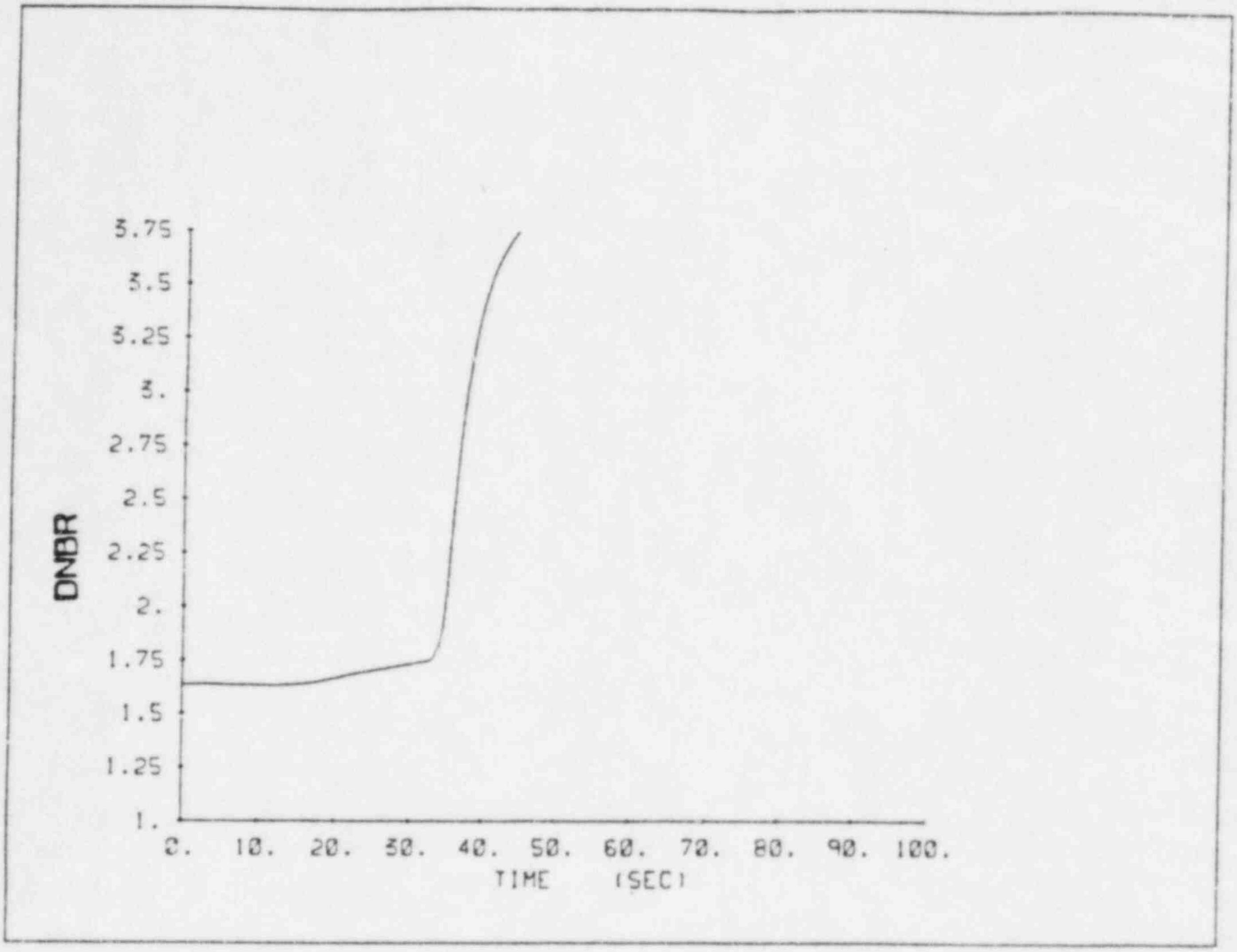

Figure 15.2-17. DNBR Transient for Main Feedline Rupture With Offsite Power Available (1 of 1)

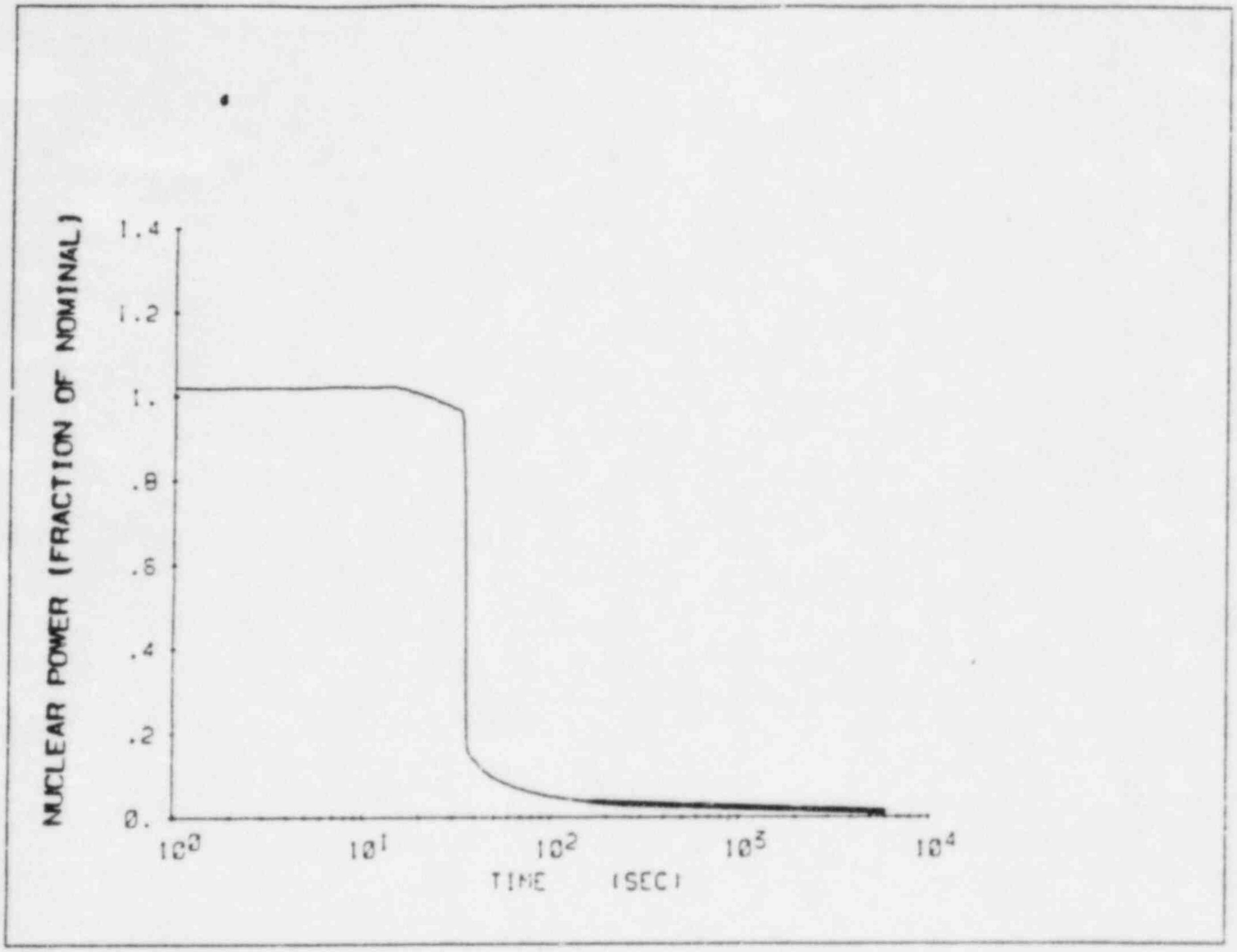

Figure 15.2-18. Nuclear Power Transient, Total Core Reactivity Transient and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power (1 of 3)

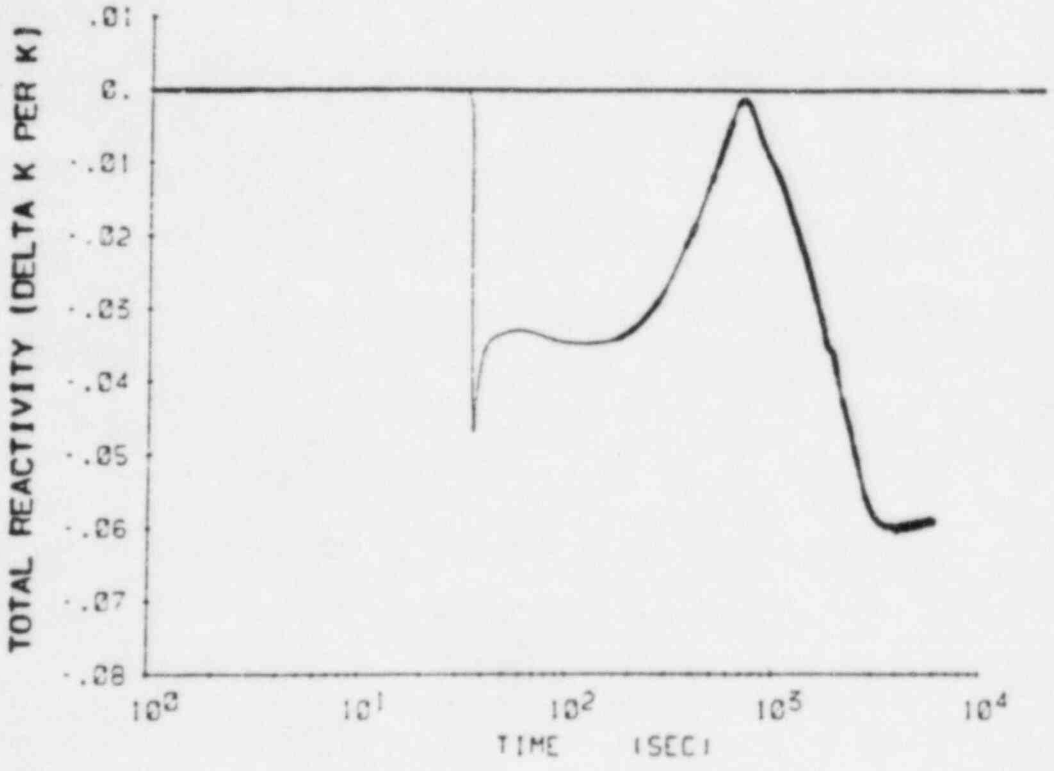

Figure 15.2-18. Nuclear Power Transient, Total Core Reactivity Transient and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power (2 of 3)

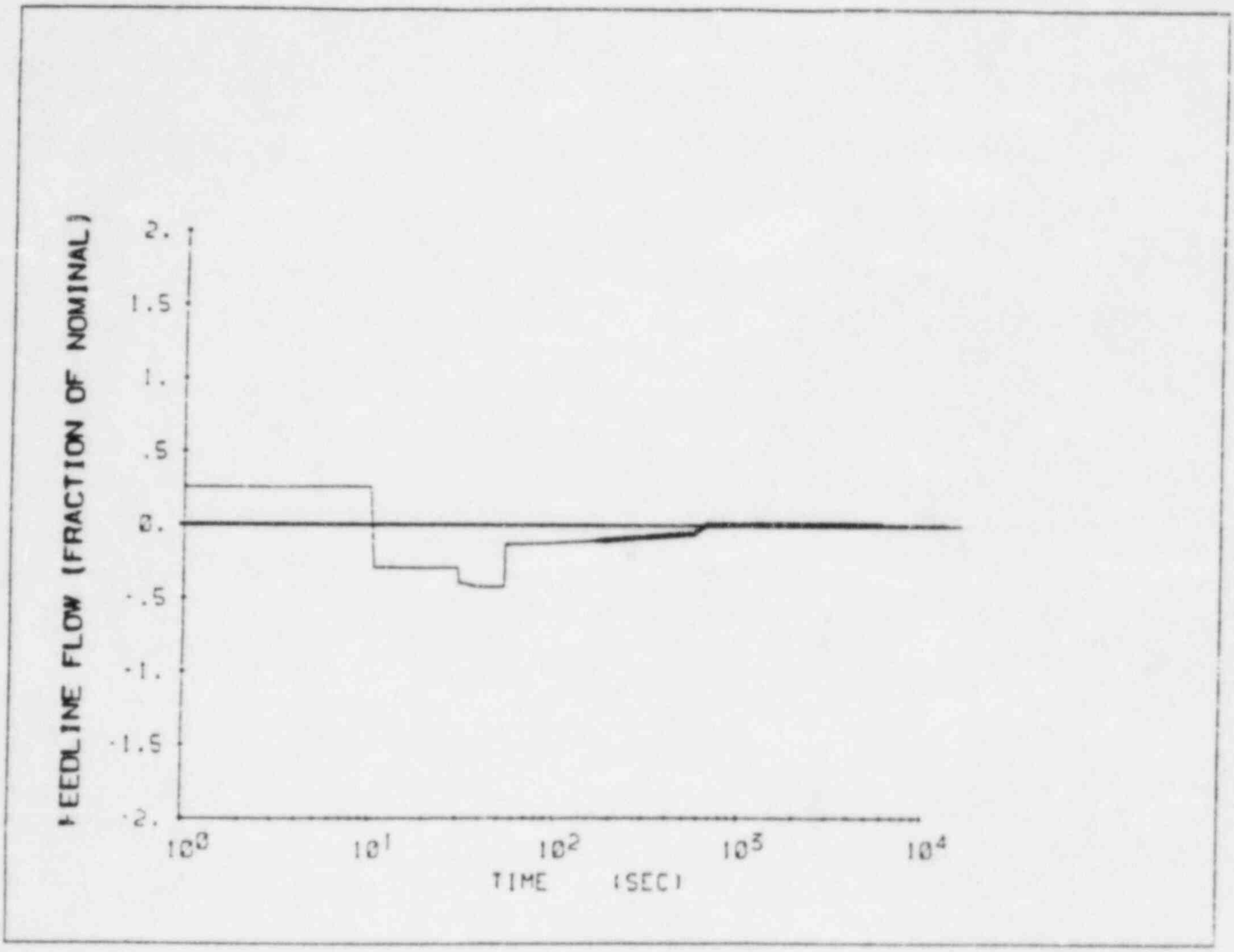

Figure 15.2-18. Nuclear Power Transient, Total Core Reactivity Transient and Feedline Break Flow Transient for Main Feedline Rupture Without Offsite Power (3 of 3)

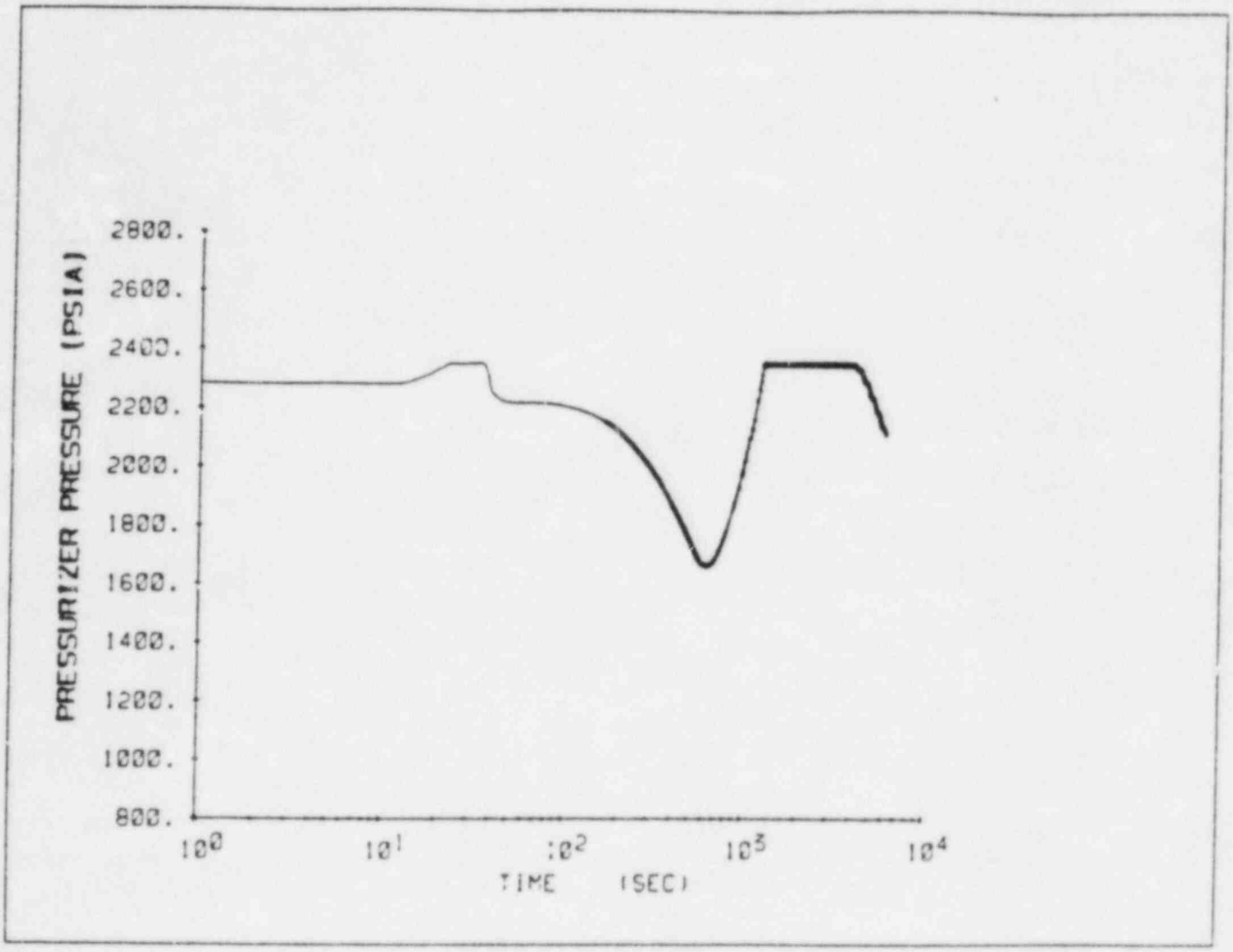

Figure 15.2-19. Pressurizer Pressure, Water Volume and Relief Rate for Main Feedline Rupture Without Offsite Power (1 of 3)

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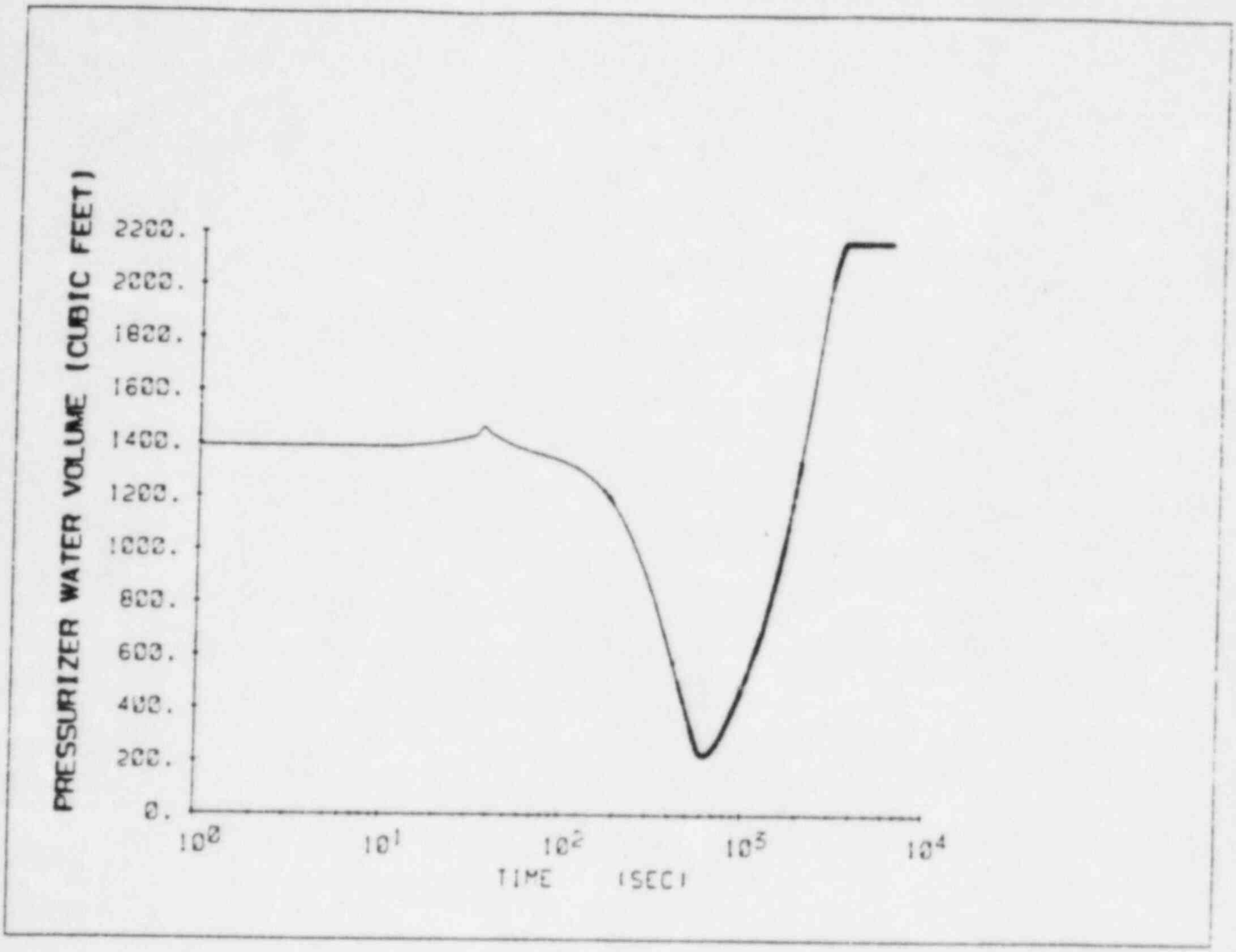

Figure 15.2-19. Pressurizer Pressure, Water Volume and Relief Rate for Main Feedline Rupture Without Offsite Power (2 of 3)

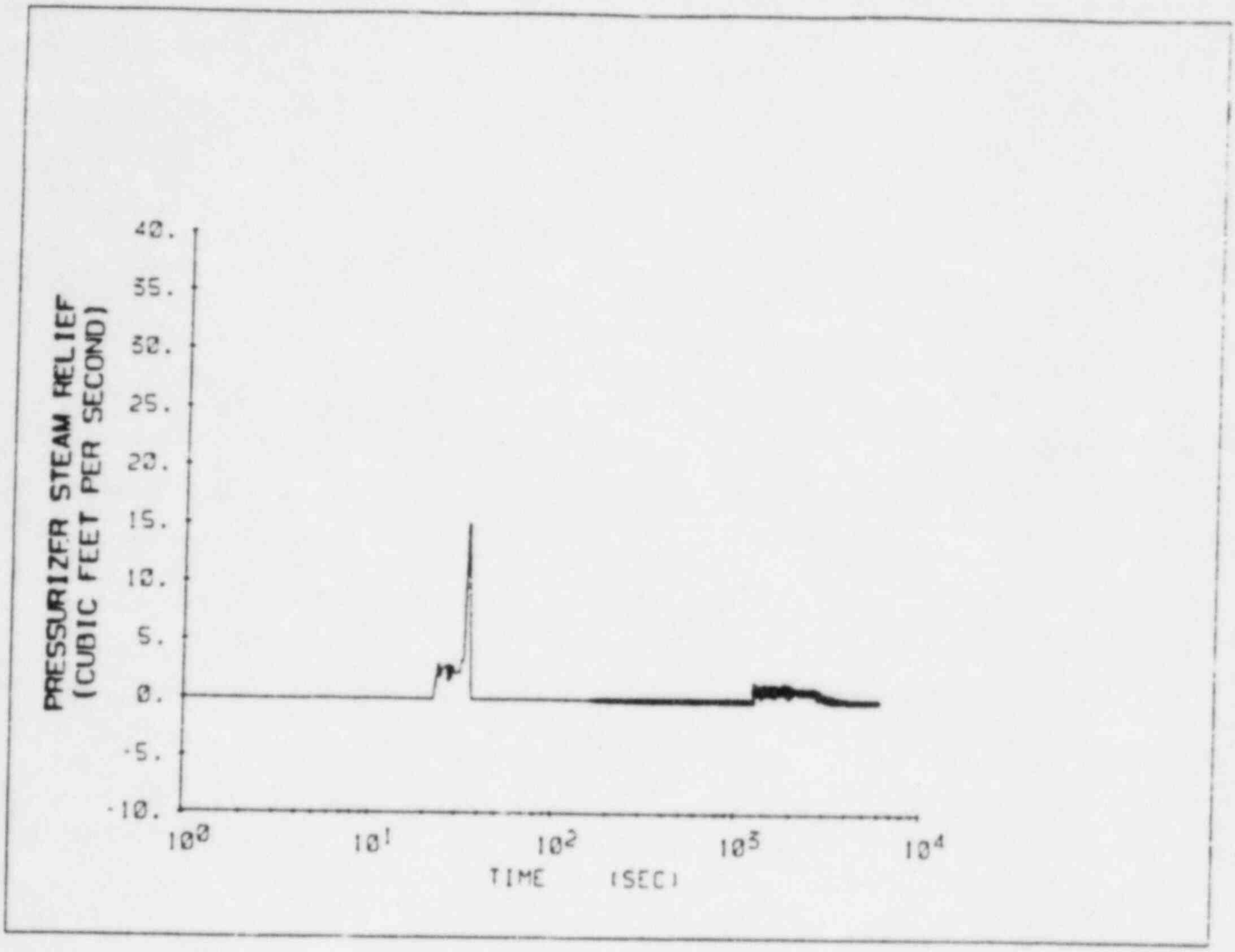

Figure 15.2-19. Pressurizer Pressure, Water Volume and Relief Rate for Main Feedline Rupture Without Offsite Power (3 of 3)

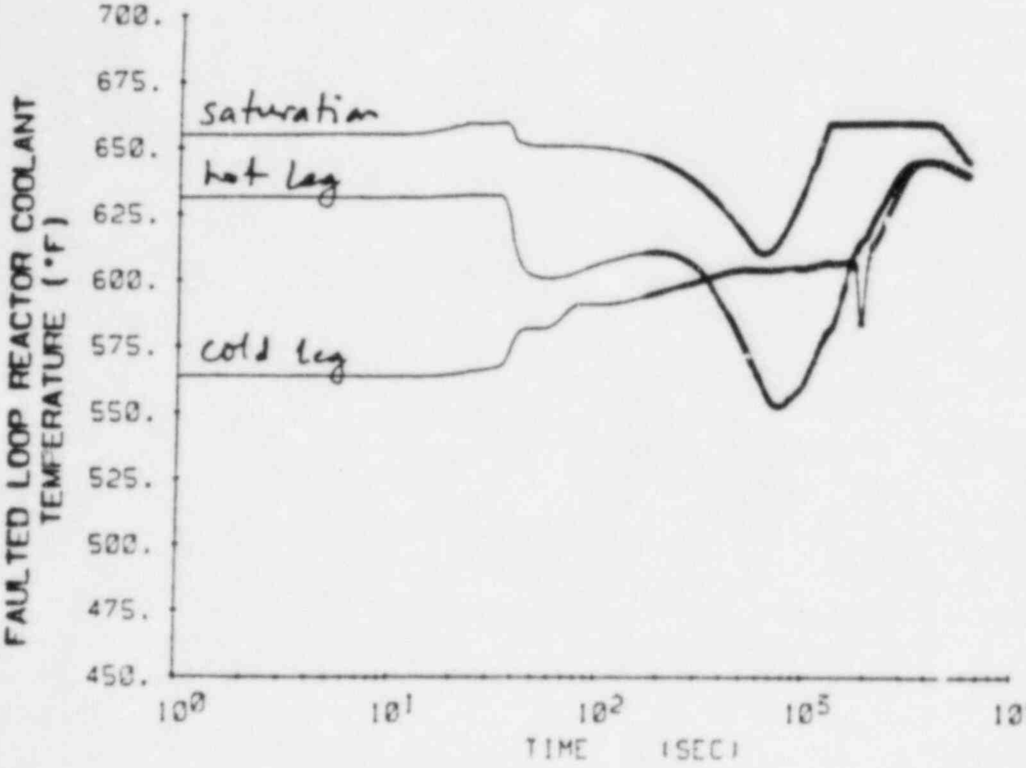

Figure 15.2-20. Reactor Coolant Temperature Transients for the Faulted Loop for Main Feedline Rupture Without Offsite Power (1 of 1)

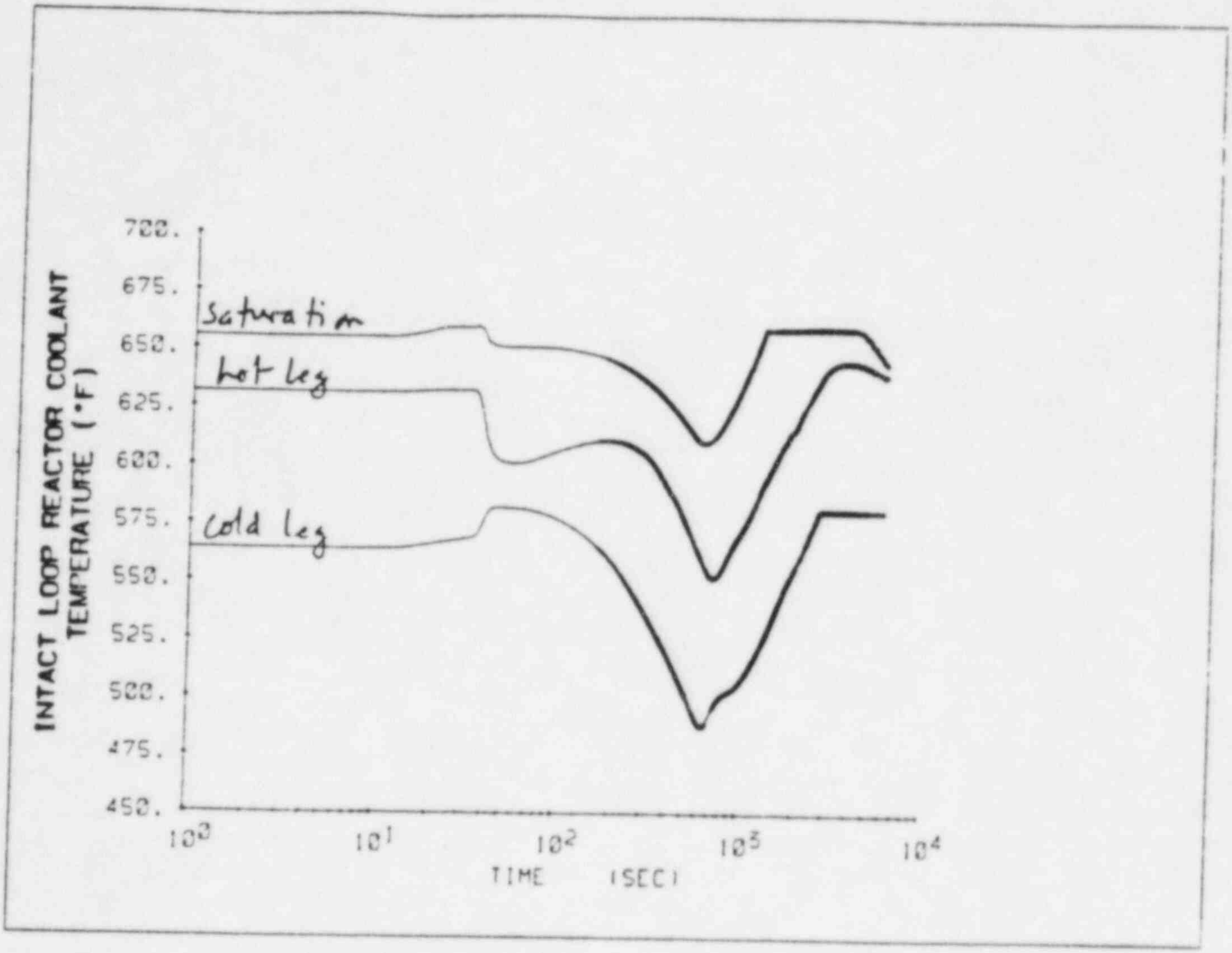

Figure 15.2-21. Reactor Coolant Temperture Transients for an Intact Loop for Main Feedline Rupture Without Offsite Power (1 of 1)

