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October 20, 1994
Refer to: RC-94-0275

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. G. F. Wunder

Gentlemen:

Subject: VIRGIL C. SUMMER NUCLEAR STATION
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
STEAM GENERATOR REPLACEMENT (SGR) TECHNICAL
SPECIFICATION CHANGE REQUEST - REVISED ANALYSES
(REM 6000, TSP 930015)

South Carolina Electric & Gas Company's (SCE&G's) submittals supporting the Technical Specification changes (TSCs) associated with SGR (dated September 3, 1992, March 12, 1993, April 30, 1993, October 29, 1993, March 11, 1994, May 18, 1994, and September 20, 1994), contain extensive analyses. Subsequent reviews of the analyses identified an input error in a portion of the Steam Line Break mass and energy releases inside containment and the use of a non-conservative delay time for Emergency Feedwater in the FSAR Chapter 15, Feedwater Line Break analyses. This submittal provides revised analysis results for these areas.

This submittal consists of three attachments. Attachment 1 and 2 provide the basis for and a description of the revised Steam Line Break and Feedwater Line Break analyses, respectively. In order to provide one complete document for the licensing submittal, Attachment 3 documents the results of the analyses as revised replacement pages to SCE&G's October 29, 1993, letter.

The revised analyses described in the attachments contain acceptable results. The pressures and temperatures inside containment during a Steam Line Break are lower than those previously reported and all acceptance criteria for the FSAR Chapter 15, Feedwater Line Break analyses are met. Consequently, no additional changes to the Virgil C. Summer Nuclear Station (VCSNS) Technical Specifications or revisions to the Technical Specification changes identified in SCE&G's prior submittals are required as a result of the revised analyses. In addition, the No Significant Hazards Determination (10CFR50.92) provided in SCE&G's letter dated March 11, 1993, remains applicable for SGR.

Also, as a result of issuance of Amendment 116 to the Technical Specifications, one revised Technical Specification page is provided to incorporate the approved amendment and the requested SGR TSC. This change is purely administrative and does not affect the No Significant Hazards Determination.

ADOI

This letter completes SCE&G's licensing submittal supporting SGR. I declare that the statements and matters set forth herein are true and correct to the best of my knowledge, information, and belief.

SCE&G appreciates NRC's efforts to support the SGR schedule. If you have any questions or require additional information to complete your review of the revised analyses, please contact Ms. April Rice at (803) 345-4232.

Very truly yours,

John L. Skolds

ARR:lcd
Attachments

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ATTACHMENT 1

Steam Line Break (SLB) Analyses

Sections 3.4.3.1 and 3.4.4.1 of SCE&G's Licensing Submittal contain mass and energy releases and resulting pressure and temperature conditions during postulated SLBs inside containment. These analyses formed the primary basis for the Technical Specification change request to increase the maximum peak calculated pressure during an accident from 47.1 to 53.5 psig. Subsequent reviews of the SLB analyses has identified an error in the mass and energy releases which affects the following analyzed cases:

- 1.4 ft^2 Double Ended Rupture (DER) @ 75% Power
- 1.4 ft^2 Double Ended Rupture (DER) @ 50% Power
- 1.4 ft^2 Double Ended Rupture (DER) @ 25% Power
- 1.4 ft^2 Double Ended Rupture (DER) @ 0% Power

In these cases, the quality of the break flow from the intact Steam Generators (SGs) was incorrectly specified as 1.0 (dry steam release). Flow from the intact SGs, which is terminated following closure of the main steam isolation valves (MSIVs), contains substantial liquid (quality less than 1.0) which, when properly reflected in the analyses, will lead to an increase in the total mass release to the containment and a subsequent increase in the containment pressure.

Additional SLB analyses have been performed and updates to SCE&G's Licensing Submittal (dated 10/29/93) are provided in Attachment 3. In addition to correcting the error discovered in the mass and energy releases, the revised analyses reflect four additional changes:

Main Feedwater Flow: In the prior analyses for the large DERs, main feedwater (MFW) flow to the faulted SG was assumed to increase to 3.5 times the nominal full power value at event initiation and to remain at this value until the feedwater lines are completely isolated via closure of the feedwater isolation and control valves. These assumptions were based on generic analyses performed previously by Westinghouse. To offset the impact of the error correction, conservatisms in MFW flow simulation were relaxed based on plant specific analyses. For the plant specific analyses, the mass of feedwater added to the faulted SG was evaluated under the following assumptions:

- All 3 MFW pumps were assumed to be operating for Mode 1 initial conditions irregardless of the initial power level.
- Although pump trip signals on high SG level and low steam line pressure will occur, the MFW pumps were assumed to remain at full speed.
- No credit for the increased flow resistance due to partial closure of the feedwater flow control or isolation valves was assumed.
- All 3 MFW pumps were assumed to runout in response to the decreasing SG pressure caused by the postulated SLB.

These assumptions maximize main feedwater addition to the faulted SG and show that MFW flow continuously increases until feedwater isolation is achieved. Results from the plant specific analyses were conservatively incorporated into the revised mass and energy

calculations by assuming MFW flow to the faulted SG increases to the maximum value calculated (approximately 1.7 times the nominal full power value) at event initiation and remains at this value until the feedwater lines are completely isolated via closure of the feedwater isolation and control valves.

Operator Action Time: The Emergency Feedwater (EFW) System is designed to automatically isolate EFW to the faulted SG. Upon detection of high flow to a depressurized SG, flow control valves at the discharge of the motor and turbine driven EFW pumps are closed , redirecting flow from the faulted SG to the intact SGs. Failure to automatically close a flow control valve will allow continuous EFW flow (up to 1000 gpm) to the faulted SG until operator action is credited. The revised SLB analyses extend the assumed operator action time from 20 to 30 minutes for those analyses which assume a failure affecting the automatic EFW isolation function.

Electrical Channel A Failure: The consequences of this worst case, electrical channel failure were more explicitly analyzed in the revised analyses. As a result of this failure, the prior calculation assumed the loss of an emergency diesel, one CHG/SI pump, one motor driven emergency feedwater pump, one RB spray pump, one RB cooling unit, and the ability to automatically isolate emergency feedwater to the faulted SG. The revised analyses continue to simulate these consequences but also credits the loss of power to the main steam isolation and feedwater control and isolation valves. This change leads to a small decrease in the time to main steam and feed line isolation during the SLB.

Reactor Building Cooling Units (RBCUs): In the prior analyses, the RBCU heat removal rate was reduced by greater than 50% below current licensing bases assumptions to allow for future potential degradation in those units. To offset the impact of the error correction, the reduction in RBCU capacity was limited to 40% in the revised analysis.

Due to the changes described above, the analyses for the 1.4 ft^2 DER, including the power level and single failure studies, were repeated. The remaining SLB cases within the licensing submittal remain applicable and provide conservative estimates of the impact of break size and type on the resulting containment temperatures and pressures. The results of the revised analyses, documented as revision pages to the licensing submittal in Attachment 3, demonstrates that the 1.4 ft^2 DER at 100% power with a failure of a MSIV remains the worst case from a RB temperature standpoint, the 1.4 ft2 DER at 25% power with failure of Electrical Channel A remains the worst case from a RB pressure standpoint, and the maximum pressure and temperature conditions are less limiting than those previously reported.

ATTACHMENT 2

Feedwater Line Break (FWLB) Analyses

VCSNS's Emergency Feedwater (EFW) System is designed to automatically isolate EFW to the faulted SG during a FWLB and to supply a minimum of 380 gpm to the intact SGs assuming a single failure and loss of offsite power. The automatic isolation function is based on the premise that the EFW system will preferentially feed the faulted SG as it continues to depressurize following closure of the main steam isolation valves. Upon detection of high flow to a depressurized SG, flow control valves at the discharge of the motor and turbine driven EFW pumps are automatically closed redirecting flow from the faulted SG to the intact SGs.

Section 15.4.2.2.1 of Appendix 6 to the SCE&G's Licensing Submittal (dated 10/29/93) presents the analyses of a FWLB. Within this analysis, the Emergency Feedwater (EFW) system was actuated by the low-low SG water level signal and assumed to supply a total of 380 gpm to the intact SGs within 60 seconds. The use of 380 gpm to 2/3 SGs within 60 seconds has been determined to be non-conservative for VCSNS operation with Delta-75 SGs.

Additional FWLB analyses have thus been performed and are documented as revisions to the Licensing Submittal provided in Attachment 3. Within the revised analyses, the delay time for delivery of 380 gpm to the intact SGs was increased taking into account the following allowances:

- 60 seconds to account for instrumentation delays and startup of the diesels and EFW feed pumps.
- A 30 second time delay following detection of high flow to the faulted (depressurized) SG.
- 18 seconds to close the EFW flow control valves.

For the cases analyzed, high flow would be detected at 60 seconds following EFW initiation, thus providing a total delay time of 108 seconds. Although substantial EFW will be added to the steam generators prior to the 108 second delay, no credit is taken for emergency feedwater flow prior to complete closure of the flow control valves associated with the faulted SG. The revised analyses continue to support the original conclusions of the licensing submittal and demonstrate that the EFW system is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to prevent uncovering the reactor core.

ATTACHMENT 3

PAGE REVISIONS TO SCE&G'S LICENSING SUBMITTAL

This attachment contains page revisions to SCE&G's prior Steam Generator Replacement Licensing Submittal (dated 10/29/93) which reflect the results of the Steam Line Break and Feedwater Line Break Analyses described in Attachments 1 & 2. The pages affected are as follows:

SECTION	TABLE OR FIGURE #	PAGE
LIST OF ACRONYMS & ABBREVIATIONS	-	xi
	-	xii
3.4.3.1	-	3.4-30
	-	3.4-31
	Table 3.4.3-1	3.4-32
	Table 3.4.3-2	3.4-33
3.4.4.1		3.4-41
		3.4-42
		3-4-43
	Table 3.4.4-2	3.4-48
	Table 3.4.4-5	3.4-51
	Table 3.4.4-5	3.4-52
	Table 3.4.4-6	3.4-53
	Table 3.4.4-7	3.4-54
	Table 3.4.4-8	3.4-55
	Table 3.4.4-9	3.4-56
	Figure 3.4.4-7	3.4-63
	Figure 3.4.4-8	3.4-64
	Figure 3.4.4-9	3.4-65
	Figure 3.4.4-10	3.4-66
	Figure 3.4.4-12	3.4-68

ATTACHMENT 3
(Continued)

PAGE REVISIONS TO SCE&G'S LICENSING SUBMITTAL

SECTION	TABLE OR FIGURE #	PAGE
3.4.4.1	Figure 3.4.4-14	3.4-70
	Figure 3.4.4-15	3.4-71
	Figure 3.4.4-16	3.4-72
	Figure 3.4.4-17	3.4-73
Appendix 6, Section 15.4.2.2		A complete replacement for this section is provided.
Appendix 6	Table 15.4-19	Page containing Feedwater Line Break Sequence of Events
	Figure 15.4-83	-
	Figure 15.4-84	-
	Figure 15.4-85	-
	Figure 15.4-86	-
	Figure 15.4-87	-
	Figure 15.4-88	-
	Figure 15.4-89	-
	Figure 15.4-90	-

LIST OF ACRONYMS & ABBREVIATION

ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
BEF	Best Estimate Flow
CCWS	Component Cooling Water System
CHG/SI	Charging/Safety Injection
CH	Channel
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CS	Condensate System
CVCS	Chemical and Volume Control System
DCD	Document Control Desk
DECL	Double-Ended Cold Leg
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EFPM	Effective Full Power Months
EOP	Emergency Operating Procedure
ECT	Eddy Current Testing
EFW	Emergency Feedwater
EPAA	East Penetration Access Area
$F\Delta H$	Hot Channel Enthalpy Rise Factor
F_Q	Total Peaking Factor
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
FWCV	Feedwater Control Valve
FWIV	Feedwater Isolation Valve
GPM	Gallons per Minute
HELB	High Energy Line Break
HZP	Hot Zero Power
IB	Intermediate Building
IFBA	Integral Fuel Burnable Absorbers
IFM	Integrated Flow Mixing
LB	Large Break
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LPZ	Low Population Zone
M/E or M&E	Mass and Energy
MMF	Minimum Measured Flow

LIST OF ACRONYMS & ABBREVIATION

MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MWt	Megawatt Thermal
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPΔT	Overpower Delta T
OTΔT	Overtemperature Delta T
PCT	Peak Clad Temperature
PLOF	Partial Loss of Flow
PORV	Power Operated Relief Valve
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RB	Reactor Building
RBCU	Reactor Building Cooling Unit
RC	Reactor Coolant
RCC	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RSG	Replacement Steam Generator
RSR	Relative Stability Ratio
RTDP	Revised Thermal Design Procedure
RTP	Rated Thermal Power
RWST	Refueling Water Storage Tank
SCE&G	South Carolina Electric & Gas Company
SER	Safety Evaluation Report
SB	Small Break
SI	Safety Injection
SIS	Safety Injection System
SG	Steam Generator
SGR	Steam Generator Replacement
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SWS	Service Water System
TA	Total Allowance
T_{AVG}	RCS Average Temperature
T_{HOT}	Vessel Outlet Temperature
T_{COLD}	Vessel Inlet Temperature
TDF	Thermal Design Flow
V+	VANTAGE+
VCSNS	Virgil C. Summer Nuclear Station
WPAA	West Penetration Access Area

3.4.3 Main Steamline Break Mass/Energy Releases

In support of operation at the uprated power level with the Δ75 replacement SGs and associated revised operating conditions, mass and energy releases following the rupture of a steam line have been performed. Mass and energy releases for breaks occurring both inside, as well as outside of containment, have been calculated and are applicable for core power levels up to 2900 MWt.

Steamline breaks inside of containment result in pressures and temperatures within the containment building which could result in challenges to the equipment inside of containment as well as to the integrity of the containment structure itself. The increased mass inventory of the Δ75 steam generator in comparison to the D3 result in a more severe containment pressure and temperature response than has currently been analyzed. Furthermore, the uprated power level results in a greater RCS stored energy which will be transferred to the steam generators and blown down into containment.

Steamline breaks occurring outside of containment may result in excessive compartment temperatures and pressures which could challenge the ability of equipment to function under the adverse environment. In the analysis of these breaks, the impact of superheated steam in the blowdown due to heat transfer from uncovered tubes has been included. Since the Δ75 replacement steam generator has a much taller tube bundle than the D3 generator, the potential for earlier tube uncovery than is predicted for the D3 exists. Earlier tube uncovery would result in earlier onset of superheat, as well as a potentially greater degree of superheat.

3.4.3.1 Inside Containment

Steamline break mass and energy releases inside of containment have been calculated for a wide range of power levels and break sizes. These analyses included the effect of entrainment in the blowdown fluid for those power level/break size combinations for which entrainment was predicted to occur.

Initial power levels of 102%, 75%, 50%, 25% and 0% (HZP) were analyzed as well as a wide range of break sizes. For the containment pressure/temperature analysis, the following break sizes were considered:

1. A full double ended (guillotine) rupture of a main steam line. The presence of an integral flow restrictor in the steam generator limits the maximum possible line break size to 1.4 ft^2.

2. Small double ended ruptures with entrainment in the blowdown.

3. Small double ended ruptures with no entrainment in the blowdown.

4. Small split ruptures which represent the largest breaks for which the isolation signals are generated first by the high containment pressure signals and result in no entrainment in the blowdown.

Analysis assumptions are chosen to maximize the mass and energy release to the RB: conservative reactor trip setpoint and delay times are utilized to maximize the energy transfer from primary side to the faulted SG; limiting SG level control errors are utilized to maximize the initial water mass in the faulted SG; the amount of main feedwater added to the faulted SG, prior to closure of the FWIV and/or FWCV, is conservatively maximized; higher than expected EFW flows to the faulted SG are used for the duration of the transient; and, the effects of one or more of the following single failures are included.

