Table A3.2-1 (Sheet 1 of 2)

AP1000 PRA SUCCESS SEQUENCES WITH AUTOMATIC ADS AND IRWST INJECTION

Success Sequence Designator	CMT Success Criterion		ADS Success Criterion		IRWST Success Criterion		Other
	Name	Specification	Name	Specification	Name	Specification	
MLO-OK2	XCM2NL CM2NL/RCN	1 CMT	ADM	3 stage 4	IW2AB	1 line, 1 path	Containment isolated
MLO-OK3	XCM2NL CM2NL/RCN	1 CMT	ADM	3 stage 4	IW2AB	1 line, 1 path	Containment not isolated
CMT-OK2	XCM1A CM1A/RCN	1 CMT	ADM	3 stage 4	IW2AB	1 line, 1 path	Containment isolated
CMT-OK3	XCM1A CM1A/RCN	1 CMT	ADM	3 stage 4	IW2AB	1 line, 1 path	Containment not isolated
SIL-OK1	XCM1A CM1A/RCN	1 CMT	ADM	3 stage 4	IW1A	1 line, 1 path	Multiple paths with containment isolation successful or failed
SLO-OK2	XCM2SL CM2SL/RCL	1 CMT	ADS	3 stage 4	IW2AB	1 line, 1 path	PRHR successful Containment isolated
SLO-OK3	XCM2SL CM2SL/RCL	1 CMT	ADS	3 stage 4	IW2AB	1 line, 1 path	PRHR successful Containment not isolated
SLO-OK6	XCM2SL CM2SL/RCL	1 CMT	ADA	1 stage 2,3 and 3 stage 4	IW2AB	1 line, 1 path	PRHR failed Containment isolated
SLO-OK7	XCM2SL CM2SL/RCL	1 CMT	ADA	1 stage 2 or 3 and 3 stage 4	IW2AB	1 line, 1 path	PRHR failed Containment not isolated

Table A3.2-1 (Sheet 2 of 2)

AP1000 PRA SUCCESS SEQUENCES WITH AUTOMATIC ADS AND IRWST INJECTION

Success Sequence Designator	CMT Success Criterion		ADS Success Criterion		IRWST Success Criterion		Other
	Name	Specification	Name	Specification	Name	Specification	
SGR-OK4	XCM2SL CM2SL/RCL	1 CMT	ADS	3 stage 4	IW2AB	1 line, 1 path	
TRA-OK5	XCM2AB	1 CMT	ADA	1 stage 2 or 3 and 3 stage 4	IW2AB	1 line, 1 path	Failure of heat sinks, including MFW, SFW, PRHR
LSP-OK5	CM2P	1 CMT	ADAL	1 stage 2 or 3 and 3 stage 4	IW2ABP	1 line, 1 path	
SBO-OK2	CM2P	1 CMT	ADAB	1 stage 2 or 3 and 3 stage 4	IW2ABB	1 line, 1 path	
SLB-OK6	XCM2SL CM2SL/RCL	1 CMT	ADA	1 stage 2 or 3 and 3 stage 4	IW2AB	1 line, 1 path	PRHR failed

Table A3.2-2

AP1000 PRA SUCCESS SEQUENCES WITH AUTOMATIC ADS AND RNS INJECTION

Success Sequence Designator	CMT Success Criterion		ADS Success Criterion		RNS Success Criterion		Other
	Name	Specification	Name	Specification	Name	Specification	
MLO-OK4	XCM2NL CM2NL/RCN	1 CMT	ADU	2 stage 2,3 or 1 stage 4	RNR	1 pump	
CMT-OK4	XCM1A CM1A/RCN	1 CMT	ADU	2 stage 2,3 or 1 stage 4	RNR	1 pump	
SLO-OK4	XCM2SL CM2SL/RCL	1 CMT	ADV	2 stage 2,3 or 1 stage 4	RNR	1 pump	PRHR successful
SLO-OK8	XCM2SL CM2SL/RCL	1 CMT	AD1A	2 stage 2,3 or 1 stage 4	RNR	1 pump	PRHR failed
SGR-OK3	XCM2SL CM2SL/RCL	1 CMT	ADV AD1A	2 stage 2,3 or 1 stage 4	RNR	1 pump	Multiple paths with PRHR successful or failed
TRA-OK4	XCM2AB	1 CMT	AD1A	2 stage 2,3 or 1 stage 4	RNR	1 pump	Failure of heat sinks, including MFW, SFW, PRHR
LSP-OK4	CM2P	1 CMT	ADRA	2 stage 2,3 or 1 stage 4	RNR	1 pump	
SLB-OK5	XCM2SL CM2SL/RCL	1 CMT	AD1A	2 stage 2,3 or 1 stage 4	RNR	1 pump	PRHR failed

Table A3.3-1

AP1000 PRA SUCCESS SEQUENCES WITH MANUAL ADS AND IRWST INJECTION

Success Sequence Designator	Accumulator Success Criterion		ADS Success Criterion		IRWST Success Criterion		Other
	Name	Specification	Name	Specification	Name	Specification	
MLO-OK6	AC2AB	1 Accum	ADQ	3 stage 4	IW2AB	1 line, 1 path	Containment isolated
MLO-OK7	AC2AB	1 Accum	ADQ	3 stage 4	IW2AB	1 line, 1 path	Containment not isolated
CMT-OK6	AC2AB	1 Accum	ADQ	3 stage 4	IW2AB	1 line, 1 path	Containment isolated
CMT-OK7	AC2AB	1 Accum	ADQ	3 stage 4	IW2AB	1 line, 1 path	Containment not isolated
SIL-OK2	AC1A	1 Accum	ADQ	3 stage 4	IW1AM	1 line, 1 path	Multiple paths with containment isolation successful or failed
SLO-OK10	AC2AB	1 Accum	ADT	3 stage 4	IW2AB	1 line, 1 path	PRHR successful Containment isolated
SLO-OK11	AC2AB	1 Accum	ADT	3 stage 4	IW2AB	1 line, 1 path	PRHR successful Containment not isolated
SLO-OK14	AC2AB	1 Accum	ADT	3 stage 4	IW2AB	1 line, 1 path	PRHR failed Containment isolated
SLO-OK15	AC2AB	1 Accum	ADT	3 stage 4	IW2AB	1 line, 1 path	PRHR failed Containment not isolated
SGR-OK6	AC2AB	1 Accum	ADT	3 stage 4	IW2AB	1 line, 1 path	Multiple paths with success or failure of PRHR and containment isolation
TRA-OK7	AC2AB	1 Accum	ADT	3 stage 4	IW2ABM	1 line, 1 path	Failure of heat sinks, including MFW, SFW, PRHR
LSP-OK7	AC2AB	1 Accum	ADL	3 stage 4	IW2ABPM	1 line, 1 path	
SBO-OK3	AC2AB	1 Accum	ADB	3 stage 4	IW2ABBM	1 line, 1 path	
SLB-OK4	AC2AB	1 Accum	ADT	3 stage 4	IW2ABM	1 line, 1 path	PRHR successful
SLB-OK8	AC2AB	1 Accum	ADT	3 stage 4	IW2ABM	1 line, 1 path	PRHR failed

Table A3.3-2

AP1000 PRA SUCCESS SEQUENCES WITH MANUAL ADS AND RNS INJECTION

Success Sequence Designator	Accumulator Success Criterion		ADS Success Criterion		RNS Success Criterion		Other
	Name	Specification	Name	Specification	Name	Specification	
MLO-OK8	AC2AB	1 Accum	ADUM	2 stage 2, 3	RNR	1 pump	
CMT-OK8	AC2AB	1 Accum	ADUM	2 stage 2, 3	RNR	1 pump	
SLO-OK12	AC2AB	1 Accum	ADZ	2 stage 2, 3	RNR	1 pump	PRHR successful
SLO-OK16	AC2AB	1 Accum	AD1	2 stage 2, 3	RNR	1 pump	PRHR failed
SGR-OK5	AC2AB	1 Accum	ADT	2 stage 2, 3	RNR	1 pump	Multiple paths with success or failure of PRHR
TRA-OK6	AC2AB	1 Accum	AD1	2 stage 2, 3	RNR	1 pump	Failure of heat sinks, including MFW, SFW, PRHR
LSP-OK6	AC2AB	1 Accum	ADR	2 stage 2, 3	RNP	1 pump	
SLB-OK3	AC2AB	1 Accum	ADZ	2 stage 2, 3	RNR	1 pump	PRHR successful
SLB-OK7	AC2AB	1 Accum	AD1	2 stage 2, 3	RNR	1 pump	PRHR failed

Table A5.1-1

DEFINITION OF SUCCESS PATH CATEGORIES FOR THERMAL/HYDRAULIC UNCERTAINTY ANALYSIS

1	OK1	More ADS-4 than design basis
2	OK2	Design basis
3	OK3	More ADS-4/less ADS-1, -2, -3 than design basis
4	OK4	Less ADS-1, -2, -3 than design basis
5	OK5A	More ADS-4/CI fails
6	OK5B	More ADS-4/CI fails/less ADS-1, -2, -3
7	OK6	Design basis ADS/CI fails
8	OK7	Two accumulators/design basis for LLOCA
9	OK8	SI-LB with auto ADS from faulted CMT
10	OK9	Loss of CMTs for smaller breaks
11	UC1	No makeup of inventory if reactor coolant system pressure greater than 700 psig
12	UC2A	One accumulator depletes prior to operator intervention
13	UC2B	Two accumulators deplete prior to operator intervention
14	UC3	No rapid inventory makeup during blowdown
15	UC4	Reduced inventory makeup during LLOCA reflood
16	UC5	No makeup when ADS is actuated at higher pressure
17	UC6	Less ADS-4 than design basis accident (i.e., < 3 of 4 ADS-4)
18	UC7	No ADS-4 for LLOCA
19	UC8	No containment isolation/design basis accident ADS
20	UC9	No containment isolation/reduced ADS

Table A5.1-2 (Sheet 1 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

	End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
>>	UC5	silb06	8.96E-07	5.37E-08	**371.66**	**275.60**	Yes	Yes	1	0	4	2-4	C	FG	
>>	UC	sad06	4.58E-07	2.75E-08	**190.05**	**140.93**	Yes	Yes	2	1	4	2-4	E	FG	
>>	UC1	silb11	3.05E-07	1.83E-08	**126.76**	**94.00**	Yes	Yes	0	1	4	2-4	A	FG	
>>	UC2B	mlo31	2.89E-07	1.73E-08	**119.85**	**88.88**	Yes	Yes	0	2	4	2-4	AB	FG	
>>	UC2B	cmt31	1.34E-07	8.05E-09	**55.67**	**41.28**	Yes	Yes	0	2	4	2-4	AB	FG	
>>	UC	sad25	9.12E-08	5.47E-09	**37.82**	**28.05**	No	Yes	2	2	4	2-4	E	G	(2)
>>	UC3	mlo11	3.01E-08	1.81E-09	**12.48**	**9.26**	Yes	Yes	2	0	4	2-4	C	FG	
>>	UC8	llo15	8.51E-09	8.51E-09	**3.53**	**43.63**	No	Yes	2	2	4	2-4	D	G	
>>	UC3	cmt26	6.42E-09	3.85E-10	**2.67**	**1.98**	Yes	Yes	1	0	4	2-4	C	FG	
>>	UC2A	mlo36	2.44E-09	1.47E-10	**1.01**	0.75	Yes	Yes	0	1	4	2-4	A	FG	
	UC5	silb08	2.09E-09	1.25E-10	0.87	0.64	Yes	Yes	1	0	3	2-4	C	F	
	UC5	silb07	1.64E-09	9.83E-11	0.68	0.50	Yes	Yes	1	0	4	0-1	C	FG	
>>	UC5	silb23	1.52E-09	1.52E-09	0.63	**7.77**	No	Yes	1	0	4	2-4	C	G	
	UC2A	cmt36	1.14E-09	6.85E-11	0.47	0.35	Yes	Yes	0	1	4	2-4	A	FG	
	UC	sad08	1.07E-09	6.42E-11	0.44	0.33	Yes	Yes	2	1	3	2-4		F	
	UC	sad07	8.40E-10	5.04E-11	0.35	0.26	Yes	Yes	2	1	4	0-1	E	FG	
	UC	sad30	7.77E-10	4.66E-11	0.32	0.24	No	Yes	2	1	4	2-4	E	G	(2)
	UC1	silb13	7.21E-10	4.32E-11	0.30	0.22	Yes	Yes	0	1	3	2-4		F	
	UC2B	mlo33	6.92E-10	4.15E-11	0.29	0.21	Yes	Yes	0	2	3	2-4		F	
	UC	sad17	6.76E-10	4.05E-11	0.28	0.21	Yes	Yes	1	1	4	2-4	E	FG	

Table A5.1-2 (Sheet 2 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

	End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
	UC2B	mlo32	6.44E-10	3.86E-11	0.27	0.20	Yes	Yes	0	2	4	0-1	AB	FG	
	UC1	silb12	6.15E-10	3.69E-11	0.26	0.19	Yes	Yes	0	1	4	0-1	A	FG	
>>	UC1	silb28	5.16E-10	5.16E-10	0.21	2.65	No	Yes	0	1	4	2-4	A	G	
>>	UC2B	mlo73	4.88E-10	4.88E-10	0.20	2.50	No	Yes	0	2	4	2-4	A	G	
	UC2B	cmt33	3.17E-10	1.90E-11	0.13	0.10	Yes	Yes	0	2	3	2-4		F	
	UC2B	cmt32	2.70E-10	1.62E-11	0.11	0.08	Yes	Yes	0	2	4	0-1	AB	FG	
	UC2B	cmt73	2.27E-10	1.36E-11	0.09	0.07	No	Yes	0	2	4	2-4	A	G	
	UC	sad27	2.13E-10	1.28E-11	0.09	0.07	No	Yes	2	2	3	2-4			
	UC	sad26	1.67E-10	1.00E-11	0.07	0.05	No	Yes	2	2	4	0-1	E	G	(2)
	UC	sad36	1.34E-10	8.04E-12	0.06	0.04	No	Yes	1	2	4	2-4	E	G	(2)
	UC3	mlo13	7.00E-11	4.20E-12	0.03	0.02	Yes	Yes	2	0	3	2-4	C	F	
	UC3	mlo12	5.49E-11	3.30E-12	0.02	0.02	Yes	Yes	2	0	4	0-1	C	FG	
	UC3	mlo53	5.08E-11	5.08E-11	0.02	0.26	No	Yes	2	0	4	2-4	C	G	
	UC3	mlo26	4.42E-11	2.65E-12	0.02	0.01	Yes	Yes	1	0	4	2-4	C	FG	
	UC8	llo17	1.99E-11	1.99E-11	0.01	0.10	No	Yes	2	2	3	2-4			
	UC8	llo16	1.56E-11	1.56E-11	0.01	0.08	No	Yes	2	2	4	0-1	D	G	
	UC3	cmt28	1.50E-11	8.97E-13	0.01	0.00	Yes	Yes	1	0	3	2-4	C	F	
	UC8	llo21	1.25E-11	1.25E-11	0.01	0.06	No	Yes	1	2	4	2-4		G	
	UC3	cmt27	1.17E-11	7.04E-13	0.00	0.00	Yes	Yes	1	0	4	0-1	C	FG	
	UC3	cmt68	1.09E-11	6.51E-13	0.00	0.00	No	Yes	1	0	4	2-4	C	G	
	UC2A	mlo38	5.86E-12	3.51E-13	0.00	0.00	Yes	Yes	0	1	3	2-4		F	