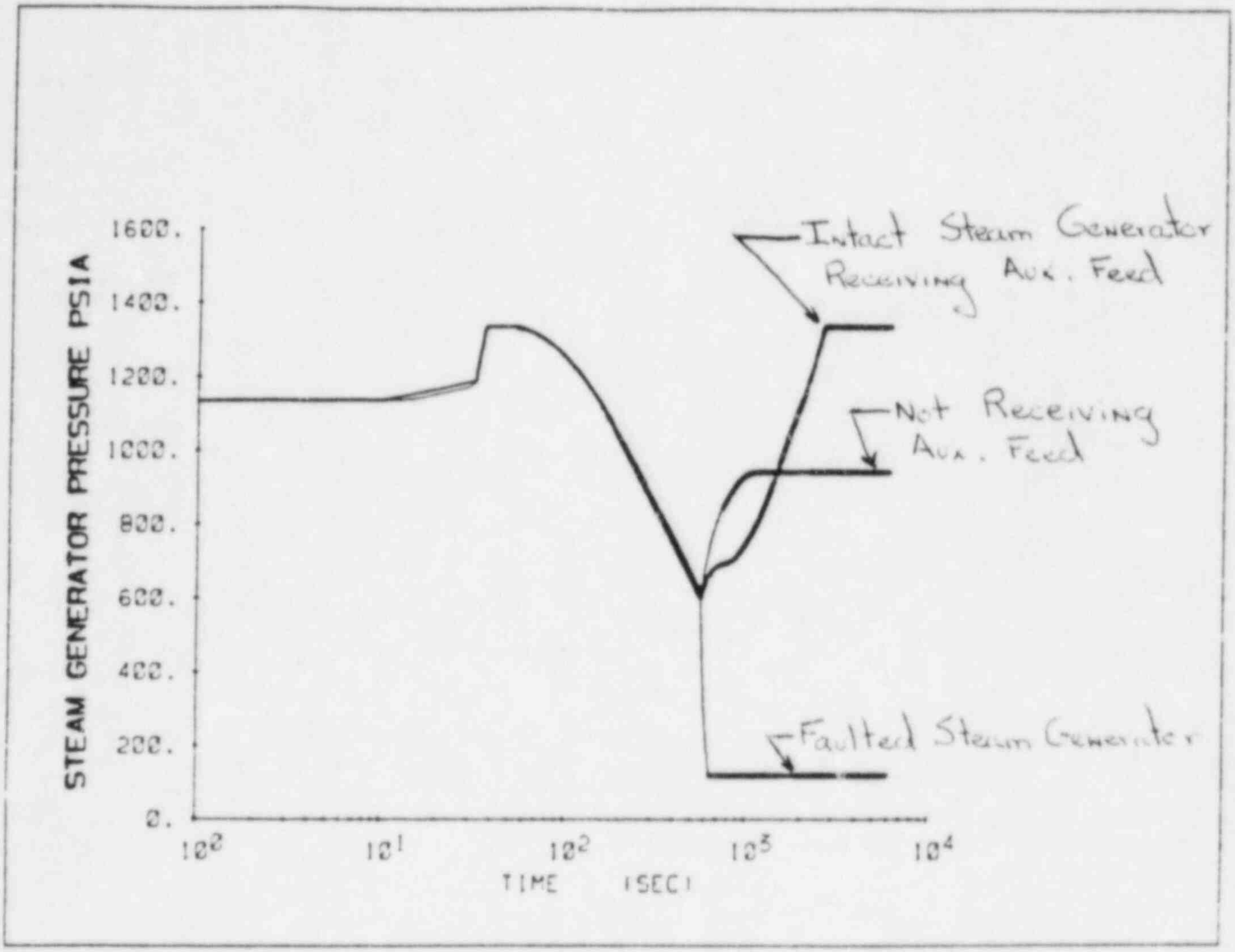

Figure 15.2-22. Steam Generator Pressure Transients for Main Feedline Rupture Without Offsite Power (1 of 1)

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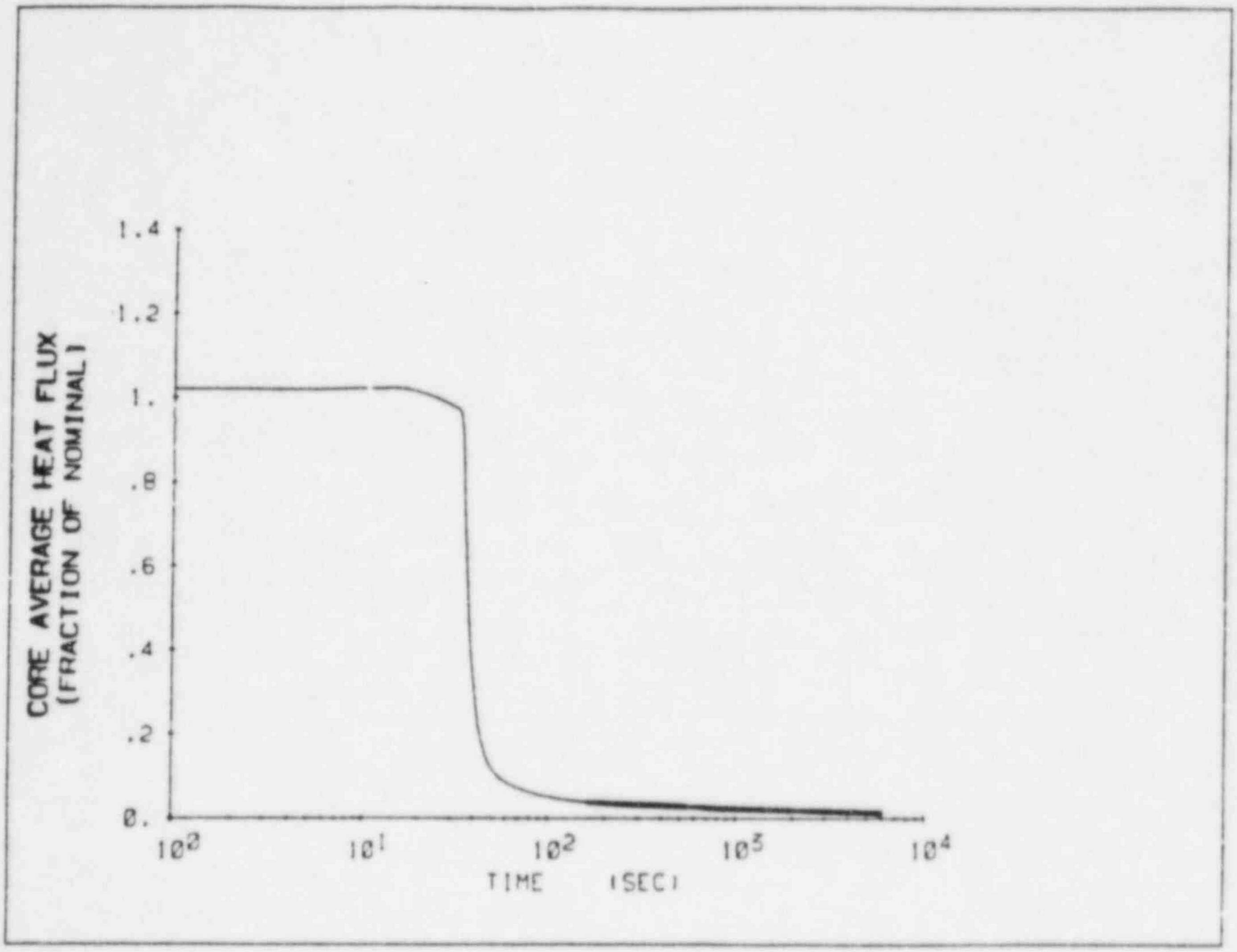

Figure 15.2-23. Core Average Heat Flux Transients for Main Feedline Rupture Without Offsite Power (1 of 1)

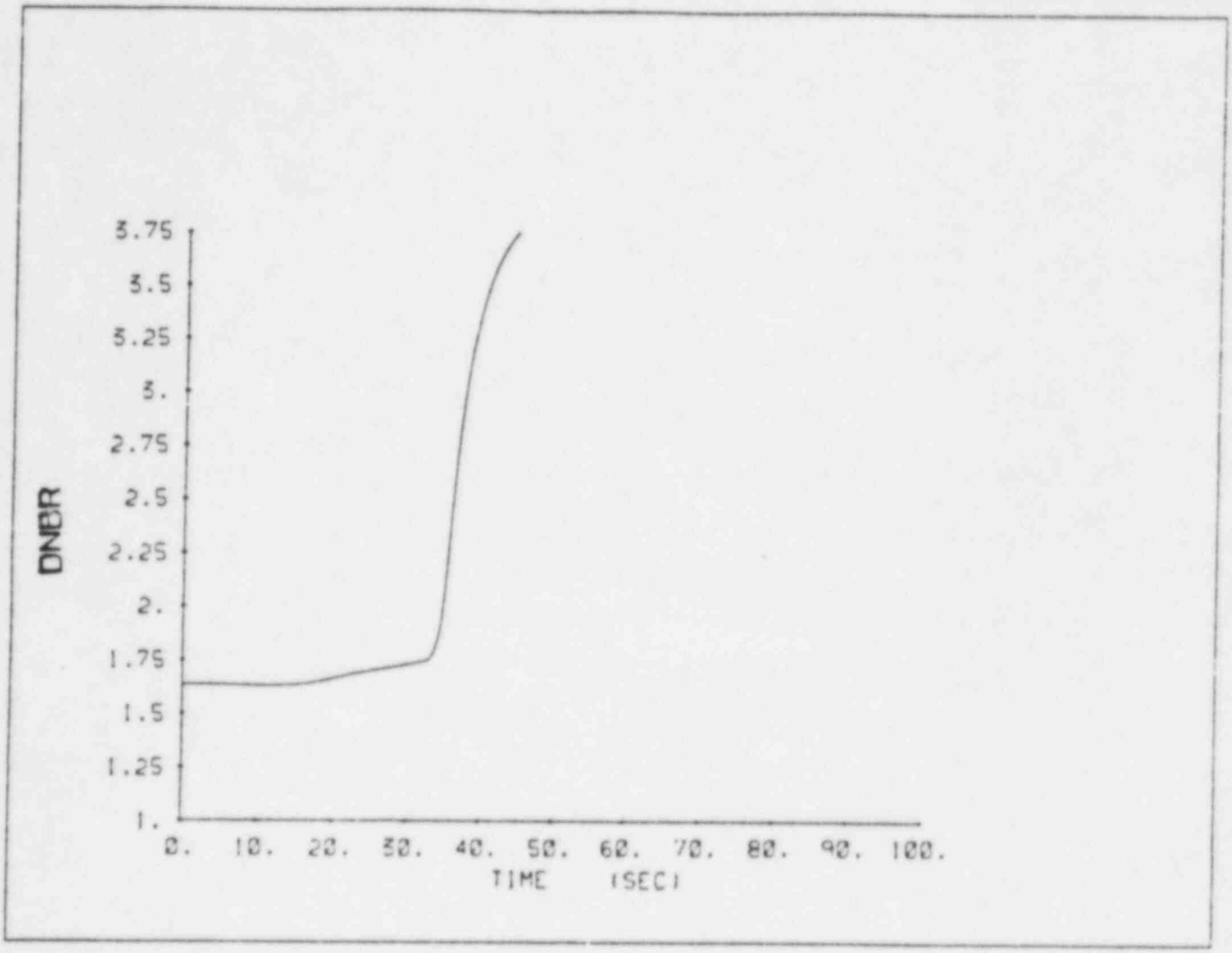

Figure 15.2-24. DNBR Transient for Main Feedline Rupture Without Offsite Power (1 of 1)

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults are postulated which could result in a decrease in reactor coolant system (RCS) flow. These events are discussed in this section. Detailed analyses are presented for the most limiting of these events.

Discussions of the following flow decrease events are presented:

1. Partial Loss of Forced Reactor Coolant Flow

2. Complete Loss of Forced Reactor Coolant Flow

3. Reactor Coolant Pump Shaft Seizure (Locked Rotor)

4. Reactor Coolant Pump Shaft Break

Item 1 above is considered to be an American Nuclear Society (ANS) Condition II event, item 2 an ANS Condition III event, and items 3 and 4 ANS Condition IV events (see Section 15.0.1).

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Accident Description. A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump or pumps supplied by a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped.

Normal power for the pumps is supplied through individual buses connected to the generator. When a generator, turbine, or reactor trip occurs, without an electrical fault, the generator circuit breaker automatically opens and back-feed of off-site power occurs through the main transformer and unit auxiliary transformer. Thus, the pumps will continue to supply coolant flow to the core. 43

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 15.0.1.

The necessary protection for a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip which is actuated by two out of three low flow signals in any reactor coolant loop. Above interlock 43
P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (interlock P-7) and the power level corresponding to interlock P-8, low flow in any two loops will actuate a reactor trip. Above interlock P-7, two or more reactor coolant pump circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip. 43

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-14. 2 Q21 6

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15.3.1.2 Analysis of Effects and Consequences.

Method of Analysis

The ~~Two~~ case~~s~~ ~~have been~~ analyzed~~s~~ is the loss

~~1. Loss~~ of one pump with four loops in operation.

~~2. Loss of one pump with three loops in operation~~

This transient is analyzed by three digital computer codes. First, the LOFTRAN code (Reference 15.3-1) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 15.3-2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Section 4.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The departure from nucleate boiling ratio (DNBR) transients presented represent the minimum of the typical or thimble cell. X

Initial Conditions

Plant characteristics and initial conditions are discussed in Section 15.0.3. Initial operating conditions assumed for this event are the most adverse with respect to the margin to DNB; i.e., maximum steady state power level, minimum steady state pressure, and maximum steady state coolant average temperature. ~~With three loops operating, the maximum power level (including errors) allowed for three loop operation is assumed.~~ The pressure uncertainty used in this analysis is 34 psi and the coolant average temperature uncertainty is 4.7°F.

Reactivity Coefficients

The most negative Doppler-only power coefficient is used (see Figure 15.0-2). This is the equivalent of a total integrated Doppler reactivity from 0 to 100 percent of 0.016 percent Δk.

The least negative moderator temperature coefficient (see Figure 15.0-6) is assumed since this results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and the pump characteristics and is based on high estimates of system pressure losses.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Figures 15.3-1 through 15.3-4 show the transient response for the loss of one reactor coolant pump with four loops in operation. Figure 15.3-4 shows the DNBR to be always greater than 1.30. 18

~~Figures 15.3-5 through 15.3-8 show the transient response for the loss of one reactor coolant pump with three loops in operation. The minimum DNBR is greater than 1.30 as shown on Figure 15.3-8.~~ 18

~~For both cases analyzed,~~ Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not significantly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events ~~for the two cases analyzed~~ is shown in Table 15.3-1. The affected reactor coolant pump will continue to coast down, and the core flow will reach a new equilibrium value corresponding to the number of pumps still in operation. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

15.3.1.3 Radiological Consequences. A partial loss of reactor coolant flow from full load would result in a reactor and turbine trip. Assuming, in addition, that the condenser is not available, atmospheric steam dump may be required.

There are only minimal radiological consequences associated with this event. Therefore this event is not limiting. The radiological consequences resulting from atmospheric steam dump are less severe than the steam line break event analyzed in Section 15.1.5 since fuel damage as a result of this transient is not postulated.

15.3.1.4 Conclusions. The analysis shows that the DNBR will not decrease below 1.30 at any time during the transient. Thus, the DNB design basis as described in Section 4.4 is met. 18

The radiological consequences of this event are not limiting.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Accident Description. A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor were not tripped promptly. 43

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. When a generator, turbine, or reactor trip occurs, without an electrical fault, the generator circuit breaker automatically opens and back-feed of off-site power occurs through the main transformer and unit auxiliary transformer. ~~Thus~~ Thus, the pumps will continue to supply coolant flow to the core. 43

This event is classified as an ANS Condition III incident (an infrequent incident) as defined in Section 15.0.1.

The following trips provide protection for a complete loss of flow accident: 43

1. Reactor coolant pump power supply undervoltage or underfrequency

2. Low reactor coolant loop flow

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps (i.e., loss of offsite power). This function is blocked below approximately 10 percent power (interlock P-7). 43

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbance on the power grid. Reference 15.3-3 provides analyses of grid frequency disturbances and the resulting Nuclear Steam Supply System (NSSS) protection requirements which are generally applicable to South Texas Project.

The reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. This function is generated by two out of three low flow signals per reactor coolant loop. Above interlock P-8, low flow in any loop will actuate a reactor trip. Between approximately 10 percent power (interlock P-7) and the 43 power level corresponding to interlock P-8, low flow in any two loops will actuate a reactor trip. If the maximum grid frequency decay rate is less than approximately 5 Hz/second, this underfrequency trip function will protect the 43 core from underfrequency events. This effect is fully described in Reference 15.3-3.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-14. Q211.6 18

15.3.2.2 Analysis of Effects and Consequences. The ~~two~~ case~~s~~ ~~have been~~ analyzed is the loss

~~1. Loss~~ of four pumps with four loops in operation.

~~2. Loss of three pumps with three loops in operation.~~

This transient is analyzed by three digital computer codes. First, the LOFTRAN code (Reference 15.3-1) is used to calculate the loop and core flow during the transient, the time of reactor trip based on the calculated flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 15.3-2) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Section 4.4) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in

Section 15.3.1, except that following the loss of power supply to all pumps at power, a reactor trip is actuated by either reactor coolant pump power supply undervoltage or underfrequency. A

Results

Figures 15.3-9 through 15.3-12 show the transient response for the loss of power to all reactor coolant pumps with four loops in operation. The reactor is again assumed to be tripped on undervoltage signal. Figure 15.3-12 shows the DNBR to be always greater than 1.30.

~~Figures 15.3-13 through 15.3-16 show the transient response for the loss of power to all reactor coolant pumps with three loops in operation. The reactor is again assumed to be tripped on undervoltage signal. The minimum DNBR is greater than 1.30, as shown on Figure 15.3-16.~~

~~For both cases analyzed,~~ since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The calculated sequence of events ~~for the two cases analyzed~~ is shown in Table 15.3-1. The reactor coolant pumps will continue to coast down, ~~coastdown~~ and natural circulation flow will eventually be established, as demonstrated in Section 15.2.6. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

15.3.2.3 Radiological Consequences. A complete loss of reactor coolant flow from full load results in a reactor and turbine trip. Assuming, in addition, that the condenser is not available, atmospheric steam dump would be required. The quantity of steam released would be the same as for a loss of offsite power.

There are only minimal radiological consequences associated with this event. Therefore, this event is not limiting. Since fuel damage is not postulated, the radiological consequences resulting from atmospheric steam dump are less severe than the steam line break, discussed in Section 15.1.5.

15.3.2.4 Conclusions. The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below 1.30 at any time during the transient. Thus, the DNB design basis as described in Section 4.4 is met. |18

15.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

15.3.3.1 Identification of Causes and Accident Description. The accident postulated is an instantaneous seizure of a reactor coolant pump rotor such as is discussed in Section 5.4. Flow through the affected reactor coolant loop is rapidly reduced, leading to an initiation of a reactor trip on a low reactor coolant flow signal. |43

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced,

(A)

One variation between this analysis and that of the previous section is that the RCCA insertion time to dashpot entry is 2.58 seconds. This is a conservative insertion time under the reduced flow conditions that exist when the RCCAs are inserted for this transient.

first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon turbine trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure reducing effect as well as the pressure reducing effect of the spray is not included in the analysis. |43

This event is classified as an ANS Condition IV incident (a limiting fault) as defined in Section 15.0.1.

15.3.3.2 Analysis of Effects and Consequences.

Method of Analysis

Three digital computer codes are used to analyze this transient. The LOFTRAN code (Reference 15.3-1) is used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN code (Reference 15.3-2), using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient. The FACTRAN code is also used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Section 4.4) is used to calculate the DNBR distribution in the core during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR distribution is used to calculate the number of rods in DNB.

~~Three~~
~~Two~~ cases are analyzed:
Two

1. Four loops operating, one locked rotor

~~2. Three loops operating, one locked rotor~~

2. Four loops operating, one locked rotor. Loss of power to the other reactor coolant pumps

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most adverse steady state operating condition (i.e., maximum steady state power level, maximum steady state pressure, and maximum steady state coolant average temperature). Plant characteristics and initial conditions are further discussed in Section 15.0.3. ~~With three loops operating, the maximum power level (including errors) allowed in that mode of operation is assumed.~~ (B)

When the peak pressure is evaluated, the initial pressure is conservatively estimated as ~~30~~ 34 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops |18

insert (A)

Insert A

seconds

For the case without offsite power available, power is lost to the unaffected pumps 2 s after reactor trip. (Note: Grid stability analyses show that the grid will remain stable and that offsite power will not be lost because of a unit trip from 100-percent power. The 2-s delay is a conservative assumption based on grid stability analyses.)

2 second

(E)

The pressure uncertainty used in these analyses is 34 psi and the coolant average temperature uncertainty is 4.7°F.

are added to the calculated pressurizer pressure. The pressure responses shown on Figure 15.3-18 ~~and 15.3-21~~ are the responses at the point in the RCS having the maximum pressure.

Evaluation of the Pressure Transient

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion begins one second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer power-operated relief valves, pressurizer spray, steam dump or controlled feedwater flow after reactor trip. Although these are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effects.

The pressurizer safety valves are full open at 2575 psia and their capacity for steam relief is as described in Section 5.4.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction.

In the evaluation, rod power at the hot spot is assumed to be 2.50 times the average rod power (i.e., $F_q = 2.50$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density and mass flow rate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) has pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 Btu/hr-ft^2-°F at the initiation of the transient. Thus the large amount of energy stored in the fuel because of the small initial value is released to the clad at the initiation of the transient.

Zirconium Steam Reaction

The zirconium steam reaction can become significant above 1800° F (clad temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium steam reaction.

$$\frac{d(w^2)}{dt} = 33 \times 10^6 \exp \frac{-(45,500)}{1.986T}$$

where:

W = amount reacted, mg/cm^2

t = time, sec

T = temperature, °F

The reaction heat is 1510 cal/gm.

The effect of zirconium steam reaction is included in the calculation of the "hot spot" clad temperature transient.

Plant systems and equipment which are available to mitigate the effects of the accident are discussed in Section 15.0.8 and listed in Table 15.0-6. No single active failure in any of these systems or equipment will adversely affect the consequences of the accident.

Results

Locked Rotor with Four Loops Operating

The transient results for this case are shown on Figures 15.3-17 through 15.3-20. The results of these calculations are also summarized in Table 15.3-24. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700° F. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. The number of rods in DNB was conservatively calculated as 7 percent of the total rods in the core. 12

X

~~Locked Rotor with Three Loops Operating~~

~~The transient results for this case are shown on Figures 15.3-21 through 15.3-24. The peak RCS pressure is slightly higher than for the previous case, but is still less than that which would cause stresses to exceed the faulted condition stress limits. The clad temperature transient is slightly more severe than for the previous case.~~

The calculated sequence of events for the ~~two~~ ~~three~~ two cases with offsite power available, analyzed is shown in Table 15.3-1. Figure~~s~~ 15.3-17 ~~and 15.3-21~~ shows that the core flow reaches a new equilibrium value by 10 seconds. With the reactor tripped, a stable plant condition will eventually be attained. ~~Normal plant shutdown may then proceed.~~

insert B

Insert B

Locked Rotor with Four Loops Operating, Loss of Power to the Remaining Pumps

The transient results for this case are shown on Figures 15.3-17 through 15.3-20. The results of these calculations are also summarized in Table 15.3-26. The peak RCS pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits. Also, the peak clad surface temperature is considerably less than 2700°F. Both the peak RCS pressure and the peak clad surface temperature for this case are similar to the 4 loop transient with power available as discussed above. The total percentage of fuel cladding damaged is the same as the with-power case, thus the conclusions of Section 15.3.3.3 are applicable to both events.

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15.3.3.3 Radiological Consequences. The postulated accidents involving release of steam from the secondary system do not result in a release of radioactivity unless there is leakage from the Reactor Coolant System (RCS) to the secondary system in the steam generators (SGs). A conservative analysis of the potential offsite doses resulting from a reactor coolant pump shaft seizure accident is presented using the technical specification limit secondary coolant concentrations. Parameters used in the analysis are listed in Table 15.3-3.

The conservative assumptions and parameters used to calculate the activity released and offsite doses for a pump shaft seizure accident are the following:

1. Prior to the accident, the primary coolant concentrations are assumed to be equal to the technical specification limit for full power operation following an iodine spike (I-131 equivalent of 60 μCi/g). These concentrations are presented in Table 15.A-4.

2. Prior to the accident, the secondary coolant specific activity is equal to the technical specification limit of 0.10 μCi/gm dose equivalent I-131. This dose equivalent specific activity is presented in Table 15.A-5.

3. Seven percent of the total core fuel cladding is damaged, which results in the release of the reactor coolant of seven percent of the total gap inventory of the core. This activity is assumed uniformly mixed in the primary coolant.

4. The primary-to-secondary leakage of 1 gal/min (technical specification limit) is assumed to continue for 8 hrs following the accident.

5. Offsite power is lost; MS condensers are not available for steam dump.

6. Eight hours after the accident, the Residual Heat Removal System (RHRS) starts operation to cool down the plant. No further steam or activity is released to the environment.

7. The iodine partition factor in the SGs is equal to 0.01.

The steam releases and meteorological parameters are given in Table 15.3-3. 4

The thyroid, gamma and beta doses for the reactor coolant pump shaft seizure accident are given in Table 15.3-4 for the Exclusion Zone Boundary (EZB) of 1430 meters and the Low Population Zone (LPZ) of 4800 meters. 43

15.3.3.4 Conclusions. Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

Since the peak clad surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700° F, the core will remain in place and intact with no loss of core cooling capability.

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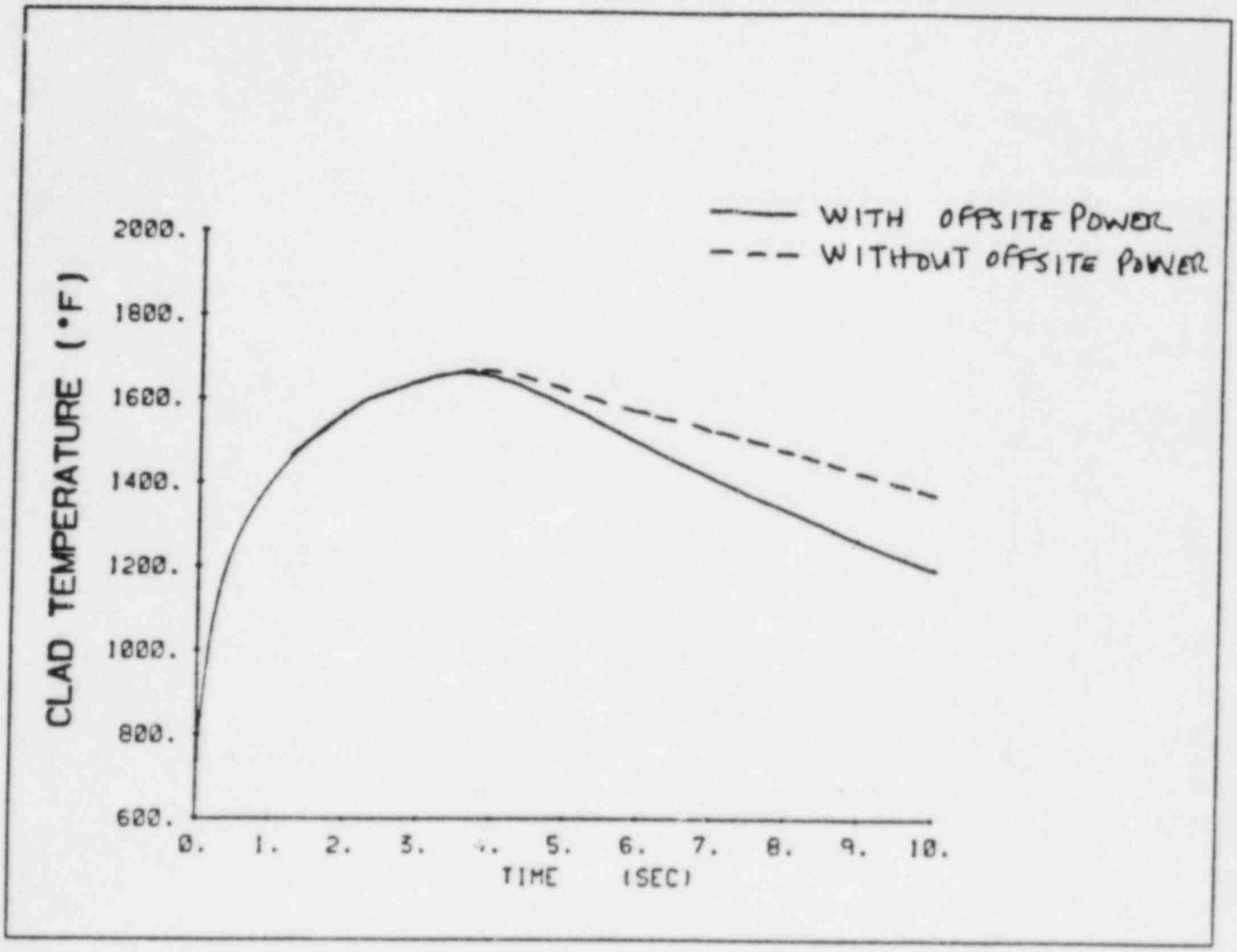

Figure 15.3-20. Maximum Clad Temperature at Hot Spot for Four Loops in Operation, One Locked Rotor (1 of 1)

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REFERENCES

SECTION 15.3

~~15.3-1 Burnett, T. W., F. McIntyre, C. C. Buker, J. C. and Rose, R. P., "LOFTRAN Code Description," WCAP-7907, June, 1972.~~

15.3-2 Hargrove, H. G., "FACTRAN, a Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod," WCAP-7908, June 1972. 43

15.3-3 Burnett, T. W., "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," WCAP-7306, April, 1969.

15.3-1 Burnett, T. W. T., et al, "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN REACTOR COOLANT SYSTEM FLOW

Accident	Event	Time (sec)	
Partial Loss of Forced Reactor Coolant Flow			
~~1.~~ Four loops operating, one pump coasting down	Coastdown begins ~~Low~~	0	
	Low reactor coolant flow trip	~~1.65~~ 1.30	43
	Rods begin to drop	~~2.65~~ 2.30	
	Minimum DNBR occurs	~~3.80~~ 3.40	18
~~2. Three loops operating, one pump coasting down~~	~~Coastdown begins Low~~	~~0~~	
	~~Low reactor coolant flow trip~~	~~3.10~~	43
	~~Rods begin to drop~~	~~4.20~~	
	~~Minimum DNBR occurs~~	~~5.20~~	18

Accident	Event	Four Loop	~~Three Loop~~	
Complete Loss of Forced Reactor Coolant Flow	All operating pumps lose power and begin coasting down	0	~~0~~	
	Reactor coolant pump undervoltage trip point reached	0	~~0~~	
	Rods begin to drop	1.5	~~1.5~~	
	Minimum DNBR occurs	~~3.4~~ 3.3	~~3.9~~	18
Reactor Coolant Pump Shaft Seizure (Locked Rotor). (with offsite power.)	Rotor on one pump locks	0	~~0~~	
	Low reactor coolant flow ~~trip~~ setpoint reached	~~0.03~~ 0.07	~~0.18~~	43

TABLE 15.3-1 (Cont'd)

Accident	Event	Time (sec) Four Loop Operation		~~Three Loop Operation~~	
	Rods begin to drop	1.07	~~[illegible]~~	~~[illegible]~~	18
	Maximum RCS pressure occurs	3.3	~~[illegible]~~	~~[illegible]~~	
	Maximum clad temperature occurs	3.7	~~[illegible]~~	~~[illegible]~~	18
Reactor Coolant Pump Shaft Seizure (Locked Rotor). (Without offsite power.)	Rotor on one pump locks	0		—	
	Low reactor coolant flow setpoint reached	0.07		—	
	Rods begin to drop	1.07		—	
	RCPs lose power, coastdown begins	3.07		—	
	Maximum RCS pressure occurs	~~3.5~~ 3.3		—	
	Maximum clad temperature occurs	~~3.8~~ 3.9			

TABLE 15.3-2a

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENTS
(with offsite power)

	4 Loops Operating Initially	~~3 Loops Operating Initially~~
Maximum Reactor Coolant System Pressure (psia)	~~[illegible]~~ 2589	~~[illegible]~~
Maximum Clad Temperature at (°F) Core Hot Spot	~~[illegible]~~ 1685	~~[illegible]~~
$Zr-H_2O$ reaction at core hot spot (% by weight)	~~[illegible]~~ .168	~~[illegible]~~

18

TABLE 15.3-2b

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT
(without offsite power)

	4 Loops Operating Initially
Maximum Reactor Coolant System Pressure (psia)	~~2604~~ 2589
Maximum Clad Temperature at Core Hot Spot (°F)	~~[illegible]~~ 1680
$Zr-H_2O$ reaction at Core Hot Spot (% by weight)	~~[illegible]~~ .188

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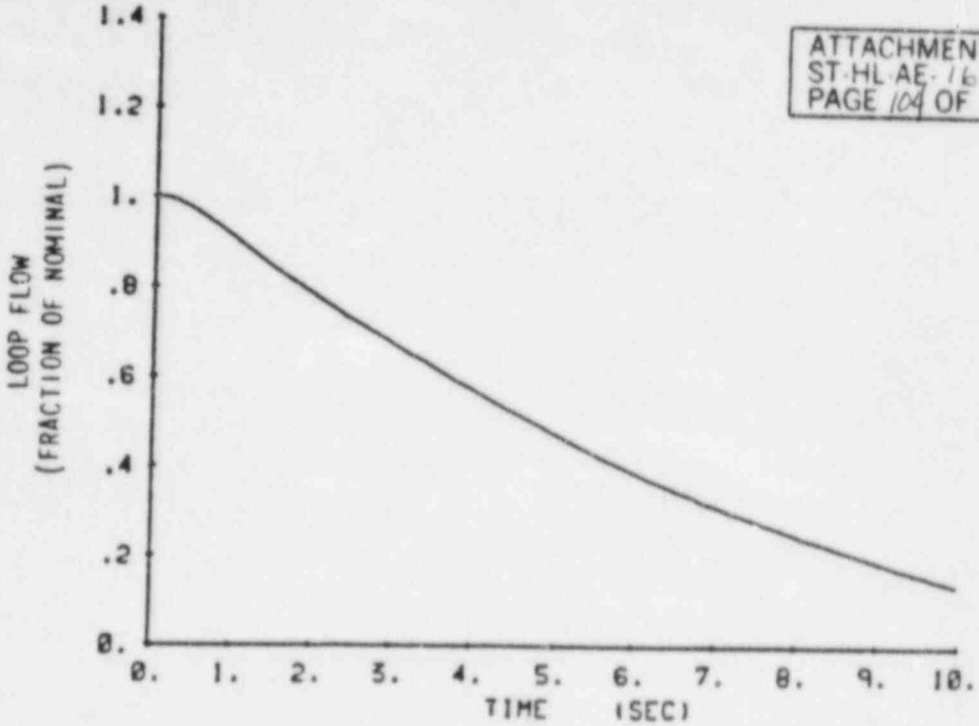
TABLE 15.3-3

PARAMETERS USED IN RC PUMP SHAFT SEIZURE ACCIDENT ANALYSIS

Parameters	
Core thermal power, MWt	3,800
SG tube leak rate prior to accident and initial 8 hrs following accident	1.0 gm
GWPS operating prior to accident	No
Offsite power	Lost
Fuel defects	1.0%
Primary coolant concentrations	Table 15.A-4
Secondary coolant concentrations	Table 15.A-5
Failed fuel (following accident)	7.0% of fuel rods in core
Activity released to reactor coolant from failed fuel and available for release	7% of total gap inventory of noble gases and iodines
Iodine partition factor in SG's during accident	0.01
Steam release from four SGs, lb	614,000* (0-2 hr) 1,264,000 (2-8 hr)
Meteorology	5 percentile Table 15.B-1
Dose model	Appendix 15.B

43

*Condensers assumed unavailable for steam dump.