Failure To Close The FWIV To The Faulted SG	Results in an increase in the unisolable volume of high energy fluid in the feedline of the faulted SG. Additional feedwater, contained between the isolation valve and upsteam control valve (470 ft^3), is added to the faulted SG and subsequently released to the RB.
Failure To Close The MSIV Downstream Of The Faulted SG	Results in the release of additional steam, contained in the steam lines and header downstream of the MSIVs on the intact SGs, to the RB.
Failure To Automatically Close A EFW FCV To The Faulted SG	Prevents automatic isolation of EFW to the faulted SG. Results in the addition of up to 1000 gpm of EFW to the faulted SG until operator action is credited to terminate EFW flow to the faulted SG.
Failure Of One Train Of SI	Results in the minimum capability for injection of high concentration boric acid (2300 ppm) solution thus permitting a higher return to power following trip. Leads to higher RCS temperatures and higher blowdown enthalpies from the faulted SG during return to power conditions.
Failure Of An Emergency Diesel	Results in the loss of one train of SI and one MD EFW pump. **
Failure Of Electrical Channel A	Results in loss of electrical power to the FWCVs, FWIVs, MSIVs, one MD EFW pump; prevents an emergency diesel from being started**; and, prevents automatic closure of a EFW FCV to the faulted SG. As noted above, failure to close a EFW FCV will result in the addition of up to 1000 gpm of EFW to the faulted SG until operator action is credited.

** The loss of a emergency diesel also eliminates one train of the RB Sprays and RB Cooling Units within the containment integrity analyses presented in Section 3.4.4.1.

Table 3.4.4-5 outlines the break spectrum and failure conditions which were analyzed. Tables 3.4.3-1 and 3.4.3-2 contain mass and energy release data for the SLB which has been determined to result in maximum temperature and pressure conditions in containment.

3.4.3.2 Outside Containment

Steamline break mass and energy releases outside of containment have been calculated for the uprated power level for a wide range of break sizes. These analyses included the effect of superheated steam in the blowdown should steam generator tube uncovery occur.

Mass and energy releases were calculated for initial power levels of 102% and 75% of the uprated power level. A wide range of steamline header break sizes were also examined ranging from 4.6 ft^2 down to 0.1 ft^2. This range of cases is identical to those previously analyzed in WCAP-10961, Rev. 1 (Reference 19)

which was prepared in support of the WOG program which analyzed the impact of superheated steam following a steamline break outside containment.

Tables 3.4.3-3 through 3.4.3-8 contain mass and energy release data which has been determined to result in the most severe environmental response.

TABLE 3.4.3-1
VCSNS RSG SLB M&E INSIDE CONTAINMENT
102% POWER, 1.4 FT^2 DER
FAILURES: 1 Train SI

Time (sec)	Break Flow Rate (lbm/sec)	Break Energy (10^6 btu/sec)	Integrated Mass Release (10^6 lbm)	Integrated Energy Release (10^6 btus)
0	0	0	0	0
0.2	8,622	10.28	1.724	2.057
0.4	8,400	10.02	3.404	4.062
0.6	8,255	9.856	5.055	6.033
0.8	8,117	9.695	6.679	7.972
1	7,985	9.541	8.276	9.88
2	7,399	8.856	15.9	19
3	6,908	8.28	23	27.5
4	6,635	7.958	29.7	35.54
5	6,521	7.824	36.26	43.42
6	6,416	7.699	42.72	51.16
7	6,320	7.587	49.08	58.79
8	6,228	7.478	55.34	66.32
9	6,132	7.365	61.52	73.73
10	6,022	7.234	67.58	81.02
10.6	5,944	7.142	71.17	85.32
10.8	2,074	2.491	71.58	85.82
11	2,067	2.482	71.99	86.31
15	1,859	2.236	79.85	95.75
20	1,594	1.919	88.43	106.1
30	1,264	1.522	102.5	123
40	1,101	1.326	114.2	137.1
61.5	1,002	1.206	136.4	163.9
80	978.3	1.178	154.7	185.8
100	964.8	1.161	174.1	209.2
110	959.7	1.155	183.7	220.8
120	981.2	1.181	193.5	232.5
140	981.5	1.181	213.1	256.2
160	962.5	1.158	232.7	279.7
180	250.6	0.2964	244.8	294.3
200	29.23	0.0337	246.7	296.5
206	0	0	246.8	296.6
252	48.42	0.0562	247.8	297.7
506	0	0	254.9	306
752	51.17	0.0594	261.5	313.6
1,000	0	0	268.4	321.6
1,250	49.56	0.0575	275.2	329.5
1,502	0	0	282.4	337.7
1,754	51.92	0.0603	289.1	345.5
1,800	0	0	290.3	346.9
1,808	13.43	0.0155	290.7	347.3
1,810	0	0	290.7	347.3

Steam Line isolation: 10.6 sec

Feedline isolation: 10.6 sec

TABLE 3.4.3-2
VCSNS RSG SLB M&E INSIDE CONTAINMENT
25% POWER, 1.4 FT^2 DER
FAILURES: ELECTRICAL CH-A

Time (sec)	Break Flow Rate (lbm/sec)	Break Energy (10^6 btu/sec)	Integrated Mass Release (10^3 lbm)	Integrated Energy Release (10^6 btus)
0	0	0	0	0
0.2	9,441	11.23	1.888	2.245
0.4	9,207	10.96	3.73	4.437
0.6	9,052	10.78	5.54	6.592
0.8	8,903	10.61	7.321	8.713
1	8,758	10.44	9.072	10.8
2	8,125	9.704	17.44	20.79
3	16,729	12.14	29.66	31.58
4	18,151	12.35	47.31	43.87
5	17,156	11.83	64.86	55.91
6	16,214	11.33	81.45	67.43
7	14,980	10.74	96.93	78.4
7.2	4,711	3.373	97.87	79.08
8	4,397	3.217	101.5	81.7
9	4,009	3.048	105.6	84.81
10	3,691	2.917	109.5	87.77
15	2,562	2.366	124.7	100.9
20	1,540	1.831	134.9	111.3
30	1,205	1.451	148.3	127.4
40	1,064	1.281	159.5	140.9
61.5	1,002	1.206	181.4	167.3
80	987.6	1.189	199.8	189.4
100	980	1.18	219.4	213.1
120	982.1	1.182	239	236.7
140	979.7	1.179	258.7	260.3
160	976	1.175	278.2	283.8
180	299.1	0.3558	291	299.2
200	297.1	0.3525	297.2	306.5
220	458.4	0.5478	303.6	314.2
240	236.6	0.2812	309.1	320.7
260	186.6	0.2205	313.8	326.3
280	143.9	0.169	317.3	330.3
300	141	0.1653	320.1	333.7
340	137.5	0.1612	325.6	340.1
1,800	137.5	0.1612	526.4	575.5
1,802	138	0.1619	526.7	575.9
1,828	0	0	527.5	576.8

Steamline isolation: 7.0 sec

Feedline isolation: 8.5 sec

in a short term superheated temperature transient for dry steam blowdowns until containment spray is actuated.

Reactor Building Sprays

The Virgil C. Summer Nuclear Station has two independent, 100% capacity Reactor Building Spray trains which are actuated upon a high containment pressure of 12.31 psia. The assumed performance characteristics of the Reactor Building sprays are shown in Table 3.4.4-2. Consistent with the current licensing basis analyses assumptions (Section 6.2.1.3.4.2 of the FSAR), the sprays are assumed to be 100% efficient in removing heat from the Reactor Building. RB sprays are initiated within 53.1 seconds consistent with the current licensing basis analysis for the large SLBs. For smaller breaks, RB sprays are initiated based on receipt of the RB Hi-3 pressure setpoint plus 43.1 seconds for pump start and header fill consistent with FSAR Table 6.2-48B..

Reactor Building Cooling Units

The Reactor Building Cooling Units (RBCUs) are modeled in CONTEMPT-LT26 as described in FSAR section 6.2.1.3.4.3. It is conservatively assumed that the RBCUs start at 86.5 seconds (time that the fan coolers are operating at full design capability) following accident initiation for the large DERs. The RBCU heat removal rate has been conservatively reduced by 40% below current licensing bases analysis assumptions (FSAR Figure 6.2-15) to allow for future potential degradation in those units. For smaller breaks the RBCUs are actuated based on receipt of the RB Hi-1 pressure setpoint plus 76.5 seconds.

__Scope Of Analysis__

Reactor Building pressure and temperature analyses have been performed for the spectrum of breaks outlined in Table 3.4.4-5. This spectrum is similar to that currently shown in FSAR Table 6.2-1 and includes the following:

1. 1.4 ft^2 Double Ended Ruptures at 102% power with various single failures. These analyses establish the limiting single failure which maximizes the resulting SLB P&T conditions.

2. 1.4 ft^2 Double Ended Ruptures at 102, 75, 50, 25 and 0% power. These analyses show the impact of power level on the reactor building P&T utilizing the most limiting single failure.

3. Small Double Ended Ruptures at 102% power. These analyses utilize multiple single failures and show the impact of break size on the reactor building consequences.

4. Smallest Double Ended Ruptures at 75, 50, 25, and 0% which result in liquid entrainment during the initial blowdown of the SG. These analyses are conducted with multiple single failures and show the impact of break size and power level on RB pressure.

5. Largest Double Ended Ruptures at 75, 50, 25, and 0% power which result in __no__ liquid entrainment during the initial blowdown of the SG. These analyses are conducted with multiple single failures and show the effects of break size and power level on the RB temperature.

6. Largest split breaks at 102, 75, 50, 25, and 0% power which result in no liquid entrainment during the initial blowdown of the SG or secondary side isolation signal. These analyses assume multiple

single failures, rely upon Reactor Building pressure signals for steam and feedwater isolation, and are conducted to quantify the impact on RB temperature.

As indicated above, the effects of various single failures on the resulting SLB pressure and temperature conditions are examined. These single failures were selected to either maximize the mass and energy release from the faulted SG or to minimize the performance of the RB cooling systems. They include the following:

1. MSIV Failure: The MSIV nearest the break and faulted SG is assumed to not close. This allows mass and energy in unisolated steam piping downstream of the isolation valves, including the steam line header, cross-connecting lines and steam dump piping, to be released to the Reactor Building. This additional mass and energy is added to the mass and energy described in Section 3.4.3.1.

2. FWIV Failure: FW isolation within 10 seconds of an actuating single is achieved via closure of the FWIVs (closest to containment) and FCVs. Failure of the FWIV on the faulted SG is assumed. This allows the additional mass and energy located between the control and isolation valve to be added to the faulted SG and ultimately to the Reactor Building. This failure is accounted for in the M&E analyses as discussed in Section 3.4.3.1.

3. EFW FCV Failure: The EFW System is designed to automatically isolate EFW to the faulted SG. Upon detection of high flow to a depressurized SG, FCVs at the discharge of the motor and turbine driven EFW pumps are closed, redirecting flow from the faulted SG to the intact SGs. Failure to automatically close a FCV, will allow continuous EFW flow (up to 1000 gpm) to the faulted SG until operator action is credited to terminate EFW flow to the faulted SG. This failure is accounted for in the M&E analyses as discussed in Section 3.4.3.1.

4. Diesel Failure: A failure of an emergency diesel to start is postulated resulting in the loss of one Chg/SI pump and one motor driven EFW pump, both of which is accounted for in the M&E analyses, and the loss of one train of RB Sprays and RB Cooling Units.

5. Electrical Channel A Failure: A failure of the worst electrical channel is assumed. Within the M&E release analyses (Section 3.4.3.1), this failure results in the loss of power to the FWCVs, FWIVs, MSIVs, and one motor driven EFW pump and prevents automatic closure of a EFW FCV to the faulted SG. As noted above, failure to automatically close a FCV results in the continuous addition of EFW (at up to 1000 gpm) to the faulted SG until operator action is credited to terminate EFW flow to the faulted SG. In addition, this failure also prevents an emergency diesel from being started thus making available only one of the two trains of RB Sprays and RB Cooling Units during the RB pressure and temperature evaluation.

In addition to the above, a SI Pump failure is also assumed when calculating the SLB mass and energy release data used for these Reactor Building pressure and temperature calculations as discussed in Section 3.4.3.1.

Reactor Building Pressure and Temperature Results

Tables 3.4.4-6 to 3.4.4-9 present peak calculated Reactor Building pressures and temperatures for the spectrum of SLB analyzed. These results show the following:

1. Analyses of the 1.4 ft^2 Double Ended Rupture at 102% and 25% power show that RB pressures are maximized during a SLB with a coincident failure of Electrical Channel A as shown in Figure 3.4.4-7. This limiting single failure maximizes the mass addition to the faulted SG while eliminating one train of RB cooling.

2. The spectrum analyses of Double Ended Ruptures at 102% power show that peak RB pressures decrease with decreasing break size given the same single failure assumptions.

3. Maximum RB pressures occur for the large Double Ended Ruptures. The sensitivity to power level is shown in Figures 3.4.4-8 and 3.4.4-9.

4. The peak Reactor Building pressure occurs for the 1.4 ft^2 Double Ended Rupture at 25% power. The calculated peak pressure is 53.0 psig at 1800 seconds as shown in Figure 3.4.4-10. A 4 psig margin to the RB design pressure of 57 psig. is maintained.

5. Superheated temperature conditions occur for the SLBs with dry blowdowns. Figures 3.4.4-11 through 3.4.4-16 show the sensitivity of peak RB temperatures for the break spectrum analyzed. Limiting conditions occur for the large Double Ended Rupture from 102% power.

6. The peak RB temperature occurs for the 1.4 ft^2 Double Ended Rupture at 102% assuming a MSIV failure. The calculated peak temperature is 372.7 °F as shown in Figure 3.4.4-17. The superheated temperatures within the RB are limited in magnitude and duration via MSIV closure (stops the initial blowdown of two of the three SGs) and spray actuation. Following spray actuation, the Reactor Building remains saturated in the long term and below the RB design temperature of 283 °F.

<u>Impact On Current Design Basis</u>

Current RB design basis conditions for VCSNS during a SLB with the Model D3 SGs are (per Table 6.2-3 of the FSAR) as follows:

1. The Peak RB Pressure is 45.96 psig for a 1.4 ft2 Double Ended Rupture at an initial core power of 102% of 2775 MWt.

2. The Peak RB Temperature is 321.5 F for a 0.645 ft2 Split Break at an initial core power of 102 % of 2775 MWt.

The Replacement SGs, which will operate with a larger secondary water mass than the current D3 SGs, are shown to result in an increase of approximately 7.0 psi in the maximum calculated RB pressure. From a temperature standpoint, the Replacement SGs result in a increase of approximately 51.2 °F in the peak RB temperature and a shift in the limiting break size from a small split break to the full Double Ended Rupture. This shift in break size is the result of no entrainment being predicted for the Delta 75 SGs for SLBs from initial conditions corresponding to 102% power.

<u>Conclusions</u>

The SLB Reactor Building pressure and temperature analyses show increases in both RB pressure and temperature conditions during a postulated SLB with the Δ75 SG. The calculated peak pressure increases

approximately 7.0 psig to 53 psig; approximately 4 psig margin is, however, maintained relative to the RB design pressure of 57 psig. The calculated peak temperature increases approximately 51.2 °F to 372.7 °F. Superheated conditions within the Reactor Building are predicted but are of short duration. Following spray actuation, the Reactor Building remains saturated in the long term and below the RB design temperature of 283 °F. Impacts of the new SLB pressure and temperature conditions inside the RB will be reconciled from an equipment qualification standpoint as discussed in Section 3.4.6.

3.4.4.2 LOCA Reactor Building Integrity Analysis

Introduction:

Analyses have been completed to determine the Reactor Building pressure and temperature response during postulated LOCAs using mass and energy release which incorporates the Replacement Steam Generators (RSGs) and revised design power capability parameters for the VCSNS. The results of these analyses demonstrate that the RSGs have a small impact on LOCA consequences within the Reactor Building and that the Reactor Building design conditions remain bounding.