Table A5.1-2 (Sheet 3 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded Short/Long-Term		
UC2A	mlo37	5.45E-12	3.27E-13	0.00	0.00	Yes	Yes	0	1	4	0-1	A	FG	
UC2A	mlo78	4.13E-12	4.13E-12	0.00	0.02	No	Yes	0	1	4	2-4	A	G	
UC5	silb09	3.85E-12	2.31E-13	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC5	silb25	3.53E-12	3.53E-12	0.00	0.02	No	Yes	1	0	3	2-4	C		
UC5	silb24	2.77E-12	2.77E-12	0.00	0.01	No	Yes	1	0	4	0-1	C	G	
UC2A	cmt38	2.69E-12	1.62E-13	0.00	0.00	Yes	Yes	0	1	3	2-4		F	
UC2A	cmt37	2.30E-12	1.38E-13	0.00	0.00	Yes	Yes	0	1	4	0-1	A	FG	
UC	sad09	1.97E-12	1.18E-13	0.00	0.00	Yes	Yes	2	1	3	0-1		F	
UC2A	cmt78	1.93E-12	1.16E-13	0.00	0.00	No	Yes	0	1	4	2-4	A	G	
UC	sad32	1.81E-12	1.09E-13	0.00	0.00	No	Yes	2	1	3	2-4			
UC2B	mlo34	1.78E-12	1.07E-13	0.00	0.00	Yes	Yes	0	2	3	0-1		F	
UC1	silb14	1.57E-12	9.45E-14	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC	sad19	1.57E-12	9.44E-14	0.00	0.00	Yes	Yes	1	1	3	2-4		F	
UC	sad31	1.42E-12	8.51E-14	0.00	0.00	No	Yes	2	1	4	0-1	E	G	(2)
UC	sad18	1.23E-12	7.40E-14	0.00	0.00	Yes	Yes	1	1	4	0-1	E	FG	
UC1	silb30	1.22E-12	1.22E-12	0.00	0.01	No	Yes	0	1	3	2-4			
UC2B	mlo75	1.17E-12	1.17E-12	0.00	0.01	No	Yes	0	2	3	2-4			
UC	sad41	1.14E-12	6.85E-14	0.00	0.00	No	Yes	1	1	4	2-4	E	G	(2)
UC2B	mlo74	1.09E-12	1.09E-12	0.00	0.01	No	Yes	0	2	4	0-1	A	G	
UC1	silb29	1.04E-12	1.04E-12	0.00	0.01	No	Yes	0	1	4	0-1	A	G	
UC2B	cmt34	6.91E-13	4.15E-14	0.00	0.00	Yes	Yes	0	2	3	0-1		F	

Table A5.1-2 (Sheet 4 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
UC2B	cmt75	5.35E-13	3.21E-14	0.00	0.00	No	Yes	0	2	3	2-4			
UC2B	cmt74	4.56E-13	2.74E-14	0.00	0.00	No	Yes	0	2	4	0-1	A	G	
UC	sad28	3.91E-13	2.35E-14	0.00	0.00	No	Yes	2	2	3	0-1			
UC	sad38	3.12E-13	1.87E-14	0.00	0.00	No	Yes	1	2	3	2-4			
UC	sad37	2.45E-13	1.47E-14	0.00	0.00	No	Yes	1	2	4	0-1	E	G	(2)
UC3	mlo14	1.29E-13	7.71E-15	0.00	0.00	Yes	Yes	2	0	3	0-1	C	F	
UC3	mlo55	1.18E-13	1.18E-13	0.00	0.00	No	Yes	2	0	3	2-4	C		
UC3	mlo28	1.03E-13	6.17E-15	0.00	0.00	Yes	Yes	1	0	3	2-4	C	F	
UC3	mlo54	9.24E-14	9.24E-14	0.00	0.00	No	Yes	2	0	4	0-1	C	G	
UC3	mlo27	8.04E-14	4.83E-15	0.00	0.00	Yes	Yes	1	0	4	0-1	C	FG	
UC3	mlo68	7.46E-14	7.46E-14	0.00	0.00	No	Yes	1	0	4	2-4	C	G	
UC8	llo18	3.66E-14	3.66E-14	0.00	0.00	No	Yes	2	2	3	0-1			
UC8	llo23	2.92E-14	2.92E-14	0.00	0.00	No	Yes	1	2	3	2-4			
UC3	cmt29	2.75E-14	1.65E-15	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC3	cmt70	2.52E-14	1.51E-15	0.00	0.00	No	Yes	1	0	3	2-4	C		
UC8	llo22	2.29E-14	2.29E-14	0.00	0.00	No	Yes	1	2	4	0-1		G	
UC3	cmt69	1.97E-14	1.18E-15	0.00	0.00	No	Yes	1	0	4	0-1	C	G	
UC2A	mlo39	1.36E-14	8.13E-16	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC2A	mlo80	9.01E-15	9.01E-15	0.00	0.00	No	Yes	0	1	3	2-4			
UC2A	mlo79	8.10E-15	8.10E-15	0.00	0.00	No	Yes	0	1	4	0-1	A	G	
UC5	silb26	6.28E-15	6.28E-15	0.00	0.00	No	Yes	1	0	3	0-1	C		

Table A5.1-2 (Sheet 5 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long-Term		
UC2A	cmt39	5.47E-15	3.28E-16	0.00	0.00	Yes	Yes	0	1	3	0-1		F	
UC2A	cmt80	4.33E-15	2.60E-16	0.00	0.00	No	Yes	0	1	3	2-4			
UC2A	cmt79	3.58E-15	2.15E-16	0.00	0.00	No	Yes	0	1	4	0-1	A	G	
UC	sad33	3.20E-15	1.92E-16	0.00	0.00	No	Yes	2	1	3	0-1			
UC	sad20	2.78E-15	1.67E-16	0.00	0.00	Yes	Yes	1	1	3	0-1		F	
UC	sad43	2.62E-15	1.57E-16	0.00	0.00	No	Yes	1	1	3	2-4			
UC2B	mlo76	2.30E-15	2.30E-15	0.00	0.00	No	Yes	0	2	3	0-1			
UC1	silb31	2.26E-15	2.26E-15	0.00	0.00	No	Yes	0	1	3	0-1			
UC	sad42	2.02E-15	1.21E-16	0.00	0.00	No	Yes	1	1	4	0-1	E	G	(2)
UC2B	cmt76	9.92E-16	5.95E-17	0.00	0.00	No	Yes	0	2	3	0-1			
UC	sad39	5.15E-16	3.09E-17	0.00	0.00	No	Yes	1	2	3	0-1			
UC3	mlo56	1.51E-16	1.51E-16	0.00	0.00	No	Yes	2	0	3	0-1	C		
UC3	mlo29	1.35E-16	8.09E-18	0.00	0.00	Yes	Yes	1	0	3	0-1	C	F	
UC3	mlo70	1.25E-16	1.25E-16	0.00	0.00	No	Yes	1	0	3	2-4	C		
UC3	mlo69	9.54E-17	9.54E-17	0.00	0.00	No	Yes	1	0	4	0-1	C	G	
UC8	llo24	5.09E-17	5.09E-17	0.00	0.00	No	Yes	1	2	3	0-1			
UC3	cmt71	3.23E-17	1.94E-18	0.00	0.00	No	Yes	1	0	3	0-1	C		
UC	sad44	0.00E+00	0.00E+00	0.00	0.00	No	Yes	1	1	3	0-1			
UC2A	cmt81	0.00E+00	0.00E+00	0.00	0.00	No	Yes	0	1	3	0-1			

Table A5.1-2 (Sheet 6 of 6)

LOW MARGIN (UC) SEQUENCES SORTED BY CORE DAMAGE FREQUENCY

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2,3	Bounded By Short/Long Term
Totals/% of total		2.24E-06	1.45E-07	100.00	100.00		<< all 102 cases listed above					
		2.23E-06	1.44E-07	99.86	99.85		<< for 58 cases bounded by analyzed T/H uncertainty cases					
		2.22E-06	1.44E-07	99.44	99.40		<< for 13 cases bounded by dominant cases					

Notes:

1. Sequences with >= 1-percent contribution to either CDF or LRF are highlighted by outline of percentage CDF/LRF. The remaining sequences are to be treated as "residue."
2. Spurious ADS 4 case was analyzed with containment isolation; because of large PCT margin, case without CI is expected to be OK.

Table A5.1-3

RISK-IMPORTANT UC SEQUENCES

End State	Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2, 3
UC5	silb06	8.96E-07	5.37E-08	371.66	275.60	Yes	Yes	1	0	4	2-4
UC	sad06	4.58E-07	2.75E-08	190.05	140.93	Yes	Yes	2	1	4	2-4
UC1	silb11	3.05E-07	1.83E-08	126.76	94.00	Yes	Yes	0	1	4	2-4
UC2B	mlo31	2.89E-07	1.73E-08	119.85	88.88	Yes	Yes	0	2	4	2-4
UC2B	cmt31	1.34E-07	8.05E-09	55.67	41.28	Yes	Yes	0	2	4	2-4
UC	sad25	9.12E-08	5.47E-09	37.82	28.05	No	Yes	2	2	4	2-4
UC3	mlo11	3.01E-08	1.81E-09	12.48	9.26	Yes	Yes	2	0	4	2-4
UC8	llo15	8.51E-09	8.51E-09	3.53	43.63	No	Yes	2	2	4	2-4
UC3	cmt26	6.42E-09	3.85E-10	2.67	1.98	Yes	Yes	1	0	4	2-4
UC2A	mlo36	2.44E-09	1.47E-10	1.01	0.75	Yes	Yes	0	1	4	2-4
UC5	silb23	1.52E-09	1.52E-09	0.63	7.77	No	Yes	1	0	4	2-4
UC1	silb28	5.16E-10	5.16E-10	0.21	2.65	No	Yes	0	1	4	2-4
UC2B	mlo73	4.88E-10	4.88E-10	0.20	2.50	No	Yes	0	2	4	2-4
Totals =		**2.22E-06**	**1.44E-07**								
		1.26E-08	8.62E-10	5.22	4.42	=	Residue from UC Sequences not selected				

Table A5.1-4

SEQUENCES WITH CORE MAKEUP TANK AND PASSIVE RESIDUAL HEAT REMOVAL FAILURE

Sequence Name	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	CI	IRWST & RECIRC	CMT	PRHR
silb11p	6.54E-11	3.92E-12	0.03	0.02	Yes	Yes	0	No
silb28p	1.12E-13	1.12E-13	0.00	0.00	No	Yes	0	No
mlo31p	6.29E-11	3.77E-12	0.03	0.02	Yes	Yes	0	No
mlo36p	5.31E-13	3.19E-14	0.00	0.00	Yes	Yes	0	No
mlo73p	1.07E-13	1.07E-13	0.00	0.00	No	Yes	0	No
cmt31p	2 87E-11	1.72E-12	0.01	0.01	Yes	Yes	0	No
Sum =	**1.58E-10**	**9.67E-12**	**0.07**	**0.05**				

Table A5.1-5

AP1000 THERMAL/HYDRAULIC UNCERTAINTY LOW MARGIN/RISK-IMPORTANT SEQUENCES

Case	Initiating Event	CI	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2/3	PRHR	Sequence CDF	Sequence LRF	Percentage CDF	Percentage LRF	Short-Term	Long-Term
1	SILB	Yes	Yes	1	0	4	2-4	N/A	8.96E-07	5.37E-08	371.7	275.6	C	F
2	SADS	Yes	Yes	2	1	4	2-4	N/A	4.58E-07	2.75E-08	190.1	140.9	E	F
3	SILB	Yes	Yes	0	1	4	2-4	Yes	3.05E-07	1.83E-08	126.8	94.0	A	F
4	MLOCA	Yes	Yes	0	2	4	2-4	Yes	2.89E-07	1.73E-08	119.9	88.9	B	F
5	CMT	Yes	Yes	0	2	4	2-4	Yes	1.34E-07	8.05E-09	55.7	41.3	B	F
6	SADS	No	Yes	2	2	4	2-4	N/A	9.12E-08	5.47E-09	37.8	28.0	E	G
7	MLOCA	Yes	Yes	2	0	4	2-4	N/A	3.01E-08	1.81E-09	12.5	9.3	C	F
8	LLOCA	No	Yes	2	2	4	2-4	N/A	8.51E-09	8.51E-09	3.5	43.6	D	G
9	CMT	Yes	Yes	1	0	4	2-4	N/A	6.42E-09	3.85E-10	2.7	2.0	C	F
10	MLOCA	Yes	Yes	0	1	4	2-4	Yes	2.44E-09	1.47E-10	1.0	0.8	A	F
11	SILB	No	Yes	1	0	4	2-4	N/A	1.52E-09	1.52E-09	0.6	7.8	C	G
12	SILB	No	Yes	0	1	4	2-4	Yes	5.16E-10	5.16E-10	0.2	2.6	A	G
13	MLOCA	No	Yes	0	2	4	2-4	Yes	4.88E-10	4.88E-10	0.2	2.5	A	G
								Totals =	**2.22E-06**	**1.44E-07**				
								Residue from UC Sequences not selected	1.26E-08	8.62E-10	5.2	4.4		
								Residue from sequences with PRHR failure	1.58E-10	9.67E-12	0.1	0.0		

Table A5.1-6

LOW MARGIN (UC) SEQUENCES ANALYZED FOR THERMAL/HYDRAULIC UNCERTAINTY EVALUATION

Analysis Case	Initiating Event(1)	Cont. Isol.	IRWST & RECIRC	CMT	ACCUM	ADS 4	ADS 2/3	PRHR	Bounds Dominant Case
Short-Term Cooling									
A	Reactor coolant system hot leg (3.0")	No	Yes	0	1	4	0	Yes	3,10,12,13
B	Double-ended CMT balance line (6.8")	Yes	Yes	0	2	4	0	Yes	4,5
C	Double-ended DVI line (4")	No	Yes	1	0	3	0	No	1,7,9,11
D	Double-ended cold-leg LLOCA	No	Yes	2	2	4	0	Yes	8
E	Spurious ADS-4(2)	No	Yes	1	1	4	0	Yes	2,6
Long-Term Cooling									
F	Double-ended DVI	Yes	1/1&1/1(3)	1	0	3	0	No	1-5,7,9,10
G	Double-ended DVI	No	1/1&2/1(3)	1	0	4	0	No	6,8,11-13

<u>Notes</u>
1. Break sizes are effective sizes (inside diameter or orifice; not outside pipe diameter).
2. Spurious ADS assumes all four ADS stage 4 valves open at same time as initiating event.
3. Indicates number of valves open/number of flow paths open.