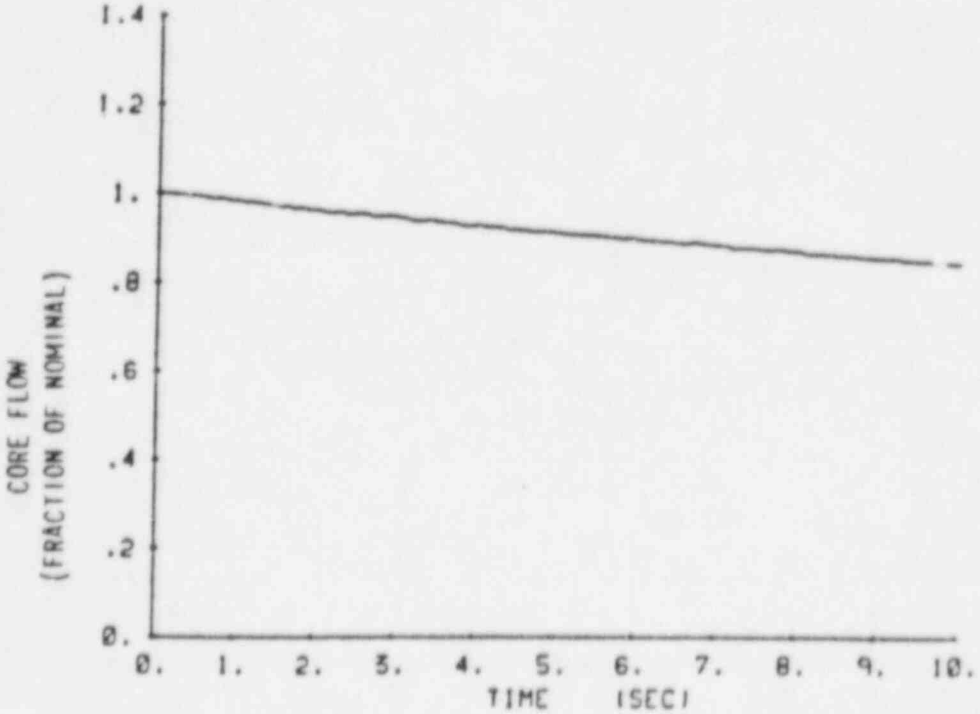


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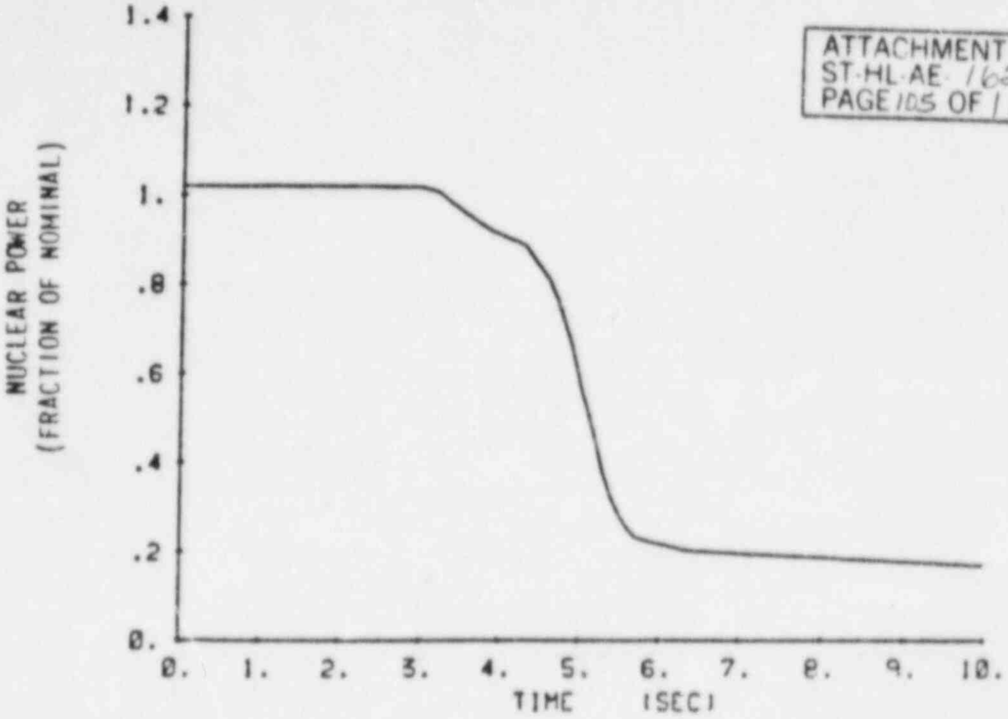
Figure 15.3-1.
Flow Transients for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down



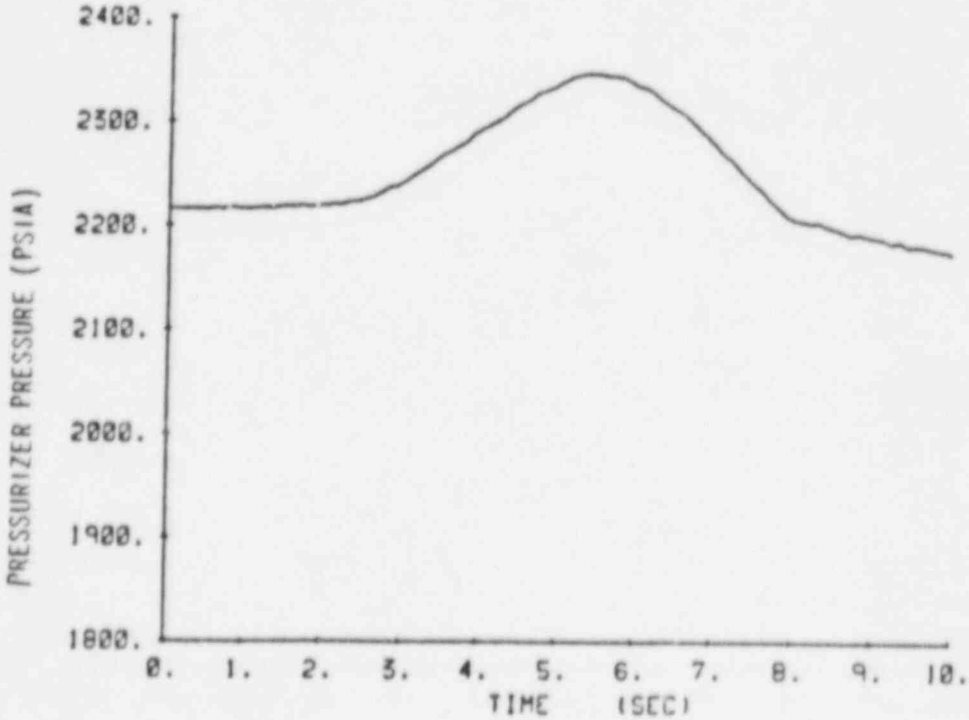


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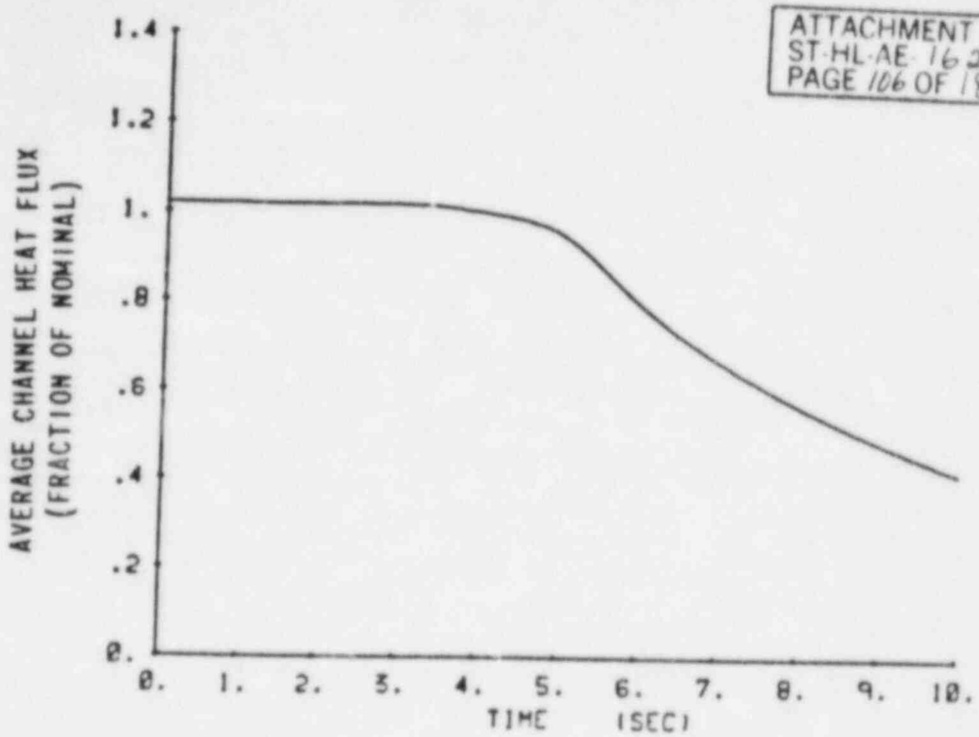
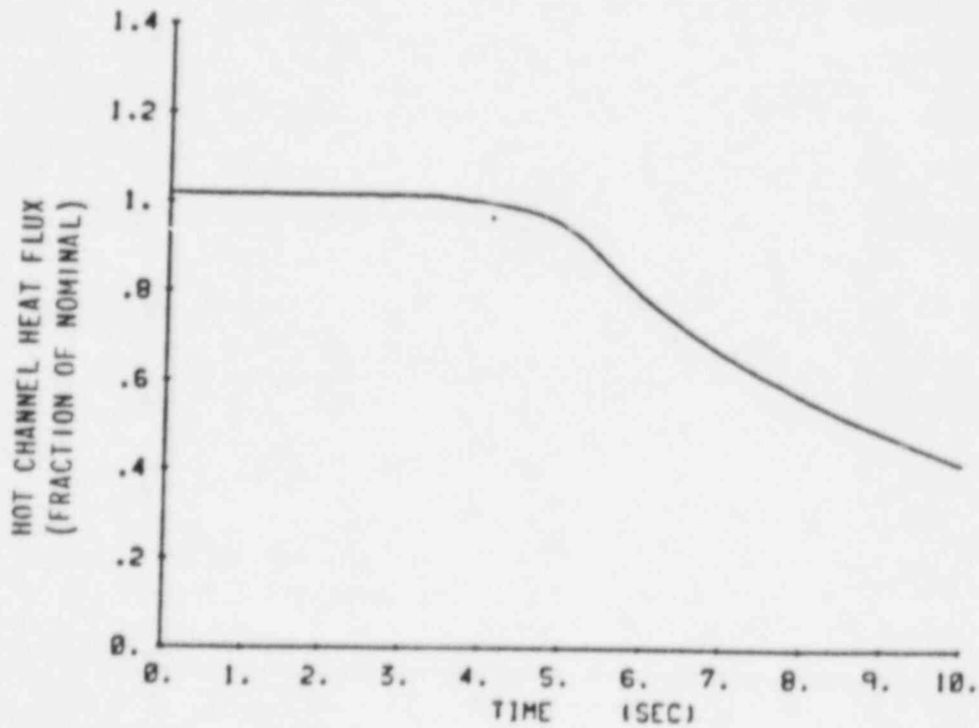
Figure 15.3-2
Nuclear Power Transient and Pressurizer Pressure Transient for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down





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UNITS 1 & 2

Average and Hot Channel Heat Flux Transients for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down

Figure 15.3.3

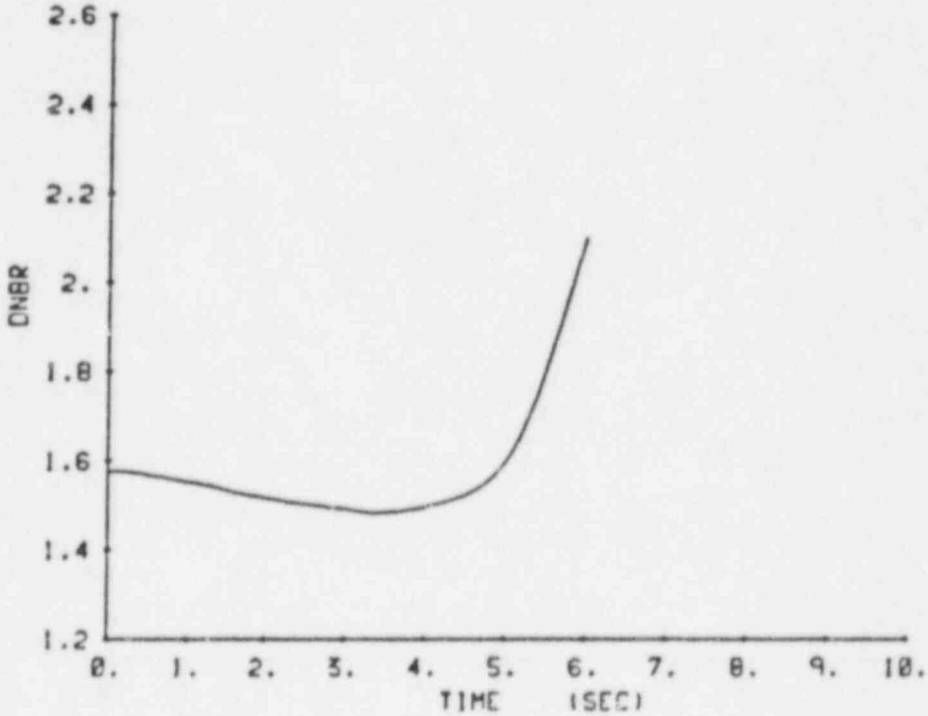

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Figure 15.3-4
DNBR versus Time for Partial Loss of Flow, Four Loops in Operation, One Pump Coasting Down

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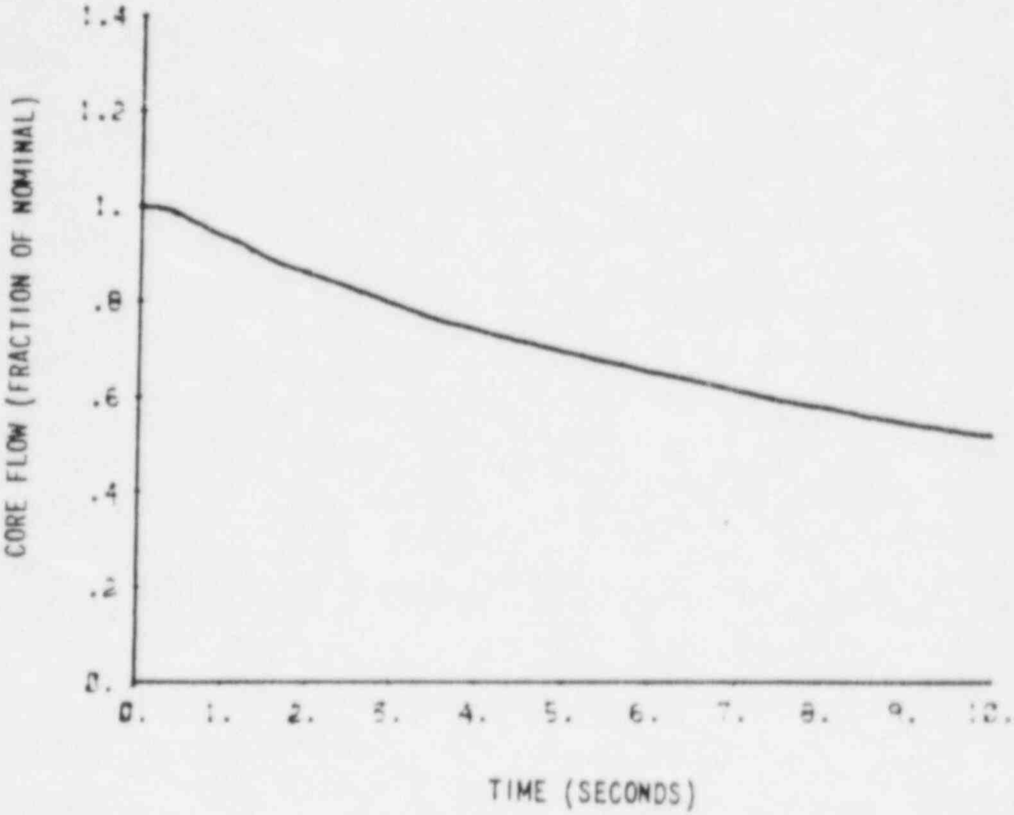

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Figure 15.3-9.
Core Flow Coastdown versus Time for Four Loops in Operation, Four Pumps Coasting Down, Complete Loss of Flow

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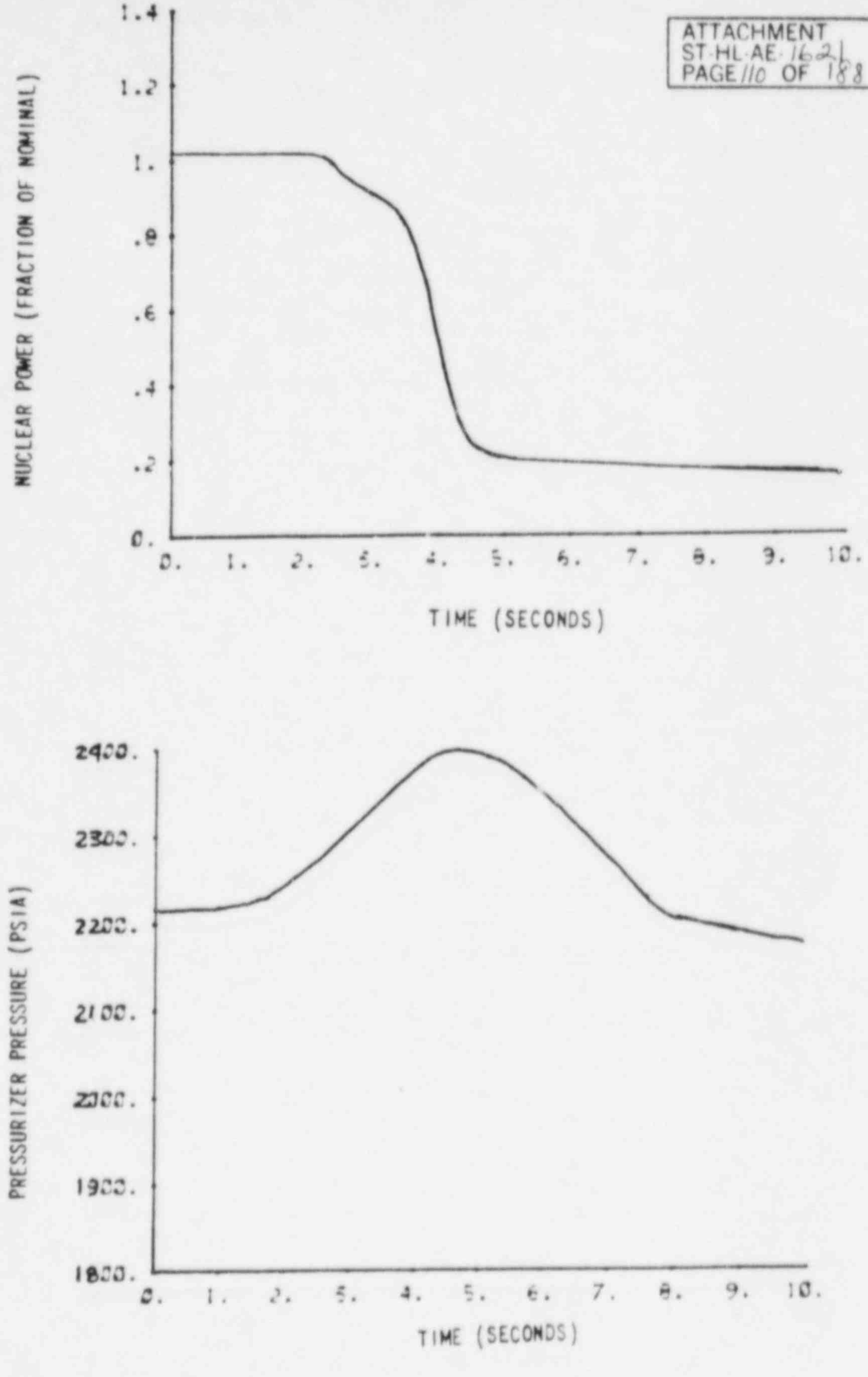

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Figure 15.3.10
Nuclear Power Transient and Pressurizer Pressure Transient for Four Loops in Operation, Four Pumps Coasting Down, Complete Loss of Flow

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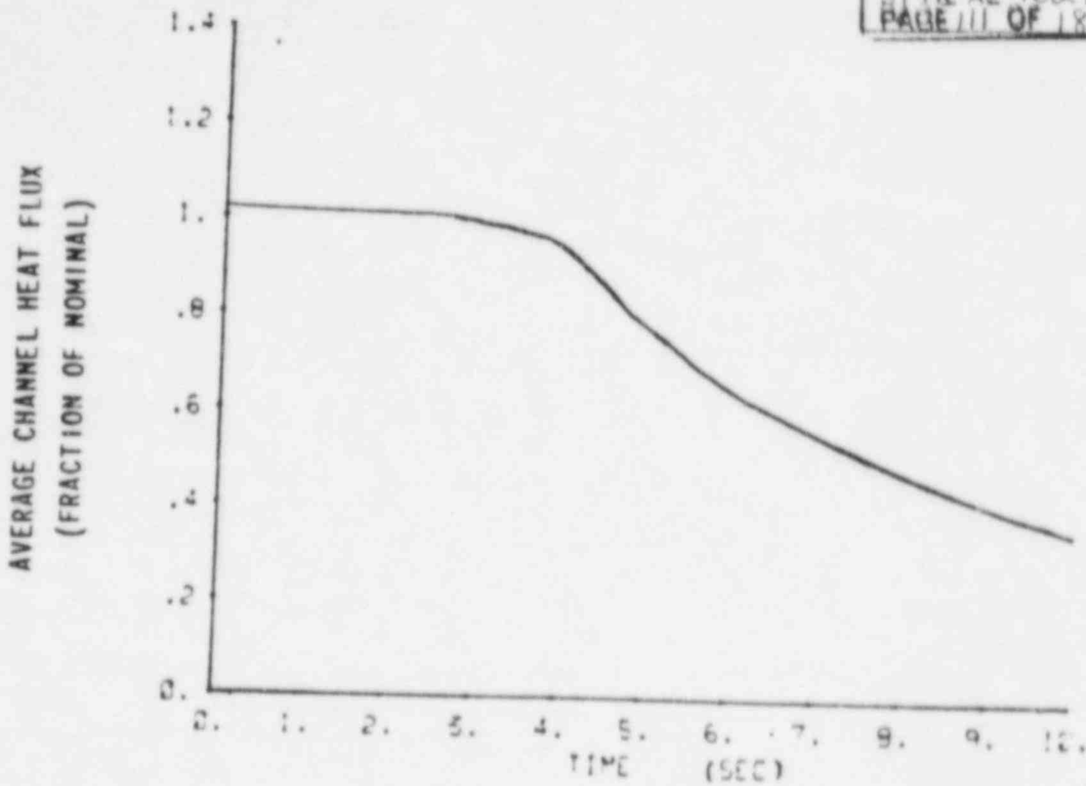

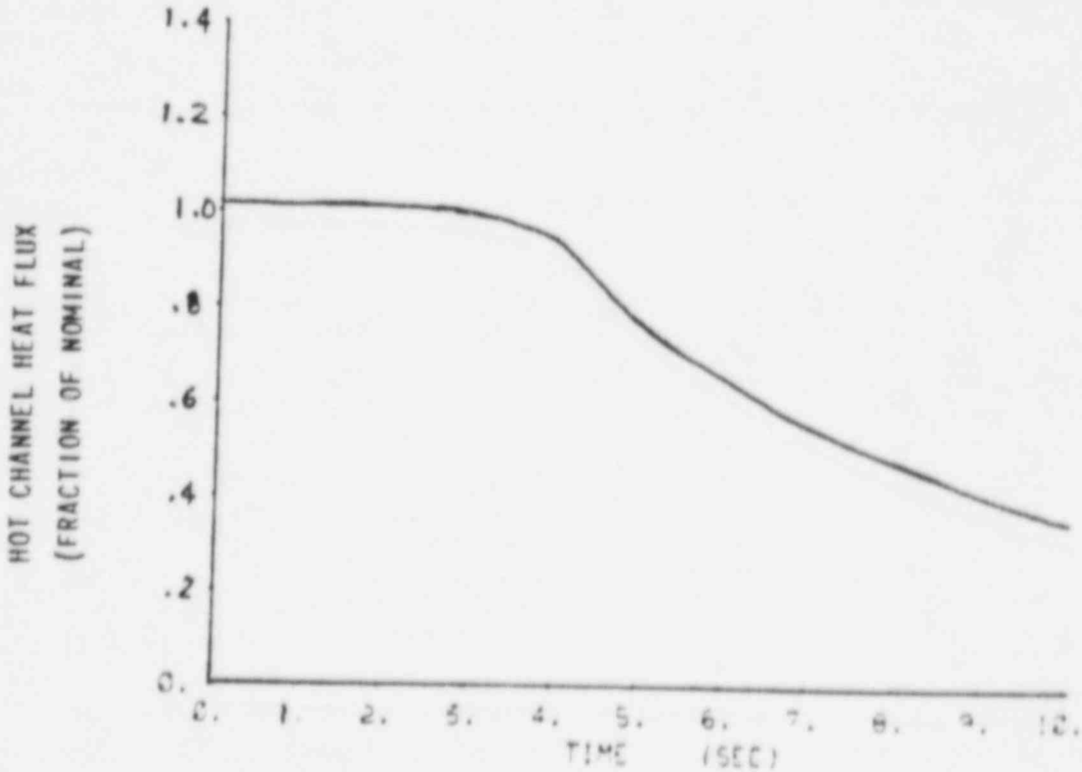

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Average and Hot Channel Heat Flux Transients for Four Loops in Operation, Four Pumps Coasting Down, Complete Loss of Flow

Figure 15.3-11

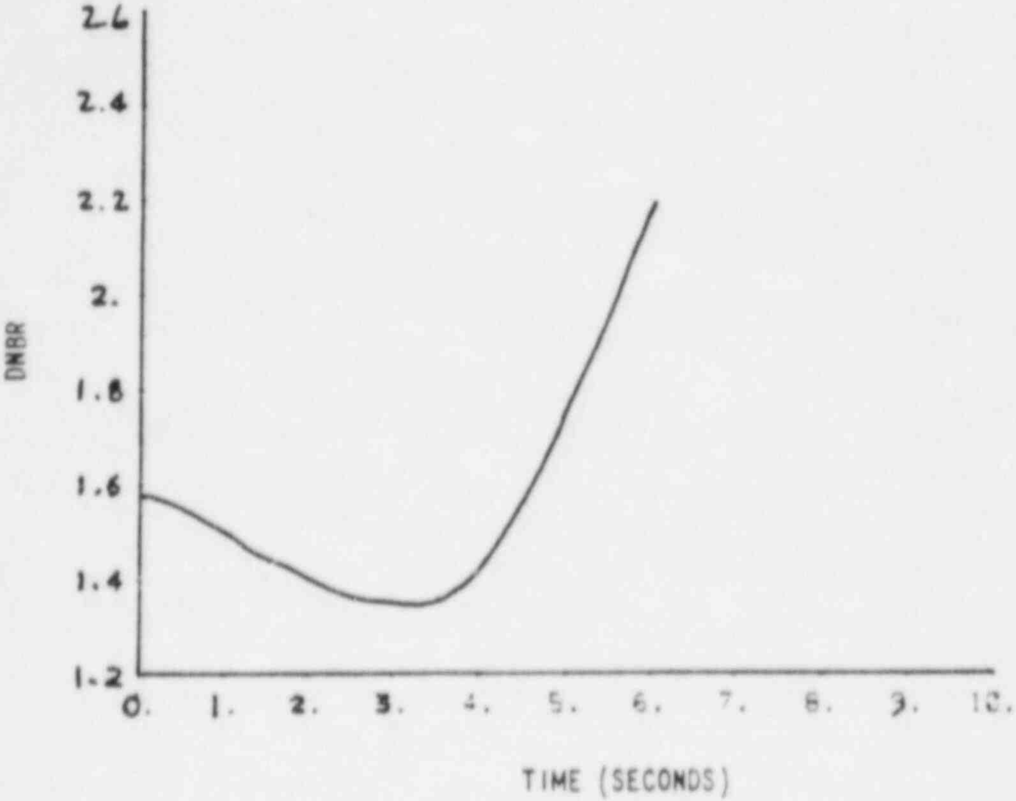

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Figure 15.3-12
DNBR versus Time for Four Loops in Operation
Four Pumps Coasting Down, Complete Loss of Flow

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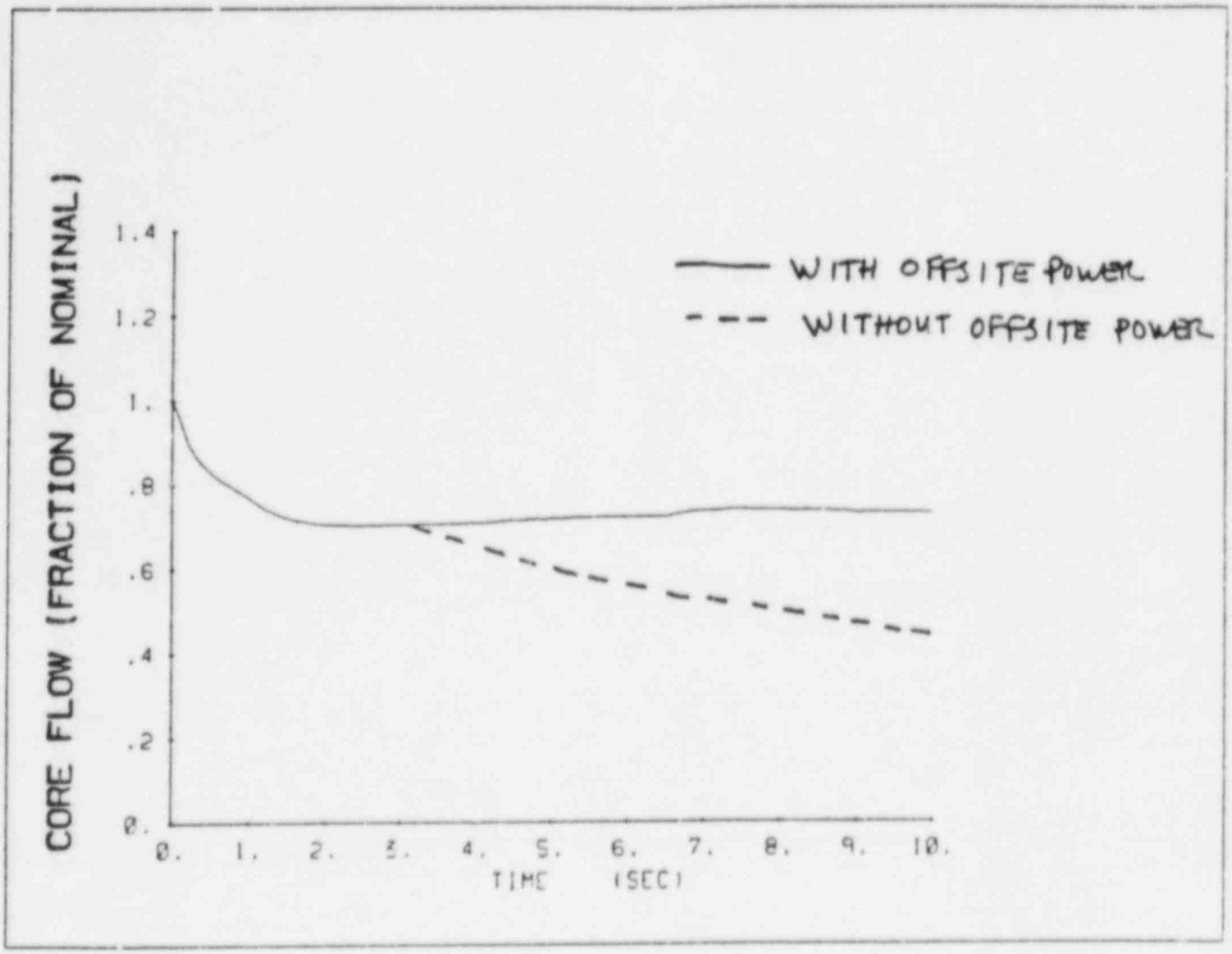