Analytical Approach:

Method of Analysis

The Reactor Building pressure and temperature response is calculated using the CONTEMPT-LT26 (Reference 11) computer code. This is a deviation from the current LOCA licensing basis analyses (FSAR Section 6.2) which used CONTEMPT-LT22 (Reference 12). Calculations for the same mass and energy releases and modeling assumptions, however, show that both CONTEMPT-LT codes predict

TABLE 3.4.4-2

GENERAL CONTAINMENT DESIGN AND EVALUATION PARAMETERS

General Design Information

Maximum Internal Design Pressure, psig	57
Maximum External Design Pressure, psig	3.5 psig
Design Temperature, °F	283
Free Volume, ft^3	1.84×10^6
Design Leak Rate, max. allowable, %/day	0.2

Engineered Safety Features	Full Capacity	Value Used for Analysis
Passive Safety Injection:		
Number of Accumulators	3	2 or 3
Pressure Setpoint, psig	600	600
Active Safety Injection:		
Residual Heat Removal Flow Rate, lb/sec	900.4	460.2 or 990.4
Reactor Building Spray System:		
Number of Lines	2	1 or 2
Number of Pumps	2	1 or 2
Number of Headers	6	3 or 6
Design Flow, gpm	5,000	2,500 or 5,000
Reactor Building Cooling Units:		
Number	4	1 or 2
Air Side Flow Rate, cfm/unit	60,270[(1)]	54,200[(1)]
Heat Removal Rate at 283°F, Btu/hr/unit	125×10^6	75×10^6[(2)]

Notes:

(1) This parameter is used only in the Chapter 15 Radiological Consequence Analysis for particulate iodine removal post-LOCA. This parameter is not used in the Chapter 6 Pressure/Temperature Analyses.

(2) Corresponds to the maximum capacity (60%) assumed in the SLB analyses.

TABLE 3.4.4-5

MAIN STEAM LINE BREAKS ANALYZED

Double Ended Ruptures[1] (No Entrainment) To Assess Impact of Single Failures

Area/Power	Failures(s) Assumed [2]
1.4 ft^2/102%	MSIV [3]
1.4 ft^2/102%	FWIV[4]
1.4 ft^2/102%	EFW FCV[5]
1.4 ft^2/102%	Diesel[6]
1.4 ft^2/102%	CH-A [7]
1.4 ft^2/25%	FWIV[4]

Double Ended Ruptures (With Entrainment) to Assess Impact of Power Level

Area/Power	Failures(s) Assumed[2]
1.4 ft^2/75%	CH-A
1.4 ft^2/50%	CH-A
1.4 ft^2/25%	CH-A
1.4 ft^2/ 0%	CH-A

Double Ended Ruptures (No Entrainment) to Assess Impact of Break Size

Area/Power	Failures(s) Assumed[2]
1.2 ft^2/102%	FWIV, MSIV, EFW FCV, Diesel
1.1 ft^2/102%	FWIV, MSIV, EFW FCV, Diesel

Small Double Ended Ruptures (With Entrainment)

Area/Power	Failures(s) Assumed[2]
1.1 ft^2/75%	FWIV, MSIV, EFW FCV, Diesel
0.8 ft^2/50%	FWIV, MSIV, EFW FCV, Diesel
0.6 ft^2/25%	FWIV, MSIV, EFW FCV, Diesel
0.2 ft^2/ 0%	FWIV, MSIV, EFW FCV, Diesel

TABLE 3.4.4-5

MAIN STEAM LINE BREAKS ANALYZED

Small Double Ended Ruptures (No Entrainment)

Area/Power	Failures(s) Assumed[2]
1.0 ft^2/75%	FWIV, MSIV, EFW FCV, Diesel
0.7 ft^2/50%	FWIV, MSIV, EFW FCV, Diesel
0.5 ft^2/25%	FWIV, MSIV, EFW FCV, Diesel
0.1 ft^2/ 0%	FWIV, MSIV, EFW FCV, Diesel

Split Ruptures (No Entrainment)

Area/Power	Failures(s) Assumed[2]
0.878 ft^2/102%	FWIV, EFW FCV, Diesel
0.871 ft^2/ 75%	FWIV, EFW FCV, Diesel
0.863 ft^2/ 50%	FWIV, EFW FCV, Diesel
0.849 ft^2/ 25%	FWIV, EFW FCV, Diesel
0.772 ft^2/ 0%	FWIV, EFW FCV, Diesel

Notes:

1. Effective break area for broken loop is 1.4 ft^2.
2. Failure of one Chg/SI pump is assumed in all analyses.
3. MSIV = Main steam isolation valve fails to close.
4. FWIV = Main feedwater isolation valve fails to close.
5. EFW FCV = EFW control valve, which terminates flow to the faulted SG, fails to close.
6. Diesel = One emergency diesel fails to start.
7. CH-A = Failure of Electrical Channel A.

TABLE 3.4.4-6
REACTOR BUILDING PEAK PRESSURE/TEMPERATURE
FOR A SPECTRUM OF LARGE MSLB'S

BREAK SIZE, FT^2 & % POWER	1.4/102 % (CASE 1G)	1.4/102 % (CASE 1H)	1.4/102 % (CASE 1I)	1.4/102 % (CASE 1J)	1.4/102 % (CASE 1K)	1.4/75 % (CASE 2E)	1.4/50 % (CASE 3E)	1.4/25 % (CASE 4E)	1.4/25 % (CASE 4G)	1.4/0 % (CASE 5F)
R B SPRAY	2	2	2	1	1	1	1	1	2	1
RBCU'S	2	2	2	1	1	1	1	1	2	1
FAILURES	FWIV SI PUMP	MSIV SI PUMP	EFW FCV SI PUMP	DIESEL	CH-A	CH-A	CH-A	CH-A	FWIV SI PUMP	CH-A
PEAK PRESSURE (PSIG)	50.4	48.1	47.9	48.2	49.3	49.6	50.9	53	52.6	51.7
TIME TO PEAK PRESSURE (SEC)	204	177	185	173	1,800	1,800	1,800	1,800	193	1,900
PEAK TEMPERATURE (°F)	360.8	372.7	360.9	361	352.1	283.7	275.4	278.1	277.8	276.4
TIME TO PEAK TEMPERATURE (SEC)	31	19	31	30	53	4	1,800	1,800	193	1,900

TABLE 3.4.4-7
REACTOR BUILDING PEAK PRESSURE/TEMPERATURE FOR A SPECTRUM OF SMALL MSLB'S WITH ENTRAINMENT

BREAK SIZE, FT^2 /POWER LEVEL %	1.1/75% (CASE 2B)	0.8/50% (CASE 3B)	0.6/25% (CASE 4B)	0.2/0% (CASE 5B)
R B SPRAY	1	1	1	1
RBCU'S	1	1	1	1
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)
PEAK PRESSURE, PSIG	51.3	48.7	49.9	25.9
TIME TO PEAK PRESSURE, SEC.	1,200	1,200	1,200	1,200
PEAK TEMPERATURE, °F	276	272.4	274.1	232.8
TIME TO PEAK TEMPERATURE, SEC.	1,200	1,200	1,200	1,200

Notes: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel.

TABLE 3.4.4-8
REACTOR BUILDING PEAK PRESSURE/TEMPERATURE
FOR A SPECTRUM OF SMALLER MSLB'S WITHOUT ENTRAINMENT

BREAK SIZE, FT^2 /POWER LEVEL %	1.2/102 (CASE 1B)	1.1/102% (CASE 1C)	1.0/75% (CASE 2C)	0.7/50% (CASE 3C)	0.5/25% (CASE 4C)	0.1/0% (CASE 5C)
R B SPRAY	1	1	1	1	1	1
RBCU'S	1	1	1	1	1	1
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)
PEAK PRESSURE, PSIG (Note 2)	51.6	50.3	50.2	51.8	49.7	13.7
TIME TO PEAK PRESSURE, SEC.	1,200	1,200	1,200	1,200	1,200	1,200
PEAK TEMPERATURE, °F	350.5	347.2	341.1	322	303.8	251.6
TIME TO PEAK TEMPERATURE, SEC.	49	53	53	58	67	480

Notes: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel.

TABLE 3.4.4-9
REACTOR BUILDING PEAK PRESSURE/TEMPERATURE FOR A SPECTRUM OF SPLIT MSLB'S WITHOUT ENTRAINMENT

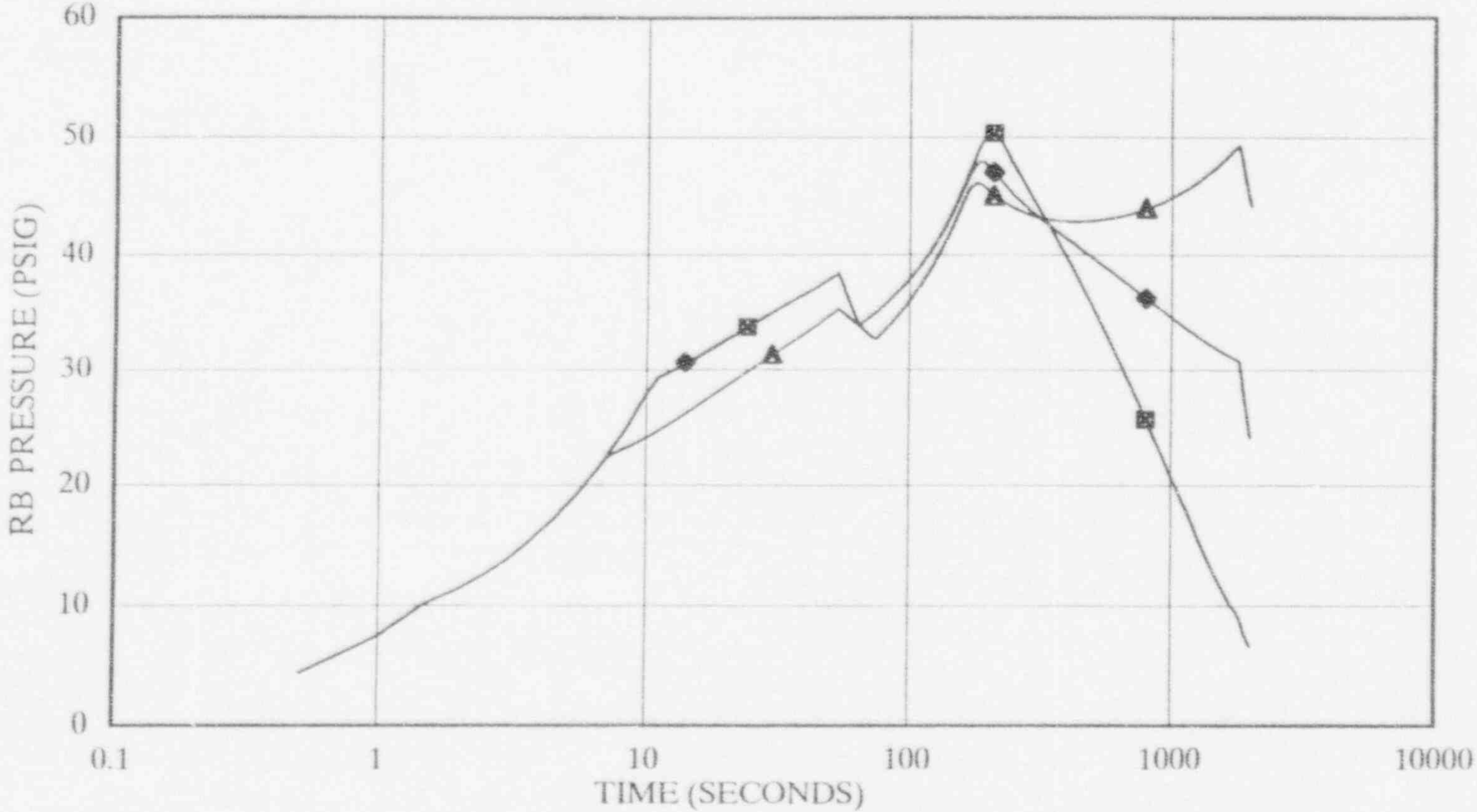
BREAK SIZE, FT^2 /POWER LEVEL %	.878/102% (CASE 1B)	0.871/75% (CASE 2B)	0.863/50% (CASE 3B)	0.849/25% (CASE 4B)	0.772/0% (CASE 5B)
R B SPRAY	1	1	1	1	1
RBCU'S	1	1	1	1	1
OTHER FAILURES	(Note 1)	(Note 1)	(Note 1)	(Note 1)	(Note 1)
PEAK PRESSURE, PSIG	35	35	38.1	37.8	35.7
TIME TO PEAK PRESSURE, SEC.	230	280	320	245	400.0
PEAK TEMPERATURE, °F	331.9	329.5	327.9	325.9	318.1
TIME TO PEAK TEMPERATURE, SEC.	67	67	67	67	67

Notes: 1. Multiple failures assumed: MSIV, FWIV, EFW FCV, and Diesel.

FIGURE 3.4.4-7

V. C. SUMMER NUCLEAR STATION

MSLB RB PRESSURE – SINGLE FAILURE SENSITIVITY



FWIV FAILURE

EFW FCV

ELECTRICAL CHANNEL "A" FAILURE

1.4 FT DER 102% POWER SINGLE FAILURE SENSITIVITY

FIGURE 3.4.4-8

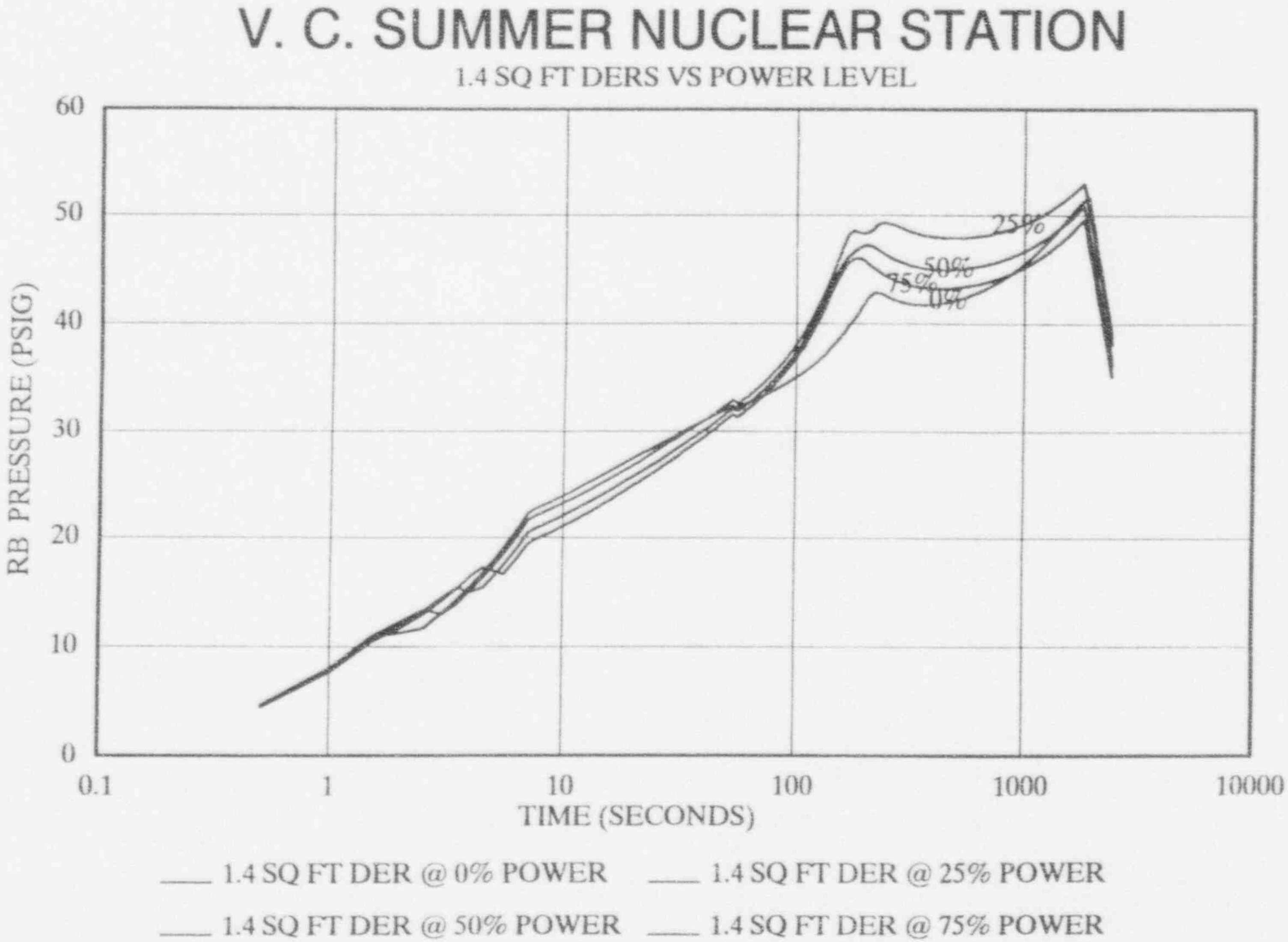

V. C. SUMMER NUCLEAR STATION
1.4 SQ FT DERS VS POWER LEVEL
RB PRESSURE (PSIG)
60
50
40
30
20
10
0
0.1
1
10
100
1000
10000
TIME (SECONDS)
25%
50%
75%
0%
1.4 SQ FT DER @ 0% POWER
1.4 SQ FT DER @ 25% POWER
1.4 SQ FT DER @ 50% POWER
1.4 SQ FT DER @ 75% POWER