Table A5.2-1

CASE A – SEQUENCE OF EVENTS (3.0-INCH HOT-LEG BREAK)

Event	Time (sec)
Break Opens	0.0
Reactor Trip Signal	39
"S" Signal	47
Steam Turbine Stop Valves Close	48
Main Feed Isolation Valves Begin to Close	48
Reactor Coolant Pumps Start Coastdown	63
Top of Core Uncovers	1150
ADS Stage 4 Opens	1247
Accumulator Injection Starts	1200
Top of Core Recovers	1250
IRWST Injection Starts	1500
Accumulator Empties	1430

Table A5.2-2

CASE B – SEQUENCE OF EVENTS (DE CMT BALANCE LINE)

Event	Time (sec)
Break Opens	0.0
Reactor Trip Signal	10.7
Steam Turbine Stop Valves Close	16.7
Main Feed Isolation Valves Begin to Close	16.7
"S" Signal	16.7
Reactor Coolant Pumps Start Coastdown	32.7
Accumulator Injection Starts	290
Accumulators Empty	1350
ADS Stage 4 Opens	1217
IRWST Injection Starts	1450

Table A5.2-3

CASE C – SEQUENCE OF EVENTS (DE DVI LINE BREAK)

Event	Time (sec)
Break Opens	0.0
Reactor Trip Signal	13.4
"S" Signal	20
Steam Turbine Stop Valves Close	14.4
Main Feed Isolation Valves Begin to Close	14.4
Reactor Coolant Pumps Start Coastdown	36
ADS Stage 4 Opens	1380
Intact-Loop CMT Empty	1700
Top of Core Uncovers	1870
IRWST Injection Starts	1960
Top of Core Recovers	2890

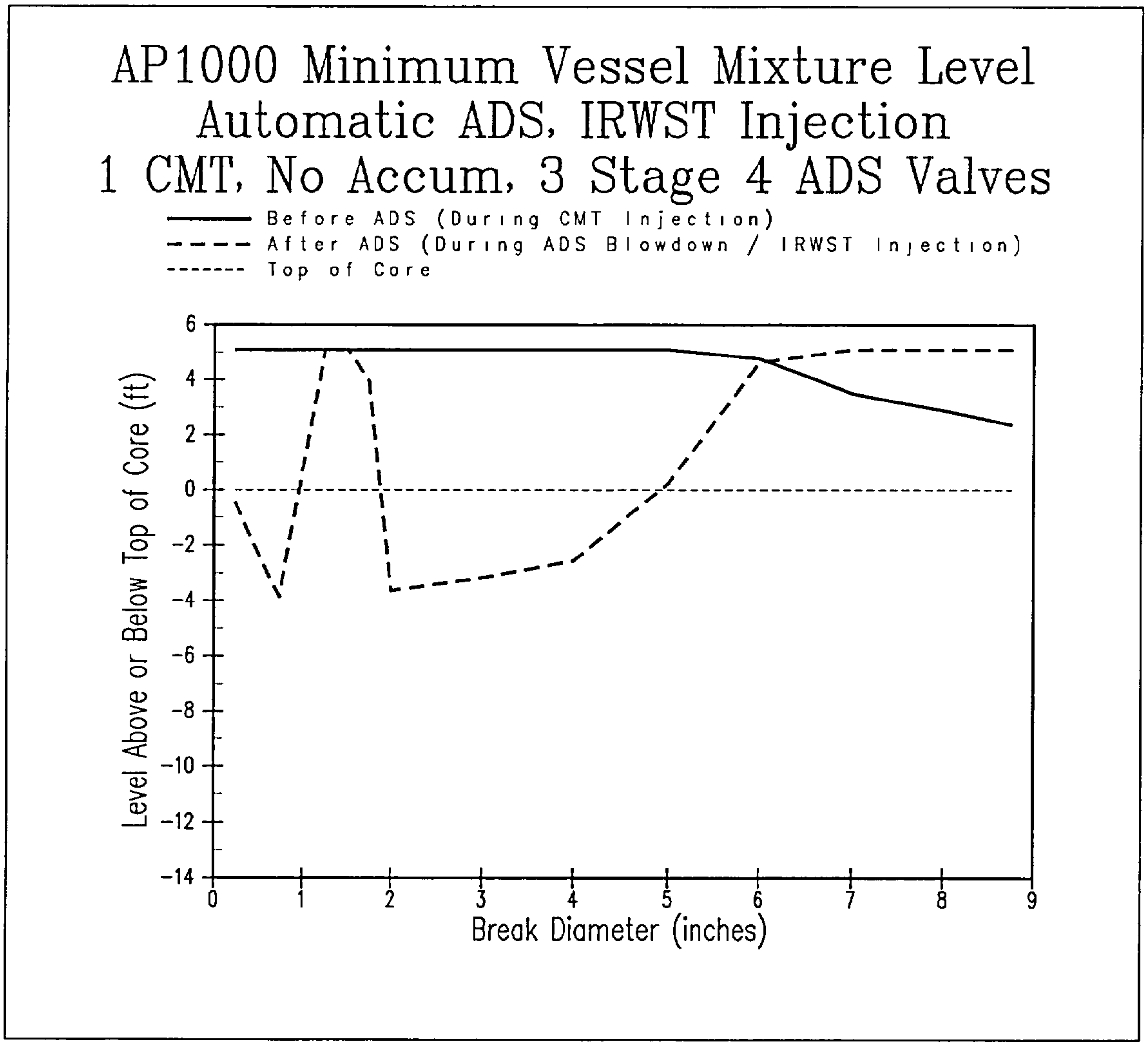


Figure A3.2-1

Minimum Core Mixture Level for Spectrum of Break Sizes (with IRWST Injection)