Figure 15.3-17. Flow Transients for Four Loops in Operation, One Locked Rotor (1 of 2)

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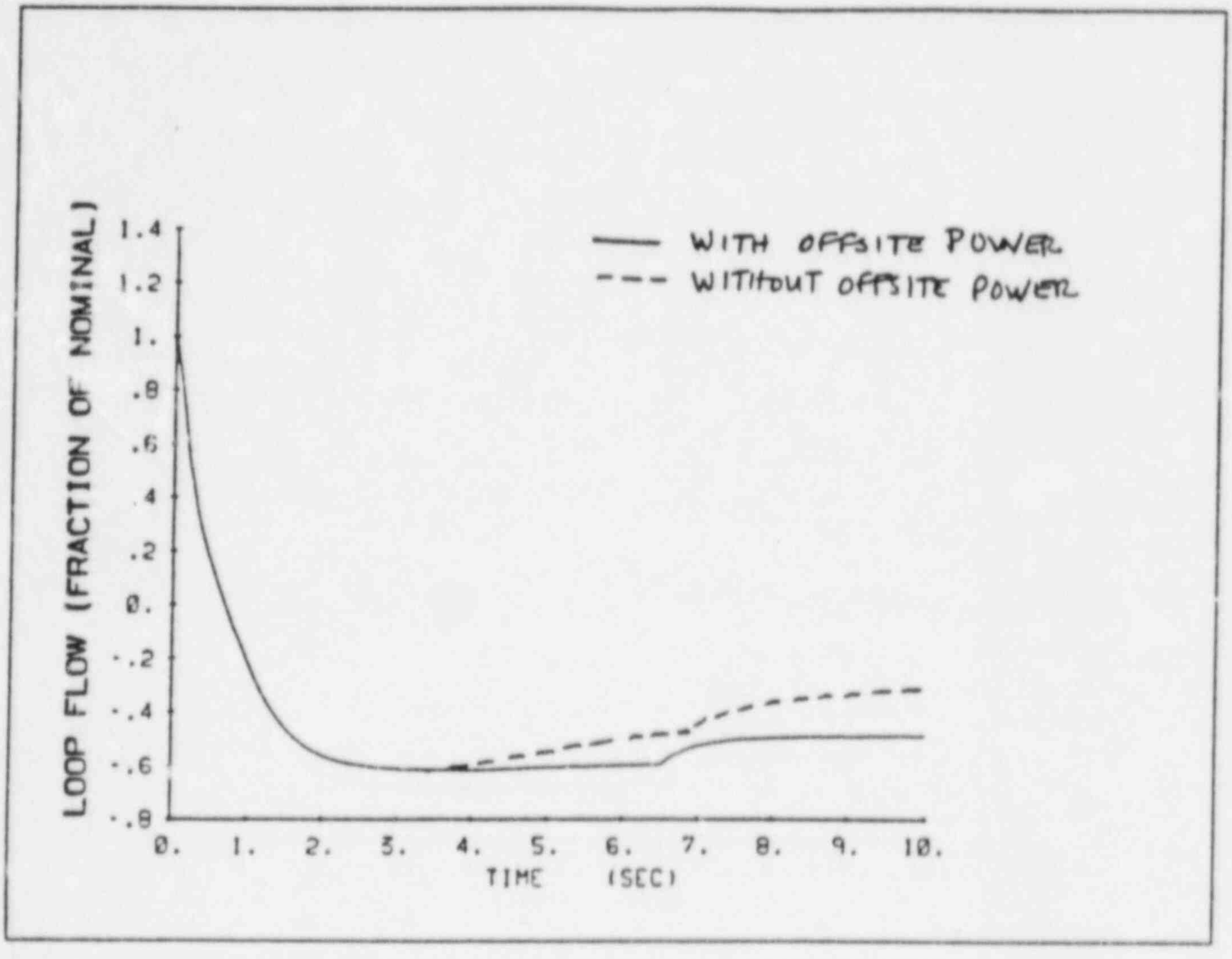

Figure 15.3-17. Flow Transients for Four Loops in Operation, One Locked Rotor (2 of 2)

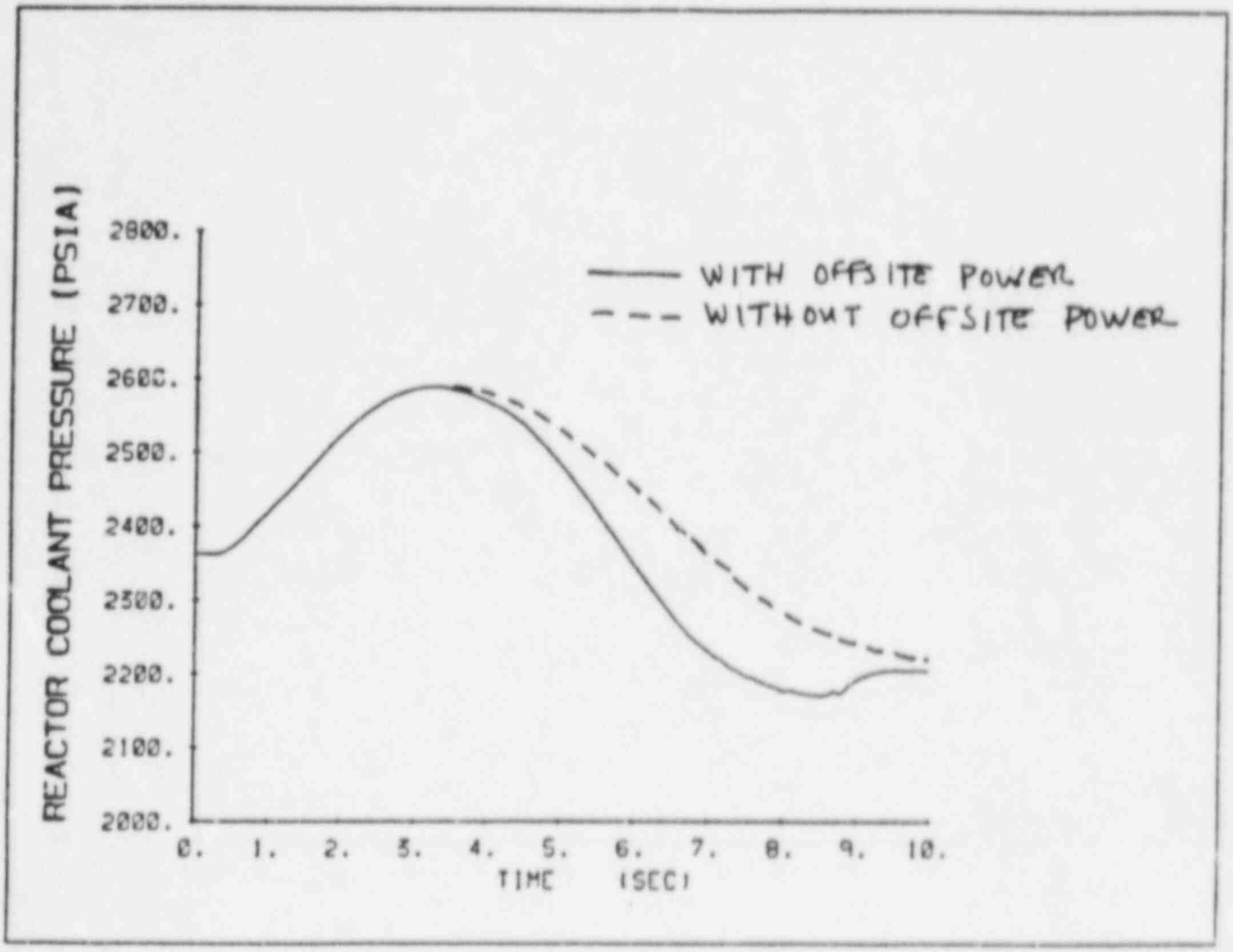

Figure 15.3-18. Reactor Coolant System Pressure Transients for Four Loops in Operation, One Locked Rotor (Sheet 1 of 1)

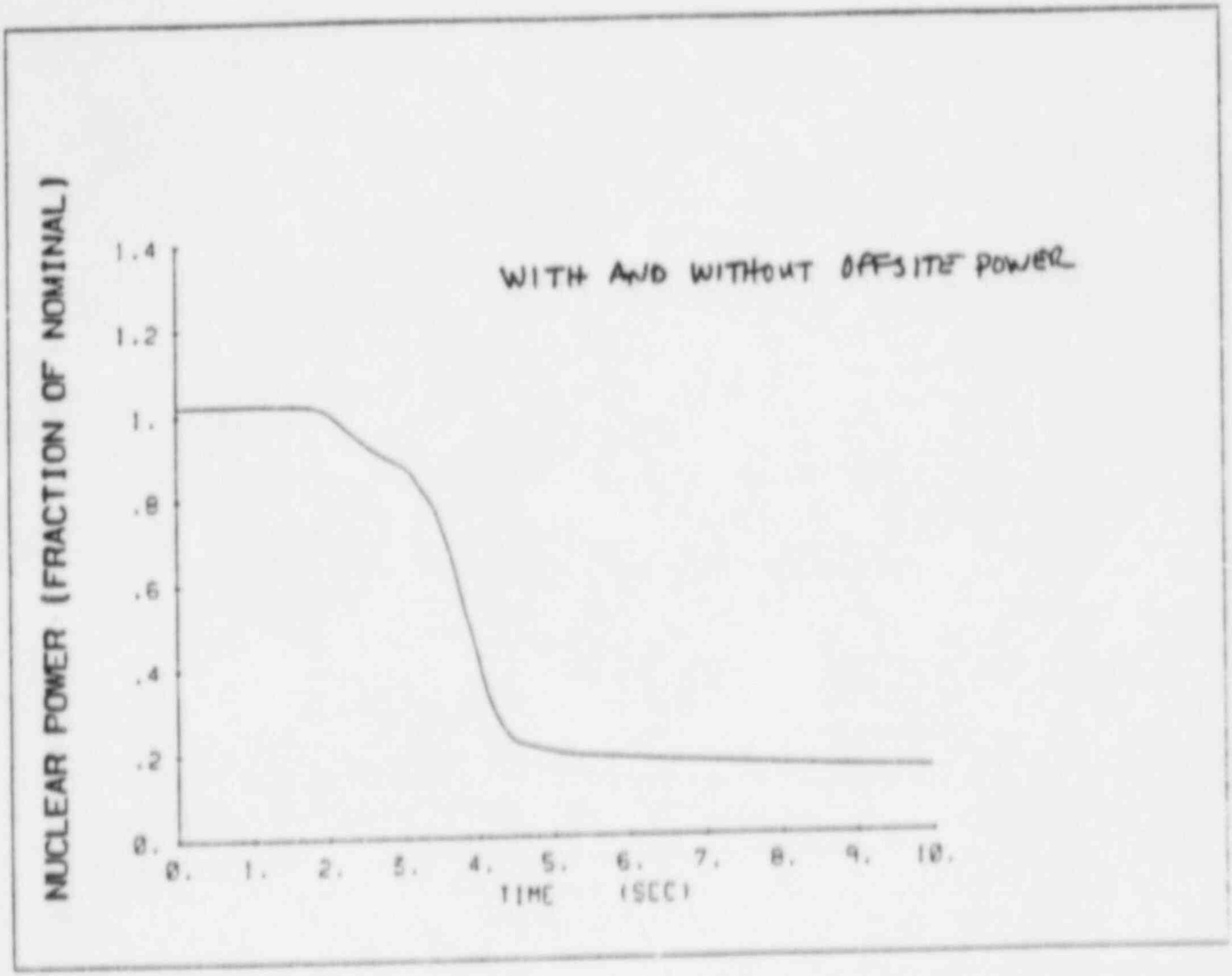

Figure 15.3-19. Nuclear Power Transient, Average and Hot Channel Heat Flux Transients for Four Loops in Operation, One Locked Rotor (1 of 3)

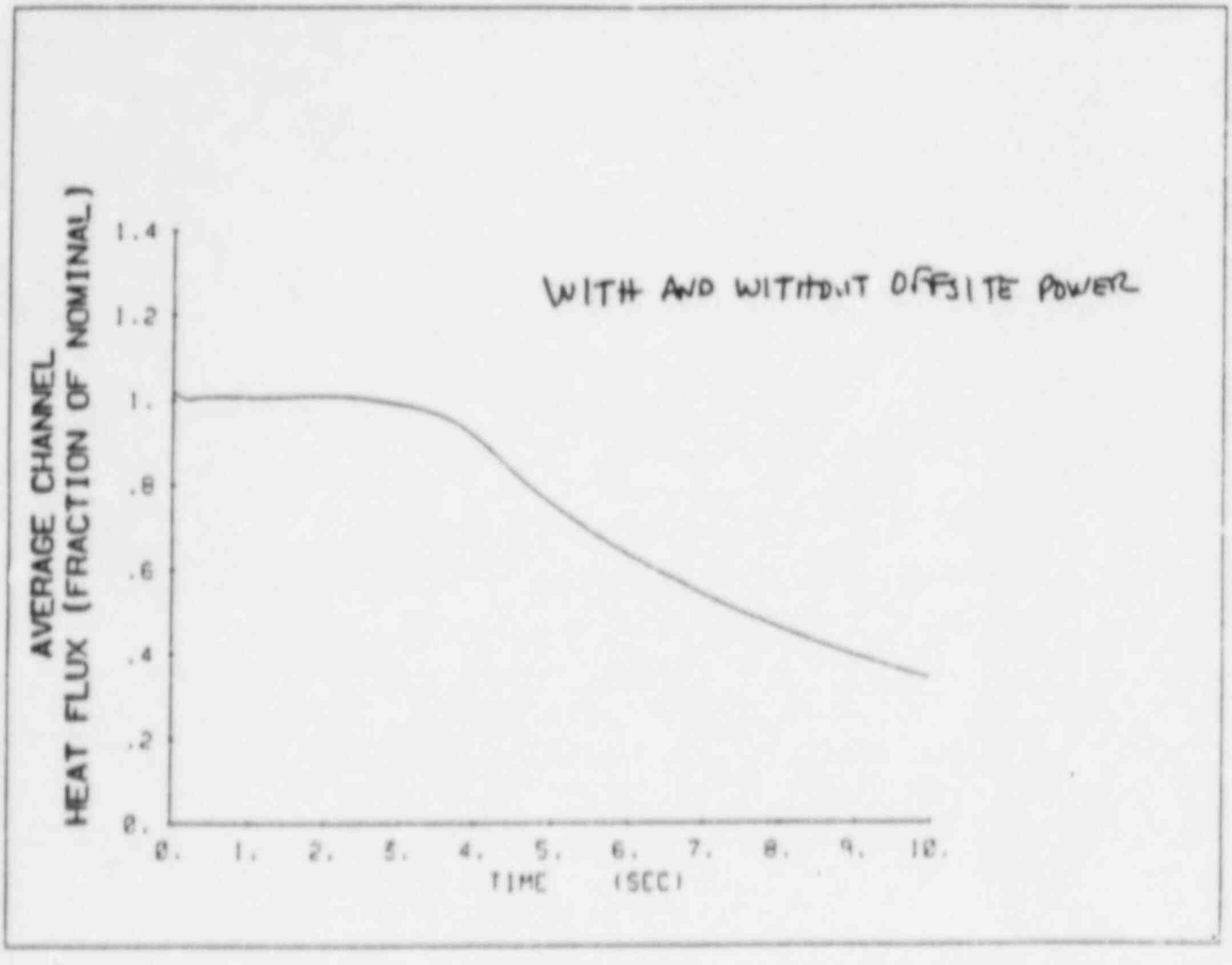

Figure 15.3-19. Nuclear Power Transient, Average and Hot Channel Heat Flux Transients for Four Loops in Operation, One Locked Rotor (2 of 3)

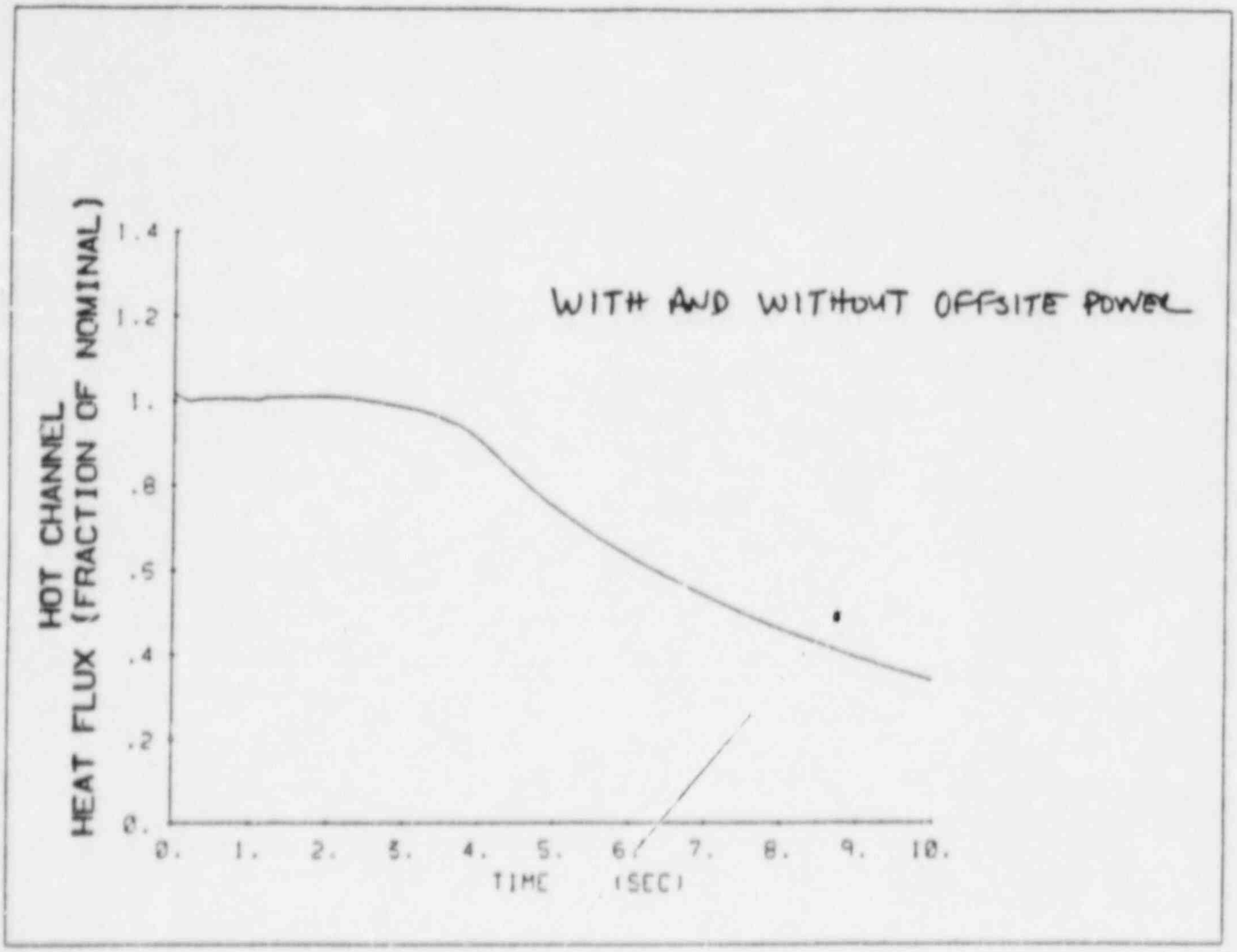

Figure 15.3-19. Nuclear Power Transient, Average and Hot Channel Heat Flux Transients for Four Loops in Operation, One Locked Rotor (3 of 3)

Figures 15.3-21 through 15.3-24 have been deleted.

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APPENDIX 10A

AUXILIARY FEEDWATER SYSTEM

RELIABILITY EVALUATION

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10A.1 INTRODUCTION

10A.1.1 Purpose

This Appendix describes the reliability evaluation of the STP auxiliary feedwater system. The evaluation was performed in a manner consistent with NUREG-0611 to allow a comparison to other plants of the reliability of the STP system for specific initiating events. The results of the evaluation show the system compares favorably with other designs and has a high reliability for the initiating events considered.

This reliability evaluation reflects the auxiliary feedwater system design at the time it was performed. Subsequent modifications will not result in revision of this appendix unless they could have a significant impact on the results presented.

10A.1.2 Objectives

The objectives of the evaluation are:

- o To perform an analysis to evaluate the reliability of the AFWS in accordance with the guidelines contained in NUREG 0611.
- o To provide indication of the contributors of the auxiliary feedwater system unavailability for the initiating events described in NUREG-0611.

10A.1.3 Scope

Three initiating events are analyzed:

Case I: Loss of main feedwater (LMFW)

Case II: Loss of main feedwater coincident with loss of offsite power (LMFW/LOOP)

Case III: Loss of main feedwater coincident with loss of all AC power (LMFW/LOAC)

10A.1.4 General Approach

The principal technique used in the quantitative evaluation is the construction and analysis of fault trees which represent the AFWS' failure logic. A summary of the basic tasks in the evaluation is presented in Figure 10A-1.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

Fault trees representing the AFWS failure logic are presented in Section 10A.3.2. AFWS unavailability is based on the Boolean logic associated with the system fault trees. The fault trees are reduced to a list of cut-sets to identify the failure modes. Failure rate data (see Section 10A.3.4) are inserted to evaluate system unavailability. Although the failure data are derived primarily from NUREG-0611, secondary sources of failure data are WASH-1400 (Ref. 2), NUREG/CR-1362 (Ref. 3), and the Zion Probabilistic Safety Assessment (Ref. 6).

Fault tree development is consistent with the procedures and data available in NUREG-0611, and is limited to AFWS unavailability per demand. STP technical specifications allow continued operation of the plant with AFWS Train A out-of-service for an indefinite period of time. ~~A Therefore, this evaluation assumes that Train A would not be available at the time of AFW initiation. In reality, it is expected that Train A would have an availability similar to the other three trains of AFW. This assumption results in system unavailability being conservative by at least 33% for Cases 1 and 2.~~ Insert 1 In this appendix, unavailability is synonymous with unreliability, and the terms are used interchangeably. The importance of specific failure modes is examined, as are the interrelationships between and significance of hardware failure, test and maintenance outages, and human errors.

In addition to the quantitative evaluation described above, a qualitative evaluation is performed in a manner consistent with NUREG-0611. This evaluation rates system reliability based on design features such as equipment redundancy, manual versus auto actuation, single-point failure vulnerability, and technical specification limits on train outage time. The rating is done to compare the South Texas design with other U.S. plants using a Westinghouse nuclear steam supply system.

The success criteria used for LMFW, LMFW/LOOP, and LMFW/LOAC require that there be a minimum flow of ~~516~~ 540 GPM delivered to at least one steam generator. There are four AFW trains, each of which is dedicated to a single steam generator. Three of the AFW trains (Trains A, B, and C) are motor driven; the fourth (Train D) is turbine driven. Each AFW train is designed to deliver 550 GPM within one minute of actuation. Only the 'D' Train is operable under LOAC. Translating the success criteria in the preceding paragraph into failure criteria for fault tree development, "failure" reduces to "no flow to any SG" in the case of LMFW and LMFW/LOOP, and "no flow to SG D" in the case of LMFW/LOAC.

10A.1.5 Assumptions

Assumptions used in this evaluation are consistent with those specified in NUREG-0611. Specific assumptions used in the evaluation are:

1. Hardware and Human Error Failure Data

   The hardware and human error failure data, taken primarily from NUREG-0611, are used in the evaluation of basic events in this study. These data are presented in Section 10A.3.4.

INSERT 1 PAGE 6 SECOND PARAGRAPH

The Train A pump is identical in design and installation to the Train B and C pumps and thus would have similar operating characteristics and failure modes. Operational needs (minimize the potential for Steam Generator A to dry out) result in similar maintenance and outage practices. Thus, it is expected that Train A would have an availability similar to the other three trains of AFW.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

2. Test and Maintenance Outage Contribution

The study uses the calculational approach and the outage duration data presented in Table III-2 of NUREG-0611. These data are presented in Section 10A.3.4.

3. Power Availability

Consistent with NUREG-0611, the following assumptions are used to model power availability.

o Offsite power is assumed to have availability equal to 1.0 for Case I and zero for Cases II and III.

o Diesel generator availability for Case I is not relevant, since offsite power availability is 1.0. ~~For Case II, the availability of one diesel generator (Train C) is assumed equal to 1.0 (Ref. 1) and the other one (Train B) equal to 0.95 (Table 10A-2). For Cases II and III, offsite and/or emergency onsite AC power is assumed to be restored within a period of 2 hours.~~ Insert 2

o DC and battery-backed AC are assumed to have availability equal to 1.0 (Ref. 1) for all three cases.

4. Sample and Test Lines

The only sample or test line providing a significant flow diversion and/or leakage path is the pump test return line, which was considered in the human errors analysis. Since this 3-inch return line discharges to the AFWST at atmospheric pressure, significant flow may be diverted if this normally locked-closed valve is inadvertently left open after testing the pump.

5. Passive Piping Components

All piping components (e.g., pipe sections, flanges, reducers, etc.) are assumed available with a probability of 1.0. They are not considered in the fault tree development.

6. Degraded Component Failures

Degraded component failures are not considered in this evaluation; that is, components are assumed to operate properly or are treated as total failures. Component failures are assumed to occur instantaneously and completely.

INSERT 2 PAGE 7 POWER AVAILABILITY

For Case II, the unavailability of each diesel generator is calculated to be 4.8E-02/d (see Table 10A-1). For Case III, the components in Train D are independent of all AC power (the components are DC-powered). For Cases II and III, offsite and/or emergency onsite AC power is assumed to be available within a period of two hours.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

7. Uncoupling of Human Errors

This study assumes that test and maintenance activities are staggered. That is, redundant AFWS components are not tested by the same personnel on the same shift, but in general, tests and/or maintenance of redundant components involve time and/or personnel changes (e.g., different personnel and shifts, or the same personnel on a different day, etc.) In addition, a double-check procedure is assumed to assure the correct status of locked open valves after test and maintenance. This significantly reduces the probability of human error in two or more trains simultaneously. Given that test and maintenance activities are staggered and the use of a double check procedure, it is reasonable to assume that human errors for test and maintenance are uncoupled.

For the above reasons, the evaluation does not consider concurrent disabling of multiple trains because of human error in conjunction with test or maintenance to be a credible failure scenario.

8. Technical Specification

The auxiliary feedwater system design is evaluated in accordance with the STP Technical Specifications (Ref. 7).

Train A - ~~Out of Service,~~ Insert 3
Trains B, C, and D - Operable except for the scenarios illustrated in the fault trees in Section 10A.3.~~4~~2.

9. HVAC Support

The motor driven auxiliary feedwater pump rooms are cooled by safety-related HVAC units powered by their respective trains. The turbine driven pump room is cooled by a Train A HVAC unit, however, the turbine driven pump is qualified for operation following the loss of all HVAC. Consistent with NUREG-0611 methodology, HVAC support to the pumps is not considered in this evaluation.

10. Auxiliary Feedwater Storage Tank

~~Water from the AFWST is assumed to be available at all times.~~ The AFWST capacity is sufficient allow the RCS to remain at hot standby for 4 hours followed by a 10 hour cooldown ~~at~~ and an 8 hour soak period which point further RCS cooldown is performed by the residual heat removal system. If additional quantities are needed, water can be provided to the AFWST from the demineralized water storage tank, the condenser hot well, or an alternate

INSERT 3 PAGE 8 TECHNICAL SPECIFICATION

Train A - Availability is assumed to be degraded since there is no Technical Specification requirement on Train A.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

onsite source. The AFWST has level instrumentation with control room indication and annunciation to warn operators of low AFWST water inventory.

10A.2 SYSTEM DESCRIPTION

10A.2.1 Introduction

This AFWS description summarizes the more extensive description given in Section 10.4.9. Emphasis is placed on operation following the three loss of normal feedwater ~~extents~~ events covered by this reliability evaluation. The water for the AFWS is supplied from the auxiliary feedwater storage tank (AFWST). Water is supplied to the AFW inlet nozzles on the secondary side of the steam generators following a loss of normal feedwater flow as described in Section 10A.2.3. The AFWS serves as a backup to the main feedwater system during normal startup and shutdown operations.

The AFWS maintains the steam generators' water inventory during periods when the main feedwater system is unavailable. The system is a safety-related system. The AFWS is activated by an auto-start and is designed to deliver flow water to the steam generators within one minute. A minimum flow of ~~515~~ 540 gal/min must be supplied to any one steam generator on a loss of feedwater transient.

Four pump trains are utilized, each taking suction from the AFWST by separate suction lines. A P&ID for the AFWS is shown on Figure 10A-2. ~~Figure 10A-3 is a simplified reliability block diagram of this system. Figure 10A-4 is the detailed reliability block diagram from which the simplified reliability block diagram (Figure 10A-3) was derived. As mentioned earlier, this analysis conservatively assumes that Train A is out of service at the onset of the transient for Cases I & II. Subsequent discussions with respect to the quantitative analysis contained in this evaluation do not include of AFWS Train A.~~ Insert 4

Trains A, B and C of the AFWS have motor-driven pumps. Train D has a steam turbine pump. Initiation of the system is automatic upon actuation of two out of four low-low water level instrument channels in any steam generator. Crossover lines are provided downstream of the pumps to interconnect the trains when offsite power is available (case I) and are operable from the control room ~~for Case I~~. The valves connecting the crossover lines to the AFW pump discharge lines are normally closed, fail closed upon loss of instrument air and close on AFWS actuation. The crossover line valves can be opened manually from outside the control room. Insert 4A The air operated crossover valves are expected to remain operable from the control room after loss of offsite power for a period of time due to stored air in the instrument air receiver tanks. Thus, loss of offsite power does not result in instantaneous loss of crossover valve operation from the control room. However, since the instrument air system is a non safety-related system which is not immediately operable following LOOP, no credit for remote manual operation of the crossover valves is taken in the Case II evaluation. ~~The valves are assumed to be opened locally in the analysis.~~ For

Although no credit for crossover lines was assumed in the analysis,

INSERT 4 PAGE 9 THIRD PARAGRAPH

As mentioned earlier, this analysis conservatively assumes that Train A is out of service more than the other three trains. Therefore by increasing the unavailabi'ity due to maintenance of the Train A pump, the train's availability is degraded.

INSERT 4A PAGE 9 FOURTH PARAGRAPH

However, this action must be accomplished within thirty minutes after the initiating event. The ability to diagnose and implement this action outside the control room is highly unlikely; therefore, no credit is taken.

Case III (LMFW/LOAC), no crossover capability is assumed since there are three valves required to be opened locally to establish a flow path to a second steam generator.

Each AFW train provides feedwater to a single dedicated steam generator following an actuation signal. No hardware components are common between trains other than the aforementioned crossover lines. Each train, which consists of suction piping, pump/driver combination, discharge piping, cross-connect piping between trains and test and recirculation piping, is housed in a separate Seismic Category I compartment.

Pump pressure and flow testing is accomplished through a 3-inch diameter recirculation line connected to the 4-inch diameter main flow line downstream of the flow element. Flow through this line is ~~regulated~~ isolated by a normally locked-closed globe valve downstream of the recirculation connection to the main line. Opening this valve allows recirculation to the AFWST for pump testing.

10A.2.2 Component Description

1. Motor-Driven Pumps:

   The motor driven pumps are driven by AC-powered electric motors. Each motor receives power from an independent Class 1E power supply bus and its corresponding standby diesel generator. The pumps are horizontal, centrifugal, multistage units.

2. Turbine-Driven Pump:

   The turbine pump is a horizontal, centrifugal, multistage, noncondensing steam turbine-driven unit. A steam line connection is taken from the Safety Class 2 section of the Steam Generator D main steam line upstream of the main steam isolation valve. The turbine steam inlet line is provided with remote manual isolation and throttle valves. The turbine discharge steam exhausts directly to atmosphere. Overspeed of the AFW pump turbine automatically trips the turbine. Once this occurs, the mechanical overspeed trip latching mechanism must be manually reset in order to restore the turbine to an operable status. Power for all controls, valve operators, trip solenoid and other support systems is from the Train D Class 1E DC System. The major support system is the lube oil pump and cooling system. The lube oil pump is direct driven off the turbine shaft. The cooling water supply for the turbine lube oil cooler comes from a first stage bleedoff point on the turbine driven pump, passes through the lube oil heat exchanger, and ~~returns to the suction of the same pump.~~ is discharged to a drain.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

3. Piping and Valves

The safety-related AFWS piping is manufactured and installed in accordance with the ASME Code. Motor operated valves AF0048, AF0019, AF0065, MS0143, AF0085 and solenoid valve FV0143 are normally closed. Motor operated valve XMS0514 is normally open. Valves AF0065 AF0048 and AF0085 are AC powered. Valves MS0143, FV0143, AF0019, and XMS0514 are DC powered. (Insert 5)

4. Auxiliary Feedwater Storage Tank (AFWST)

The Seismic Category I auxiliary feedwater storage tank provides water to the AFW pumps. It is a concrete, stainless steel lined, ~~497,000~~ 518,000 gallon tank which has sufficient capacity to allow the RCS to remain at hot standby for 4 hours followed by a 10 hour cooldown and an 8 hour soak period ~~at which point further RCS~~ cooldown is performed by the residual heat removal system.