FIGURE 3.4.4-9
PEAK RB PRESSURE VS LOAD FOR 1.4 FT2 DE SLB

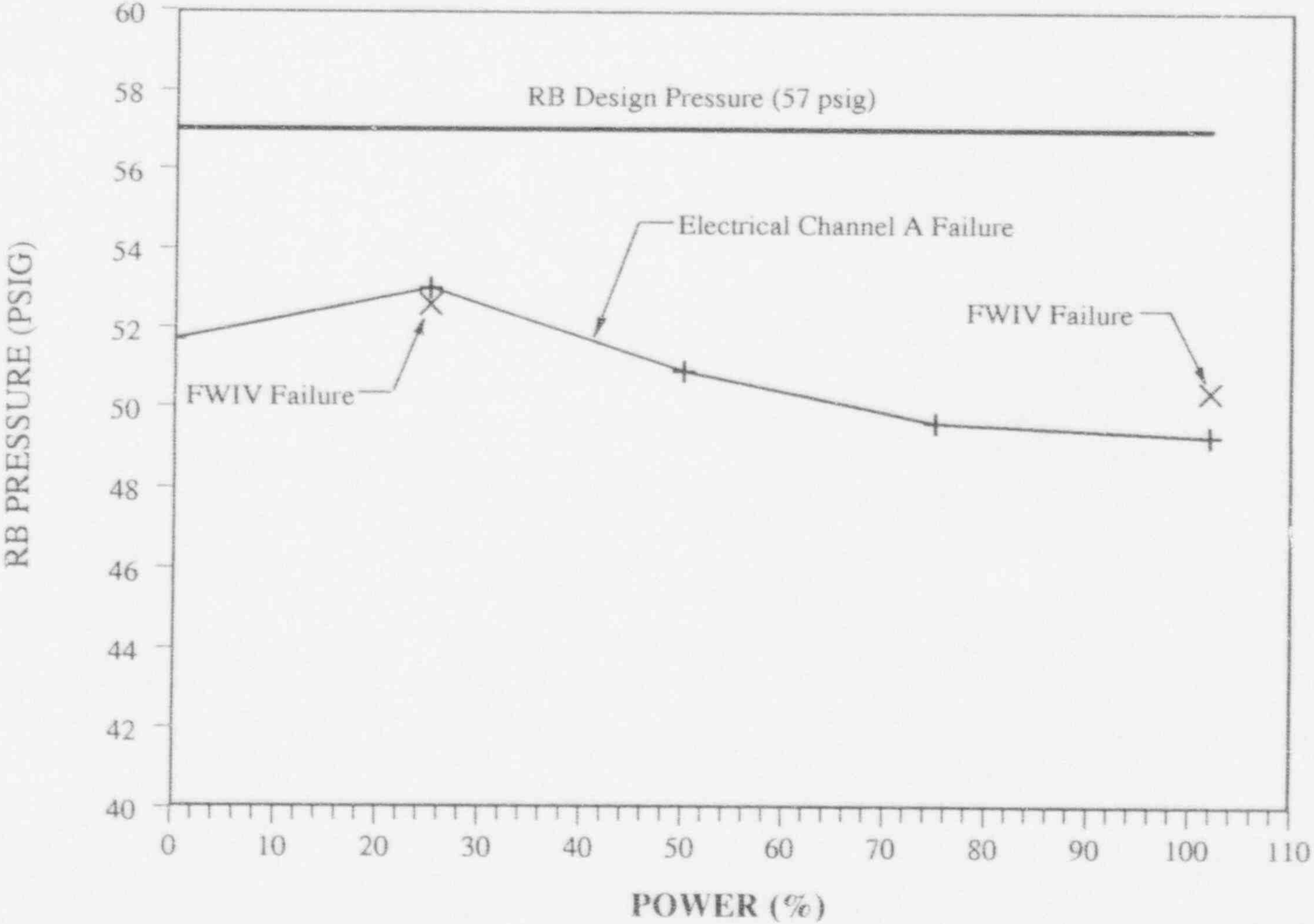


FIGURE 3.4.4-10

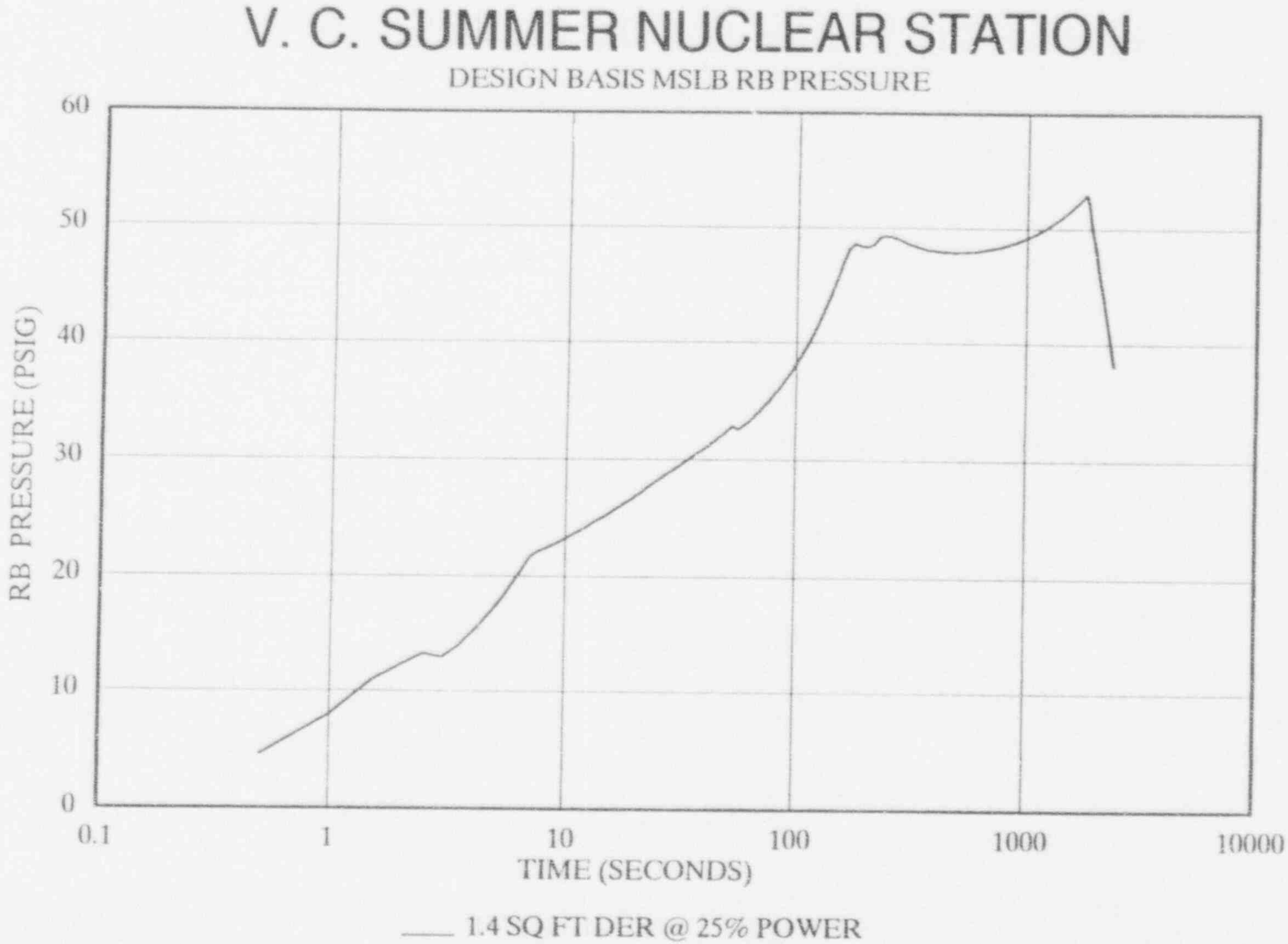


FIGURE 3.4.4-12
PEAK RB TEMP FOR SMALL DE SLB WITHOUT ENTRAINMENT

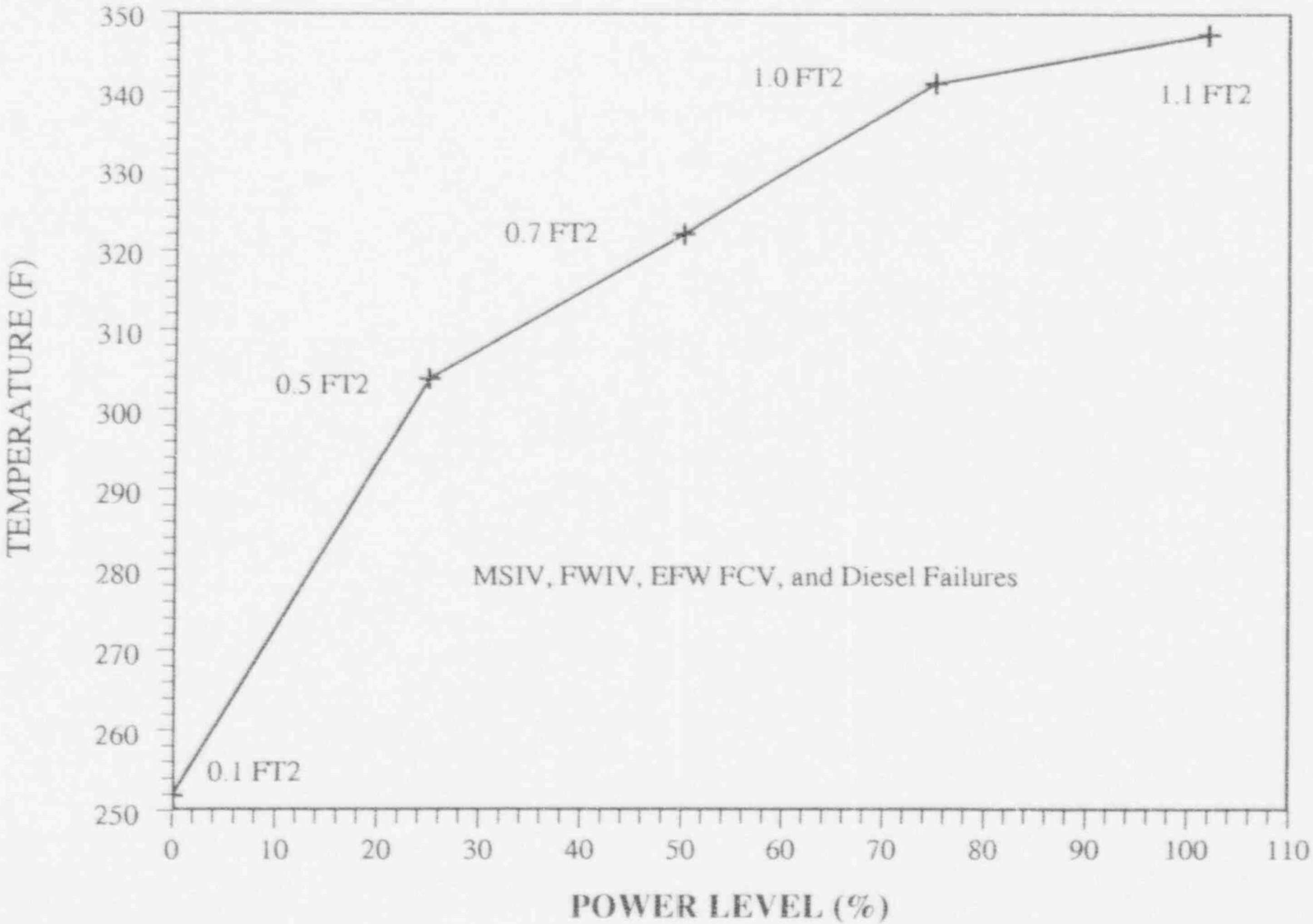


FIGURE 3.4.4-14
PEAK RB TEMP FOR SMALL SPLIT SLBs W/O ENTRAINMENT

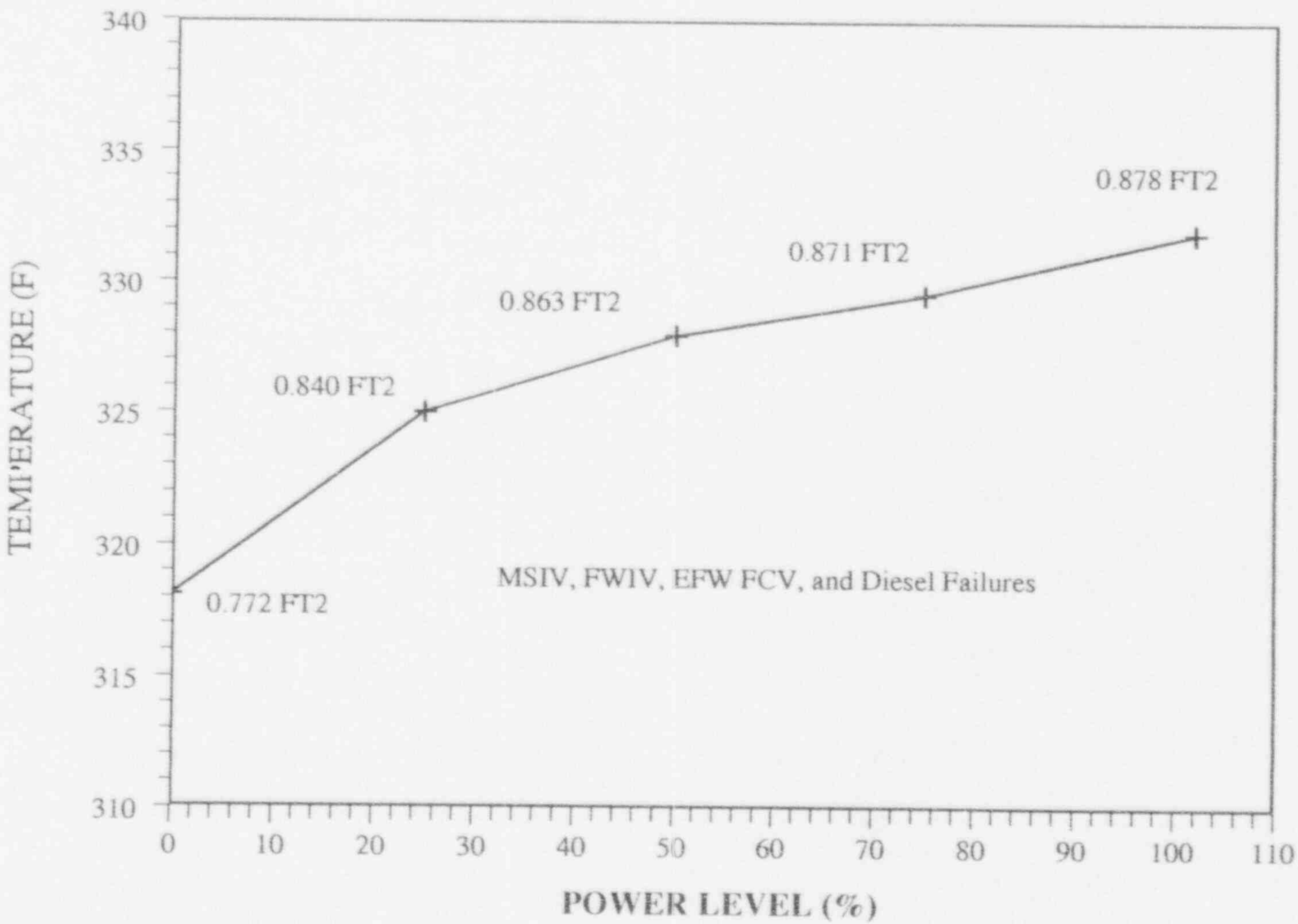


FIGURE 3.4.4-15

V. C. SUMMER NUCLEAR STATION

102% POWER MSLB'S – RB TEMPERATURE

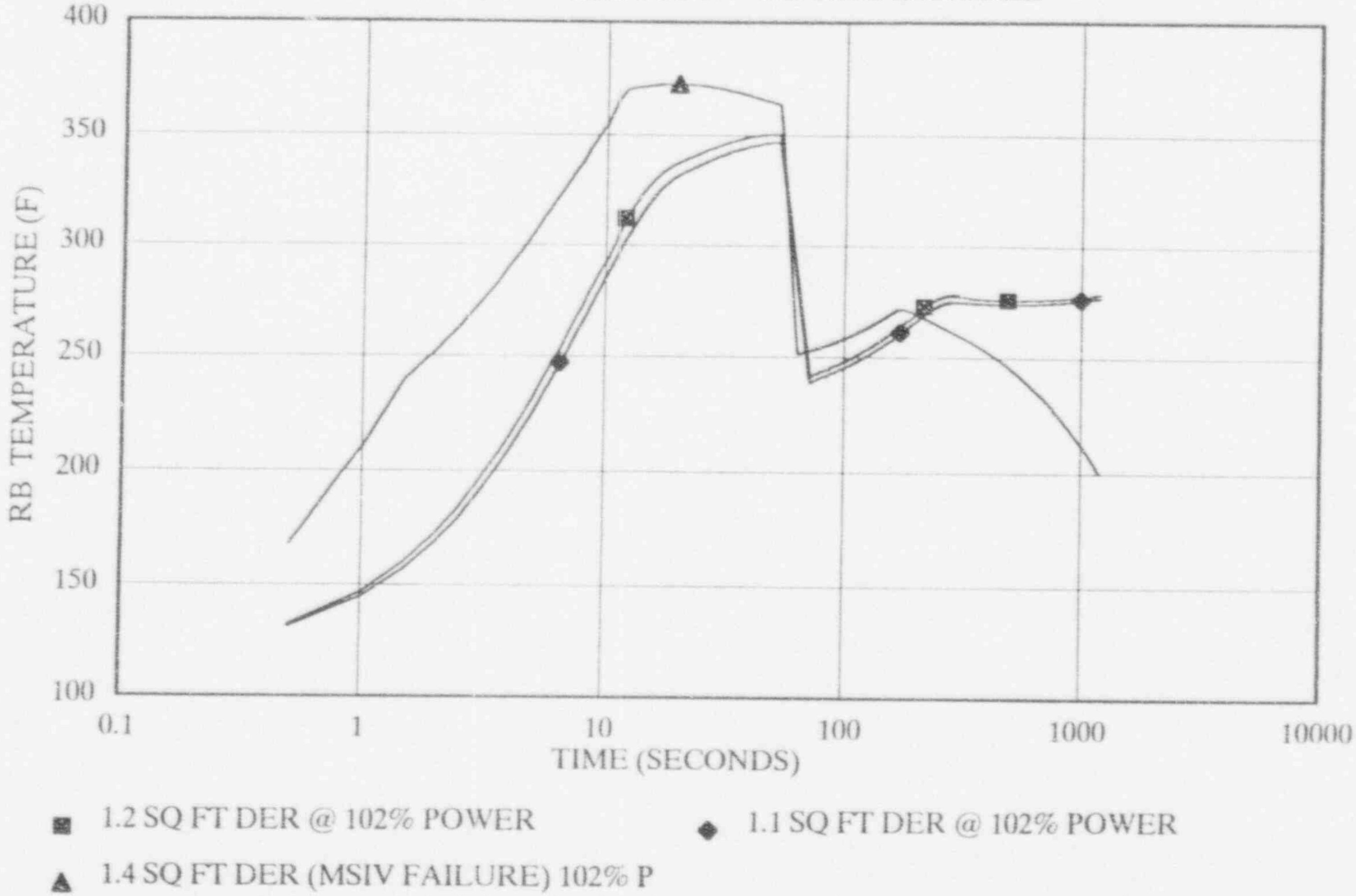


FIGURE 3.4.4-16
PEAK RB TEMP FOR DESLB @ 102% POWER

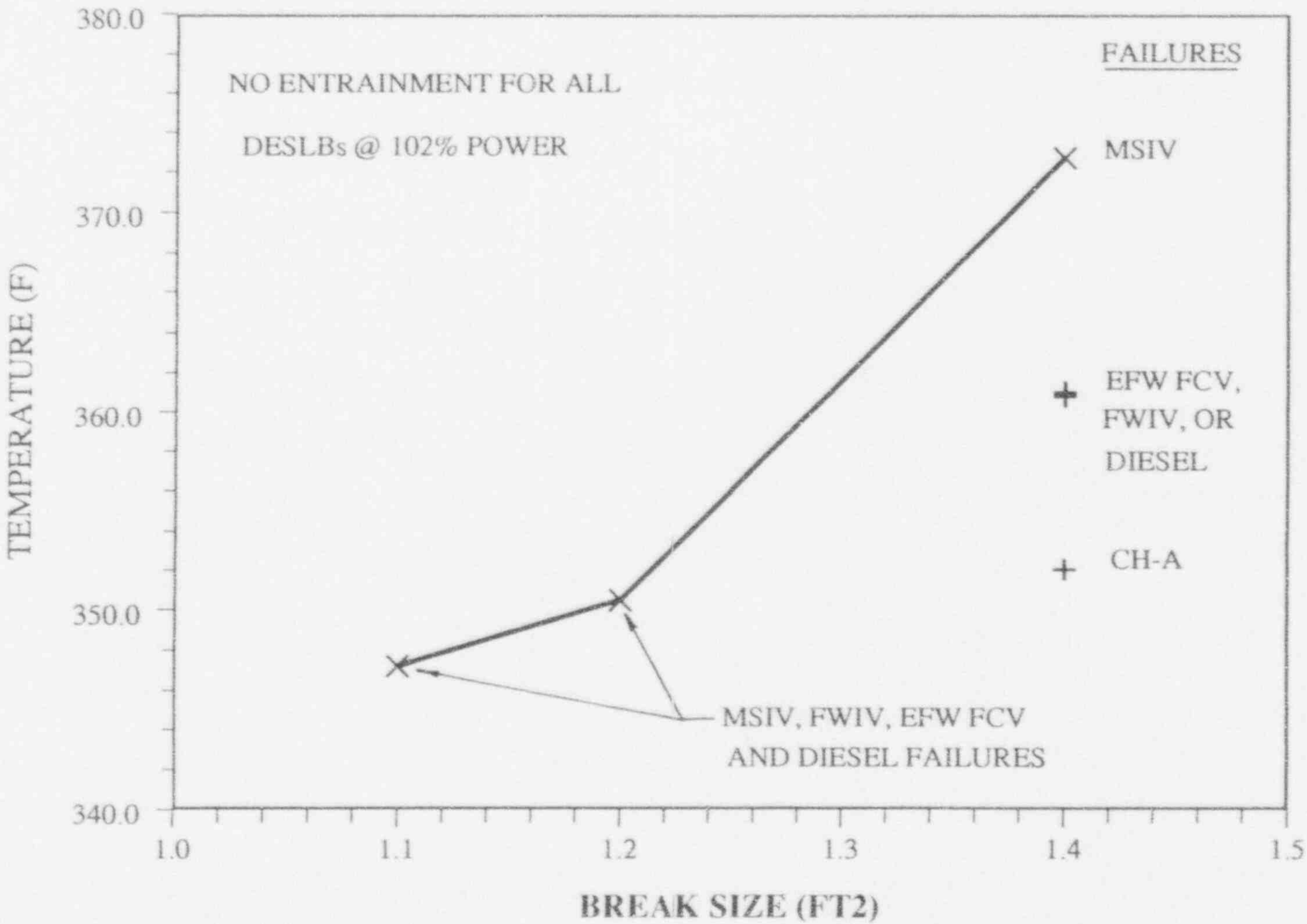
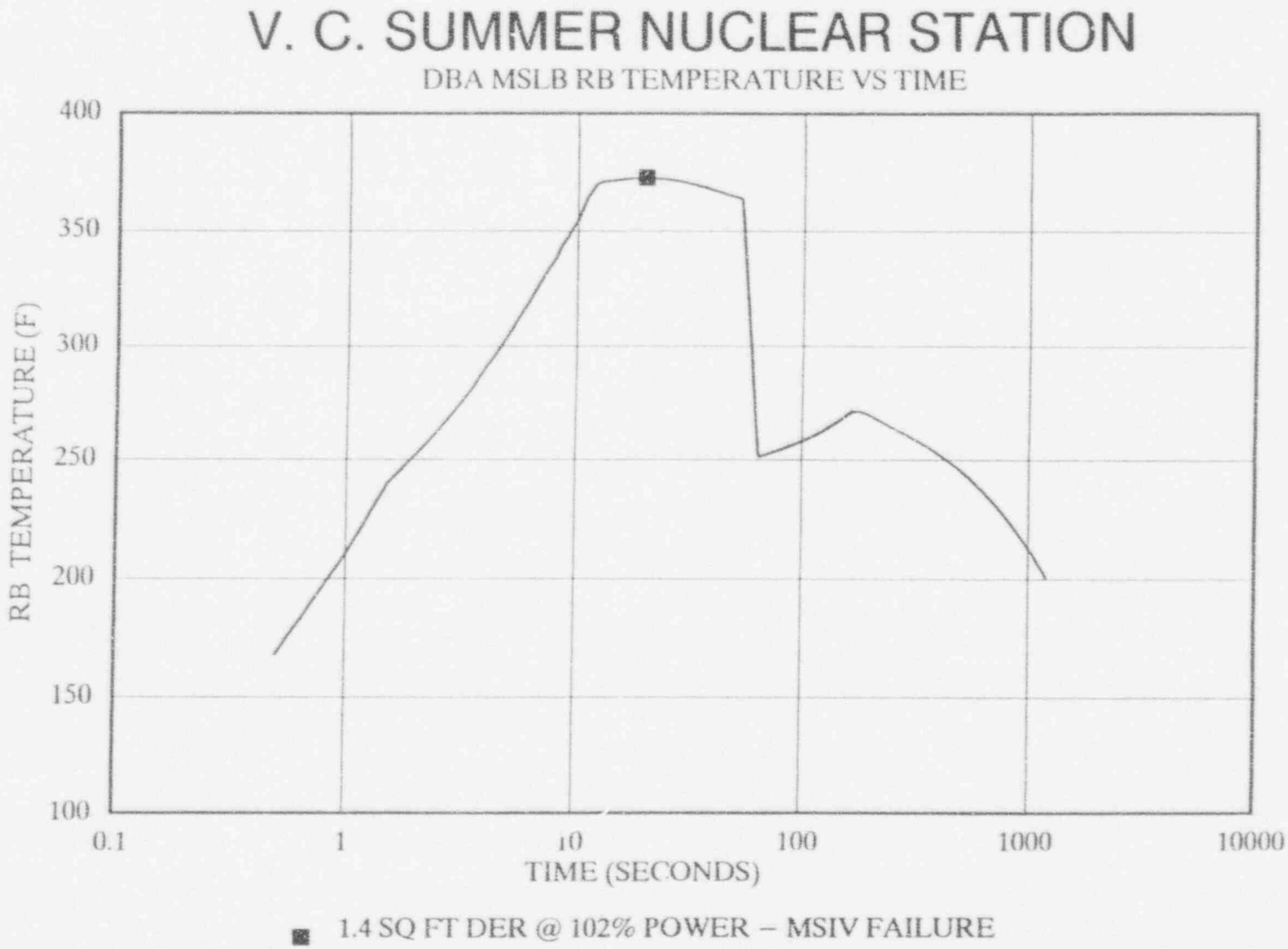


FIGURE 3.4.4-17



15.4.2.2 Major Rupture of a Main Feedwater Line

15.4.2.2.1 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 feedline 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 feedline check valve would affect the NSSS only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.8).

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break), or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in Section 15.4.2.1. Therefore, only the reactor coolant system heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the reactor coolant system because of the following reasons:

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

2. Liquid in the steam generator may be discharged through the break, and would then 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.

An emergency feedwater system is provided to assure that adequate feedwater will be available such that:

1. No substantial overpressurization of the reactor coolant system shall occur; and

2. Liquid in the reactor coolant system shall be sufficient to cover the reactor core at all times.

The following provides the necessary 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. Low steam generator level plus steam/feed flow mismatch in any steam generator.

    e. Safety-injection signals from any of the following:
        (1) Low steam line pressure.
        (2) High containment pressure (Hi-1).
        (3) High steam line differential pressure.

    (Refer to FSAR Chapter 7 for a description of the actuation system)

2. An emergency feedwater system to provide an assured source of feedwater to the steam generators

for decay heat removal. (Refer to FSAR Section 10.4.9 for a description of the emergency feedwater system.)

15.4.2.2.2 Analysis of Effects and Consequences

15.4.2.2.2.1 Method of Analysis

A detailed analysis using the LOFTRAN[25] Code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, reactor coolant system 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.

Major assumptions are:

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 above the nominal value, and the initial pressurizer pressure is 50 psi above its nominal value.

3. A conservatively high initial pressurizer level is assumed; initial steam generator water level is at the nominal value plus 5% in the faulted steam generator, and at the nominal value minus 5% in the intact steam generators.

4. No credit is taken for the pressurizer power operated relief valves or pressurizer spray.