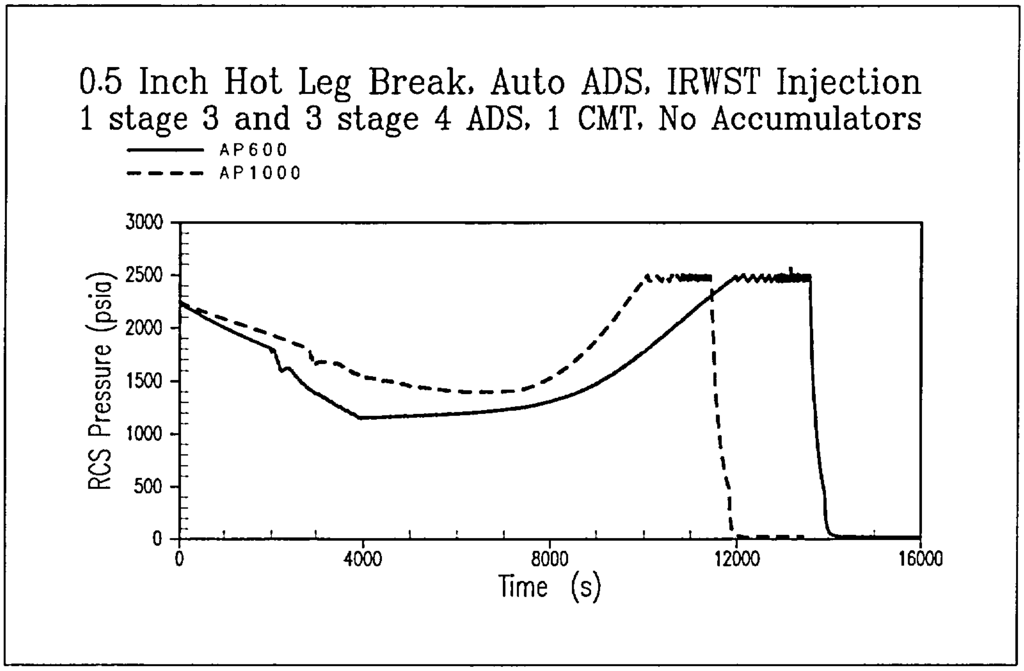


Figure A3.2-2

RCS Pressure for 0.5 Inch Break, Automatic ADS

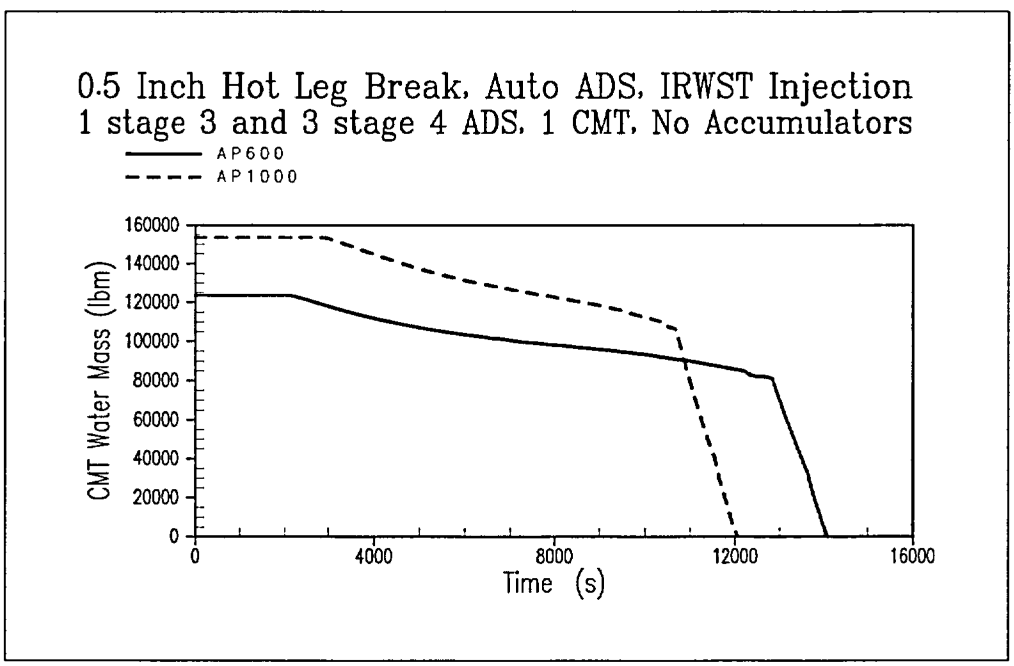


Figure A3.2-3

CMT Water Mass for 0.5 Inch Break, Automatic ADS

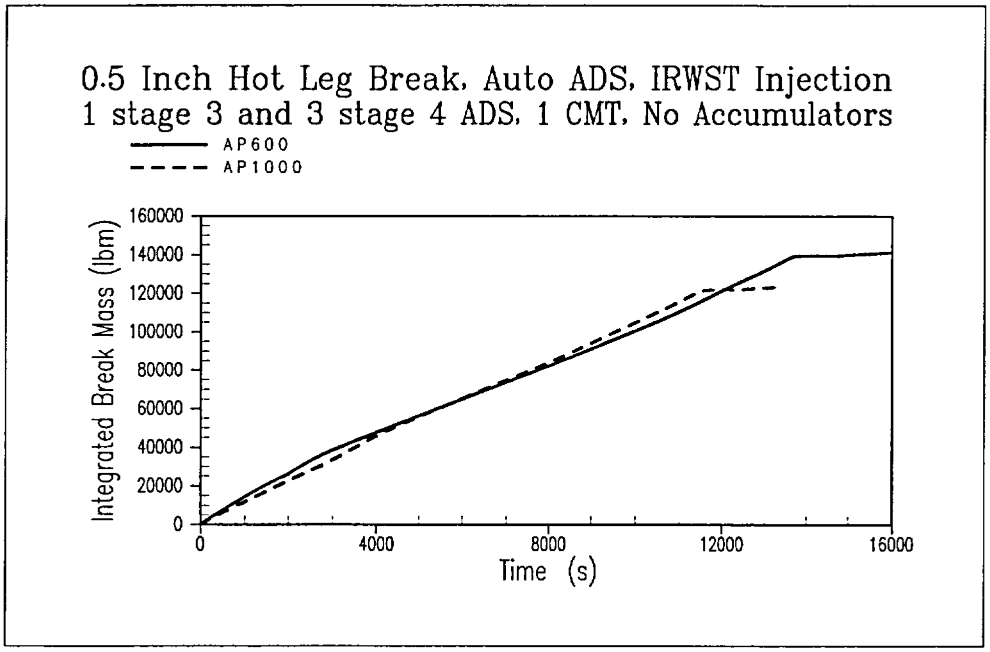


Figure A3.2-4

Integrated Break Flow for 0.5 Inch Break, Automatic ADS

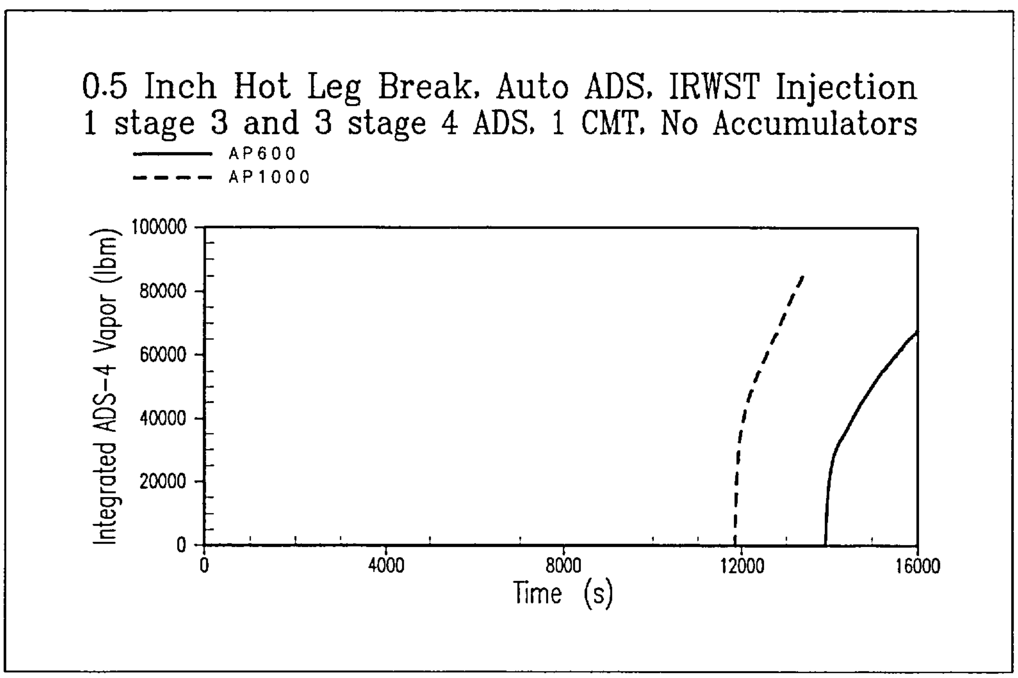


Figure A3.2-5

Integrated Stage 4 ADS Vapor Release for 0.5 Inch Break, Automatic ADS

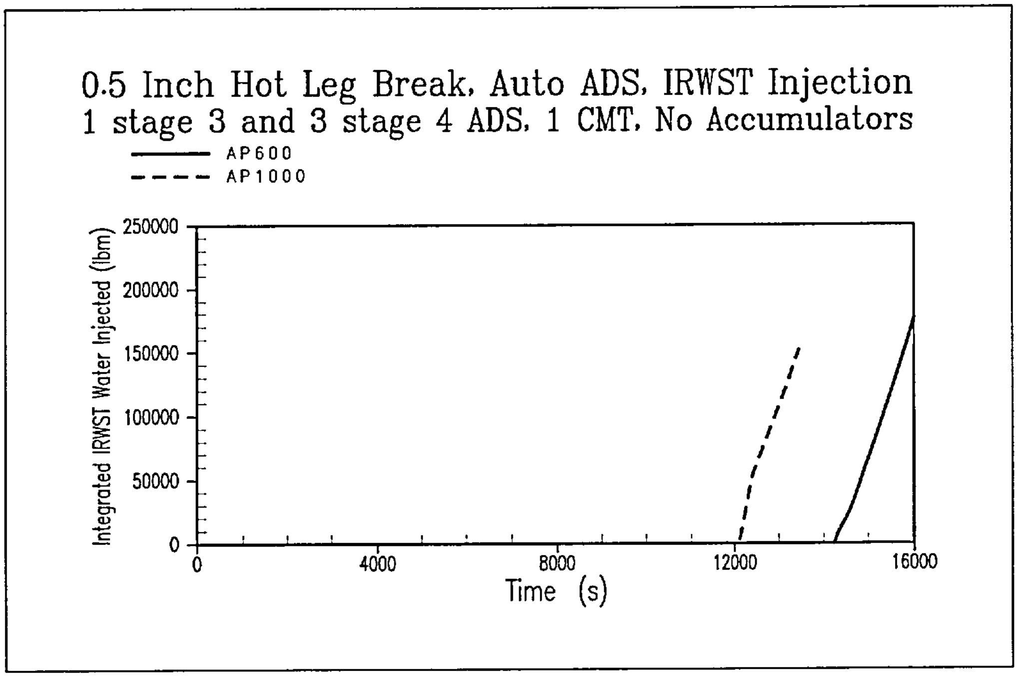


Figure A3.2-6

Integrated IRWST Water Injection for 0.5 Inch Break, Automatic ADS

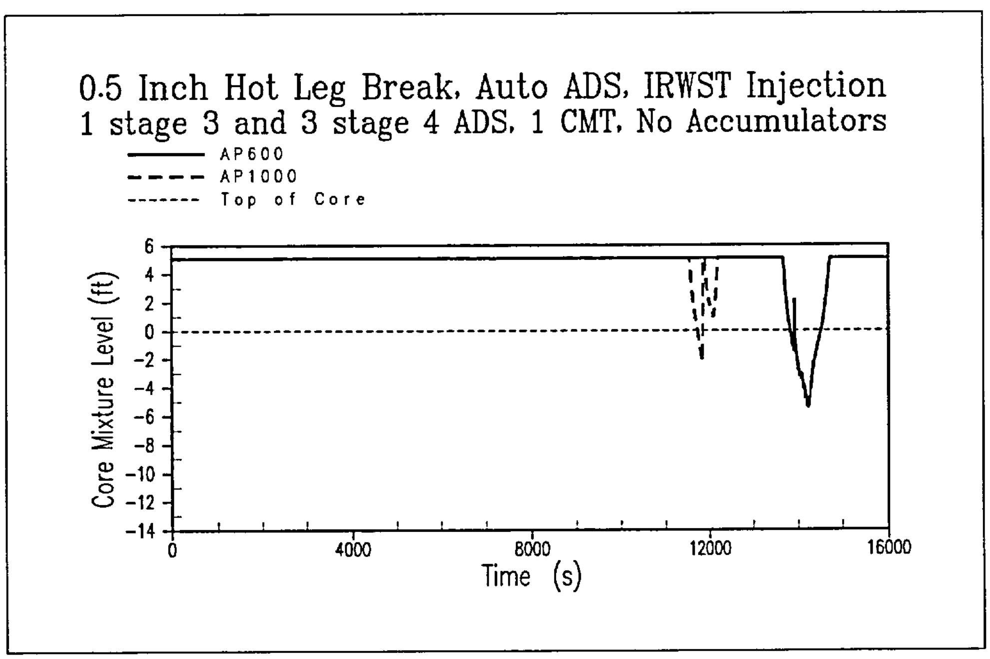


Figure A3.2-7

Core Mixture Level for 0.5 Inch Break, Automatic ADS

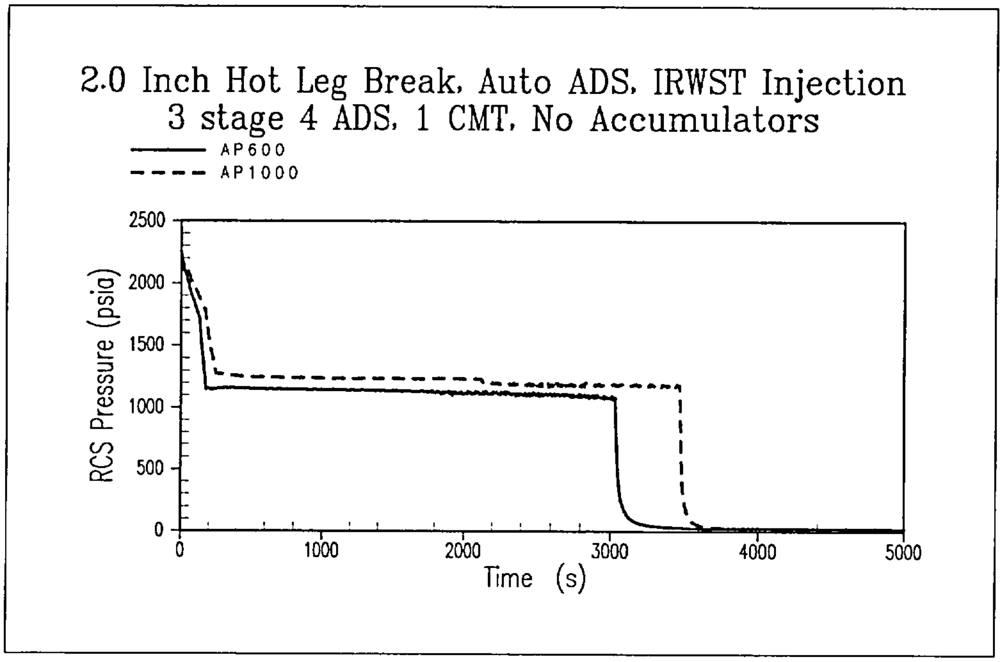


Figure A3.2-8

RCS Pressure for 2.0 Inch Break, Automatic ADS

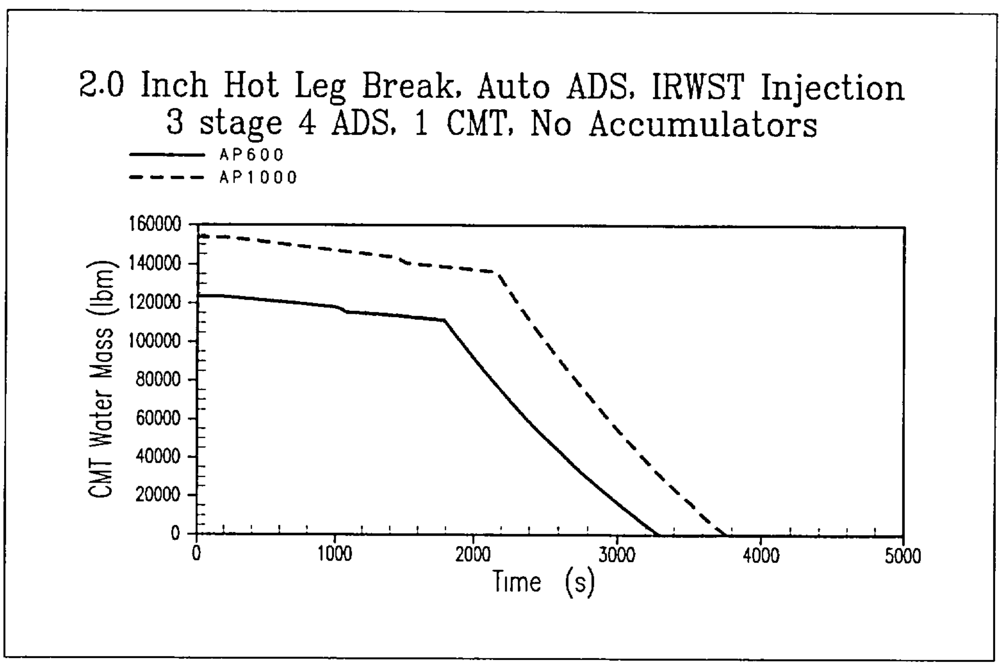


Figure A3.2-9

CMT Water Mass for 2.0 Inch Break, Automatic ADS

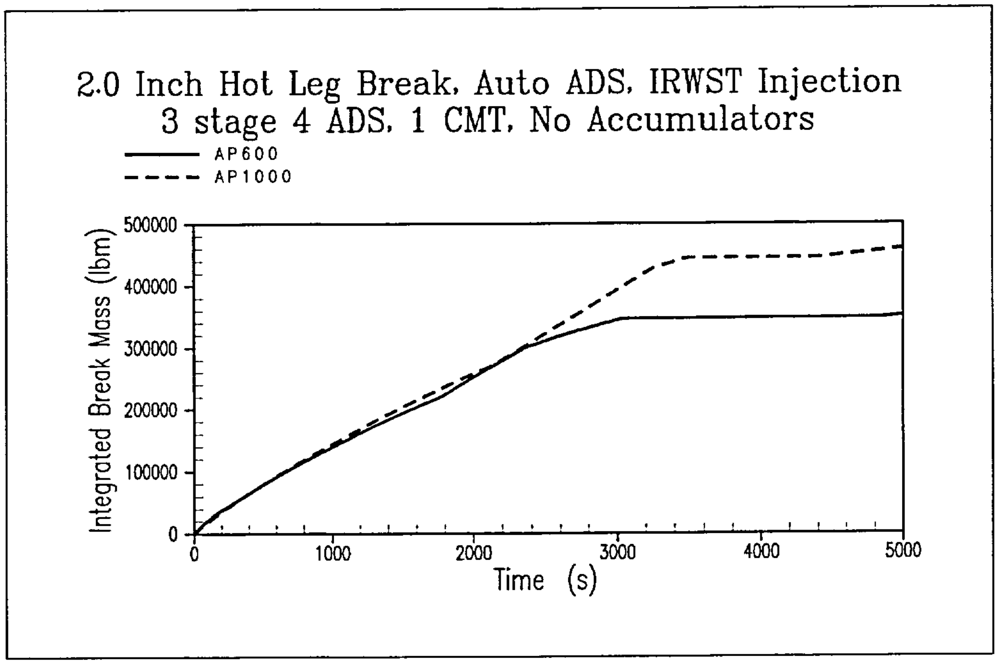


Figure A3.2-10