The AFWST is designed to withstand environmental design conditions, including floods, earthquakes, hurricanes, tornado loadings, and tornado missiles. The AFWST is designed so that no single active failure will preclude the ability to provide water to the AFW system. Each train has a dedicated suction line from the AFWST to the AFW pumps. The water level in the AFWST is indicated in the control room as well as at the auxiliary shutdown panel. A low level alarm is also provided in the control room.

10A.2.3 Emergency Operation

The AFWS is designed for automatic actuation in an emergency. Any of the following conditions automatically starts the three Class 1E motor-driven pumps:

1. Two out of four channels showing low-low water level in any steam generator

2. Safety injection signal

3. 4.16 kV bus undervoltage. The AFW pump is started in conjunction with diesel generator starting and load sequencing. Water is not automatically fed to the steam generator until condition 1 or 2 above exists.

The turbine-driven auxiliary feedwater pump starts automatically on any of the following signals:

1. Two out of four channels showing low-low water level in any steam generator

2. Safety injection signal

INSERT 5 PAGE 11 PIPING AND VALVES

Since motor-operated valves 7523, 7524, 7525 and 7526 may be in any initial position prior to AFW actuation, the valves are assumed to be closed prior to actuation.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

A one-inch bypass line with a normally closed solenoid operated valve (FV0143) and orifice is provided around the steam inlet valve (MS0143). This bypass valve (FV0143) opens upon receipt of either of the above signals to supply steam to the turbine and allow the turbine to reach governor control speed. After a time delay to allow governor control speed to be reached, the steam inlet valve is opened which allows rated steam flow to the turbine. This arrangement precludes an overspeed trip due to excessive steam flow prior to governor warmup. This bypass line is not dependent upon AC power to operate.

Automatic jog control of the auxiliary feedwater flow control valves operates to initially limit the maximum and minimum flow to any SG when the system is started by an automatic signal. The operator may assume manual flow control after resetting the system.

10A.2.4 Power Sources

The onsite AC Power Systems of Units 1 and 2 each consist of four major subsystems as follows.

1. 13.8 kV Auxiliary Power System (non-Class 1E)

2. 13.8 kV Standby Power System (non-Class 1E)

3. 138 kV Emergency Transformer Systems (non-Class 1E)

4. Onsite Standby Power System (Class 1E)

The arrangement of the AC Power Distribution Systems provides sufficient switching flexibility and equipment redundancy to ensure reliable power supply to the Class 1E and non-Class 1E plant loads during startup, normal operation, and shutdown following a design basis event.

The Onsite Standby Power Supply Systems of Units 1 and 2 each consist of three independent, physically separated, standby DGs supplying power to three associated load groups designated Train A, Train B, and Train C. Each load group consists of a 4.16 kV ESF bus and the electrical loads connected to that bus. The Onsite Standby Power Supply Systems of Units 1 and 2 operate independently of each other. Each standby DG and load group of a particular unit is also physically separated and electrically independent from the other two standhy DGs and their load groups.

Each 4.16 kV ESF bus is provided with switching that permits energization of the bus by five alternate sources:

1. The respective unit auxiliary transformer

2. No. 1 standby transformer

3. No. 2 standby transformer

4. Standby DG

5. 138 kV emergency transformer

When neither standby transformer nor the respective unit auxiliary transformer is available, the standby DGs supply the power required by the ESF loads to safely shut down the reactor. The 138 kV emergency transformer provides an additional means for supplying power to these systems if for any reason the above power sources are unavailable. The 138 kV emergency transformer is immediately available; however, its use is operator controlled.

Each standby DG is automatically started in the event of loss of offsite power or safety injection (SI) signal, and the required Class 1E loads connected to that ESF bus are automatically connected in a predetermined time sequence. Each standby DG is ready to accept load within 10 seconds after the start signal.

The Class 1E 125V DC battery systems of each unit consist of four independent, physically separated buses, each energized by two battery chargers and one battery. Emergency power required for plant protection and control is supplied without interruption by the batteries when the power from the Class 1E essential AC source is interrupted.

Each battery system also supplies power to inverters, two each for channels I and IV and one each for channels II and III. The inverters convert DC power to AC power at 118V AC, 60 Hz single phase for the vital instrumentation and protection system. The six vital AC busses supply power to instrumentation channels I, II, III, and IV which are associated with electrical trains A, D, B and C respectively. The two battery chargers associated with each of the four 125V DC busses are connected to separate Class 1E busses of the same train to enhance the reliability of each DC bus in the event that offsite power is lost. Following a loss of offsite power, AC power to the battery chargers is supplied by the standby DGs. Components in the turbine-driven train are powered from the Train D Class 1E DC system. Consistent with NUREG 0611, it is assumed that offsite and/or onsite AC power are restored within two hours to supply power to the battery chargers to restore the Train D battery to full capacity.

In the motor driven trains, the pump motors and valve actuators in each train are powered by the corresponding Class 1E train. Instrumentation and controls in each train are provided by DC or AC power from its associated Class 1E train.

10A.2.5 Testing

The AFWS inservice testing and inspection frequencies assumed in this analysis are described below. The frequencies are in agreement with Reference 7 with the exception of automatic valve position verification which is indicated as at least once every 31 days in the Technical Specifications. This increase in test frequency serves to decrease the auxiliary feedwater system hardware related unavailability (~~Table 10A-6~~) without affecting human error and test and maintenance related unavailability. The calculated total auxiliary feedwater system unavailabilities are therefore conservative.

Component Test	Test Frequency
o Motor Driven Pumps Operability	Recirculate to AFWST at least once every 92 days
o Turbine Driven Pump Operability	Recirculate to AFWST at least once every 92 days
o Automatic Valve Position	Verify position at least once every 92 days
o Non-Automatic Valve Position	Verify position at least every 31 days
o Automatic Valve Actuation	Verify actuation to correct position during each refueling shutdown
o Motor and Turbine Driven Pump Actuation	Verify pumps start on actuation signal during each refueling shutdown
o Train Operability	Verify ability to establish flow path to each steam generator following cold shutdowns greater than 30 days

10A.3 METHODOLOGY

This section presents the step-by-step procedure followed in performing the AFWS quantitative reliability evaluation.

10A.3.1 System Review

In the first step, the various drawings, P&IDs, and schematics representing the AFWS were examined. Special attention was given to identifying:

1. Inst mentation systems required for system actuation
2. Fluid systems connected directly or indirectly to the AFWS
3. Power sources for each component
4. Any obvious single-point vulnerabilities.

The reliability information described in Appendix III of NUREG-0611 was then appraised, and AFWS studies of other facilities were reviewed. With this information, the evaluation boundaries were established.

10A.3.2 Fault Tree Development and Quantification

The reliability block diagram for the AFWS (Figure 10A-4) was constructed. A simplified version of the reliability block diagram is provided in Figure 10A-3. Fault trees (Figures 10A-5 through 10) are constructed from the reliability block diagram and the P&IDs. These trees include the occurrence of individual component failures. Fault trees for test and maintenance, and human error after test and maintenance are also constructed (Figures 10A-11 and 12). From these detailed fault trees, simplified trees were constructed. The simplified trees contain the same system information, but basic events that are under a single OR-gate or AND-gate are combined into a composite event (hereafter referred to as a supercomponent). By using simplified fault trees, a tree containing a manageable number of events is constructed, yet the fault propagation within and between systems is preserved. When consolidating basic events into composite events, care is taken to assure that no basic event appearing in a composite event appears elsewhere in the tree. Definitions of composite events are given in Section 10A.3.4. Reduced fault trees (Figures 10A-13 and 14) are constructed to provide a simple illustration of the overall logic configuration for each case, but are not used in the quantification process.

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Quantification of the AFWS fault trees is done by two computer programs, FTAP and IMPORTANCE. Refer to Section 10A.3.5 for a description of these computer programs.

Three distinct contributions to AFWS unavailability are quantified in the evaluations. Unavailability due to random hardware failures is quantified using the AFWS hardware-related fault tree (Figures 10A-5 through 10). AFWS unavailability resulting from system downtime for test and maintenance is also quantified. In addition, system unavailability resulting from human errors associated with test and maintenance activities is quantified. The total AFWS unavailability (per demand) is the sum of the unavailabilities due to random hardware failure, test and maintenance, and human error.

10A.3.3 Common Cause Failure Evaluation

The evaluation and design provisions of common cause factors such as floods (Section 3.4), fires (Section 9.5.1), earthquakes (Section 3.2), sabotage and high energy pipe breaks (Section 3.6) are outside the scope of this AFWS unavailability study. The only common cause factor considered is that resulting from human errors during test and maintenance.

This evaluation assumes that human errors are statistically independent. Tests and maintenance of redundant components will involve time and/or personnel changes (e.g., different personnel and shifts or the same personnel on a different day, etc.). This assumption is also supported by Technical

INSERT 6 PAGE 15 FAULT TREE DEVELOPMENT AND QUANTIFICATION

Fault trees are constructed from the P&IDs. These trees include component failures (mechanical and control circuit), test and maintenance outages, and human errors (from testing, maintenance and accident response). The fault trees are constructed using a segment level approach. A segment is defined as the piping section between two points of intersection with other pipe segments. Failures within the segments are characterized and developed into the fault trees. The fault trees developed for each scenario are presented in Figures 10A-3 to 10A-5. A table to identify the codes used in the fault trees is shown in Table 10A-5.

Quantification of the AFWS fault trees is done by two computer codes, GRAFTER and WESCUT. Refer to Section 10A.3.5 for a description of these codes.

Each fault tree is quantified. The results of this quantification include total system unavailability and the failure combinations (cutsets) that contribute to this unavailability.

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Specification limitations on plant operation associated with coincident test and maintenance activities that reduce train availability to an unacceptable level.

10A.3.4 Failure Data

10A.3.4.1 Description of Supercomponents

The detailed reliability block diagram illustrated on Figure 10A-4 shows groupings of equipment within each train of the AFWS that function as an identifiable unit, and whose failure logic can be represented in a fault tree as basic events connected under a single OR-gate or AND-gate. These equipment groupings, referred to as supercomponents, can be used to generate simplified fault trees in which the supercomponents are used to represent basic events rather than each individual piece of equipment.

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The following symbols represent supercomponent abbreviations used in Figure 10A-4.

MB1, MC1, MD1 = hardware-related failure of motor-driven and turbine-driven AFW pumps and associated valves in Trains B, C, and D, respectively upstream of the crossover valves.

MB2, MC2, MD2 = hardware-related failure of flow elements and associated valves on the steam generator side of AFW Trains B, C, and D, respectively downstream of the crossover valves.

MD3 = hardware-related failure of valves controlling steam supply to turbine-driven AFW pump for Train D.

DG12, DG13 = hardware-related failures and test or maintenance unavailabilities causing inability of diesel generators for Trains B and C (respectively) to start.

SB = Failure of both automatic and manual backup actuation signals for Train B (ASB, MSB on Figure 10A-4).

SC = failure of both automatic and manual backup actuation signals for Train C (ASC, MSC on Figure 10A-4).

SD = failure of both automatic and manual backup actuation signals for Train D (ASA, MSD on Figure 10A-4).

BVLC = CVLC = DVLC = Human error related unavailability due to operator's failure to restore the block valves on the pump suction or discharge lines following maintenance.

~~Table 10A-1 enumerates the individual pieces of equipment included within each of the above-listed supercomponent groupings. Equipment numbers shown in the table correspond to those shown on Figure 10A-4~~.

10A.3.4.~~2~~1 Failure Rate Data

10A.3.4.~~2~~1.1 Hardware

Hardware-related failure data used in this evaluation are presented in Table 10A-~~2~~1. Unless otherwise indicated, all failure data are taken directly from NUREG-0611.

10A.3.4.~~2~~1.2 Human Error

Since the AFWS is automatically actuated, the treatment of human error is limited to mispositioning manual valves based on the human error probabilities given in NUREG-0611. Valves considered are ~~AF0024, AF0053, AF0073, AF0012, AF0059, and AF0078 and~~ the manual valves in the recirculation lines to the AFST. ~~which are not shown on the detailed reliability block diagram (Figure 10A-4)~~.

During maintenance, valves AF0024, AF0053, AF0073, AF0012, AF0059, AF0031, AF0041 and AF0078 must be closed in order to drain the water from pumps. They may inadvertently be left closed. (Insert 7) ~~A failure rate of 5 x 10-3 per demand is used in this calculation~~. During the testing of a pump, the manual valve in the recirculation line must be open. The manual valve may inadvertently be left open. A failure rate of 5×10^{-3} per demand is used in this calculation. For Train D, the trip and throttle valve overspeed trip mechanism must be manually reset after maintenance or a previous overspeed trip. A failure rate of 5×10^{-3} per demand is used for this calculation.

10A.3.4.~~2~~1.3 Test and Maintenance

The approach presented in NUREG-0611 is used. Testing and maintenance (T&M) activities that remove components and/or the system from service can be significant contributors to overall AFWS unavailability. The most common forms of valve maintenance performed during power operation are packing adjustments and repairs to the MOV and AOV control circuits and operators. Nearly all of these activities are performed with the valve in the safe position during the maintenance interval. Therefore, maintenance of MOVs and AOVs is not considered to contribute to valve unavailability, ~~except for the stop check isolation valves. Since the valves are normally closed, maintenance would disable any local control circuit, effectively failing that portion of the train.~~ Check valves and manual valves are expected to require very little maintenance. The low test and maintenance impact on this part of the AFWS is the basis for not including a human error contributor to unavailability for the manual valves in the individual steam generator flow paths. Although testing and maintenance contributions are not treated for the valves associated with the branch flowpaths to a specific steam generator, unavailability from testing and maintenance of the pump subsystem is treated.

INSERT 7 PAGE 17 HUMAN ERROR

Due to the fact that this failure mode will only occur after maintenance, that procedures require the position of these valves be double checked after maintenance as well as periodically checked (every 31 days), and that flow tests on the pump are required after maintenance, this failure mode was assumed to be insignificant.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

In the subsystem part of the fault tree, testing and maintenance are treated as a distinct composite basic event. Unavailability due to T&M is calculated using outage durations from NUREG-0611 and the test frequencies as presented in Section 10A.2.5. T&M unavailabilities for each train are comprised of contributions due to testing of the train and maintenance of the pump~~s and maintenance of the stop check isolation valve. The sum of these contributions constitute the total test and maintenance unavailability of a particular train. T&M unavailabilities are provided in Table 10A-3.~~ (Insert 8)

STP Technical Specifications (Reference 7) do not allow coincident test or maintenance of components of more than one AFW pump train. Therefore, the analysis explicitly accounts for maintenance in one train and ~~coincident hardware related failures in the remaining two trains.~~ not in the other trains by use of the "NOT" gate.

~~10A.3.4.3 Computed Unavailabilities for Composite Events~~

~~The unavailability per demand of each of the supercomponents described in Section 10A.3.4.1 is calculated by substituting the failure rate data in Tables 10A-2 and 10A-3 into the supercomponent expressions given in Table 10A-1. Unavailability per demand for each supercomponent grouping is summarized in Table 10A-4.~~

10A.3.5 Computer Programs

The following ~~Bechtel Power Corporation~~ Westinghouse Electric Corporation computer programs are used in performing the evaluation of auxiliary feedwater system unavailability.

~~10A.3.5.1 FTAP~~ (Insert 9)

~~This program is used to generate fault tree cut sets. Minimal cut set families are generated by one of three processing methods: (1) top-down, (2) bottom up, or (3) "Nelson" method. FTAP results have been verified by comparison with hand calculations.~~

~~10A.3.5.2 IMPORTANCE~~

~~This program uses the minimum cut sets generated by FTAP and basic event data, failure rates and fault duration times to determine system and subsystem unavailability. This program has been verified by comparison with hand calculations.~~

10A.4 RESULTS OF THE RELIABILITY EVALUATION

The results of the AFWS reliability evaluation are provided in two forms. The first is a general qualitative evaluation based on system design features. The second part is a quantitative evaluation based on the fault tree representation of the AFWS design.

INSERT 8 PAGE 18 TEST AND MAINTENANCE

In order to decrease the availability of Train A relative to the other trains, the maintenance outage time for the Train A pump was increased from 19 hours to 336 hours (2 weeks) per maintenance activity. This assumption is in general agreement with the Technical Specifications and is conservative. T&M unavailabilities are provided in Table 10A-2.

INSERT 9 PAGE 18 COMPUTER PROGRAMS

10A.3.5.1 GRAFTER

GRAFTER is a computer code written in FORTRAN and ASSEMBLER languages to construct fault trees interactively. It is used in conjunction with the WESCUT code to carry out fault tree analysis from the construction stage to the quantification.

The GRAFTER code can be used to construct, store, update and print fault trees interactively. GRAFTER can construct fault trees containing up to 2064 boxes (gates or basic events). A menu of commands is provided to be used to construct the fault trees. The computer keyboard is used to move to different locations within the fault tree.

10A.3.5.2 WESCUT

WESCUT is a computer code written in FORTRAN77. It identifies the minimal cutsets of a fault tree. It also quantifies the mean failure probability and variance of the top event and other specified lower level events.

For each gate specified when generating the input for cutset identification, the code will identify and print the cutsets. The cutsets are listed in order of decreasing probability. The mean probability and variance for the requested gate or gates is also calculated and printed.

The code can quantify fault trees containing up to 320 gates and 320 basic events.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

10A.4.1 Qualitative Evaluation

In the qualitative characterization of the reliability of AFW systems, NUREG-0611 assumes that the traits identified in Table 10A-3 exist for specific reliability ratings. These characterizations are reviewed for each of the three initiating events considered in NUREG-0611.

10A.4.1.1 Loss of Main Feedwater

In NUREG-0611, some of the plants whose AFWS are found to have low reliability have single-point vulnerabilities. This is due to a single manual valve through which all AFW flow passes, where a human error of failing to reopen the valve after maintenance is found to be the dominant failure contributor. The South Texas design has four lines supplying water to the four pump trains. Thus, no single human error could disable the system. The only single failure that could disable the system is rupture of the auxiliary feedwater storage tank. The unavailability due to this failure is extremely small and this event would be readily detected by tank level indication and low level alarms in the main control room.

The NUREG-0611 plants classified in the high-reliability range for this transient generally have three AFW pumps (two motor and one steam turbine driven) which are actuated automatically, with manual backup signal.

Since the South Texas AFWS design includes all these features and control room actuated crossover capability, and has four AFW pumps it receives a high reliability rating for this transient ~~even though Train A is assumed out of service~~.

10A.4.1.2 Loss of Main Feedwater with Loss of Offsite Power

The major difference between this and the previous LMFW event is that offsite power sources are not available and the system must rely on onsite power sources (i.e., diesel generators, batteries and steam).

The reliability of various AFWS designs for this event are generally found to be quite similar to those for the previous initiating event (LMFW). The major difference is that onsite AC power sources are required and the potential impact of degrading these power sources (e.g., the loss of one or more emergency diesel-generators) on the AFWS reliability is evaluated.

Compared to other Westinghouse NSSS plants evaluated in NUREG-0611, the South Texas AFWS contains a greater number of motor driven pump trains (3 versus the typical 2)~~; however, this analysis conservatively assumes that one train is out of service~~. This redundancy reduces the likelihood of AFWS unavailability during a LMFW/LOOP event.

For this reason and the local manual crossover capability, the qualitative reliability rating given the South Texas AFWS is comparable to that of other high reliability Westinghouse NSSS plants as reported in NUREG-0611.

10A.4.1.3 Loss of Main Feedwater with Loss of All AC Power

The major feature of this initiating event is the total dependency of the AFWS on steam power. Low and medium reliability classifications under this event are generally due to systems having AC power dependencies in the steam turbine-driven pump train. Such dependencies may include lube oil cooling, AC power to steam turbine admission valves, or air-operated valves which fail closed on loss of air. Those systems characterized as having a relatively high reliability are usually automatically actuated and have no potentially degrading AC power dependencies (except HVAC).

When comparing the STP AFWS to the NUREG-0611 plants which have a high reliability characterization, the STP design has a comparably high reliability because the turbine pump train has no AC dependency in order to function. However, since no credit is taken for the steam turbine driven pump to serve other than SG-D (due to absence of control room activated crossover capability and the requisite manual actuation of the stop check isolation valves in the other trains), the South Texas AFWS is rated slightly lower than some of the highest rated other Westinghouse NSSS plants as reported in NUREG-0611 (refer
10A-6 to Figure ~~10A-15~~). As noted earlier, it is possible to manually initiate crossover from outside the control room if the need should ever arise. The turbine driven pump is qualified for operation in the environment resulting from a loss of HVAC.

10A.4.1.4 Qualitative Comparison with Other Designs

10A-6 Figure ~~10A-15~~ is a reproduction of the reliability characteristic chart presented in NUREG-0611 for AFWS designs in plants using the Westinghouse NSSS. An added row presents the results of a qualitative evaluation of South Texas AFWS reliability. The figure shows the relative reliability ranking of South Texas AFWS for each of the three cases studied and compares these results to those obtained by the NRC. This qualitative evaluation is included to complement the results of the quantitative analysis.

10A.4.2 <u>Quantitative Evaluation</u>

The quantitative characterization of the South Texas AFWS reliability is developed using the methods and data provided in NUREG-0611. The system's conditional unavailability is quantified for three initiating events: LMFW, LMFW/LOOP and LMFW/LOAC. System unavailability is associated with hardware failure, human error, and test and maintenance downtime.

10A.4.2.1 Quantitative Results

The results of the quantitative evaluation are presented in Table 10A-~~6~~4.
~~Table 10A-6 identifies the individual contributions of hardware failure, human~~

~~error, and test and maintenance to the AFWS unavailability for three initiating events (LMFW, LMFW/LOOP and LMFW/LOAC).~~ System unavailability for the LMFW and LMFW/LOOP events is approximately ~~2 x 10^{-5} and 4 x 10^{-5}~~ 3.2E-6 and 3.6E-5 per demand, respectively. Even for the LMFW/LOAC event, where all AC power is lost and the system is totally dependent on the steam turbine driven pump to supply water to the steam generators, the system unavailability ~~of~~ is approximately ~~8~~ 4.5 x 10^{-2} per demand ~~is good~~. These results demonstrate that the South Texas AFWS design is reliable when compared with other designs and the USNRC acceptance criteria of 10^{-5} to 10^{-4} per demand for the LMFW transient (Ref. 5)~~, particularly when one considers that the South Texas AFWS analysis excludes one train from consideration~~.

10A.4.2.2 Failure Modes

There are many possible combinations of random hardware failures, component unavailabilities due to test or maintenance, and human error which can result in the unavailability of the AFWS. Since each system component (e.g., pump, valve) generally has a different failure rate, there are certain combinations of failure modes that contribute significantly more to the total unavailability of the AFWS than others. These are the most significant failure modes. Unavailability per demand of each of the possible combinations of failure modes is computed ~~by quantifying each of the minimal cut-sets generated~~ by the computer code ~~"FTAP"~~ 'WESCUT'. Once the unavailabilities associated with each minimal cut set have been computed, their percentage contribution to total AFWS unavailability can be determined, and significant failure modes identified.

The AFWS reliability evaluation uses the computer code ~~FTAP~~ WESCUT to generate minimal cut-sets based on Boolean expressions ~~for the random hardware failure, test and maintenance, and human error fault trees shown in Section 10A.3.2~~. In general, higher-order cut-sets contribute less to the top event than do lower order cut-sets if the failure rates of the basic events are similar. With ~~three~~ four separate pump trains, the aggregate of ~~third~~ fourth-order cut-sets (representing various combinations of pump and valve failures affecting different trains) contribute significantly to the failure of the entire AFWS. Higher order cut-sets (e.g., ~~fourth~~ fifth-order) involve other basic events with much smaller failure rates, and their aggregate contribution to total AFWS unavailabilty is numerically small.

The following sections present a summary of failure modes associated with the LMFW, LMFW/LOOP, and LMFW/LOAC failure scenarios.

10A.4.2.2.1 Loss of Main Feedwater (Case 1)

~~For the LMFW scenario (Case 1), the "FTAP" code produces 1 first-order cut-set, 0 second-order cut-sets and 10 third-order cut-sets for the hardware failure fault tree shown on Figures 10A-6 through 10A-10. The "FTAP" run for the test and maintenance fault tree shown on Figure 10A-11 results in no first-order cut-set and 34 third-order cut-sets. The human error fault trees~~ delete

(Figure 10A-12) produce no first or second-order cut-sets, and 23 third-order cut-sets. From Table 10A-6, it can be seen that the hardware failure cut-sets (in aggregate) contributes about 25 percent to total AFWS unavailability; human error cut-sets about 51 percent; and test and maintenance cut-sets about 24 percent. Within each group of cut-sets, no one particular failure mode could be characterized as a true dominant contributor. The single first-order cut-set for the hardware failure fault tree represents unavailability of the AFWST which is numerically small (approximately 3.6×10^{-8} per demand). To simplify the discussion, AFWST unavailability (here and in the following two sections) will be treated as a hardware-related failure. Third-order cut-sets for the hardware failure fault tree represent various combinations of failures of pumps and valves in different AFW trains. Because failure rates assigned to pumps and valves are numerically similar, the numerical values of the 10 third-order cut-sets are close to one another, with no single contributor being dominant. Human error is the largest contributor to AFWS unavailability for Case I (LMFW).

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10A.4.2.2.2 Loss of Main Feedwater Coincident with Loss of Offsite Power (Case II)

For the LMFW/LOOP scenario (Case II), the "FTAP" code produces 1 first-order cut-set, 0 second-order cut-sets, and 14 third-order cut-sets for the hardware failure fault tree shown on Figures 10A-6 through 10A-10. The greater number of hardware failure cut-sets for Case II versus Case I is attributable to combinations of pump and valve failures in addition to failure of the diesel generator to start (diesel generator operation is required for Case II but not Case I). From the test and maintenance and human error fault trees (Figures 10A-11 and 12), a combined total of 0 first-order cut-sets, 0 second-order cut-sets, and 65 third-order cut-sets are generated by "FTAP". Considering the aggregate contribution of hardware failure, test and maintenance, and human error to total AFWS unavailability for Case II, hardware failure cut-sets contributes 40 percent to the total unavailability, test and maintenance contributes about 15 percent, and human error contributes 45 percent (refer to Table 10A-6). As for Case I, no cut-sets belonging to the test and maintenance group are dominant contributors. In the category of hardware failure, various combinations of failures of one diesel generator affecting one train and valve failures disabling a second and third train are responsible for 66 percent of the total unavailability attributable to hardware-related failures. For the human error contribution to total AFWS unavailability, human error affecting one train, plus failure of one diesel generator disabling a second train, and a valve failure disabling a third train represent 39 percent of the total human error contribution to AFWS unavailability.

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From the quantitative analysis of Case II, it is concluded that failure of diesel generators to start, hardware failures associated with valves in the pump discharge lines, and human error are the most important factors affecting AFWS unavailability.

INSERT 10 PAGE 21

10A.4.2.2.1 Loss of Main Feedwater (Case I)

For the LMFW scenario (Case I), the AFWS unavailability was calculated as 3.2E-06/d. The dominant contributors to system unavailability are fourth-order cutsets in which motor-driven pumps B and C fail due to hardware faults, Train A pump is unavailable due to maintenance, and a motor-operated valve in Train D fails. Other contributors include combinations of a pump failure (either Train B or C), a motor-operated valve failure in Train D, a motor-operated valve failure in Train B or C (the opposite train in which the pump failure occurred) and the Train A pump unavailable due to maintenance.

Each fourth order cutset described above has a cutset probability of approximately 4E-08 and contributes approximately 1.3% to the system unavailability. Because each individual cutset has a probability close to the other cutsets, no single cutset contributor is dominant.

However, when the basic events are examined, approximately 95 percent of the failures of the system can be attributed to the Train A pump's unavailability due to maintenance in combination with other failures. (This result is expected based on the restrictions applied in the analysis.) Other dominant basic events are the failure to start and run of motor-driven pumps in Trains B and C (30.1%) and the motor operated valves (failing to open) in the discharge lines of Trains B and C (25.3%). (These basic events are present in cutsets that contribute 30.1 percent and 25.3 percent respectively to the total system unavailability.)

One first order cutset was determined for the LMFW event (failure of the AFST). However, the failure probability is 3.6E-08 and its contribution to system unavailability is approximately one percent. Thus, the conclusion can be drawn from this analysis that the South Texas AFWS is highly reliable in the event of a loss of main feedwater.

INSERT 11, PAGE 22

10A.4.2.2.2 Loss of Main Feedwater Coincident with a Loss of Offsite Power

For the LMFW/LOOP scenario (Case II) (unavailability equal to 3.6E-05/d), most of the failure combinations involve pump or valve hardware failures coupled with failure of the diesel generators (diesel generator operation is required during a loss of offsite power). The top six cutsets contributing to AFWS unavailability are combinations of two diesel generators failing (for Trains B and C) with a valve failure in Train D and the Train A pump unavailable due to maintenance. The top four cutsets have a probability of 1.66E-06 and contribute approximately 4.6 percent to the total system unavailability. The remaining two cutsets have probabilities of 1.19E-06 and 1.17E-06 and contribute approximately four percent to the unavailability. Other failure combinations determined in the evaluation include failure of three diesel generators coupled with a failure in the steam turbine driven pump Train D.

When the basic events involved in these failures are examined, the dominant contributors are the diesel generators (72.3 percent) followed by Train A motor driven pump unavailable due to maintenance (59.8%) and the motor operated valves in Train D (17.3 percent each). These basic events are coupled with other failures in cutsets that contribute that percentage to the system unavailability.

From this analysis, it can be concluded that the failure of the diesel generators and not an actual AFWS failure is the most important factor affecting AFWS availability following a loss of main feedwater coincident with a loss of offsite power.

10A.4.2.2.3 Loss of Main Feedwater Coincident with Loss of All AC Power (Case III)

AFWS unavailability for the LMFW/LOAC scenario (Case III) (unavailability = 4.5E-3/d) is attributable to any hardware-related failure, test or maintenance unavailability, or human error that could disable Train D, since this is the only AFW train which can operate independently of AC power. The percentage contribution of each to total AFWS unavailability for Case III is as follows ~~(see Table 10A-6)~~: ~~hardware-related failure, 52 percent; test and maintenance, 14 percent;~~ and ~~human error, 34 percent~~. Insert 12

10A.4.2.3 Conclusions

The quantitative evaluation of auxiliary feedwater system reliability concludes the system reliability is high and in accordance with the guidelines contained in Standard Review Plan 10.4.9, Rev. 2. The qualitative evaluation also shows the system reliability to compare favorably with that of other plants described in NUREG 0611. With the exception of the loss of the AFWST (an extremely low probability event), no single point vulnerabilities were identified in the system. Furthermore, no second order cut-sets were identified and no AC dependencies were found in Train D.

10A.5 REFERENCES

1. NUREG-0611 "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants" by USNRC January, 1980.

2. WASH-1400 "Nuclear Reactor Safety Study" Appendix III, Failure Data, by USNRC October, 1975.

3. NUREG/CR-1362, "Data Summaries of Licensee Event Reports on Diesel Generators at U.S. Commercial Nuclear Power Plants; January 1, 1976 to December 31, 1978", March 1980, by E.G.&G. Idaho, Inc.

4. NUREG-0452, Revision 4, Standard Technical Specifications for Westinghouse Pressurized Water Reactors, USNRC, Fall 1981.

5. NUREG-0800, USNRC Standard Review Plan, Section 10.4.9, July 1981.

6. Zion Probabilistic Safety Assessment; Pickard, Lowe, & Garrick; Newport Beach, CA. September, 1981.

7. Dewease, J. G. (Houston Lighting and Power) to Thompson, H. L. (USNRC), "South Texas Project Electric Generating Station Technical Specifications, Offsite Dose Calculation Manual, Process Control Program," ST-HL-AE-1271, June 17, 1985

INSERT 12 PAGE 23

10A.4.2.2.3 Loss of Main Feedwater Coincident with a Loss of All AC Power

Train D motor-operated valve failure (62%), operator error in failing to reset the trip and throttle valves or failing to close a manual valve after test (22%) and the unavailability of the turbine driven pump due to maintenance (11%).

TABLE 10A-1

CONSTITUENTS OF SUPERCOMPONENTS[a]

MB1 = AF0095 + AF0053 + MPA02 + AF0058 + AF0059

MC1 = AF0096 + AF0073 + MPA03 + AF0091 + AF0078

MD1 = AF0093 + AF0024 + MPA04 + AF0011 + AF0012

MB2 = FV7524 + AF0061 + FE7524 + AF0065 + AF0120

MC2 = FV7523 + AF0080 + FE7523 + AF0085 + AF0121

MD2 = FV7526 + AF0014 + FE7526 + AF0019 + AF0122

MD3 = Governor Valve + XMS0514 + MS0143

SB = ASB x MSB

SC = ASC x MSC

SD = ASA x MSD

delete

(a) For general description of supercomponents, refer to Section 10A.3.4.1 and Figure 10A-4.