5. No credit is taken for the high pressurizer pressure reactor trip. Note: This assumption is made for calculational convenience. Pressurizer power operated relief valves and spray could act to delay the high pressure trip. Assumptions 3 and 4 permit evaluation of one hypothetical, limiting case rather than two possible cases: one with a high pressure trip and no pressure control; and one with pressure control but no high pressure 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 which minimizes the steam generator fluid inventory at the time of trip and maximizes the blowdown discharge rate following the time of trip, and thereby maximizes the resultant heatup of the reactor coolant.

8. A conservative feedline 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 conservatively low blowdown quality is assumed until all water inventory is discharged from the affected steam generator. A low blowdown quality after trip results in a relatively small amount of energy release through the break. This increases the amount of energy which must be removed via the emergency feedwater system.

9. Reactor trip is assumed to be initiated when the low-low level setpoint in the ruptured steam generator is reached. A low-low level setpoint of 0% narrow range span is assumed.

10. The emergency feedwater system is actuated by the low-low steam generator water level signal, and is assumed to supply a total 380 gpm to the unaffected steam generators within 108 seconds. The assumed time delay includes the following allowances: 60 seconds to account for instrumentation delays and startup of the diesels and emergency feed pumps and 48 seconds to automatically isolate emergency feed flow to the faulted SG. Although substantial emergency feedwater will be added to the steam generators prior to 108 seconds, no credit is taken for emergency feedwater flow prior

to completion of the automatic actions to isolate flow to the faulted SG. Before the relatively cold (120°F) emergency feedwater enters the unaffected steam generators, additional time is also modeled to allow for the purging of 5 cubic feet of hot water contained in the emergency feedwater system lines.

11. No credit is taken for heat energy deposited in reactor coolant system metal during the reactor coolant system heatup.

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

13. Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS 5.1[28] decay heat standard plus uncertainty was used for calculation of residual decay heat levels.

14. No credit is taken for charging or letdown.

15.4.2.2.2.2 Results

Results for two feedline break cases are presented. Results for a case in which offsite power is assumed to be available are presented in Section 15.4.2.2.2.2.1. Results for a case in which offsite power is assumed to be lost following reactor trip are presented in Section 15.4.2.2.2.2.2. The calculated sequence of events for both cases is listed in Table 15.4-18.

15.4.2.2.2.2.1 Feedline Rupture with 0ffsite Power Available

The system response following a feedwater line rupture, assuming offsite power is available, is presented in Figures 15.4-83 through 15.4-86. Results presented in Figures 15.4-84 and 15.4-86 show that pressures in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to the pressurizer safety valve setpoint. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4-84 shows that the water volume in the pressurizer increases in response to the heatup, pressurizer water relief begins at
668 seconds. At approximately 2600 seconds, decay heat generation decreases to a level such that the total RCS heat generation (decay heat plus pump heat) is less than emergency feedwater heat removal capability, and RCS pressure and temperature begin to decrease.