Integrated Break Flow for 2.0 Inch Break, Automatic ADS

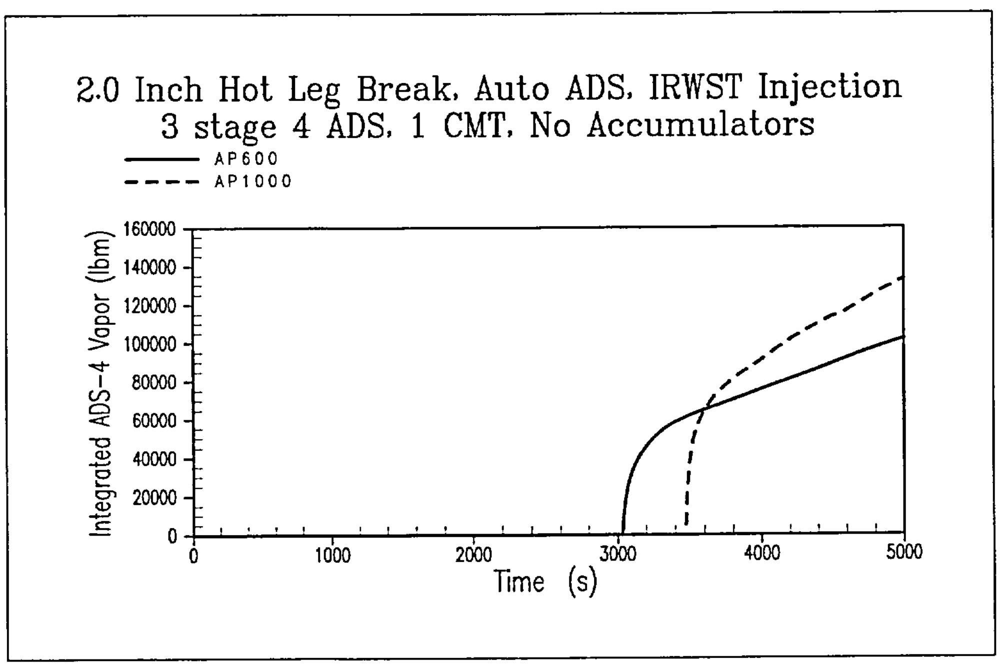


Figure A3.2-11

Integrated Stage 4 ADS Vapor Release for 2.0 Inch Break, Automatic ADS

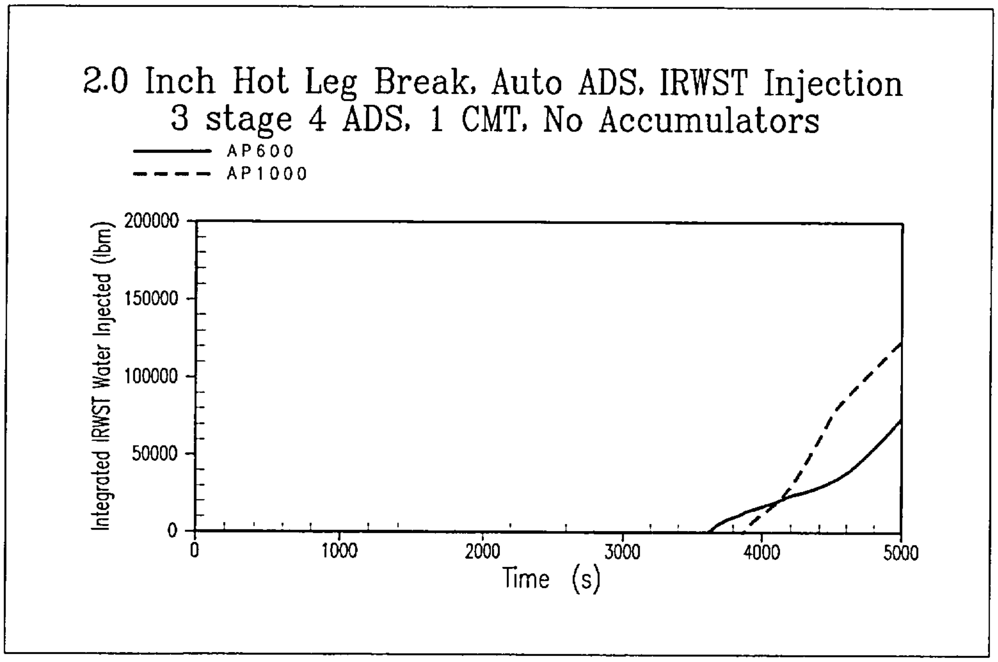


Figure A3.2-12

Integrated IRWST Water Injection for 2.0 Inch Break, Automatic ADS

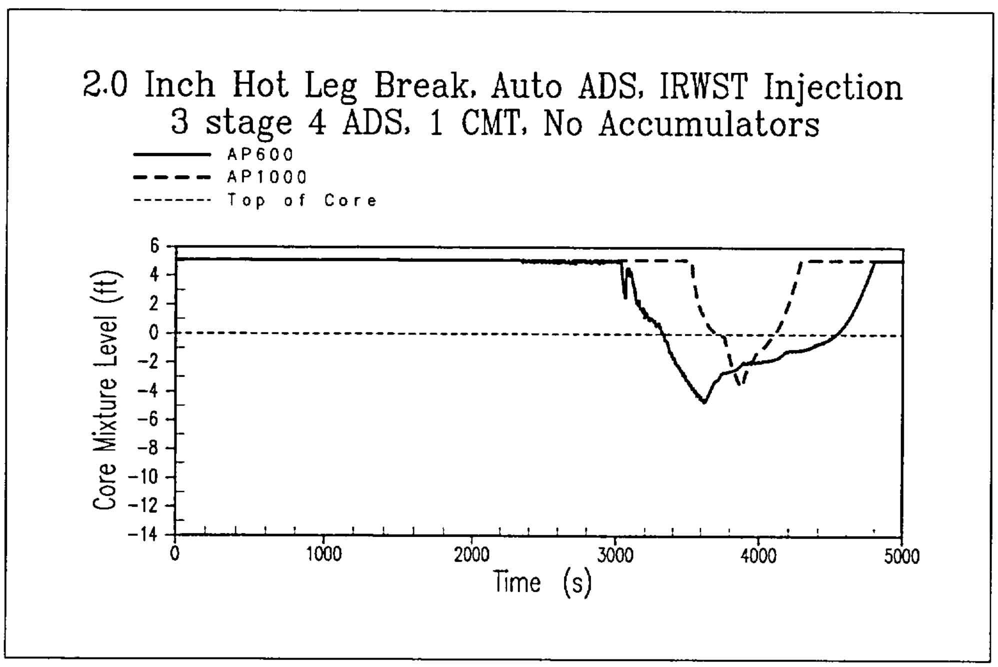


Figure A3.2-13

Core Mixture Level for 2.0 Inch Break, Automatic ADS

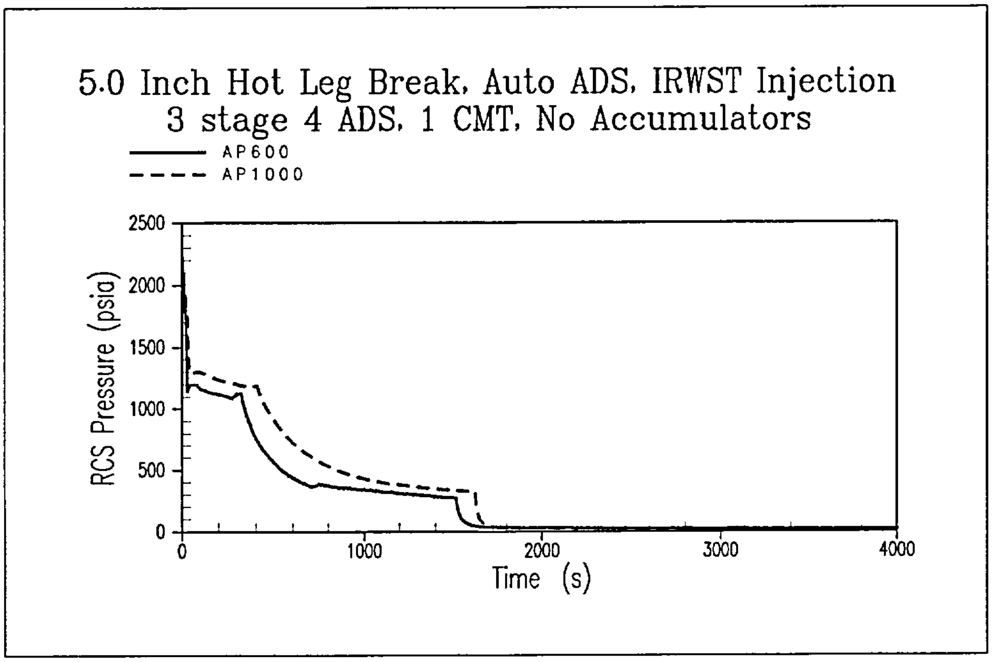


Figure A3.2-14

RCS Pressure for 5.0 Inch Break, Automatic ADS

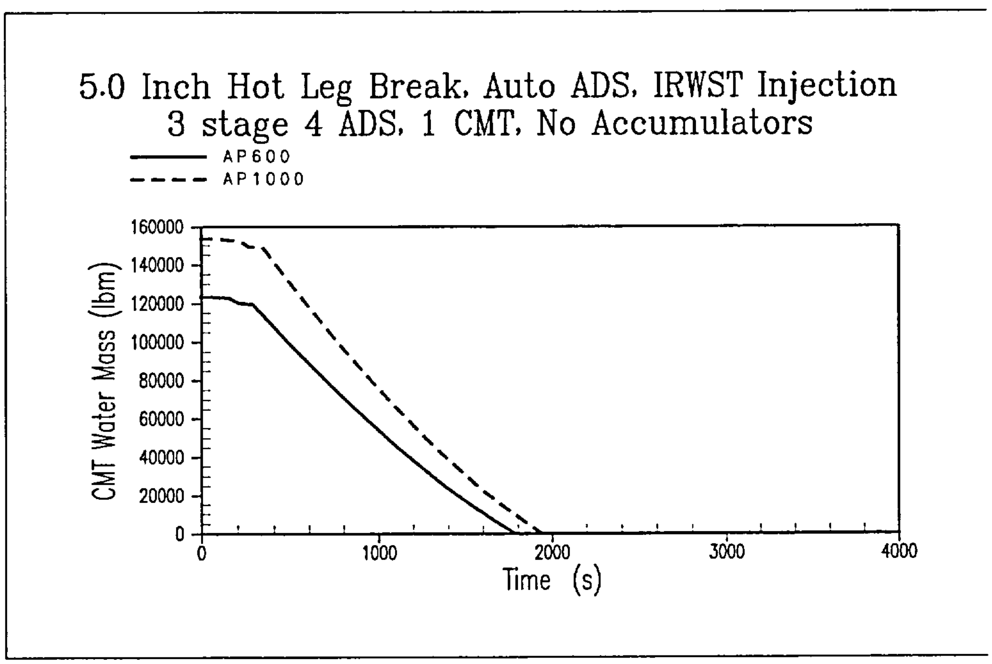


Figure A3.2-15

CMT Water Mass for 5.0 Inch Break, Automatic ADS

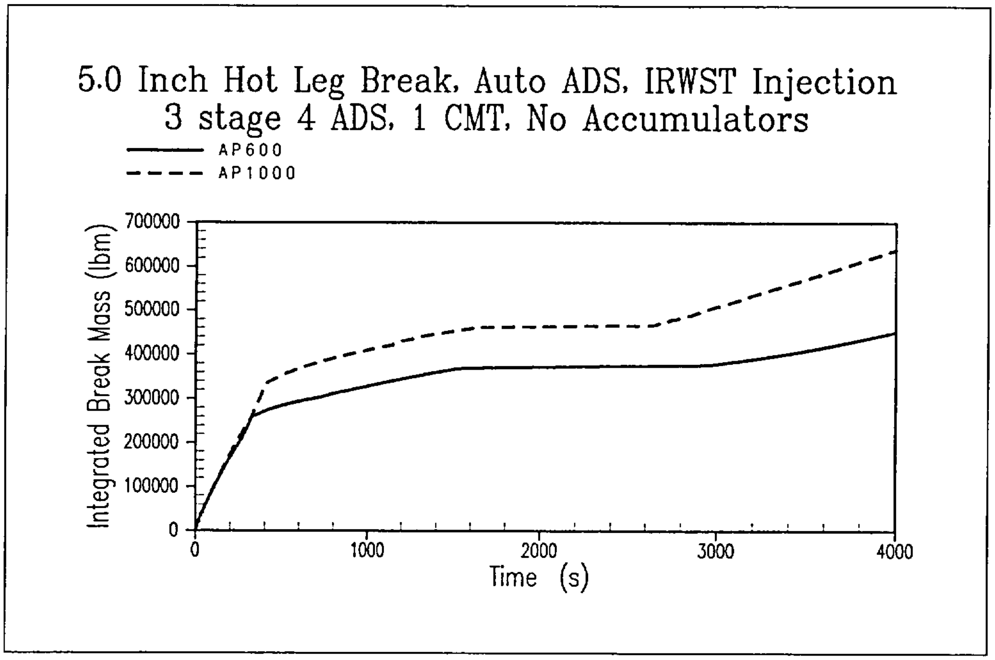


Figure A3.2-16

Integrated Break Flow for 5.0 Inch Break, Automatic ADS

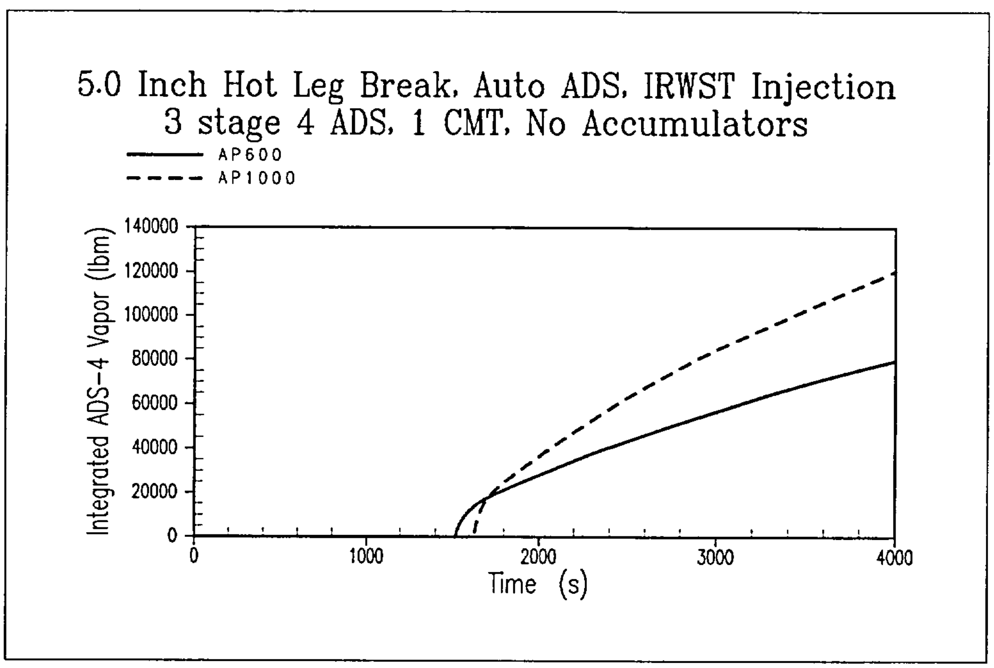


Figure A3.2-17

Integrated Stage 4 ADS Vapor Release for 5.0 Inch Break, Automatic ADS

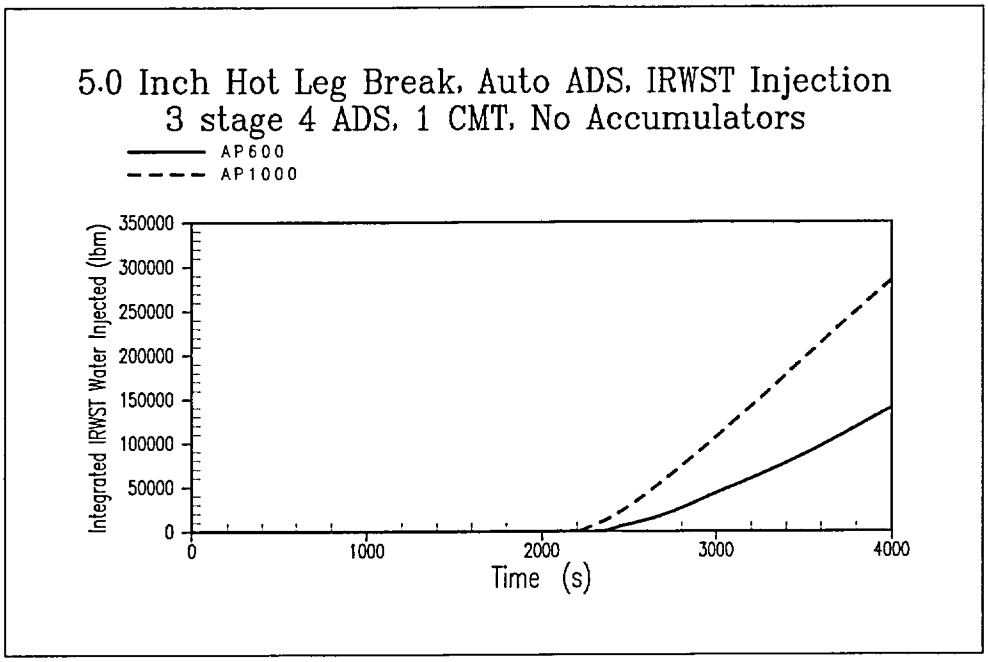


Figure A3.2-18

Integrated IRWST Water Injection for 5.0 Inch Break, Automatic ADS