Table 10A-~~2~~ 1

Component Basic Event Failure Probabilities*

1.	Check valve. Failure to open. AF0122, AF0120, AF0121, AF0119 AF0011, AF0058, AF0091, AF0036	1×10^{-4}/d(a)
2.	Automatic actuation signal. ASA, ASB, ASC	7×10^{-3}/d
3.	Manual backup signal. (Conditional probability given automatic signal fails) MSB, MSC, MSD	1×10^{-2}/d
4.	Flow element plugging. FE7526, FE7524, FE7523, FE7525 (This failure rate was taken from WASH-1400 for plugging of the flow orifice Table III 4-1).	3×10^{-4}/d
5.	Gate valve. Plugging contribution. AF0014, AF0012, AF0024, AF0093, AF0061, AF0059, AF0053, AF0095, AF0080, AF0078, AF0073, AF0096 AF0041, AF0043, AF0031, AF0094	1×10^{-4}/d
~~6.~~	~~Air operated valve (crossover valves)~~ ~~FV7515, FV7516, FV7518,~~ ~~Case I: (Control room operation)~~	
	~~Mechanical components~~	~~3×10^{-4}/d~~
	~~Plugging contribution~~	~~1×10^{-4}/d~~ delete
	~~Operator failure (Manual backup signal)~~	~~1×10^{-2}/d~~
	~~Local control circuit~~	~~6×10^{-3}/d~~
	~~Total~~	~~1.64×10^{-2}/d~~
	~~Case II: (Local Manual Operation)~~	
	~~Plugging contribution~~	~~1×10^{-4}/d~~
	~~Local manual actuation~~	~~2.34×10^{-2}/d** (Ref. 6)~~
	~~Total~~	~~2.35×10^{-2}/d~~
~~7.~~ 6.	Solenoid valve failure FV0143	
	Mechanical components	1×10^{-3}d
	Plugging contribution	1×10^{-4}/d (WASH-1400)
	Local control circuit	6×10^{-3}/d
	Total	7.1×10^{-3}/d

* Data Source, NUREG-0611 except as noted.
** The median value presented here was calculated from the mean value and the variance contained in Reference 6.

Table 10A-~~2~~ 1 (Continued)

Component Basic Event Failure Probabilities

~~8~~ 7. Motor-operated valve, failure to open.
AF0019, AF0065, AF0085, MS0143, AF0048
FV7523, FV7524, FV7526, FV7525

Mechanical components	1×10^{-3}/d
Plugging contribution	1×10^{-4}/d
Control circuit (local)	6×10^{-3}/d
Total	7.1×10^{-3}d

~~9~~ 8. Motor-driven pump.
MPA02, MPA03, MPA01

Mechanical components	1×10^{-3}/d
Control circuit (local)	7×10^{-3}/d
Total	8×10^{-3}/d

~~10~~ 9. Turbine-driven pump.
MPA04

Mechanical Components	1×10^{-3}/d
Overspeed Trip:	
Solenoid Valve Failure (See Item 7)	7.1×10^{-3}/d
Orifice Plugged	3×10^{-4}/d
Total	8.4×10^{-3}/d

~~11~~ 10. Motor-operated valve.
XMS0514 Plugging contribution. 1×10^{-4}/d

~~12~~ 11. Auxiliary feedwater storage tank (unavailability per demand estimated from that given for condensate storage tank in WASH 1400) 3.6×10^{-8}/d

~~13~~ 12. Diesel generator.

DG13	~~0~~ 4.8×10^{-2}/d
DG12	4.8×10^{-2}/d
DG11	4.8×10^{-2}/d

The hardware failure rate of diesel-generators (4×10^{-2}/demand) is taken from Ref. 3. Total diesel generator 12 unavailability is the sum of unavailabilities due to hardware failure, test, and maintenance; i.e., total unavailability = $4 \times 10^{-2} + 1.9 \times 10^{-3} + 6.4 \times 10^{-3} = 4.8 \times 10^{-2}$ (Refer to Table 10A-3).

~~14~~ 13. Governor Valve
Plugging Contribution 1×10^{-4}/d

(a)d = demand

4759c/0181c

Table 10A-~~2~~ 3

Unavailability of Components Due to Testing or Maintenance

Component	Hrs/ Test	Test/ Yr	Hrs/ Maint.	Q_{test}(a)	Q_{maint}(b)
Pump B,C,D	1.4	4	19	6.39×10^{-4}/d(c)	5.8×10^{-3}/d
Valve			7	---	2.1×10^{-3}/d
Diesel Generator	1.4	12	21	1.9 x 10-3/d	6.4×10^{-3}/d
Pump A	1.4	4	336	6.39×10^{-4}/d	1.03E-1/d (d)

(a) $Q_{test} = \frac{(\#\ hrs/test)(\#tests/year)}{(\#hrs/year)}$ [See NUREG-0611, Table III-2]

(b) $Q_{maint.} = \frac{(0.22)(\#hrs/maintenance\ activity)}{720}$ [See NUREG-0611, Table III-2]

(c) d = demand

(d) see explanation in Section 10A.3.4.1.3

4759c/0181c

Table 10A-4

Composite Event Unavailability (per Demand)

	Supercomponent	Probability of Failure
Hardware:	MB1	8.4×10^{-3}
	MC1	8.4×10^{-3}
	MD1	8.8×10^{-3}
	MB2	1.47×10^{-3}
	MC2	1.47×10^{-3}
	MD2	1.47×10^{-3}
	MD3	7.3×10^{-3}
	SB	7×10^{-5}
	SC	7×10^{-5}
	SD	7×10^{-5}
Human Error:	BVLC	1×10^{-2}
	CVLC	1×10^{-2}
	DVLC	1×10^{-2}

delete

Table 10A-3

AFWS Qualitative Reliability Characterization Traits

Low-Reliability	Medium-Reliability	High-Reliability
a. Manual system actuation	a. Auto actuation with manual backup	a. Auto actuation with manual backup
b. Two-pump system	b. System with more than two pumps	b. System with more than two pumps and reduced AC dependence
c. Single-point vulnerabilities present	c. Single-point vulnerabilities may be present	c. No single-point vulnerabilities present
d. Technical Specifications permit unlimited outage time for system maintenance, tests, etc.	d. Technical Specifications permit unlimited outage time	d. Technical Specifications do not allow unlimited outage time

Table 10A-~~8~~ 4

AFWS Unavailability (per Demand)

	LMFW	LMFW/LOOP	LMFW/LOAC
~~Hardware Failure~~	~~4.60×10^{-6}~~	~~1.57×10^{-5}~~	~~3.06×10^{-2}~~
~~Human Error~~	~~9.33×10^{-6}~~	~~1.80×10^{-5}~~	~~1.99×10^{-2}~~
~~Test and Maintenance~~	~~4.28×10^{-6}~~	~~5.87×10^{-6}~~	~~8.52×10^{-3}~~
Total	~~1.82×10^{-5}~~	~~3.96×10^{-5}~~	~~5.90×10^{-2}~~
	3.23E-6	3.57E-5	4.54E-2

4759c/0181c

INSERT 13 TABLE 10A-5

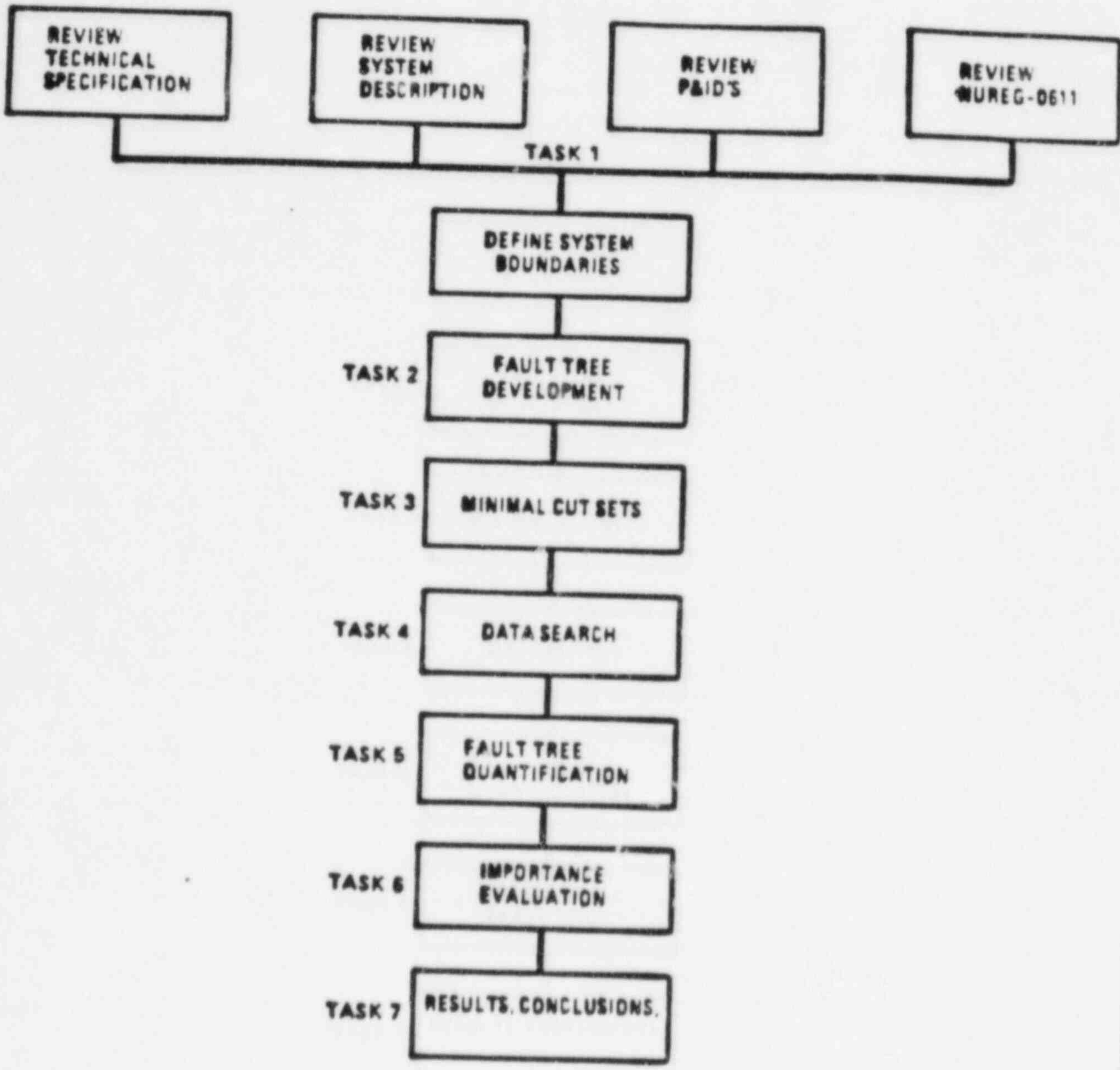
TABLE 10A-5
FAULT TREE COMPONENT IDENTIFICATION CODES

Nine or ten character codes identify component failures in the fault trees. The format of component failures in the fault trees is STCCCXXXXF where:

* S is the system identification code.
* T is the identification of the train to which the component belongs.
* CCC is the component type identification code.
* XXXX is the number designating the single component in the P&IDs.
* F is the specific component failure.

The following lists the codes used in this evaluation.

	SYSTEM
A	Auxiliary Feedwater
	TRAIN
A	Motor driven pump train A
B	Motor driven pump train B
C	Motor driven pump train C
D	Turbine driven pump train D
	COMPONENT
AFST	Auxiliary feedwater storage tank
FL	Flow element
PM	Motor driven pump
PT or TDP	Turbine driven pump
CV	Check valve
MV	Motor operated valve
XV	Manual valve
DG	Diesel generator
ESFAUTO	Automatic ESF signal
ESFMAN	Manual ESF backup signal
GV	Governor valve
	FAILURE MODE
P	Plugging
OE	Operator error
MAIN	Maintenance
TST	Test



SOUTH TEXAS PROJECT
UNITS 1 & 2

BASIC TASKS OF THE SOUTH TEXAS PROJECT AFWS RELIABILITY EVALUATION

Figure 10 A-1

ATTACHMENT
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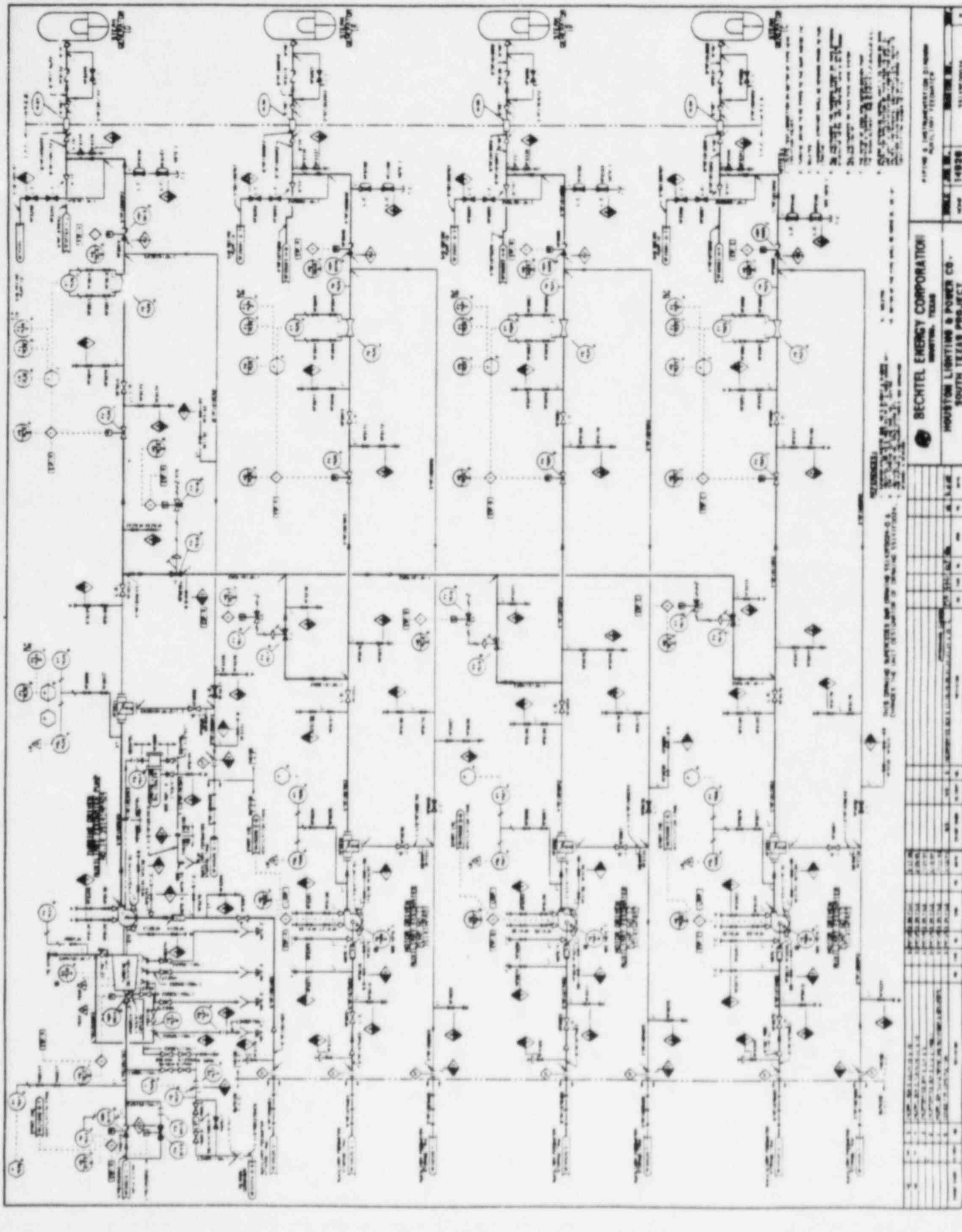
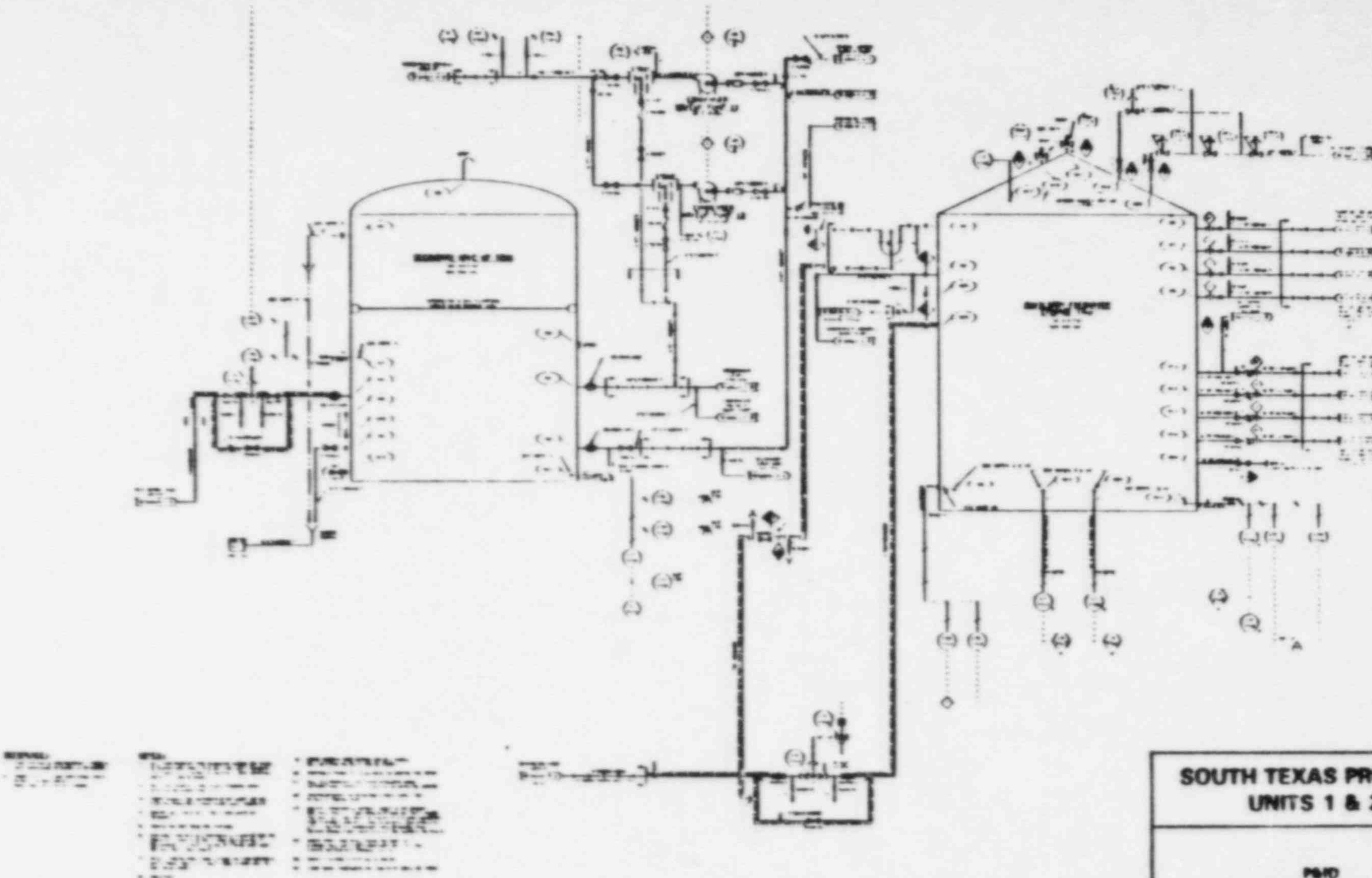


FIGURE 10A-2
SHEET 1 OF 2



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UNITS 1 & 2

P&ID
AUXILIARY FEEDWATER SYSTEM
(Sheet 2 of 2)

Dwg. No. 9519-3-F-00020

Figure 10A-2

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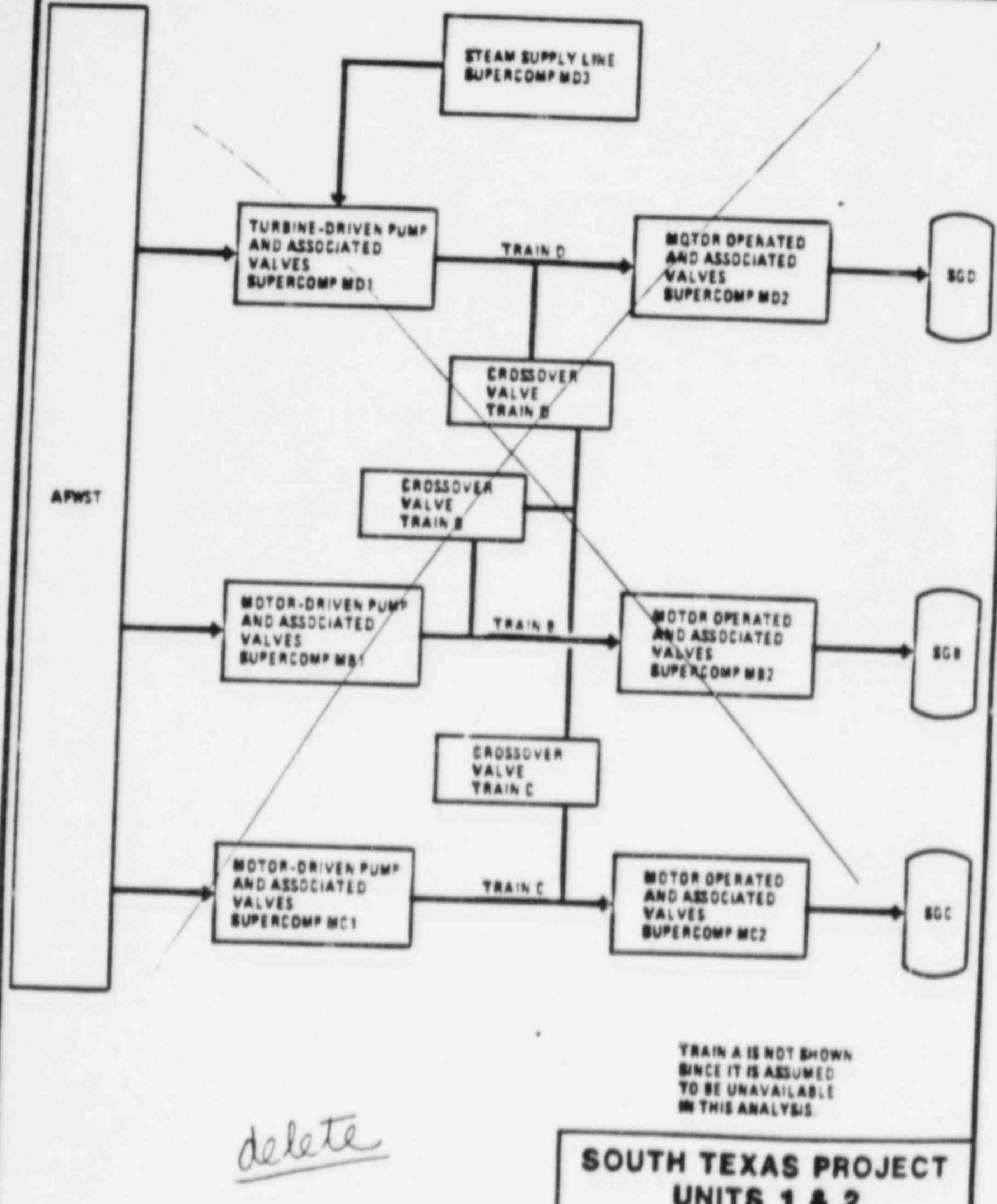
PAGE 168 OF 18X



SOUTH TEXAS PROJECT
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SOUTH TEXAS AFWS
RELIABILITY BLOCK DIAGRAM
(Simplified)

Figure 10 A-3

ATTACHMENT
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SOUTH TEXAS PROJECT
UNITS 1 & 2

DETAILED RELIABILITY BLOCK DIAGRAM SHOWING CONSTITUENTS OF SUPERCOMPONENTS

Figure 10 A.4

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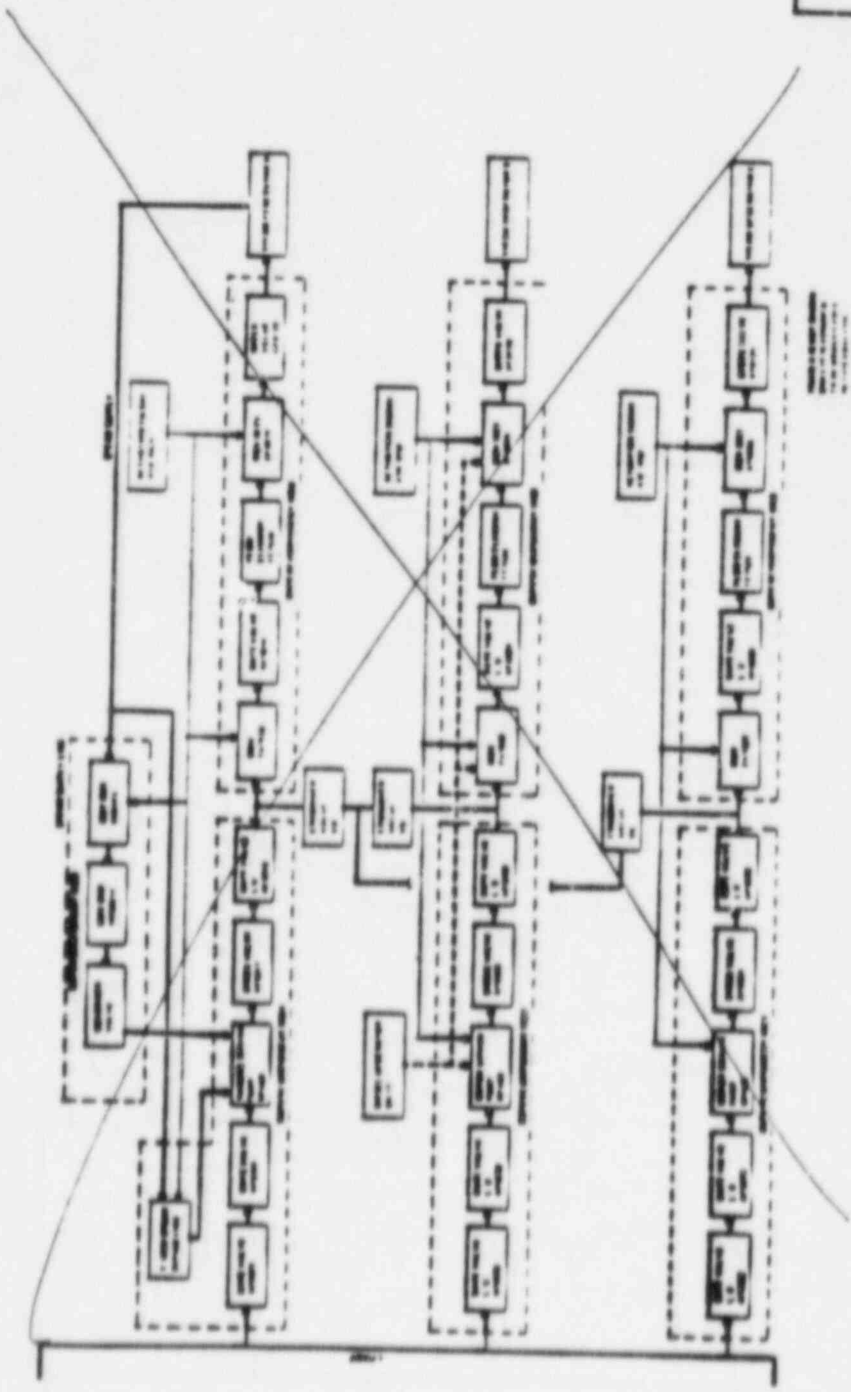

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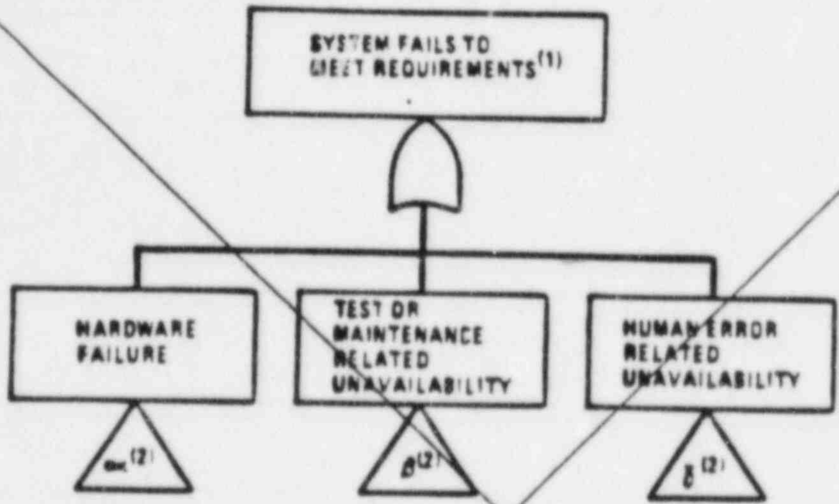

NOTES

(1) FAILURE CRITERIA

CASE I: NO FLOW TO ANY STEAM GENERATOR FOR T ≥ 20 MIN
CASE II: NO FLOW TO ANY STEAM GENERATOR FOR T ≥ 20 MIN
CASE III: NO FLOW TO STEAM GENERATOR D FOR T ≥ 20 MIN

(2)

SUBTREE	LMFW (CASE I)	LMFW LOOP (CASE II)	LMFW LOAC (CASE III)
α	A	A	AD
β	B	B	D
γ	C	C	E

SOUTH TEXAS PROJECT UNITS 1 & 2

MASTER FAULT TREE

Figure 10 A-5

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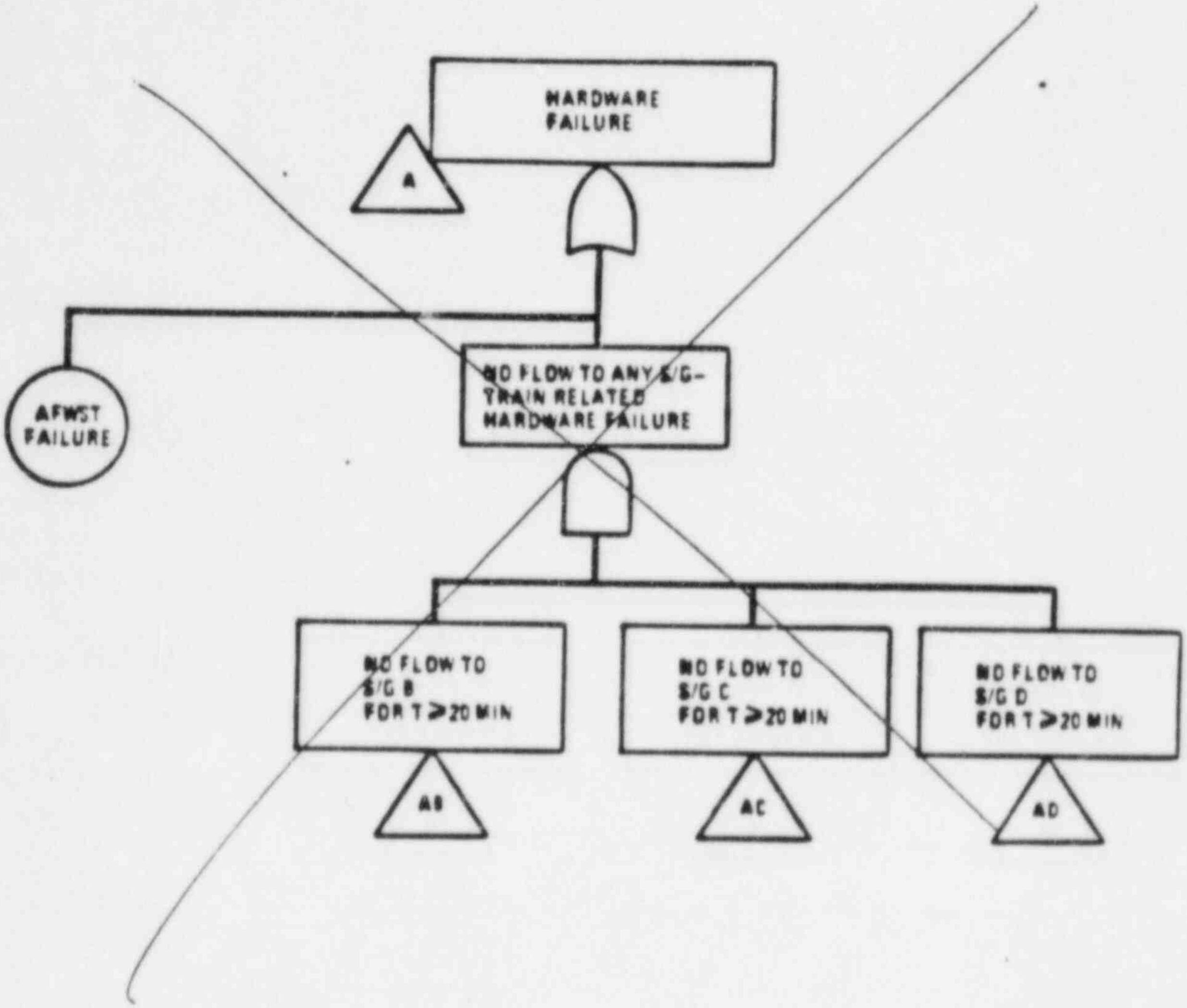