The results show that the core remains covered at all times and that no boiling occurs in the reactor coolant loops.

15.4.2.2.2.2.2 Feedline Rupture with Offsite Power Unavailable

The system response following a feedwater line rupture without offsite power available is similar to the case with offsite power available. However, as a result of the loss of offsite power (assumed to occur at reactor trip), the reactor coolant pumps coast down. This results in a reduction in total RCS heat generation by the amount produced by pump operation.

The reduction in total RCS heat generation produces a milder transient than in the case where offsite power is available. Results presented in Figures 15.4-88 and 15.4-90 show that pressure in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to a peak pressure of 2507 psia at
4056 seconds. This increase in pressure is the result of coolant expansion caused by the reduction in heat

transfer capability in the steam generators. Figure 15.4-88 shows that the water volume in the pressurizer increases in response to the heatup, pressurizer water relief begins at 1122 seconds. At approximately 1120 seconds, decay heat generation decreases to a level less than the emergency feedwater heat removal capability, and RCS temperatures begin to decrease. The results show that the core remains covered at all times since the pressurizer does not empty.

5.4.2.2.3 Conclusions

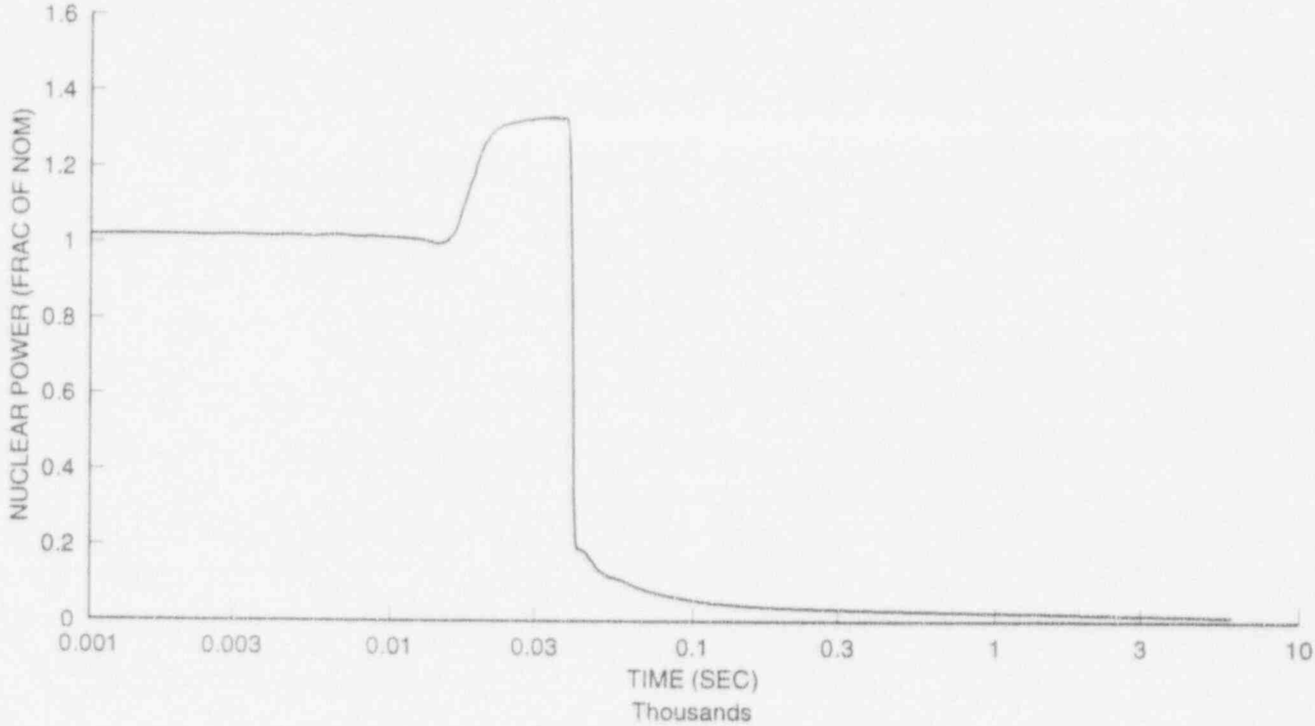
Results of the analysis show that for the postulated feedline rupture, the assumed emergency feedwater system capacity is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to prevent uncovering the reactor core. Radioactivity doses from the postulated feedline rupture are less than those previously presented for the postulated steam line break.

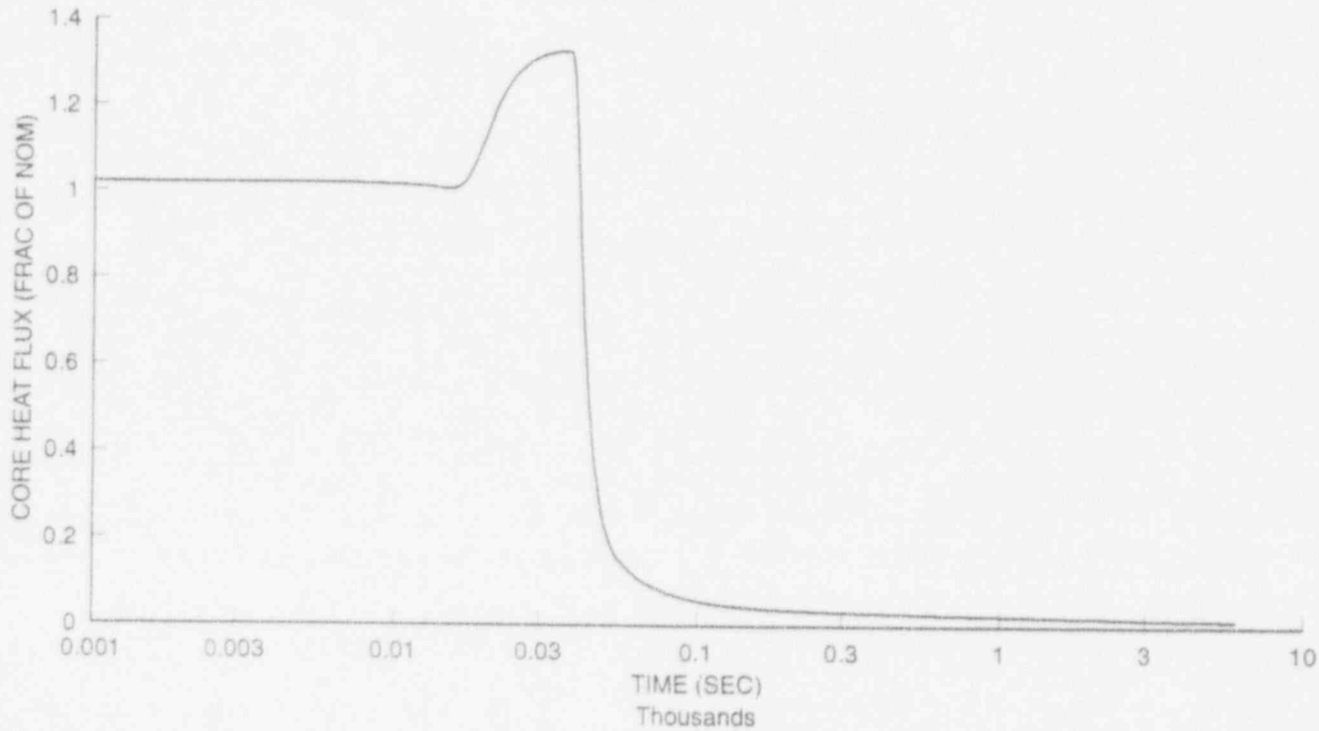
TABLE 15.4-19

TIME SEQUENCE OF EVENTS FOR

MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time (sec)
Rupture of Main Feedwater Pipe (offsite power available)	Feedline ruptures occurs	10
	Low-low SG water level setpoint reached	35.1
	Rods begin to drop	37.1
	Low steamline pressure setpoint reached	42.6
	Steamline and feedline isolation occurs	52.6
	Emergency feedwater started	95.1
	Feedwater lines are purged and emergency feedwater is delivered to two intact SGs	149
	Steam generator safety valves lift in intact loops	109
	Pressurizer water relief begins	668
	Total RCS heat generation (decay heat + pump heat) decreases to emergency feedwater heat removal capability	~2600
Rupture of Main Feedwater Pipe (offsite power unavailable)	Feedline ruptures occurs	10
	Low-low SG water level setpoint reached	35.1
	Rods begin to drop	37.1
	Reactor coolant pump coastdown	39.1
	Low steamline pressure setpoint reached	42.6
	Steamline and feedline isolation occurs	52.6
	Emergency feedwater started	95.1
	Feedwater lines are purged and emergency feedwater is delivered to two intact SGs	149
	Steam generator safety valves lift in intact loops	142
	Total RCS heat generation (decay heat + pump heat) decreases to emergency feedwater heat removal capability	~1120
	Pressurizer water relief begins	1122

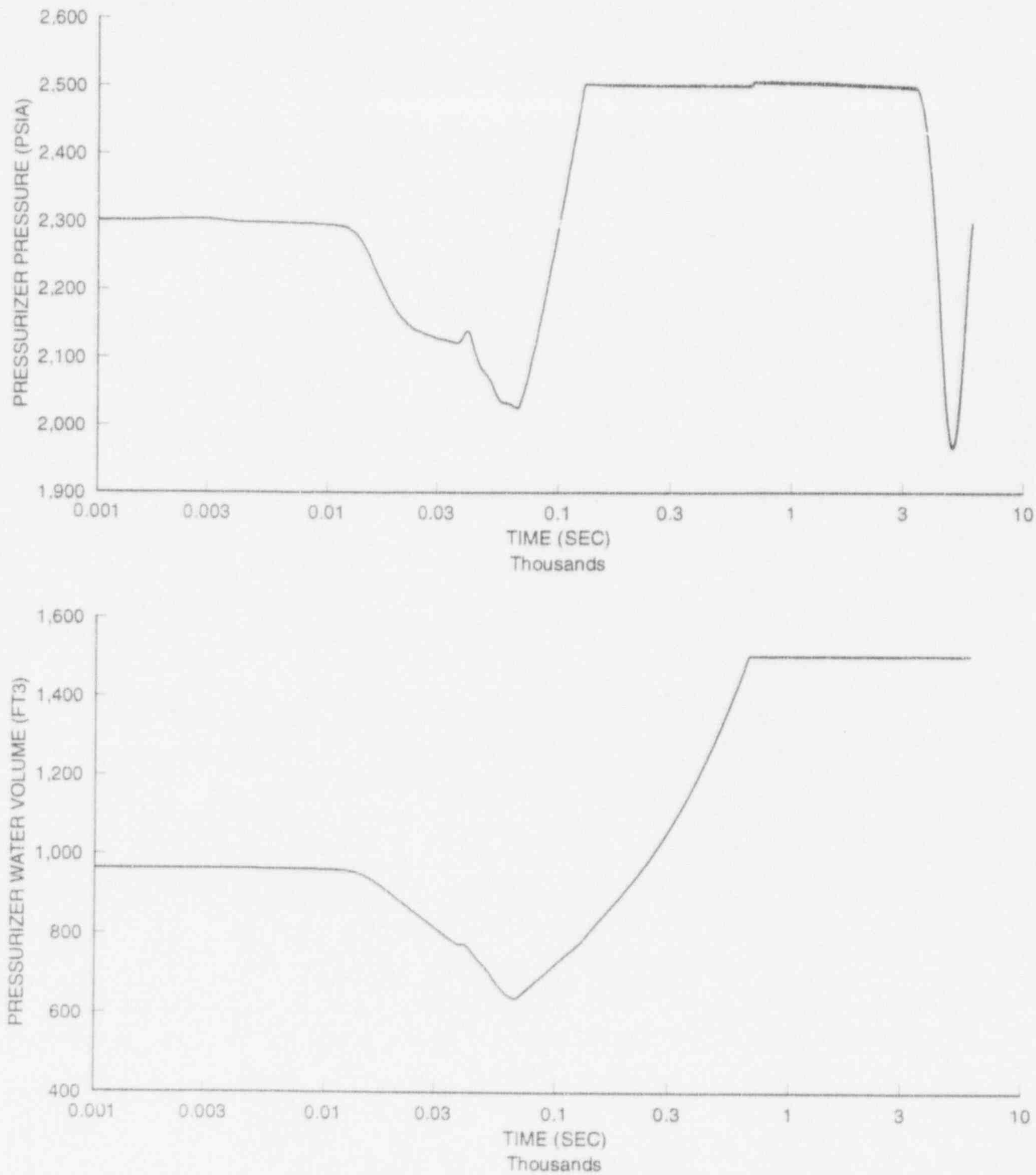




SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedline Rupture With Offsite Power Nuclear Power and Core Heat Flux vs. Time

FIGURE 15.4-83



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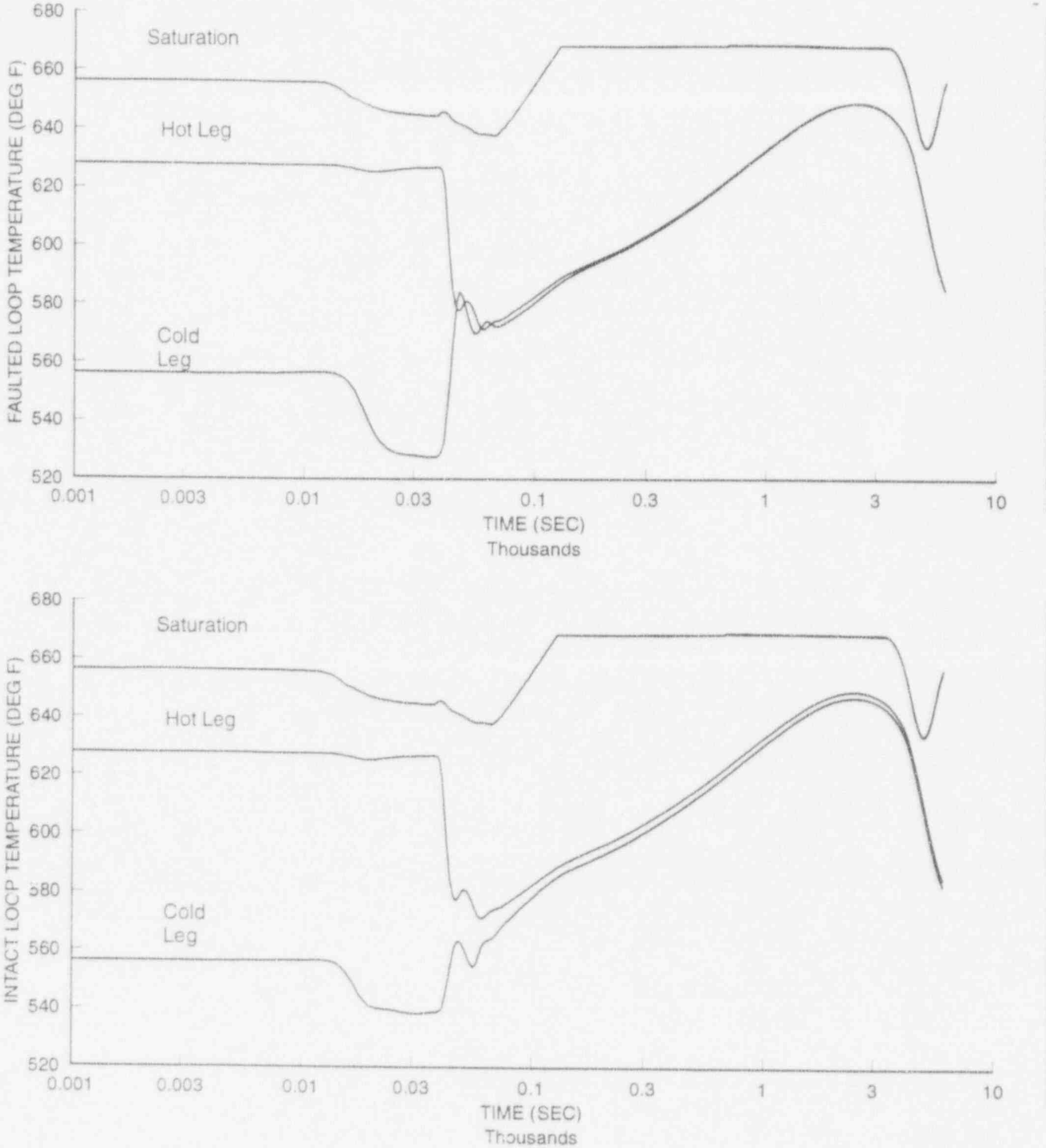
Main Feedline Rupture With Offsite Power Pressurizer Pressure and Water Volume vs. Time