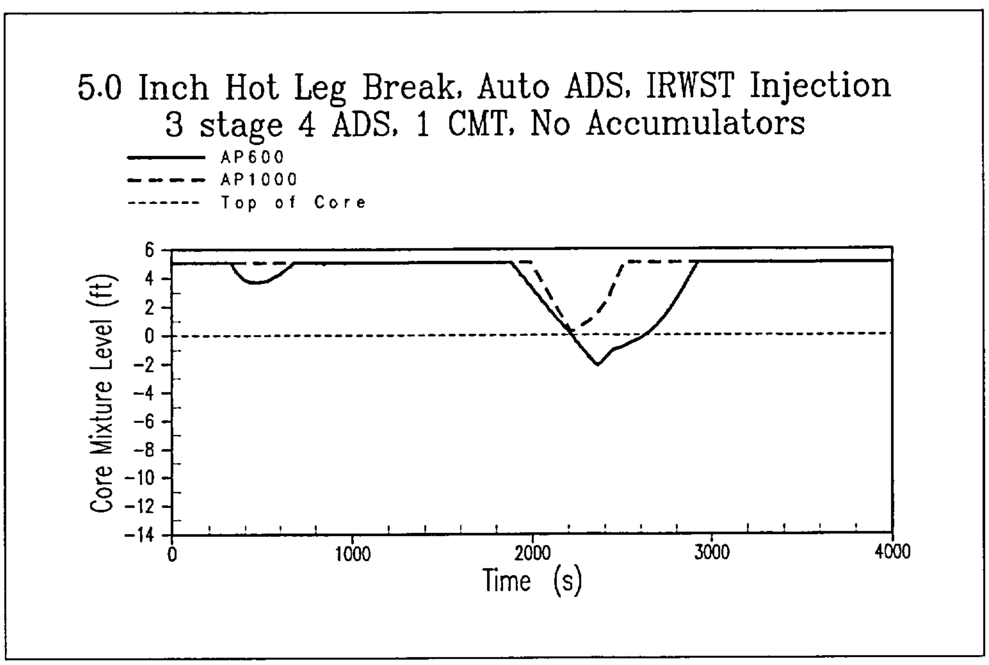


Figure A3.2-19

Core Mixture Level for 5.0 Inch Break, Automatic ADS

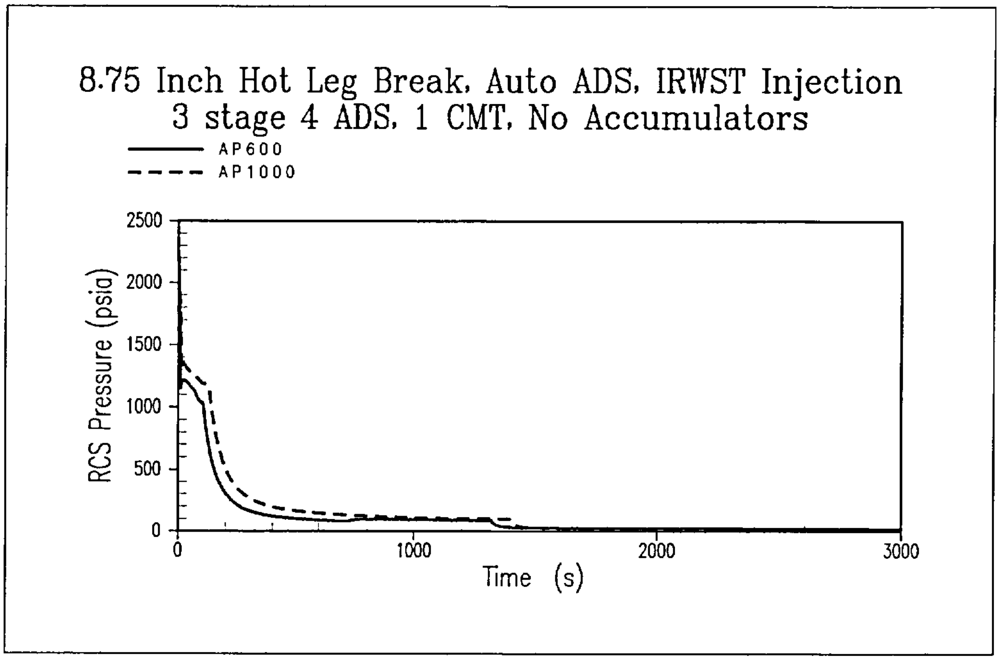


Figure A3.2-20

RCS Pressure for 8.75 Inch Break, Automatic ADS

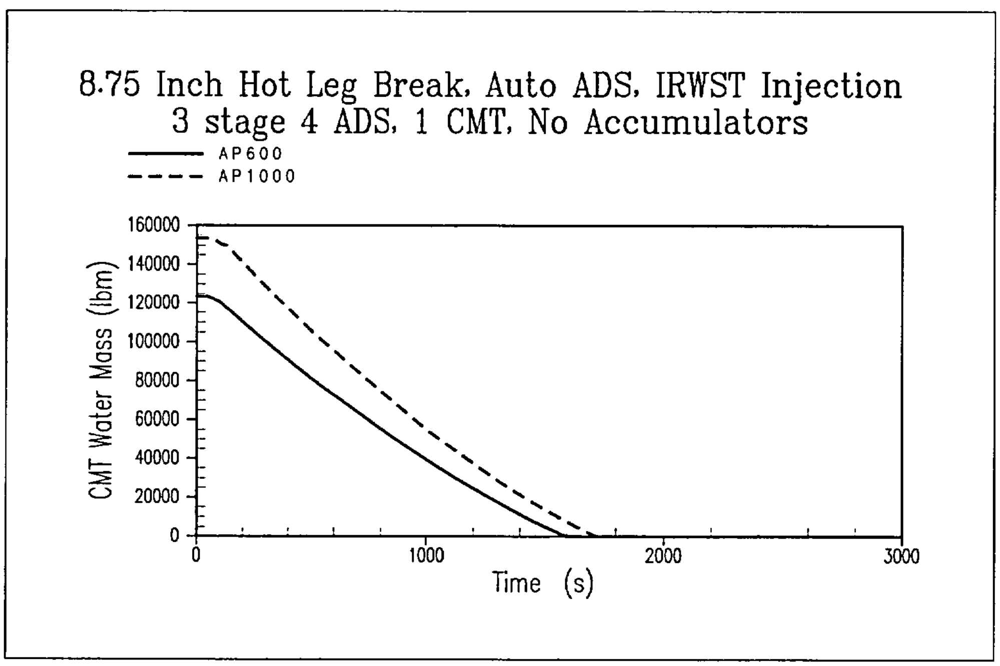


Figure A3.2-21

CMT Water Mass for 8.75 Inch Break, Automatic ADS

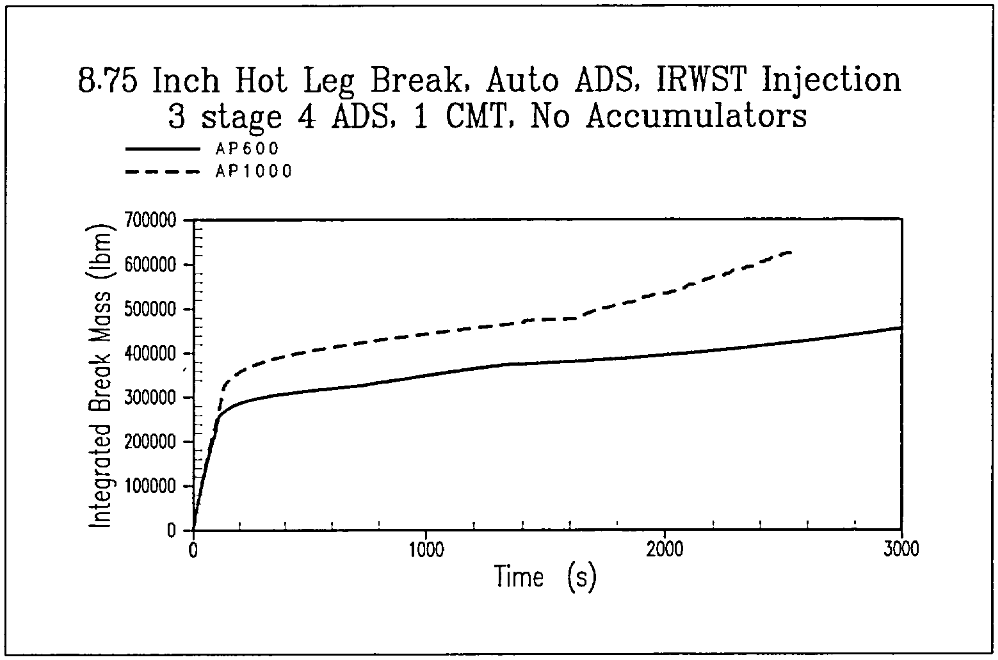


Figure A3.2-22

Integrated Break Flow for 8.75 Inch Break, Automatic ADS

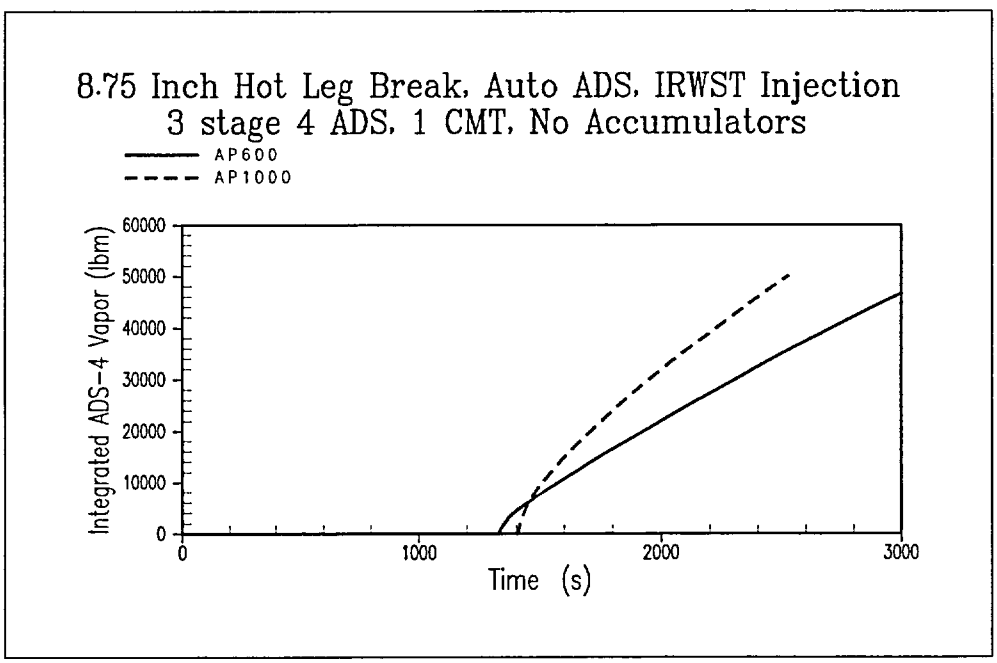


Figure A3.2-23

Integrated Stage 4 ADS Vapor Release for 8.75 Inch Break, Automatic ADS

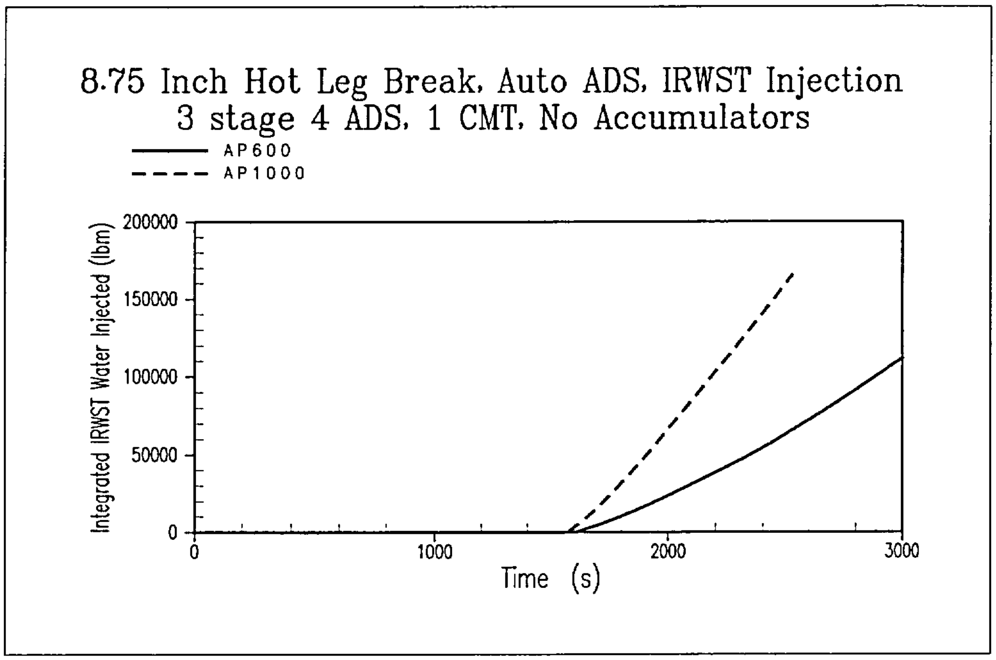


Figure A3.2-24

Integrated IRWST Water Injection for 8.75 Inch Break, Automatic ADS

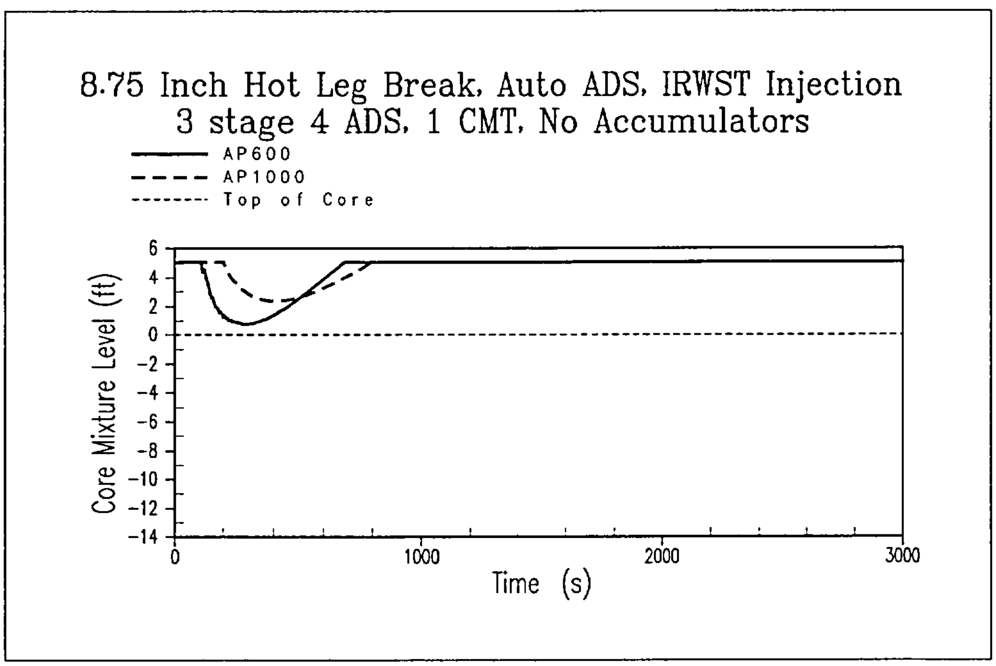


Figure A3.2-25

Core Mixture Level for 8.75 Inch Break, Automatic ADS

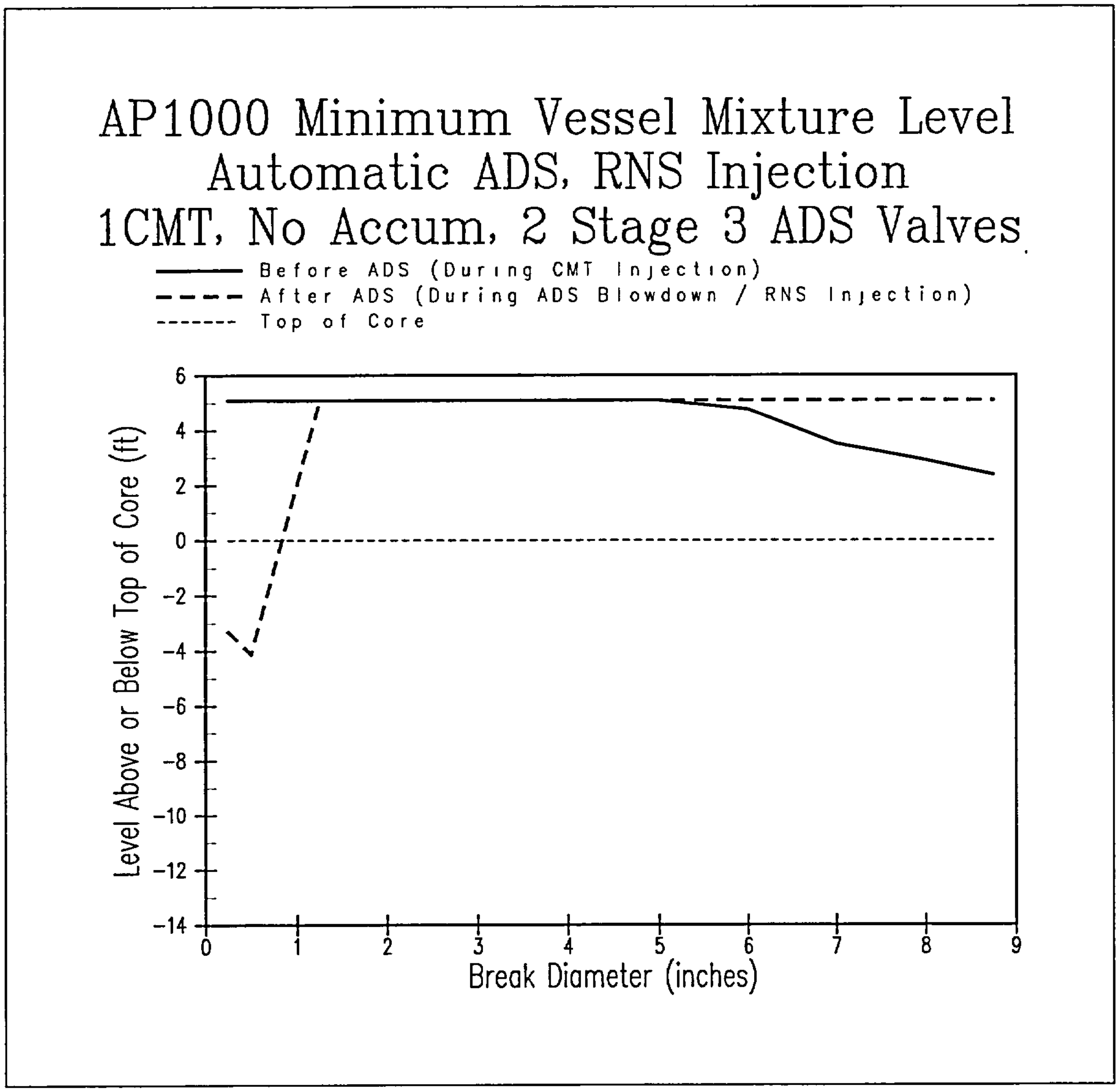


Figure A3.2-26

Minimum Core Mixture Level for Spectrum of Break Sizes (with RNS Injection)