SOUTH TEXAS PROJECT UNITS 1 & 2

HARDWARE FAULT TREE, CASES I & II

Figure 10 A-6

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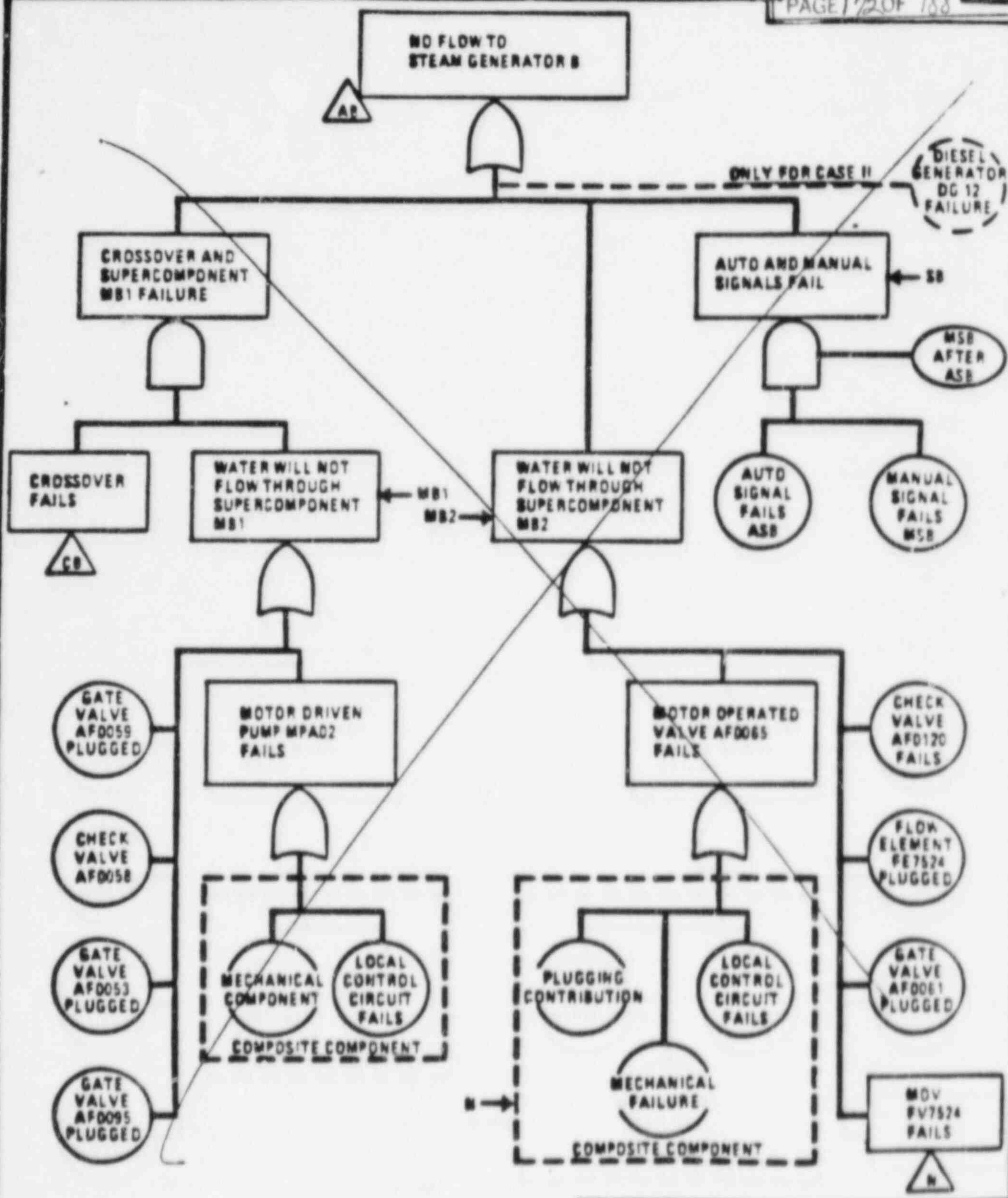

SOUTH TEXAS PROJECT
UNITS 1 & 2

TRAIN B HARDWARE
FAULT TREE

Figure 10 A-7

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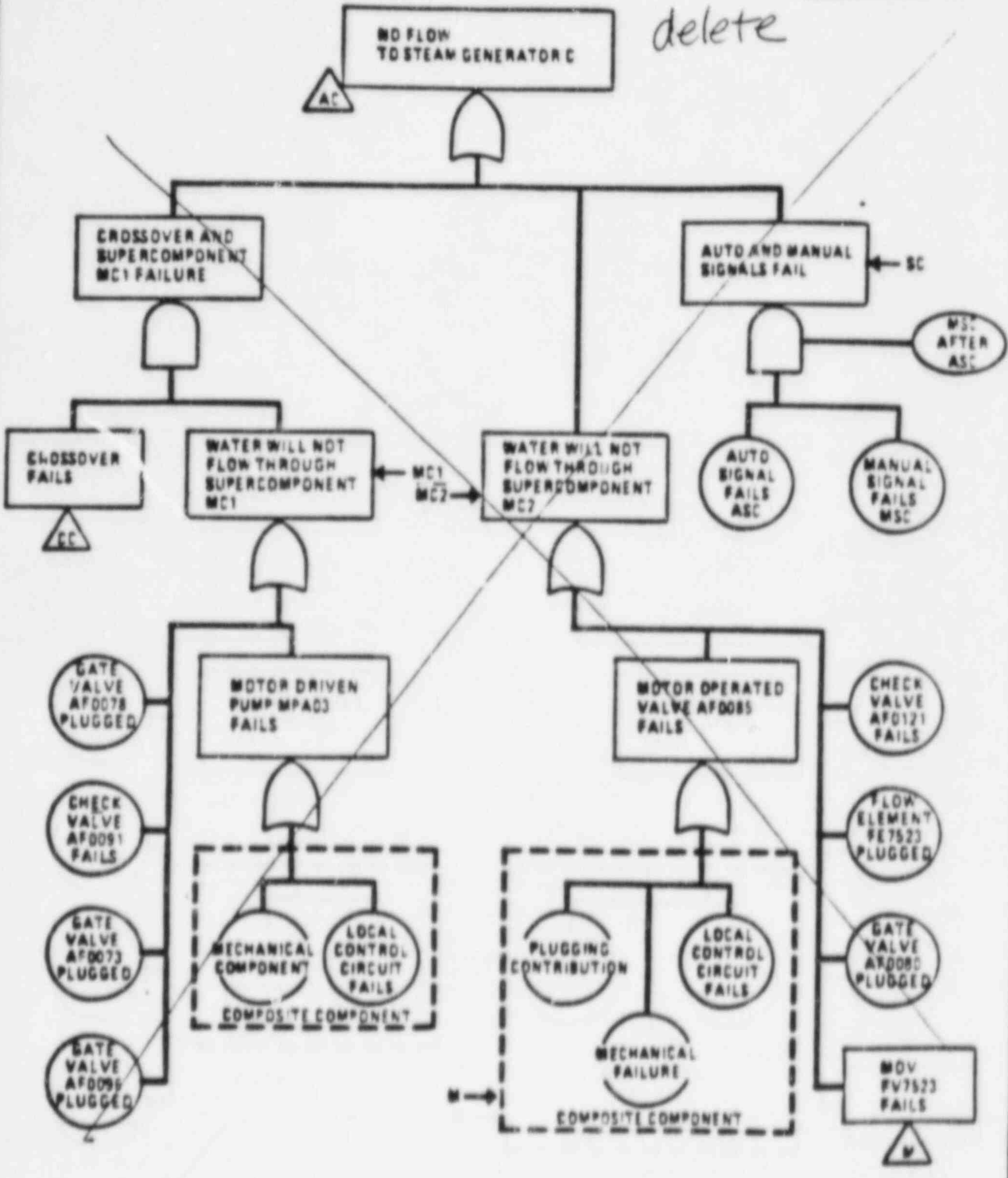

SOUTH TEXAS PROJECT UNITS 1 & 2

TRAIN C HARDWARE FAULT TREE

Figure 10 A-8

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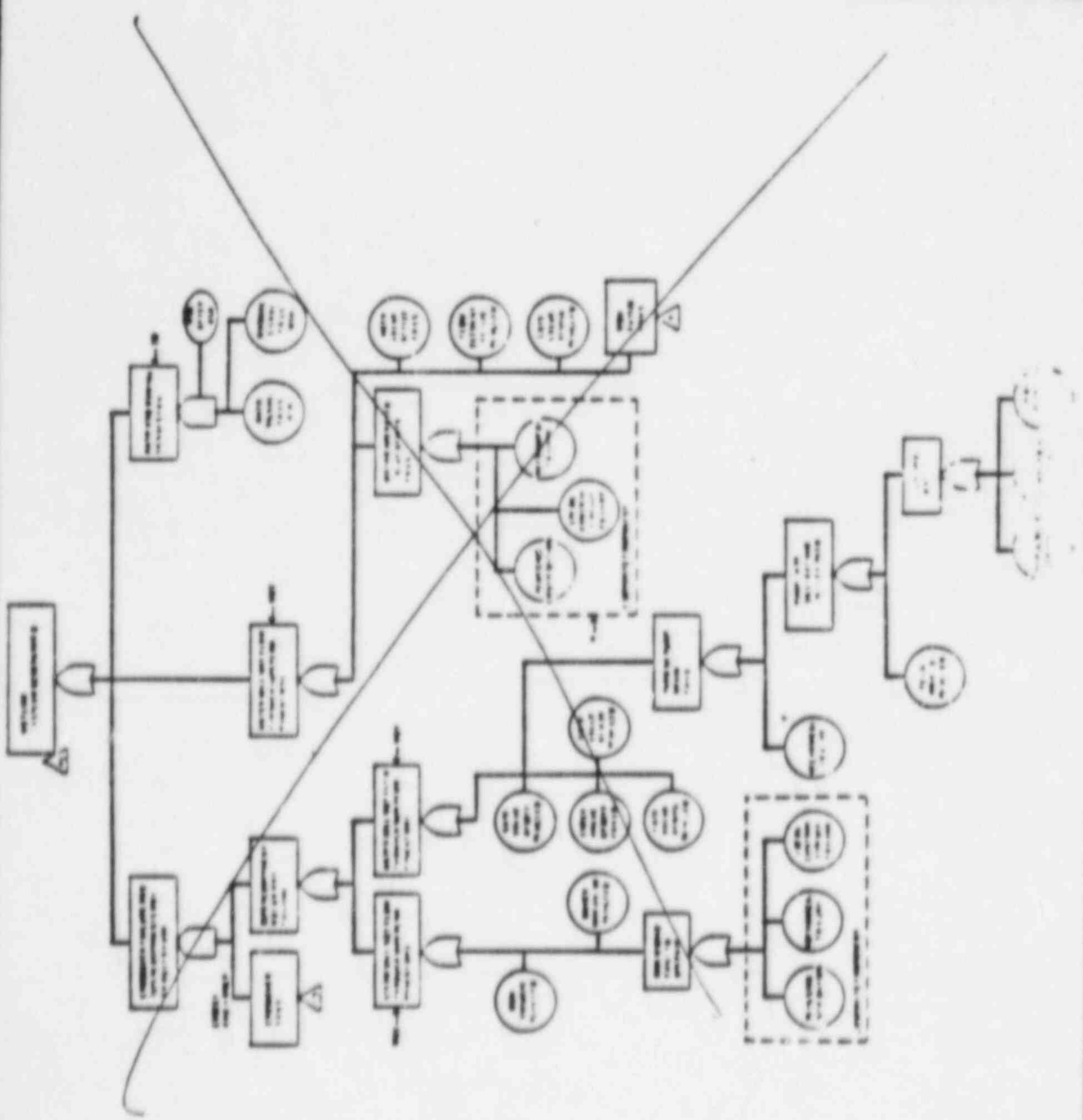

SOUTH TEXAS PROJECT
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TRAIN D HARDWARE
FAULT TREE

Figure 19A.9

ATTACHMENT
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TRAIN B CROSSOVER FAILURE

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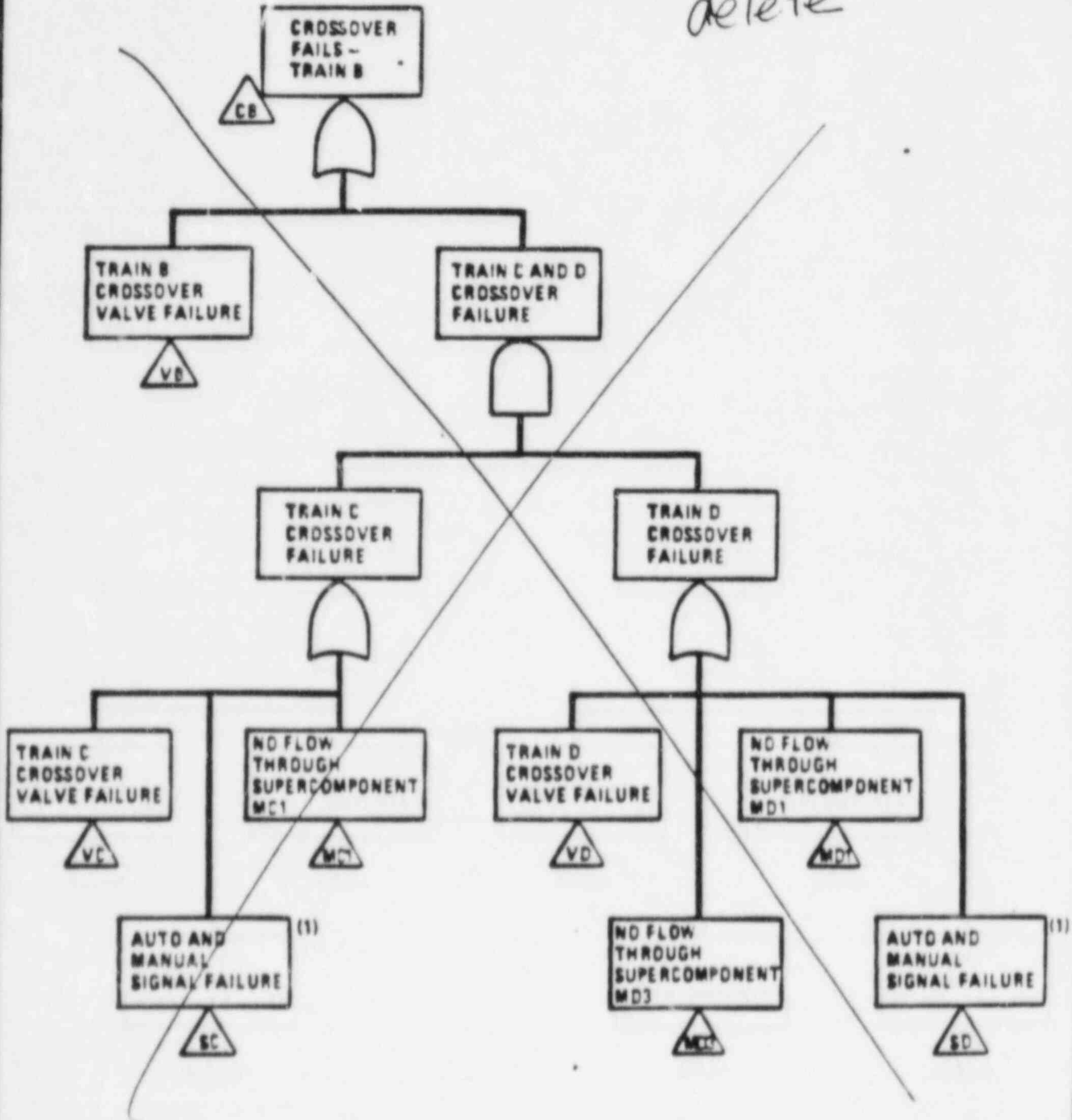

(1) THIS TERM IS NEGLIGIBLE COMPARED TO OTHER HARDWARE FAILURES AND IS NOT INCLUDED IN THE FAULT TREE QUANTIFICATION.

SOUTH TEXAS PROJECT UNITS 1 & 2

SUPPORTING HARDWARE FAULT TREES
(Sheet 1 of 4)

Figure 10 A-10

TRAIN C CROSSOVER FAILURE

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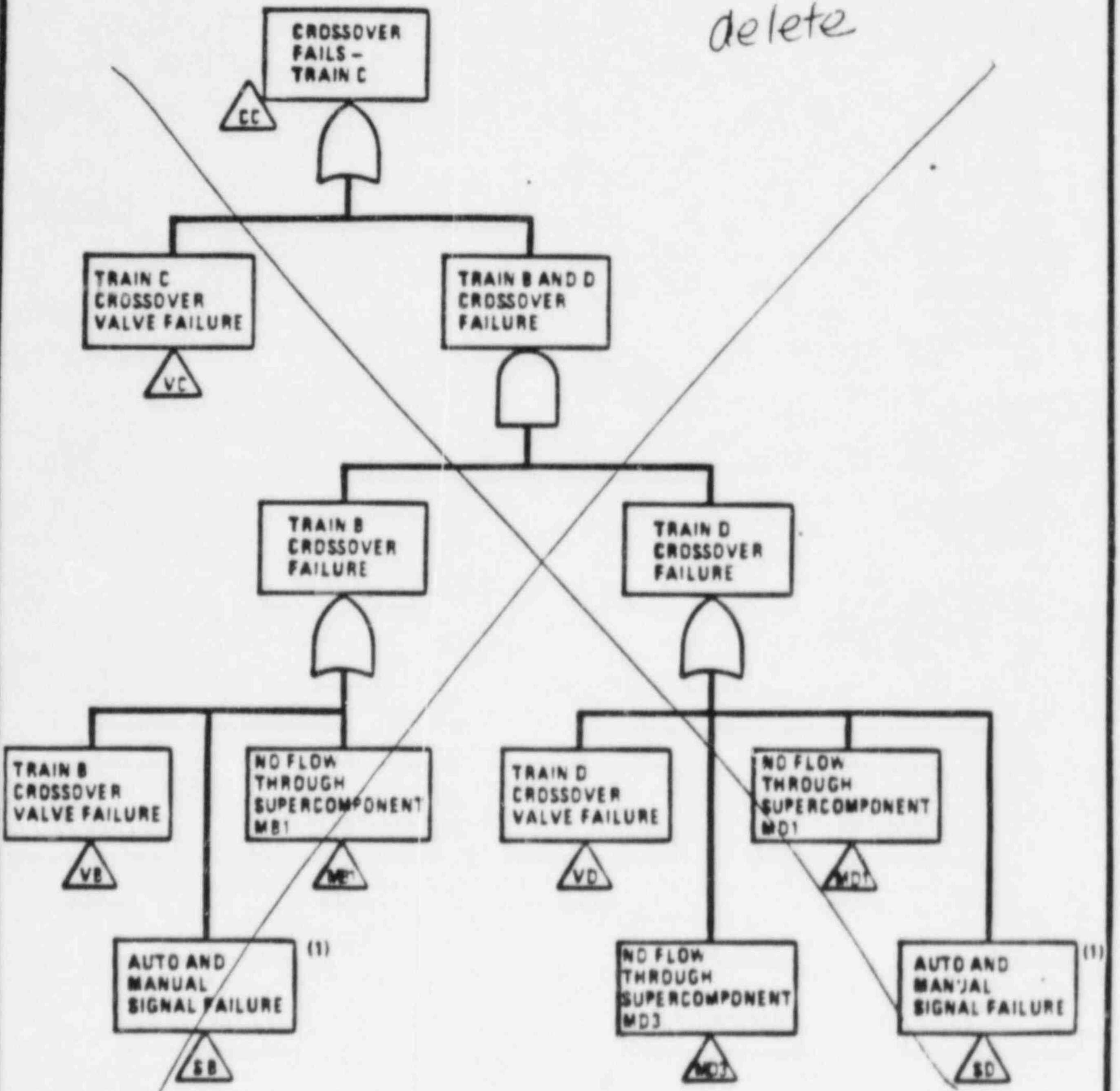

(1) THIS TERM IS NEGLIGIBLE COMPARED TO OTHER HARDWARE FAILURES AND IS NOT INCLUDED IN THE FAULT TREE QUANTIFICATION.

SOUTH TEXAS PROJECT UNITS 1 & 2

SUPPORTING HARDWARE FAULT TREES (Sheet 2 of 4)

Figure 10 A-10

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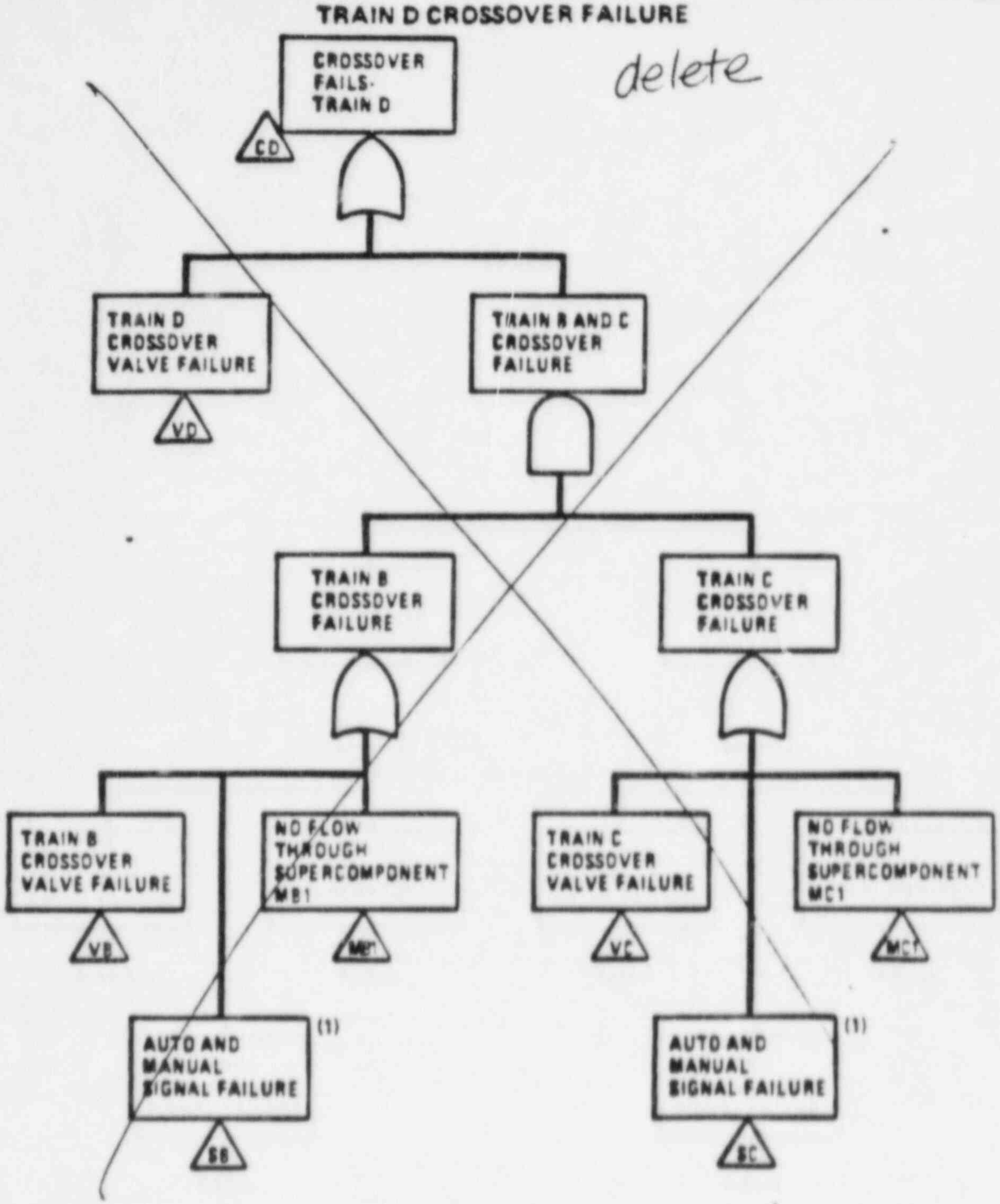

(1) THIS TERM IS NEGLIGIBLE COMPARED TO OTHER HARDWARE FAILURES AND IS NOT INCLUDED IN THE FAULT TREE QUANTIFICATION.

SOUTH TEXAS PROJECT
UNITS 1 & 2

SUPPORTING HARDWARE
FAULT TREES
(Sheet 3 of 4)

Figure 10 A-10

CROSSOVER VALVE FAILURE

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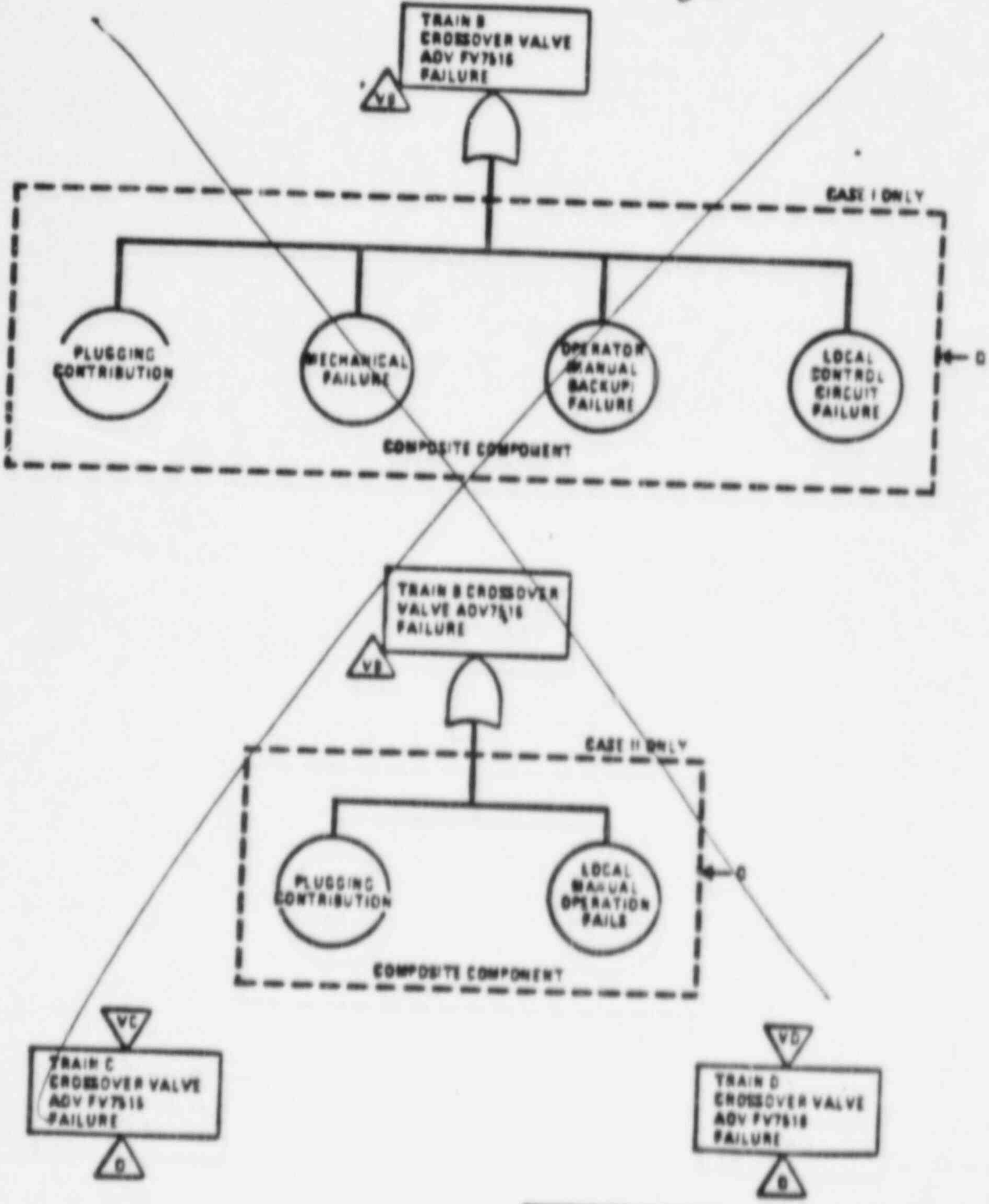

SOUTH TEXAS PROJECT
UNITS 1 & 2

SUPPORTING HARDWARE
FAULT TREES
(Sheet 4 of 4)

Figure 10 A-10

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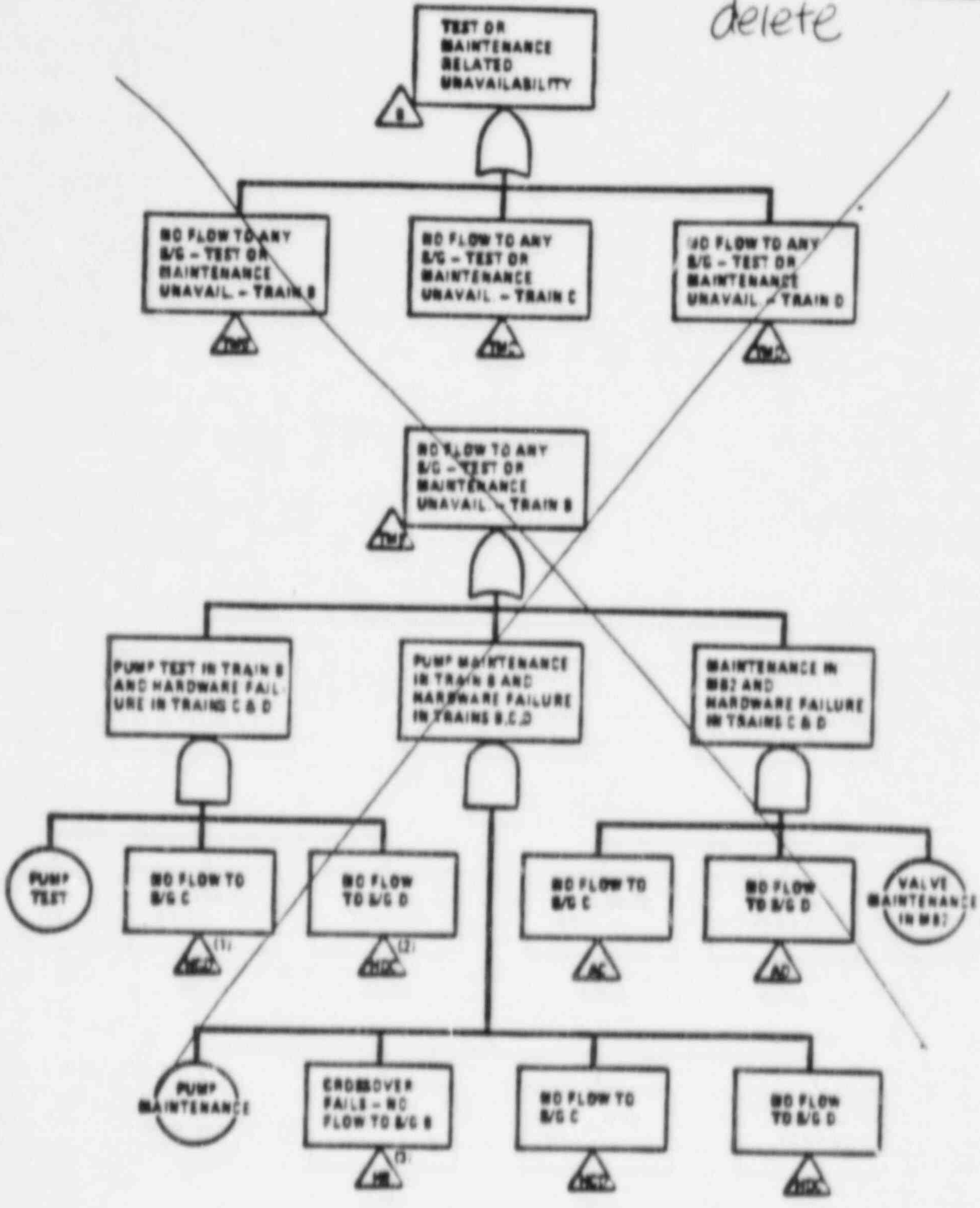

SOUTH TEXAS PROJECT
UNITS 1 & 2

TEST AND MAINTENANCE FAULT TREES
(Sheet 1 of 4)