FIGURE 15.4-84



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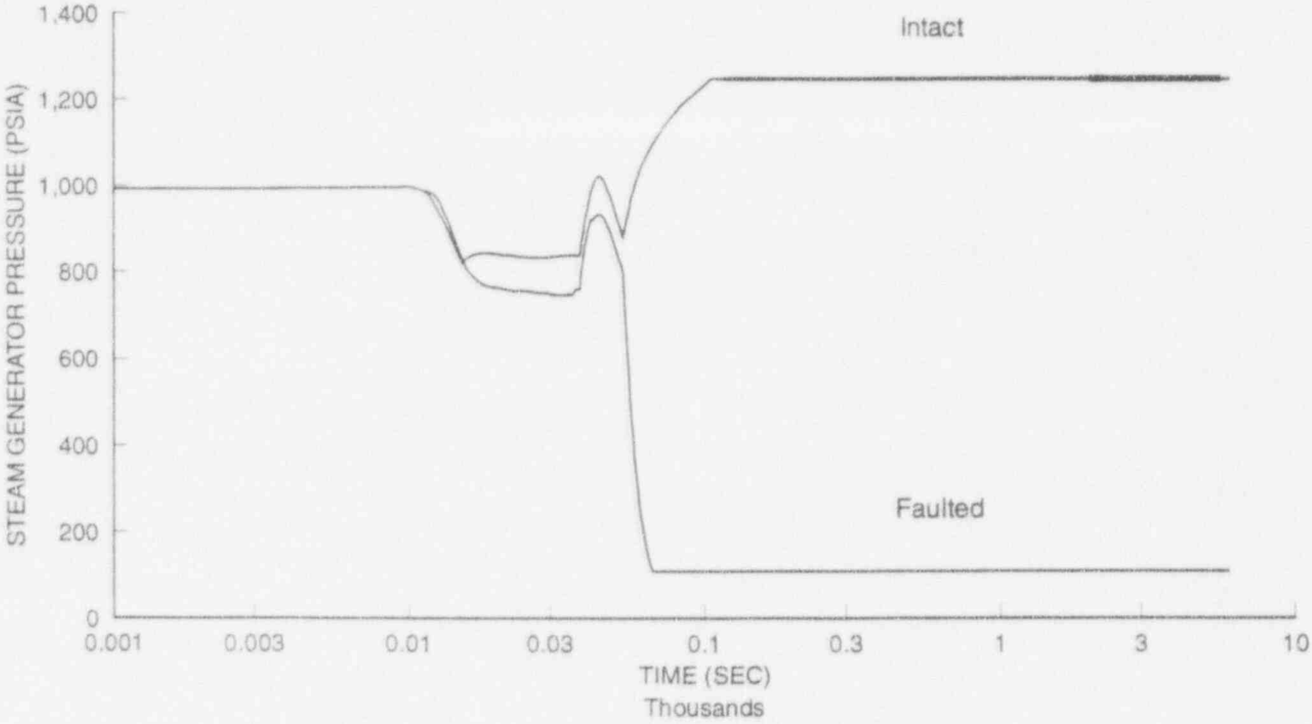
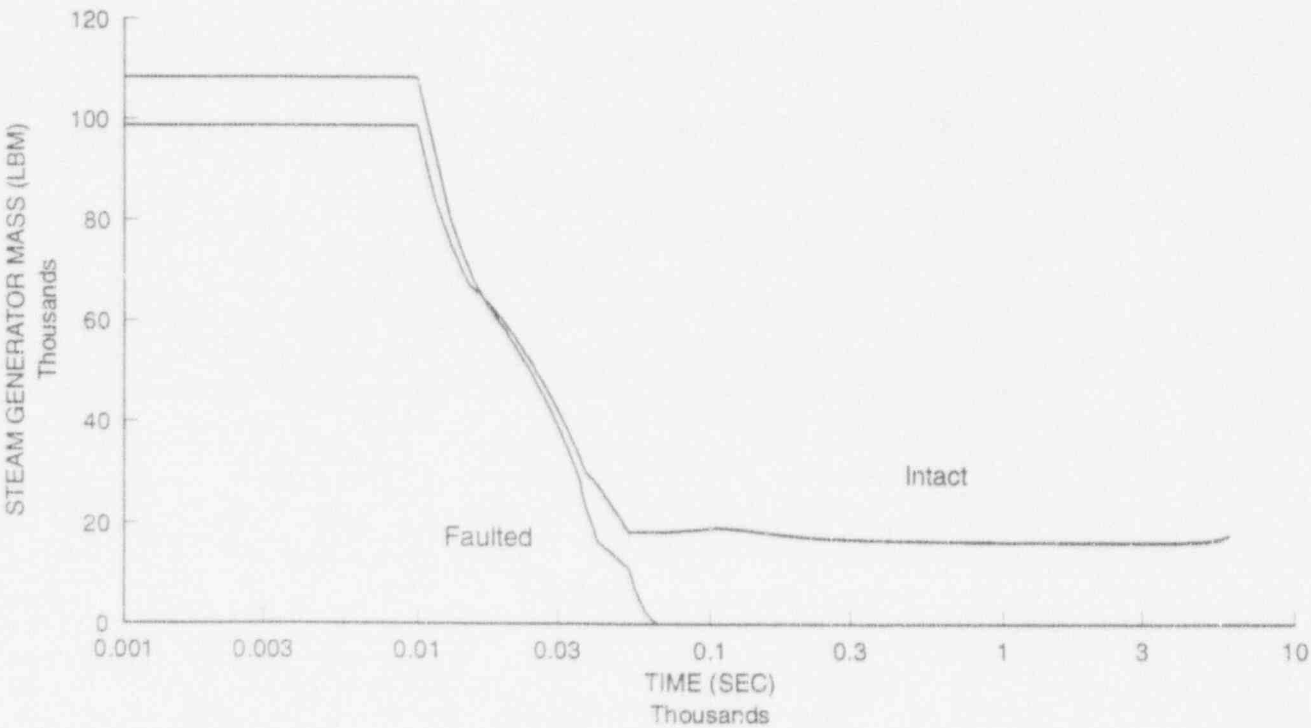
Main Feedline Rupture With Offsite Power Faulted and Intact Loop Coolant Temperature vs. Time

FIGURE 15.4-85





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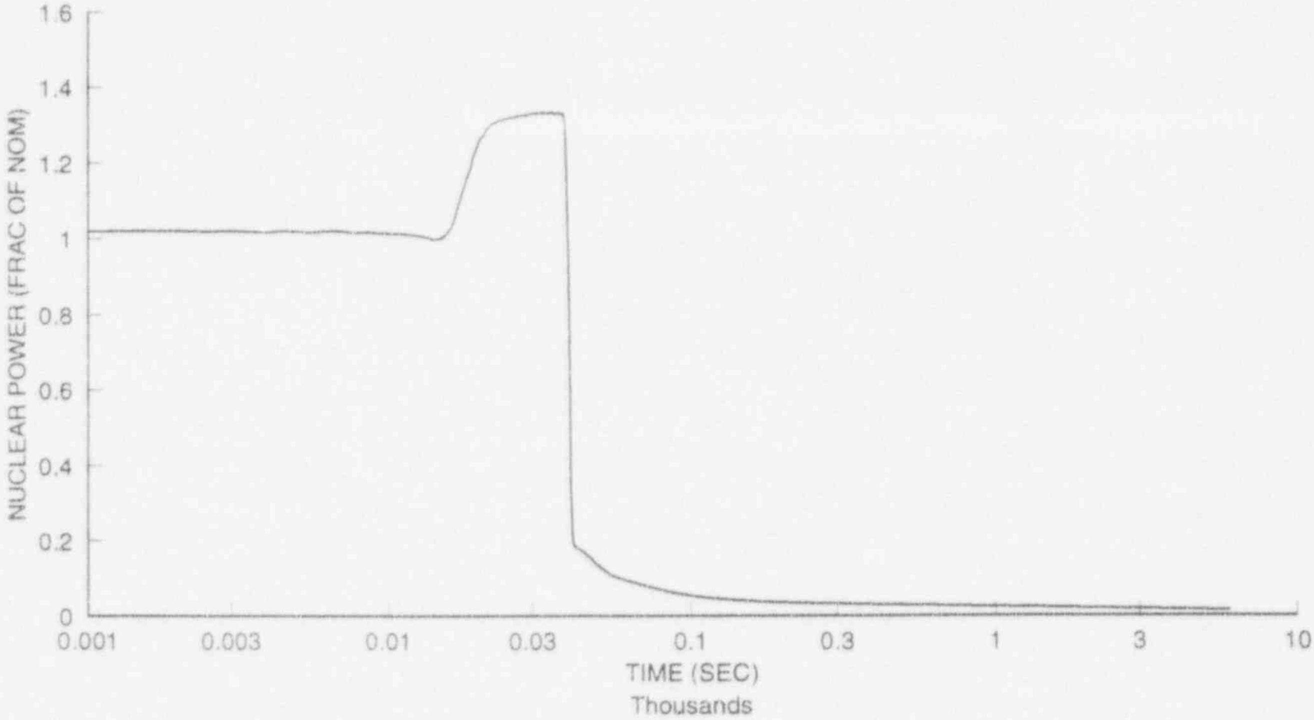
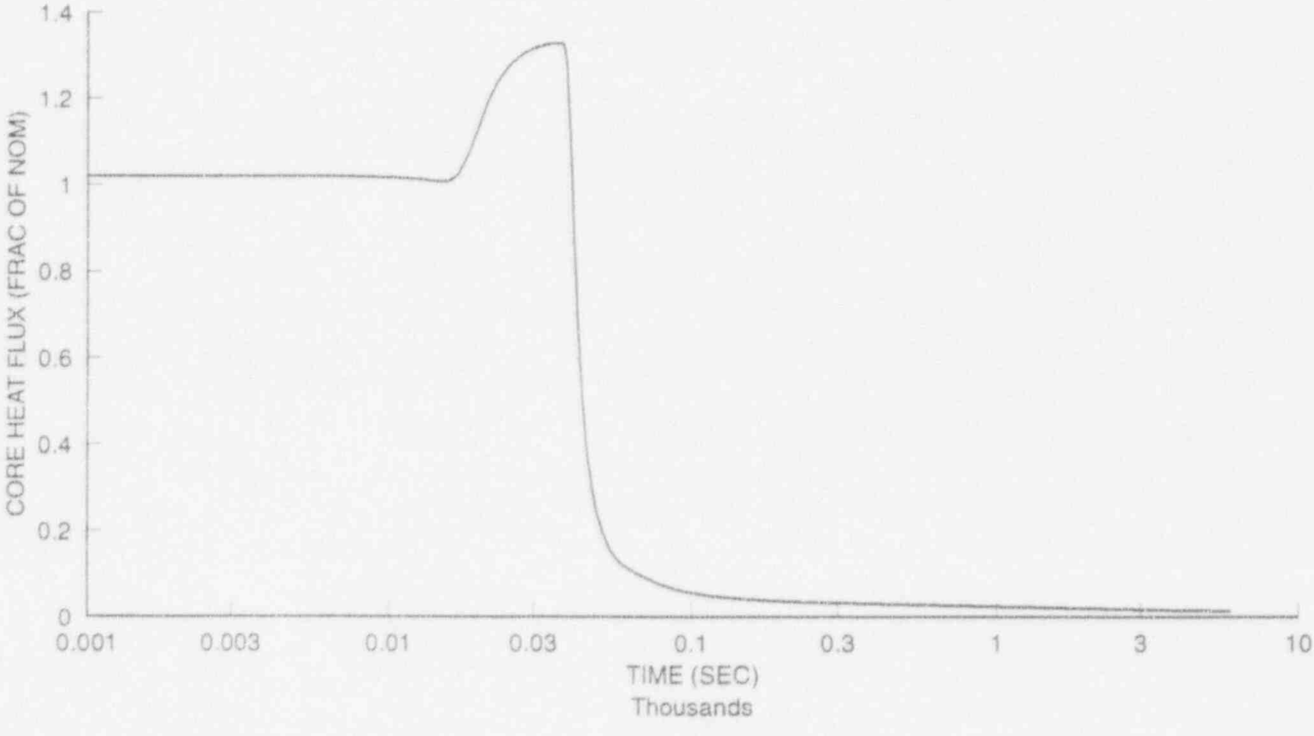
Main Feedline Rupture With Offsite Power Steam Generator Pressure and Water Mass vs. Time FIGURE 15.4-86