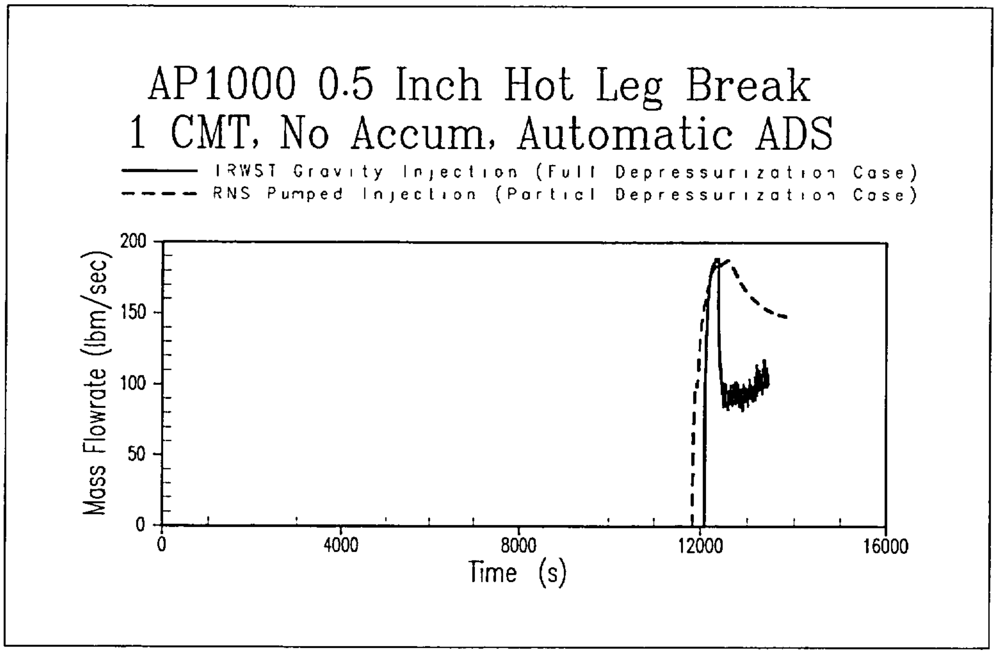


Figure A3.2-27

RNS Injection vs. IRWST Gravity Injection for 0.5 Inch Break

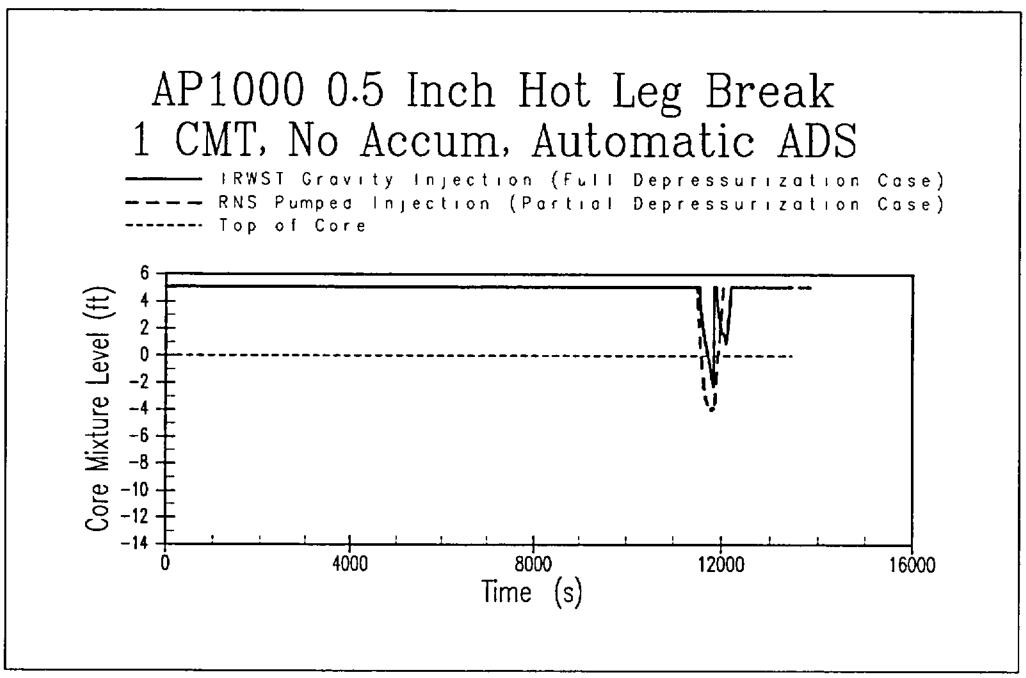


Figure A3.2-28

Core Mixture Level for 0.5 Inch Break with Automatic ADS

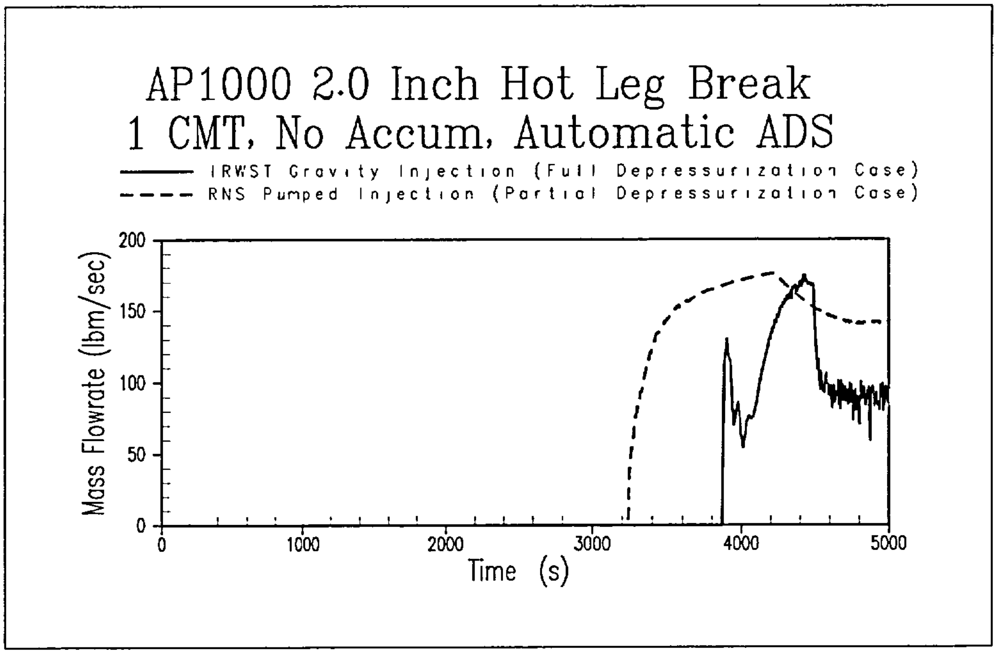


Figure A3.2-29

RNS Injection vs. IRWST Gravity Injection for 2.0 Inch Break

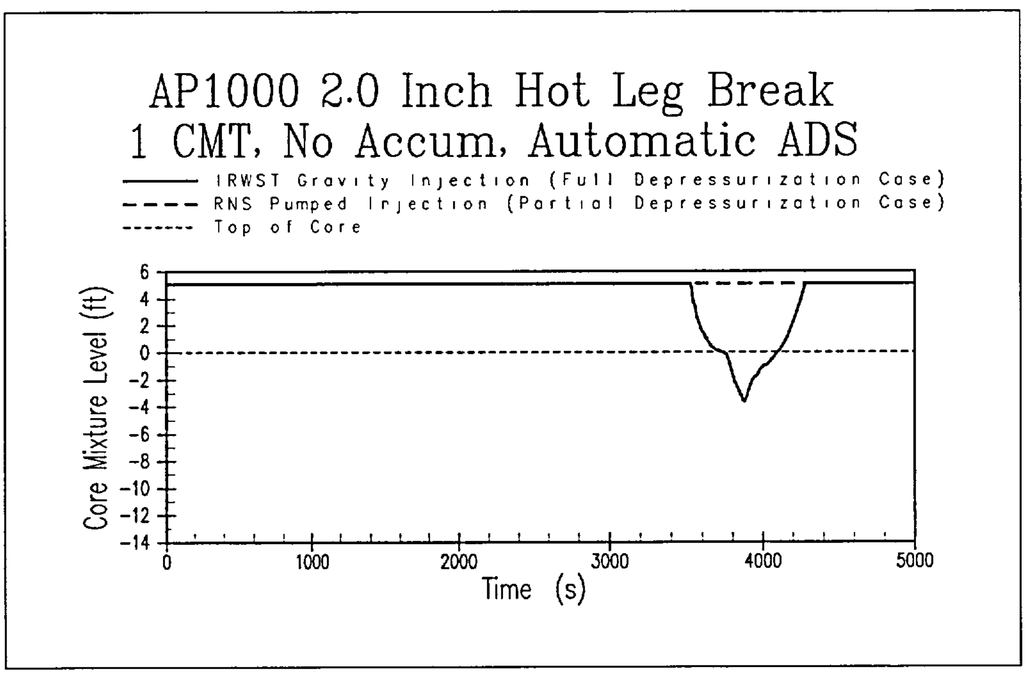


Figure A3.2-30

Core Mixture Level for 2.0 Inch Break with Automatic ADS

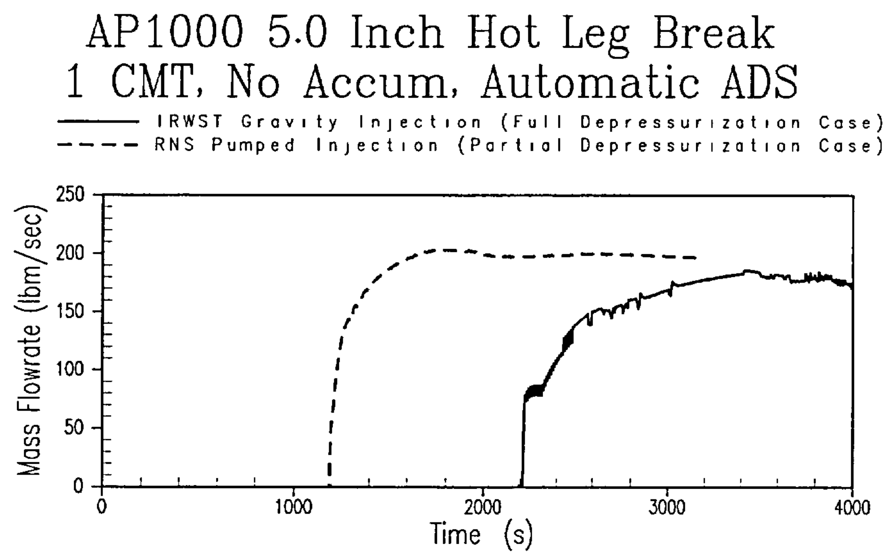


Figure A3.2-31

RNS Injection vs. IRWST Gravity Injection for 5.0 Inch Break

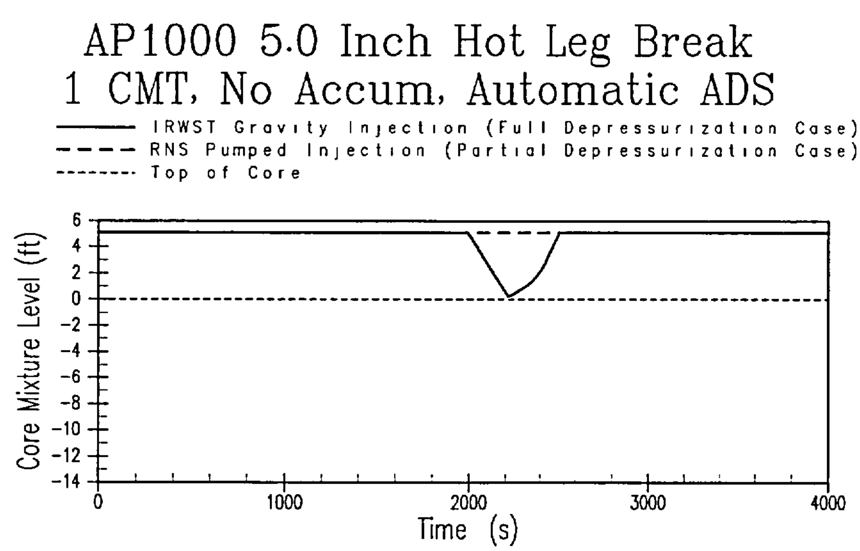


Figure A3.2-32

Core Mixture Level for 5.0 Inch Break with Automatic ADS

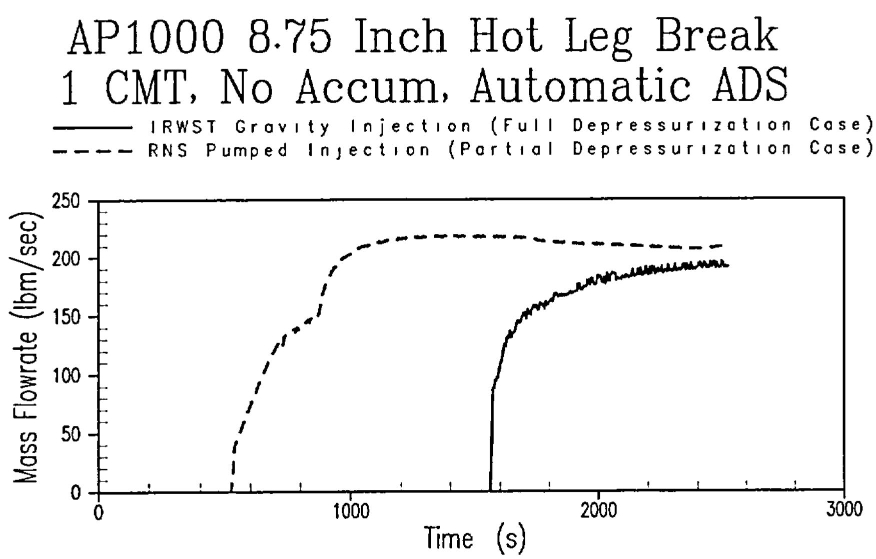


Figure A3.2-33

RNS Injection vs. IRWST Gravity Injection for 8.75 Inch Break

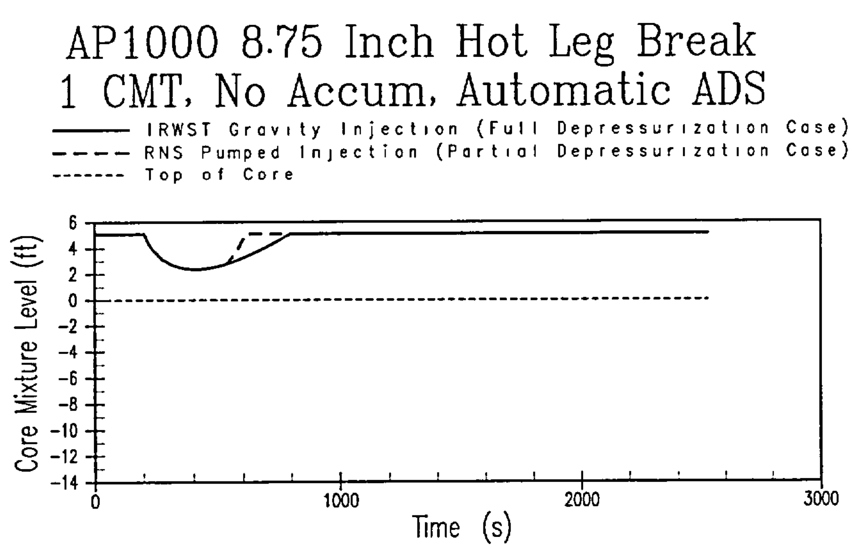