Figure 10 A-11

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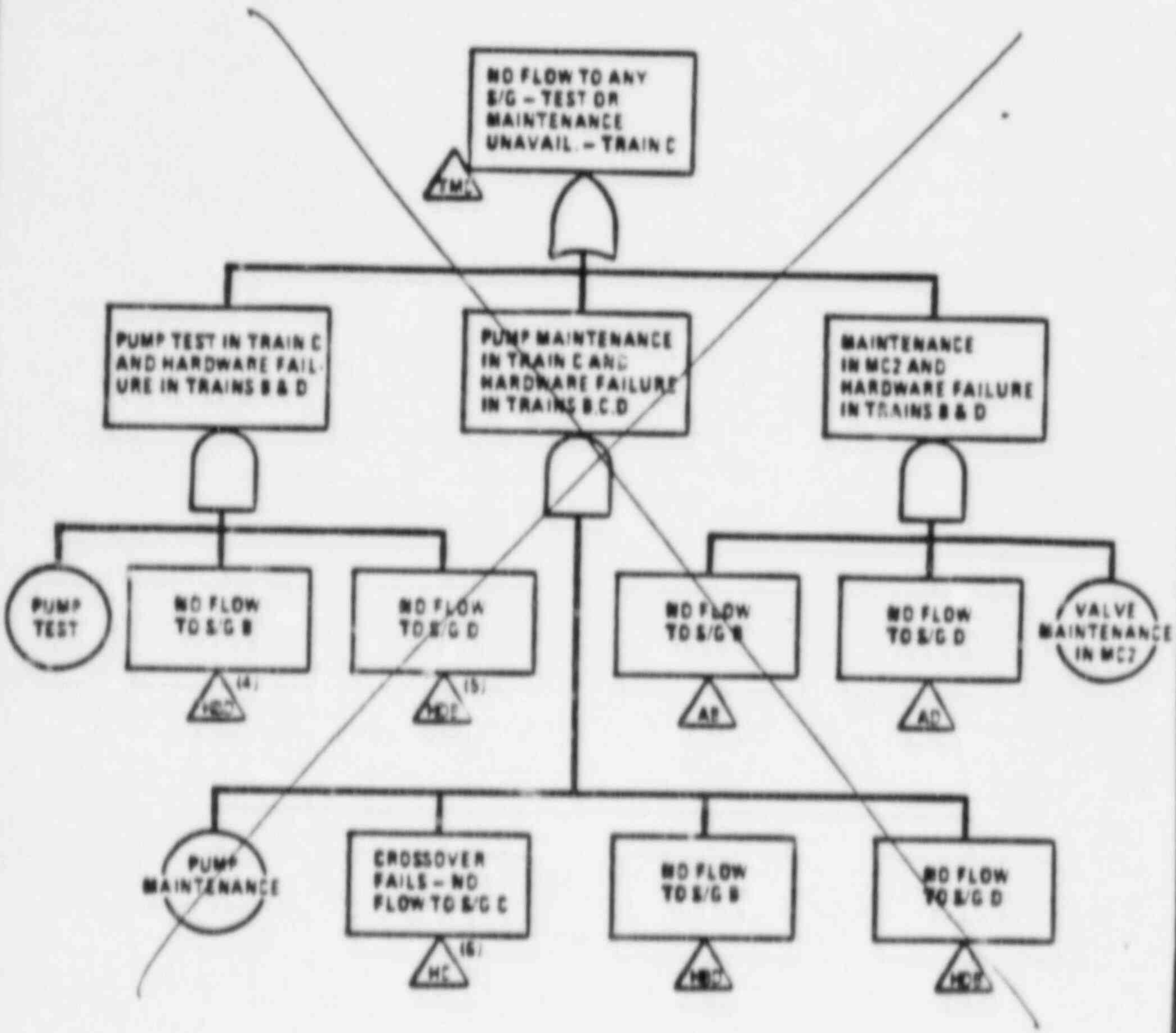

SOUTH TEXAS PROJECT
UNITS 1 & 2

TEST AND MAINTENANCE
FAULT TREES
(Sheet 2 of 4)

Figure 10 A-11

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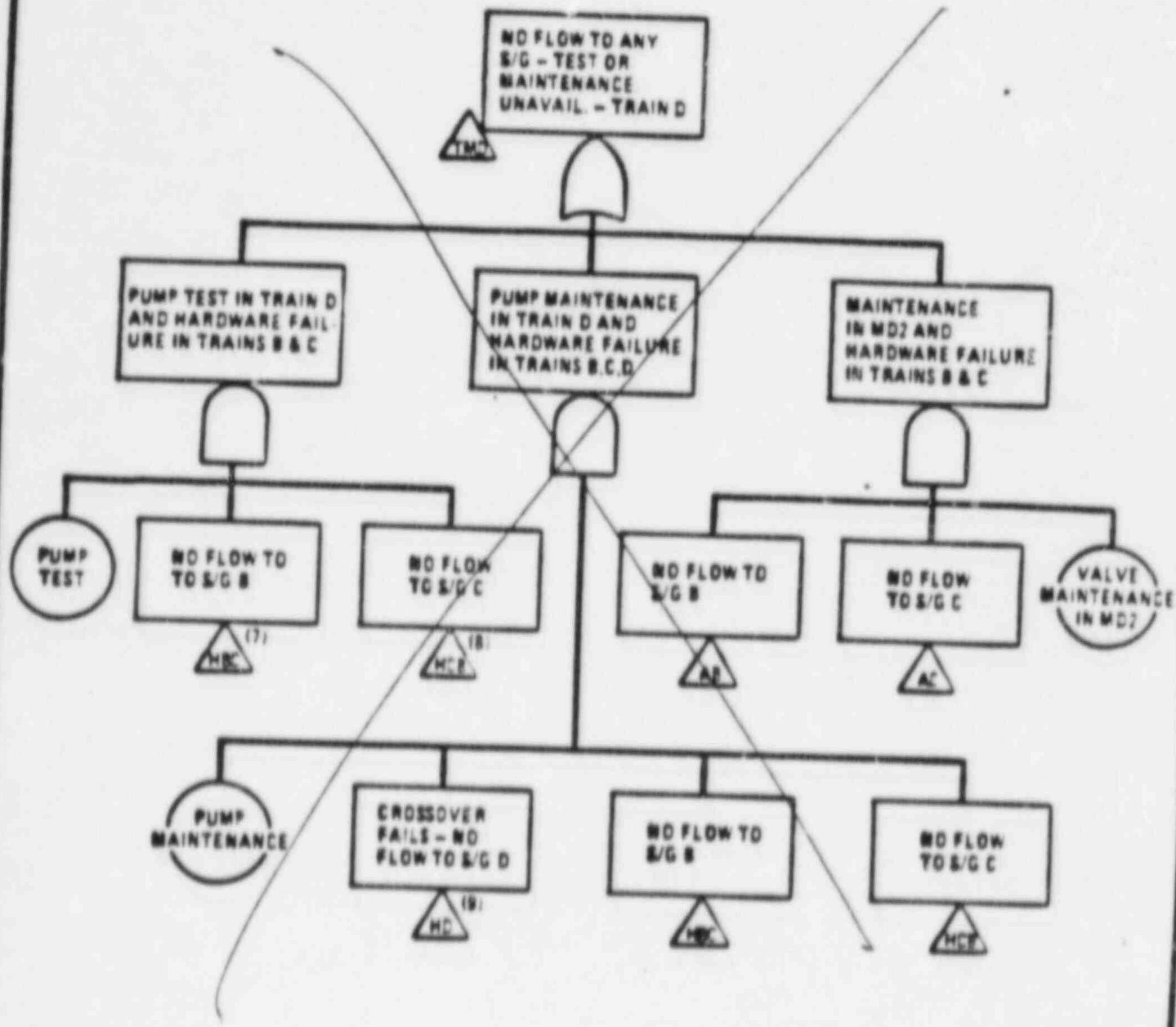

SOUTH TEXAS PROJECT
UNITS 1 & 2

TEST AND MAINTENANCE
FAULT TREES
(Sheet 3 of 4)

Figure 10 A-11

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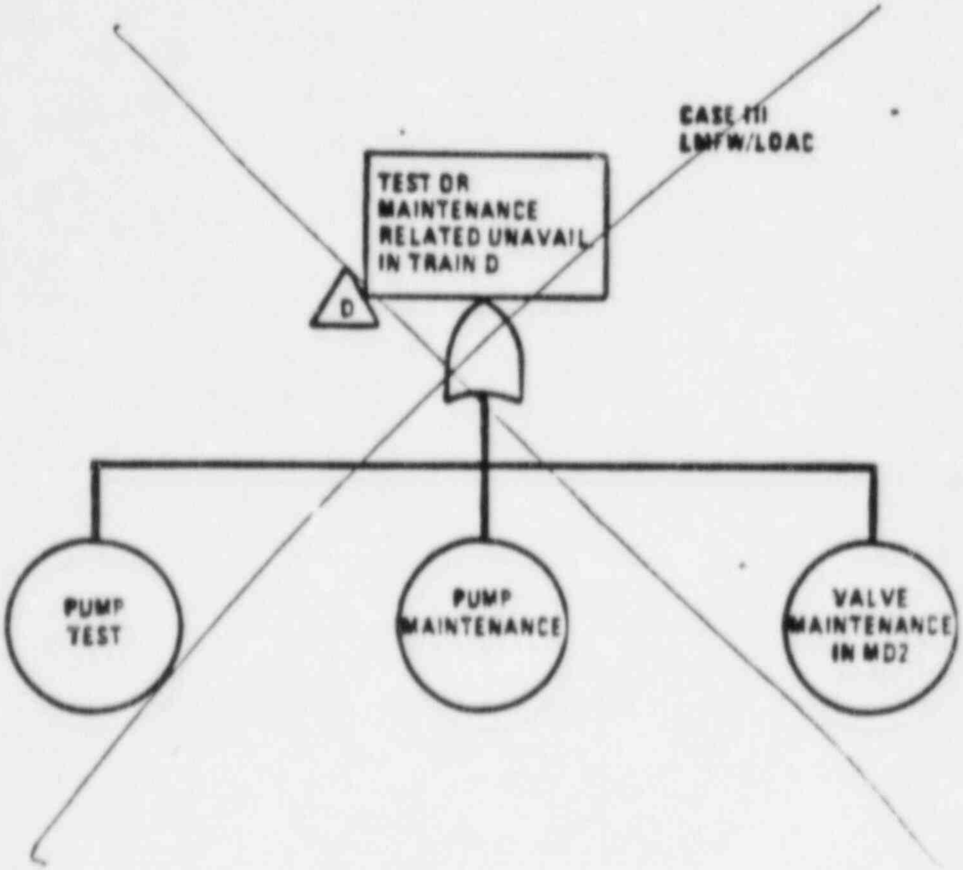

FOOTNOTES: (1) – (8)

SEE FOOTNOTES ON FIGURE 10A-12 FOR EXPLANATION OF THE BOTTOM COMPONENTS IN THESE TREES.

SOUTH TEXAS PROJECT
UNITS 1 & 2

TEST AND MAINTENANCE FAULT TREES
(Sheet 4 of 4)

Figure 10 A-11

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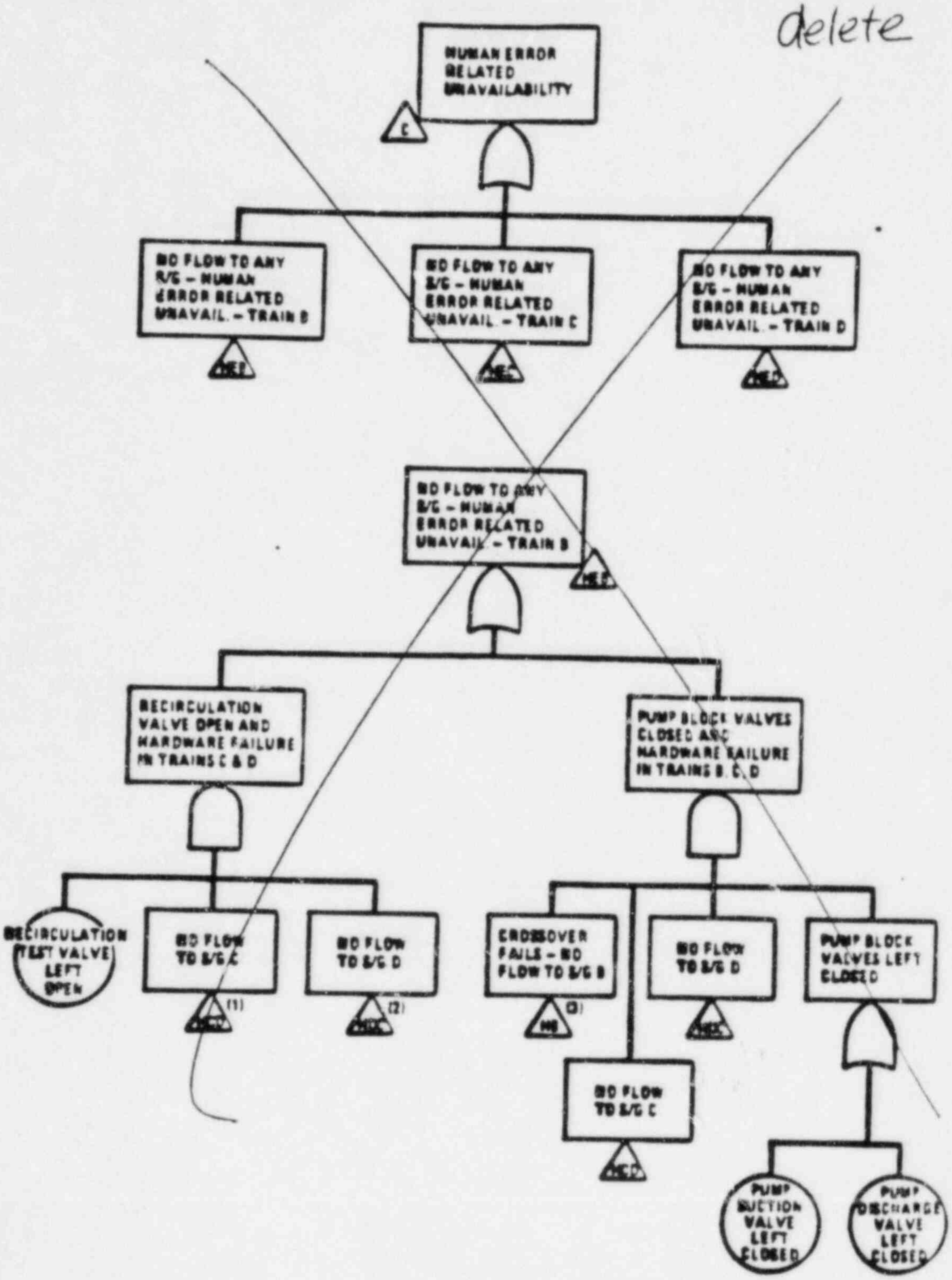

SOUTH TEXAS PROJECT
UNITS 1 & 2

HUMAN ERROR
FAULT TREES
(Sheet 1 of 4)

Figure 10 A-12

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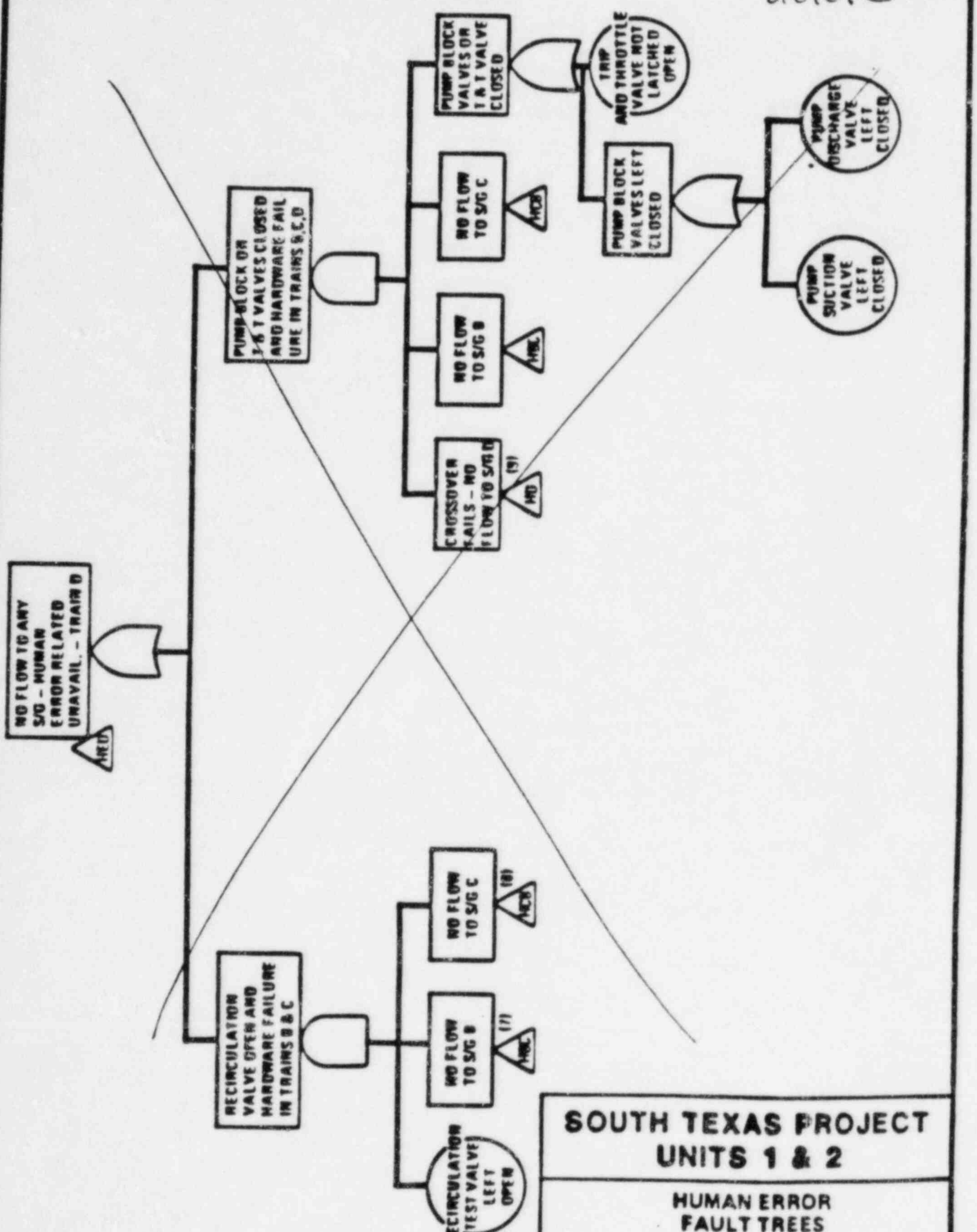

SOUTH TEXAS PROJECT
UNITS 1 & 2

HUMAN ERROR
FAULT TREES
(Sheet 2 of 4)

Figure 10 A-12

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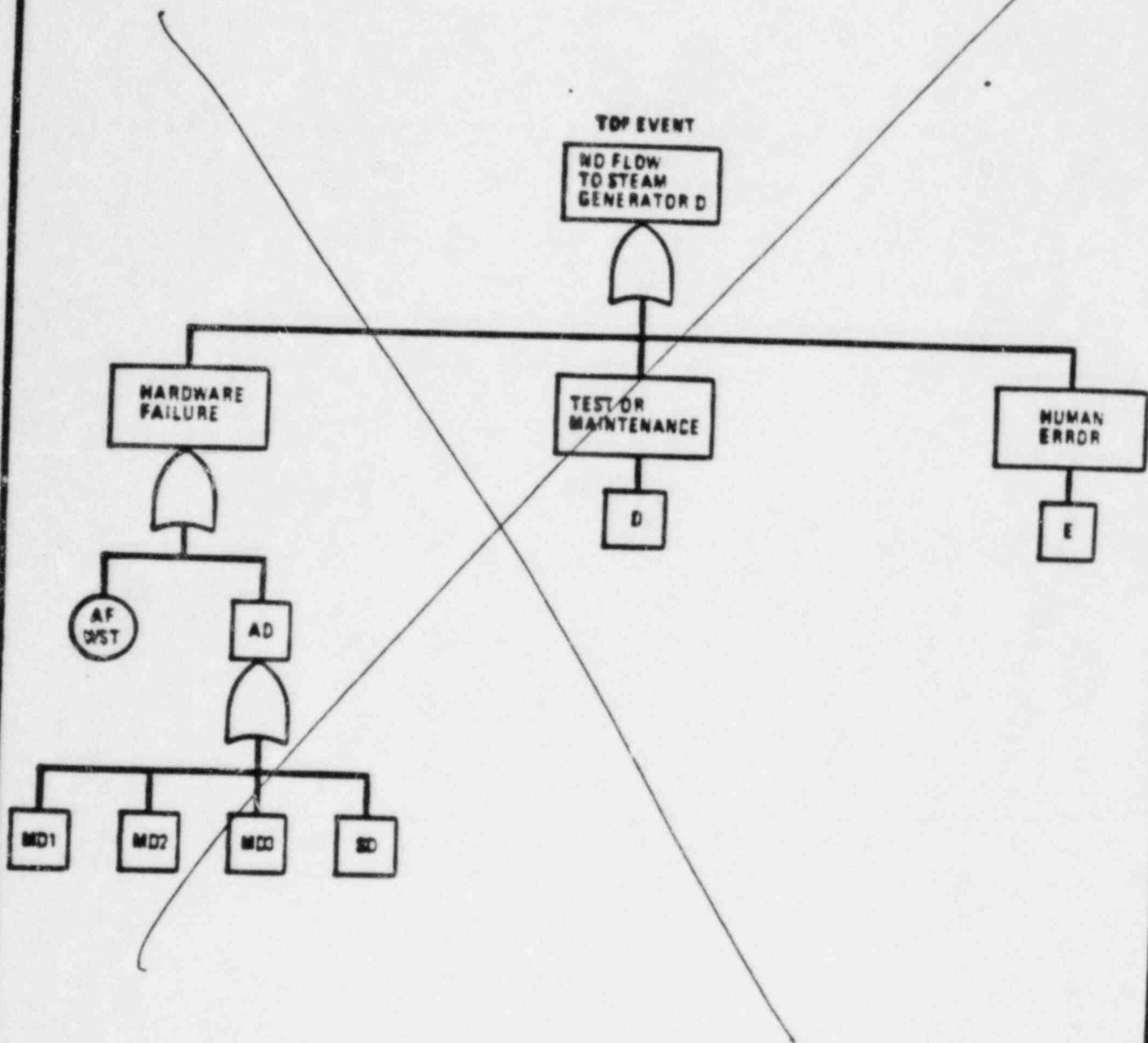

SOUTH TEXAS PROJECT UNITS 1 & 2

REDUCED FAULT TREE FOR CASE III

Figure 10 A-14

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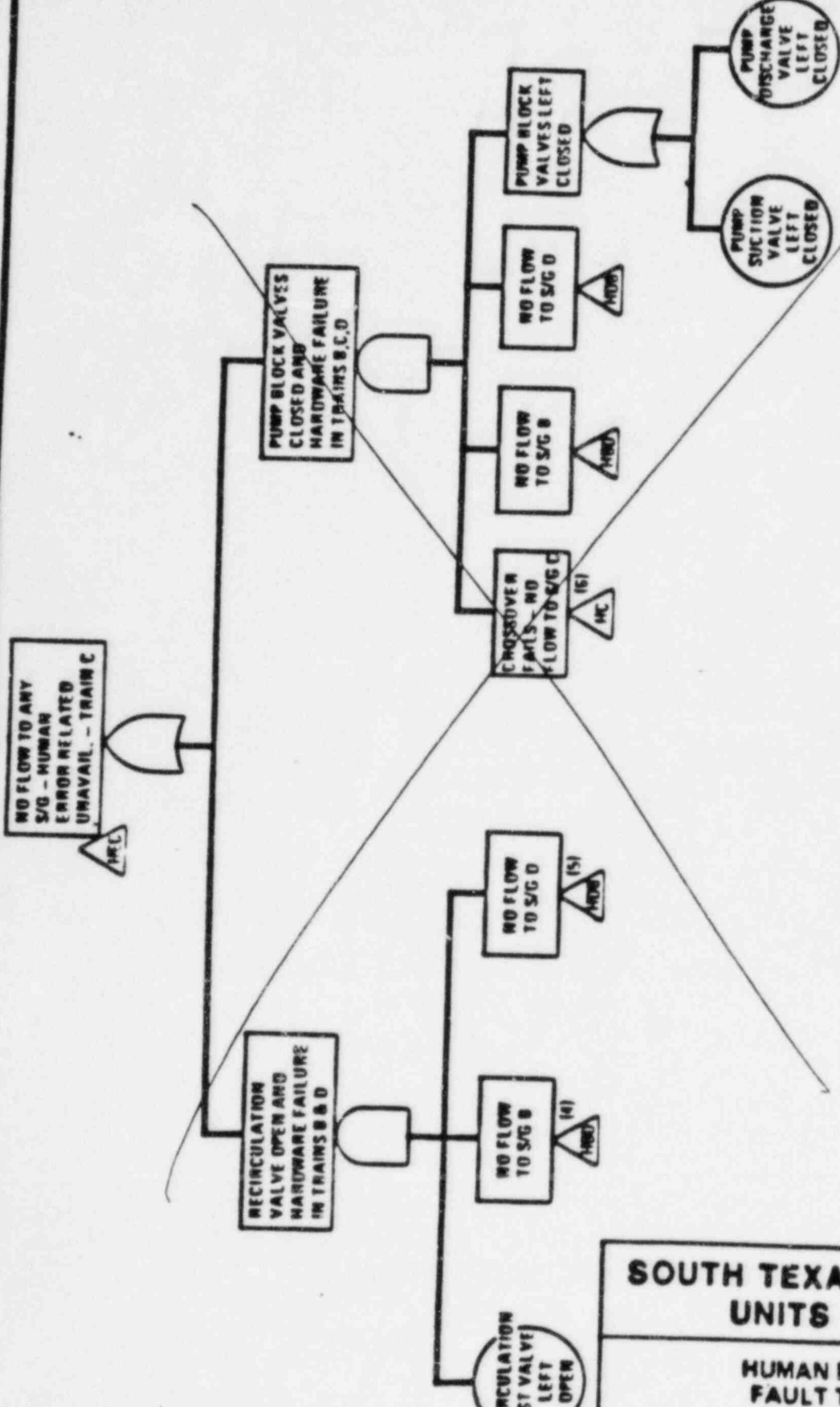

SOUTH TEXAS PROJECT
UNITS 1 & 2

HUMAN ERROR
FAULT TREES
(Sheet 3 of 4)

Figure 10 A-12

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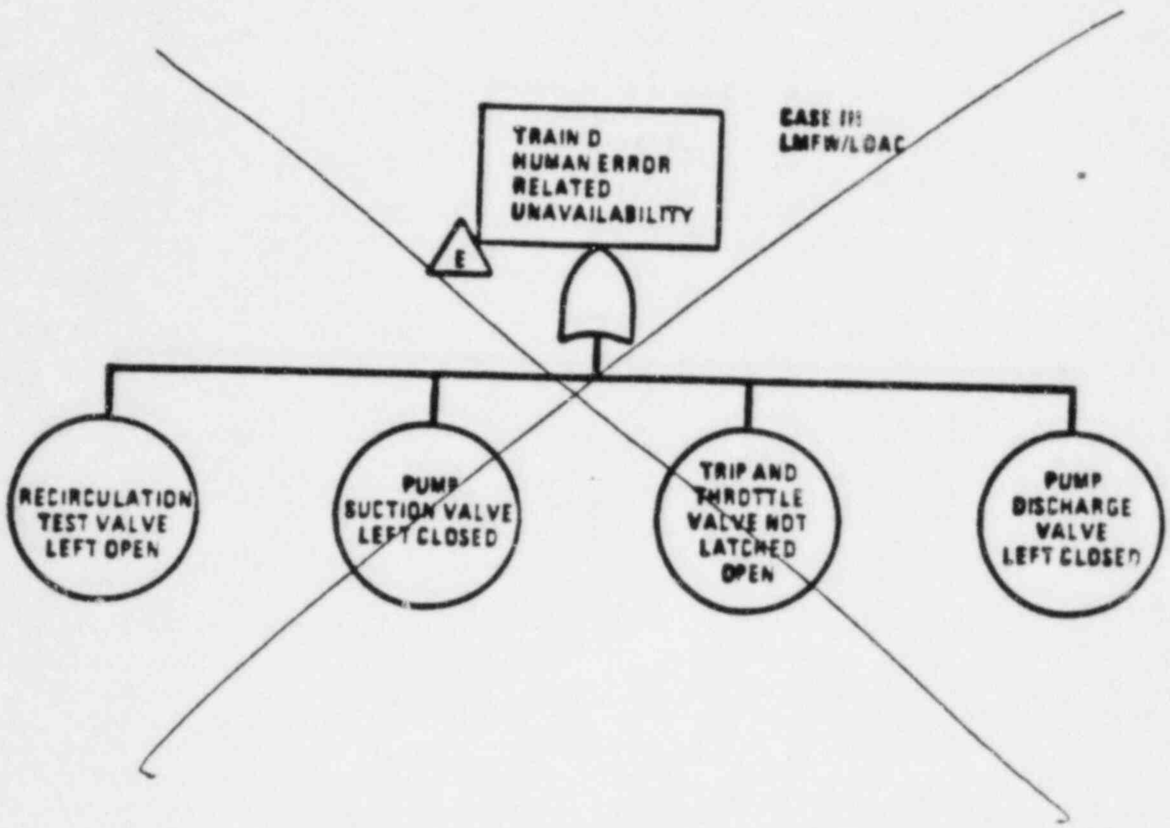

FOOTNOTES

(1) HCD IS IDENTICAL TO AC EXCEPT THAT MB1, VB, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(2) HDC IS IDENTICAL TO AD EXCEPT THAT MB1, VB, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(3) HB IS IDENTICAL TO AB EXCEPT THAT MB1, AND THE IMMEDIATELY CONNECTING "AND" GATES ARE REMOVED

(4) HBD IS IDENTICAL TO AB EXCEPT THAT MC1, VC, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(5) HDB IS IDENTICAL TO AD EXCEPT THAT MC1, VC, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(6) HC IS IDENTICAL TO AC EXCEPT THAT MC1, AND THE IMMEDIATELY CONNECTING "AND" GATES ARE REMOVED

(7) HBC IS IDENTICAL TO AB EXCEPT THAT MD1, MD3, VD, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(8) HCB IS IDENTICAL TO AC EXCEPT THAT MD1, MD3, VD, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

(9) HD IS IDENTICAL TO AD EXCEPT THAT MD1, MD3, AND THE IMMEDIATELY CONNECTING "AND" AND "OR" GATES ARE REMOVED

COMPONENTS AB AC, AND AD ARE SHOWN ON FIGURES 10A-7, 10A-8 AND 10A-9. THE CONTINUATION OF THESE TREES WHICH SHOW COMPONENTS MB1, VB, MC1, VC, MD1, MD3, AND VD ARE SHOWN ON FIGURE 10A-10. THE SIGNAL FAILURE TERMS ARE NEGLIGIBLE AND ARE NOT INCLUDED IN THE QUANTITATIVE ANALYSIS

SOUTH TEXAS PROJECT
UNITS 1 & 2

HUMAN ERROR
FAULT TREES
(Sheet 4 of 4)

Figure 10 A-12

ATTACHMENT
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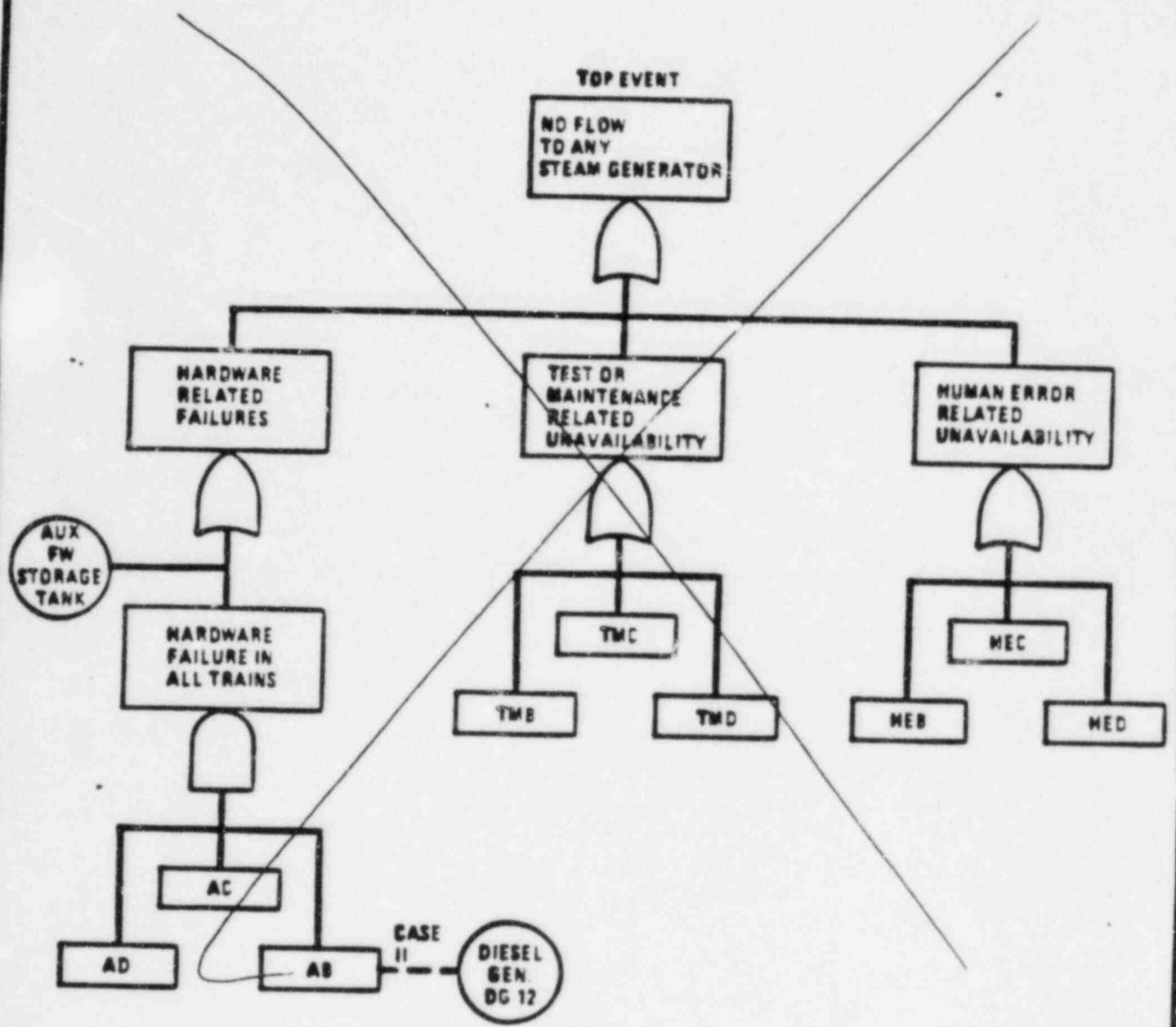

SOUTH TEXAS PROJECT
UNITS 1 & 2

REDUCED FAULT TREE
FOR CASES I AND II

Figure 10 A-13