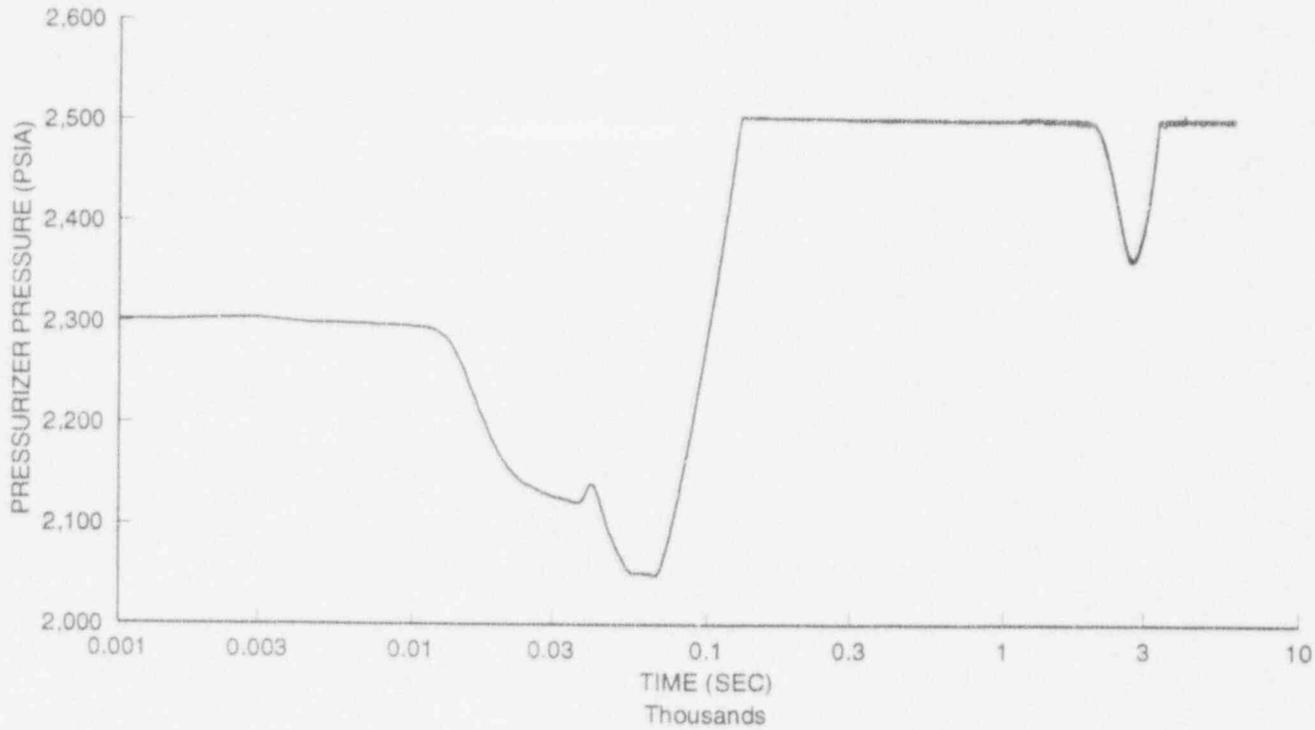


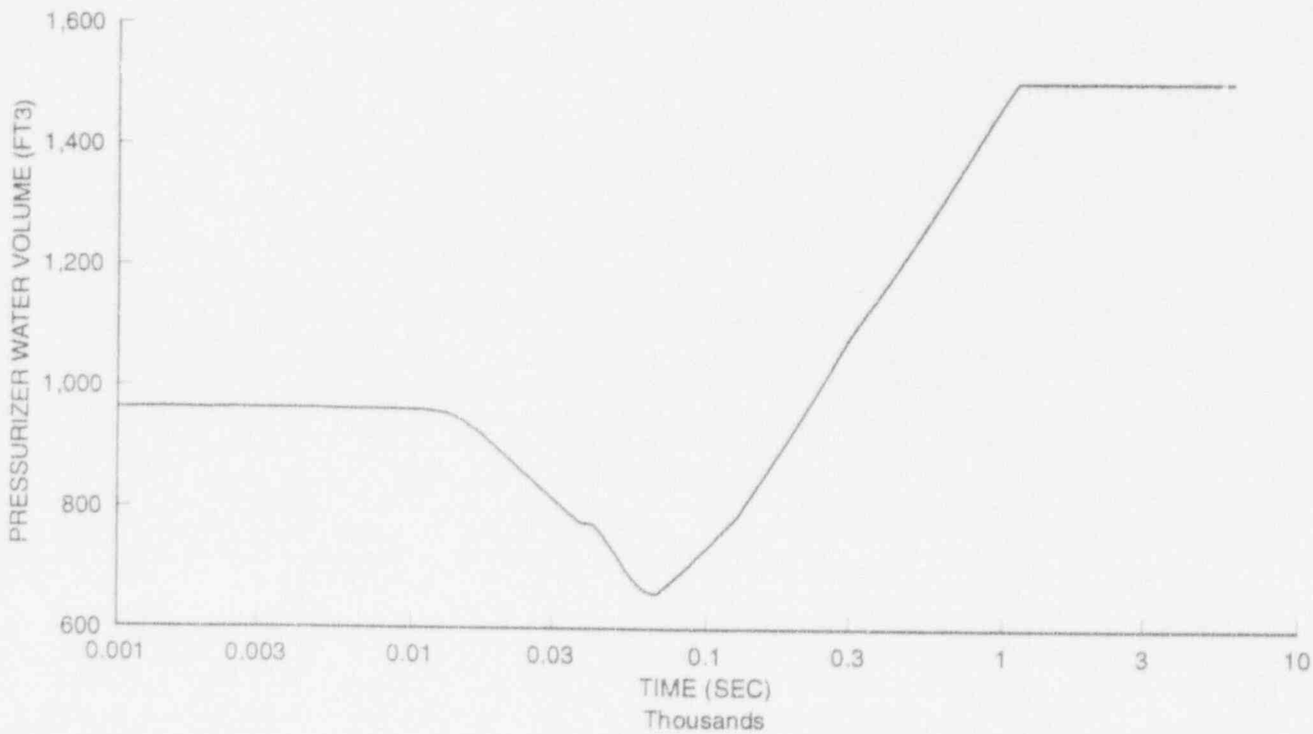


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedline Rupture Without Offsite Power Nuclear Power and Core Heat Flux vs. Time

FIGURE 15.4-87

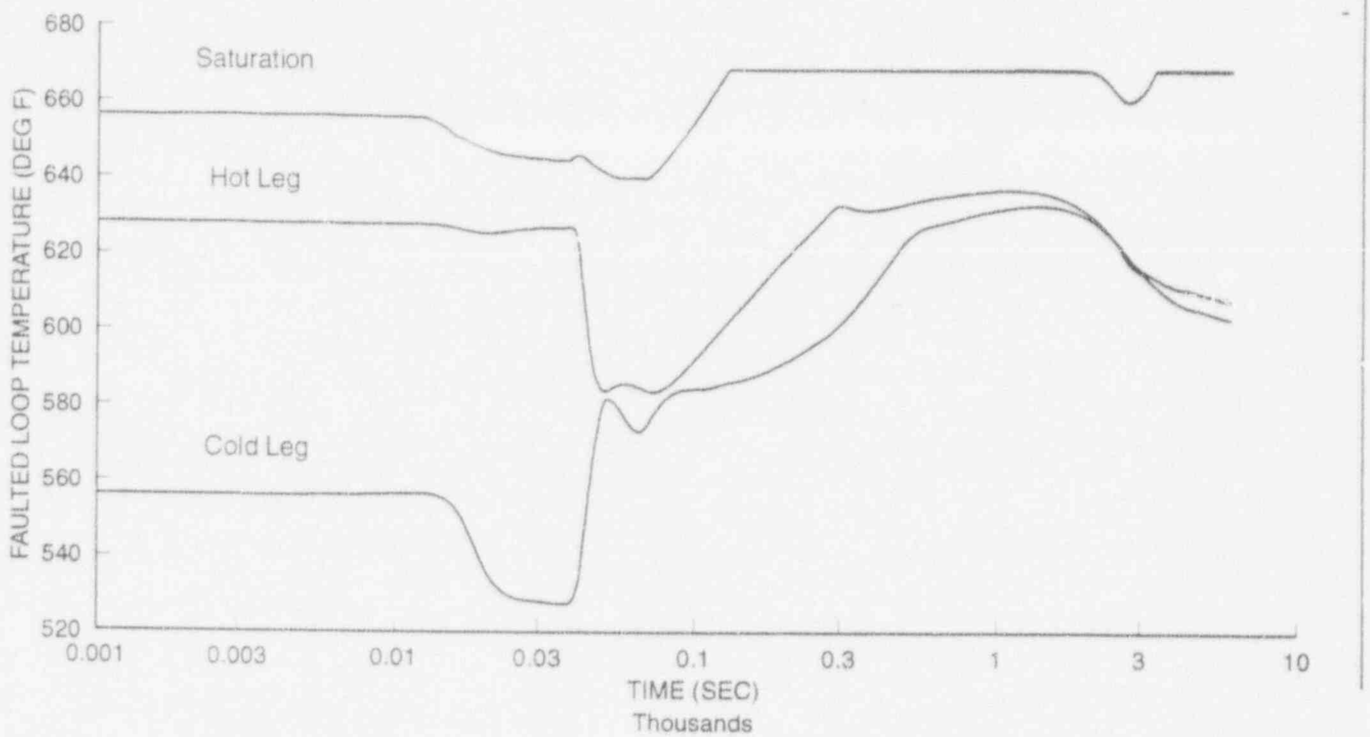


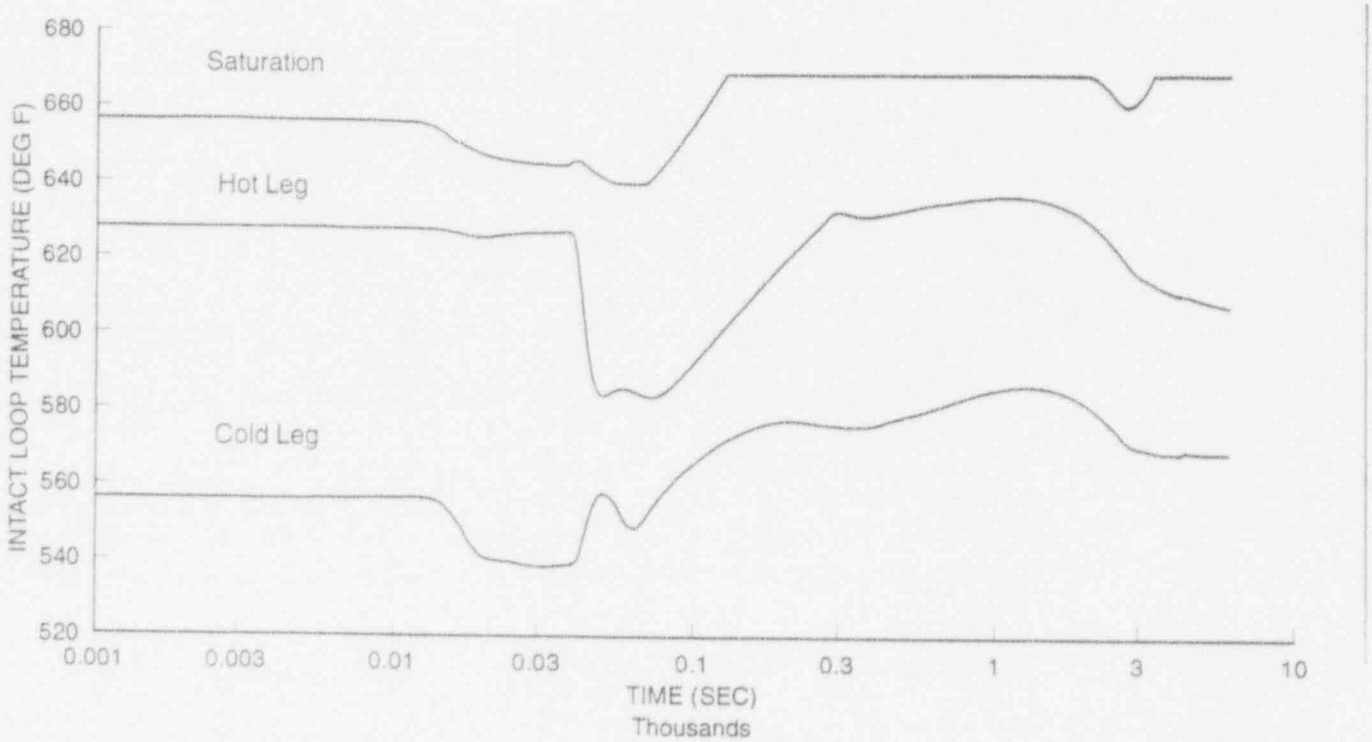


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedline Rupture Without Offsite Power Pressurizer Pressure and Water Volume vs. Time

FIGURE 15.4-88

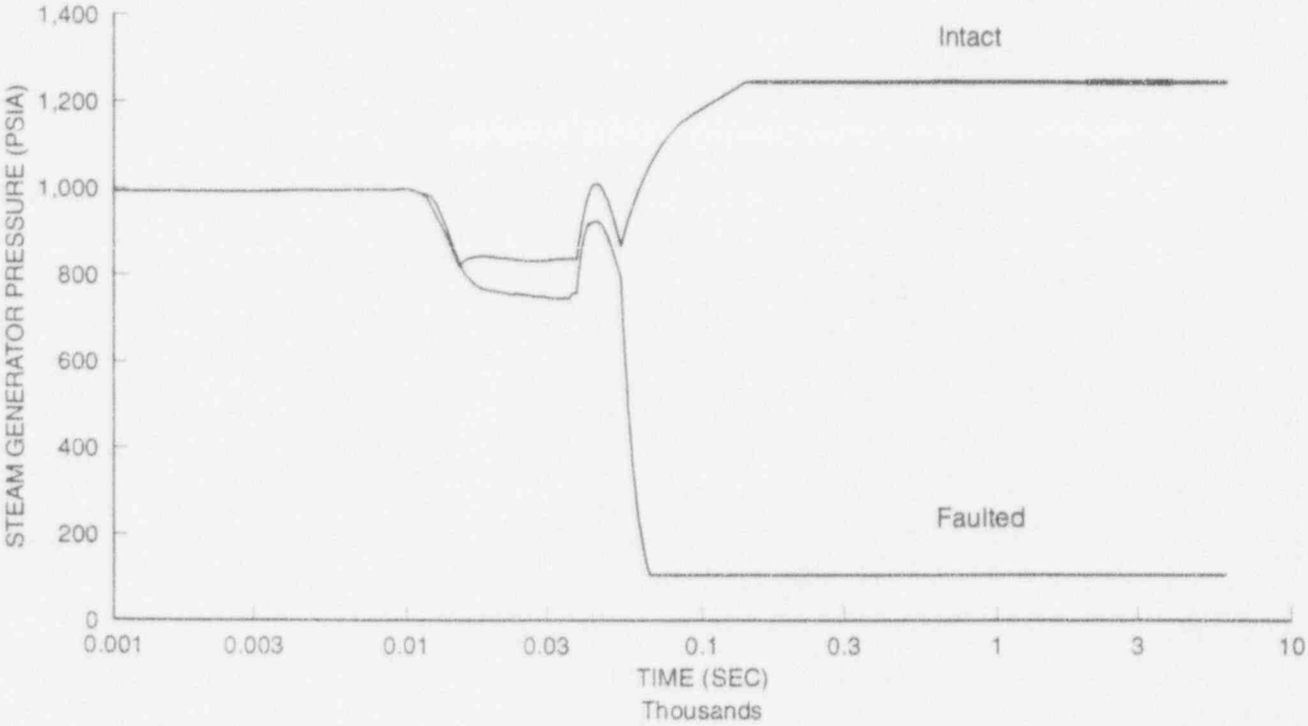


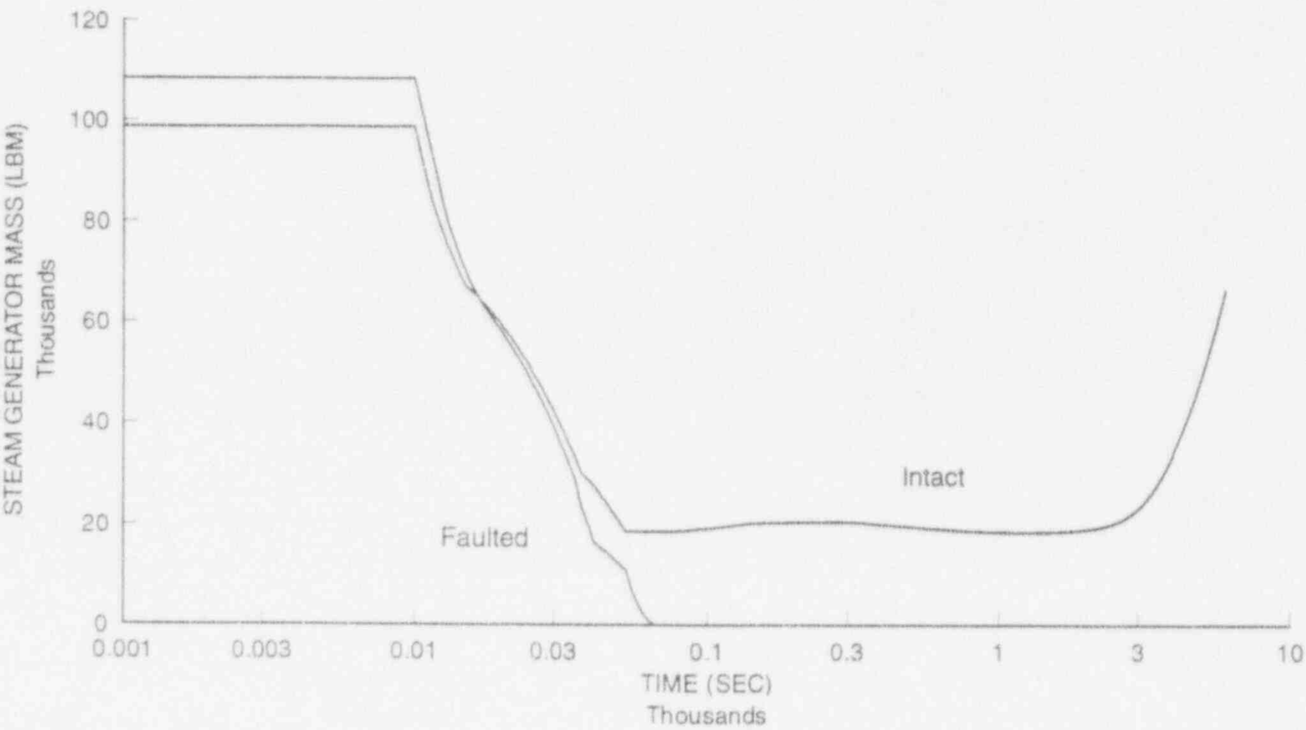


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedline Rupture Without Offsite Power Faulted and Intact Loop Coolant Temperature vs. Time

FIGURE 15.4-89





SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Main Feedline Rupture Without
Offsite Power Steam Generator
Pressure and Water Mass vs. Time

FIGURE 15.4-90