Figure A3.2-34

Core Mixture Level for 8.75 Inch Break with Automatic ADS

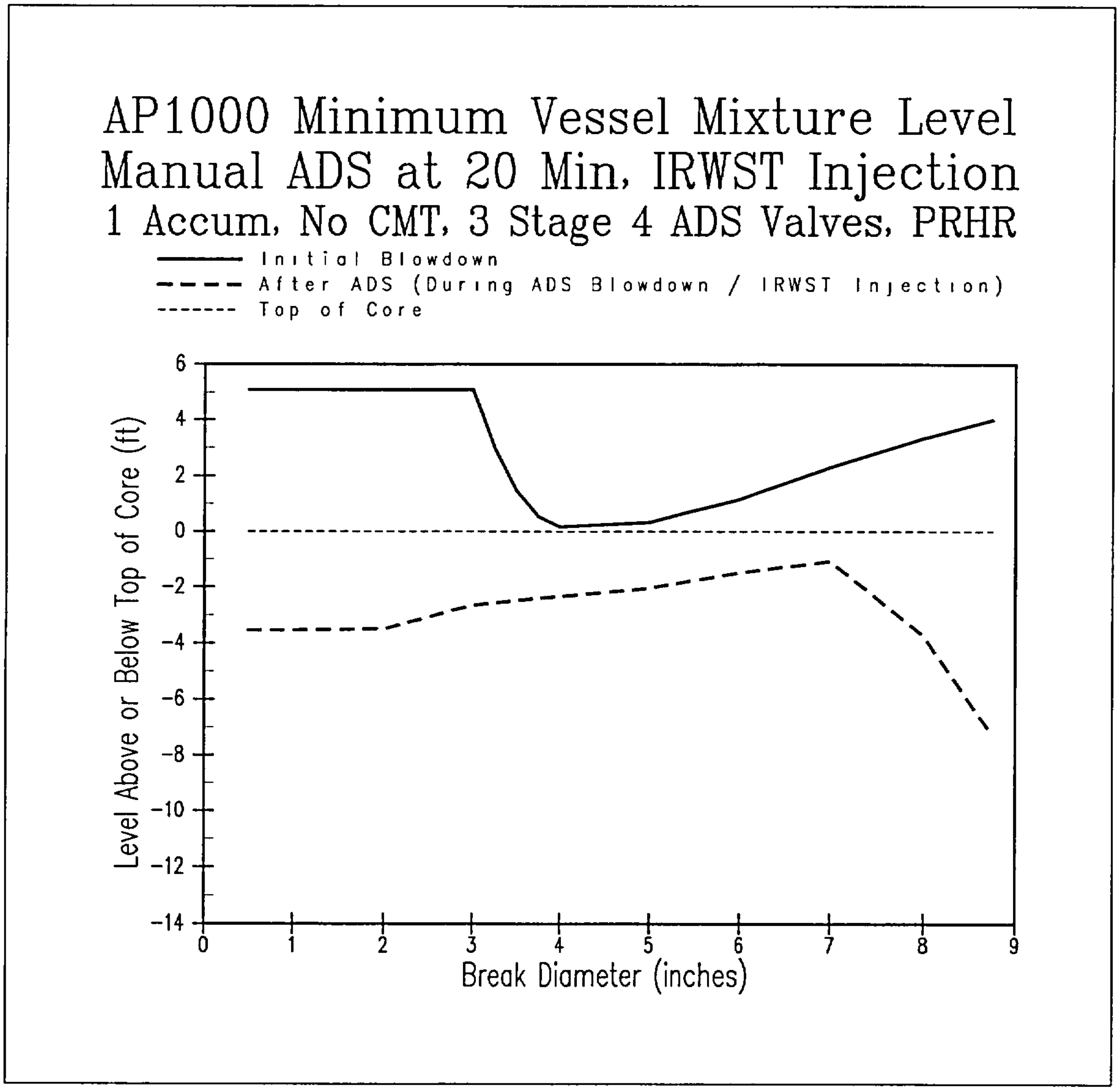


Figure A3.3-1

Minimum Core Mixture Level for Spectrum of Break Sizes (Manual ADS Actuation, IRWST Injection)

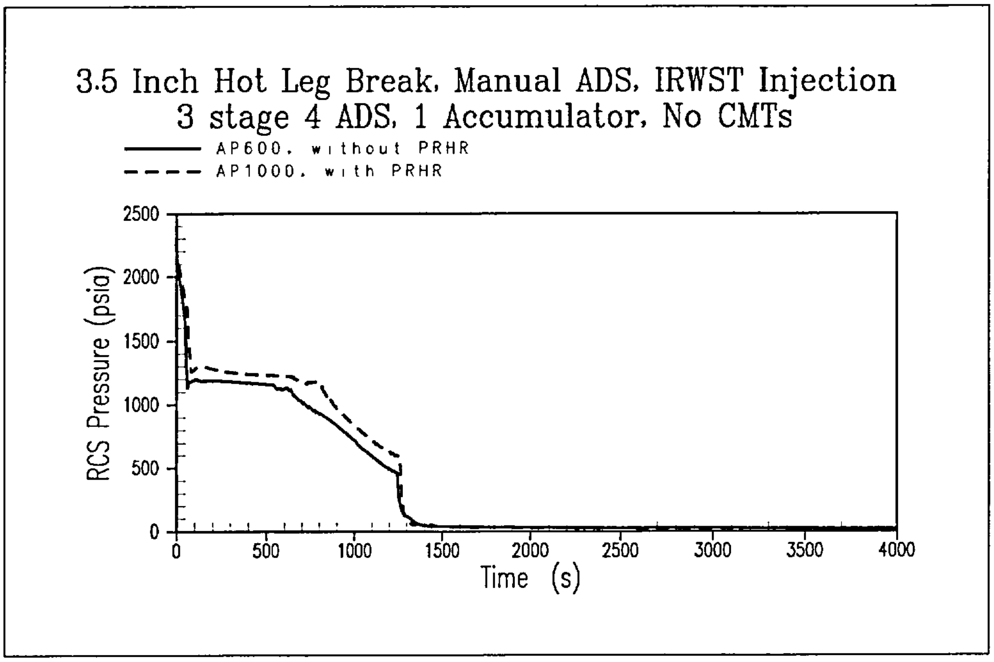


Figure A3.3-2

RCS Pressure for 3.5 Inch Break with Manual ADS at 20 Minutes

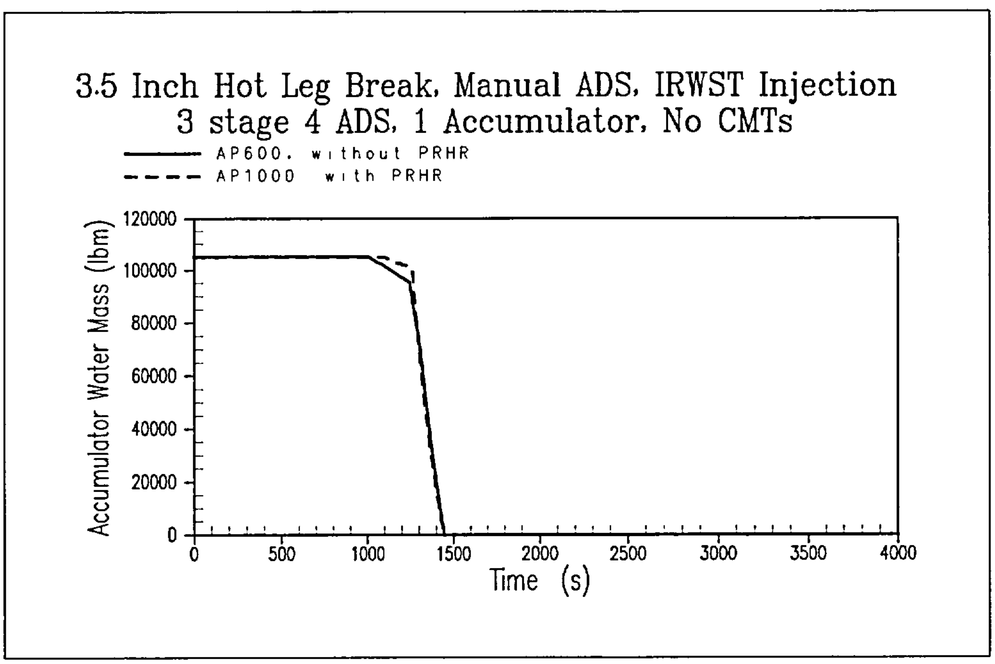


Figure A3.3-3

Accumulator Water Mass for 3.5 Inch Break with Manual ADS at 20 Minutes

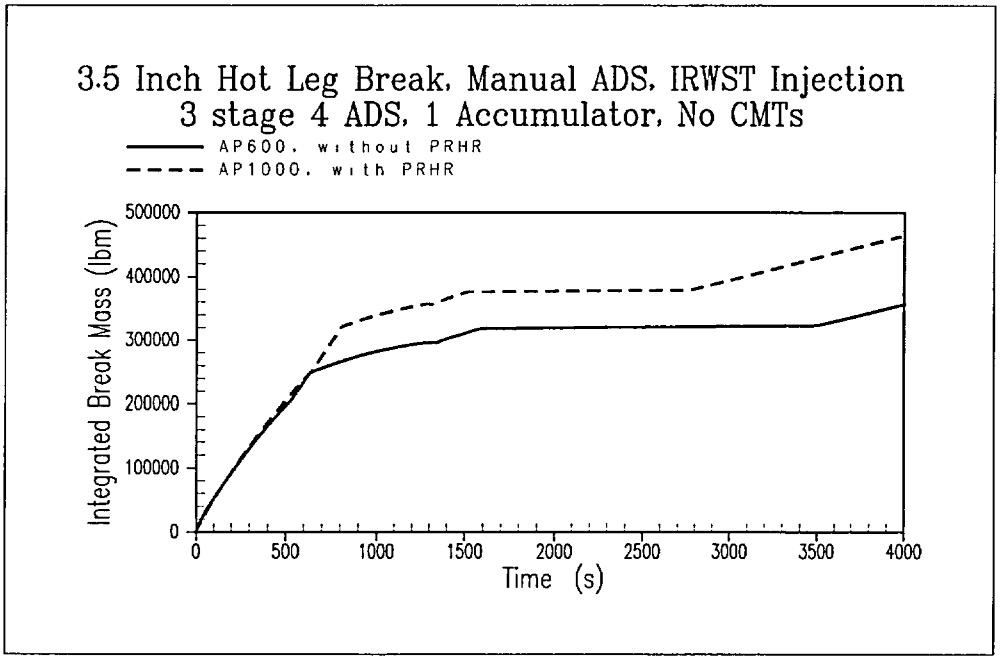


Figure A3.3-4

Integrated Break Flow for 3.5 Inch Break with Manual ADS at 20 Minutes

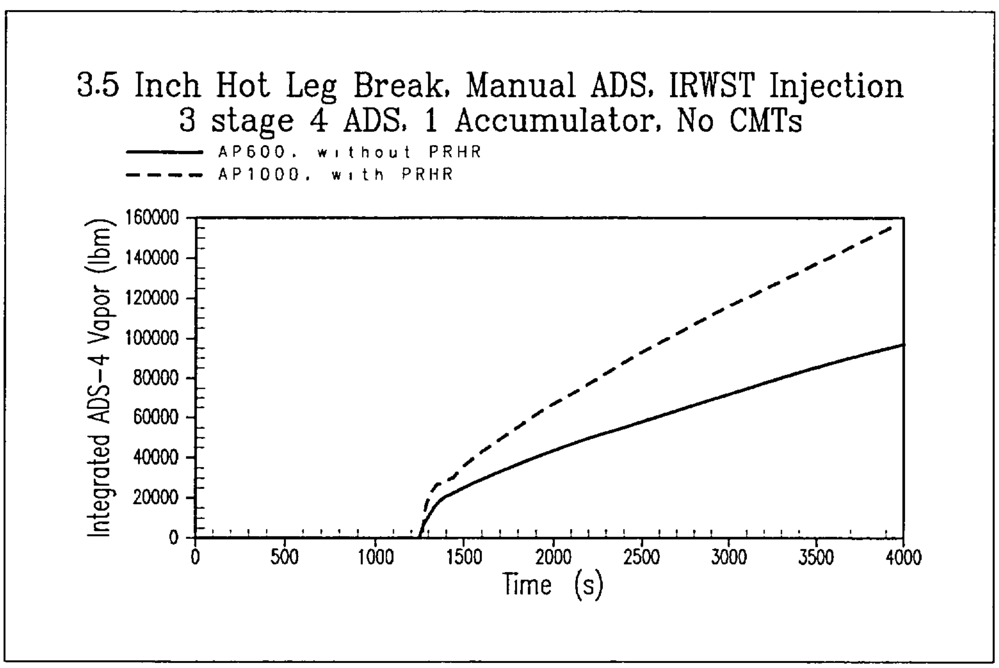


Figure A3.3-5

Integrated Stage 4 ADS Vapor for 3.5 Inch Break with Manual ADS at 20 Minutes

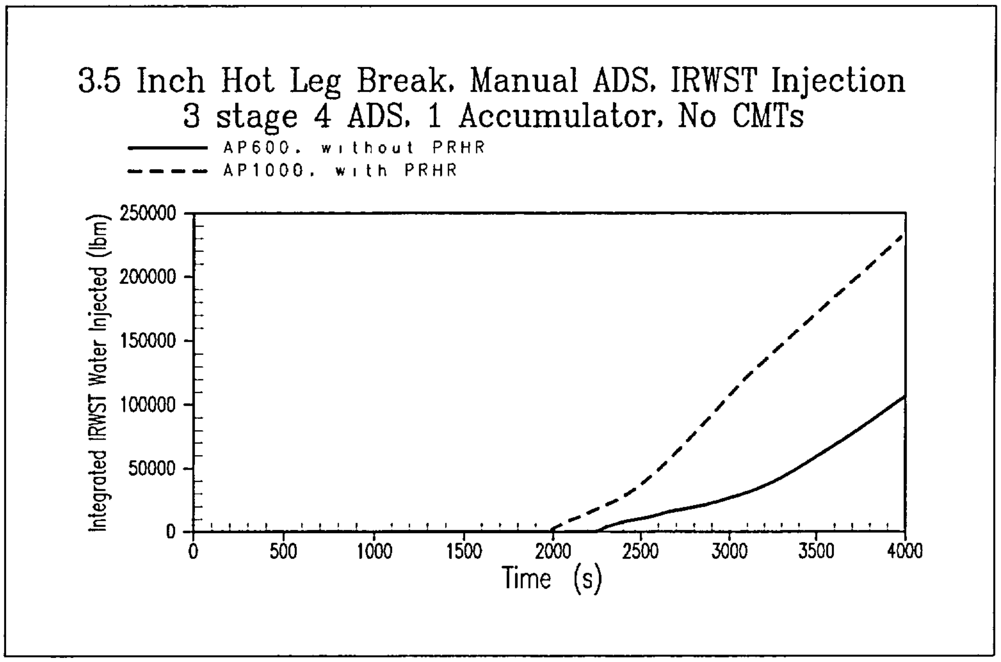


Figure A3.3-6

Integrated IRWST Water Injection for 3.5 Inch Break with Manual ADS at 20 Minutes

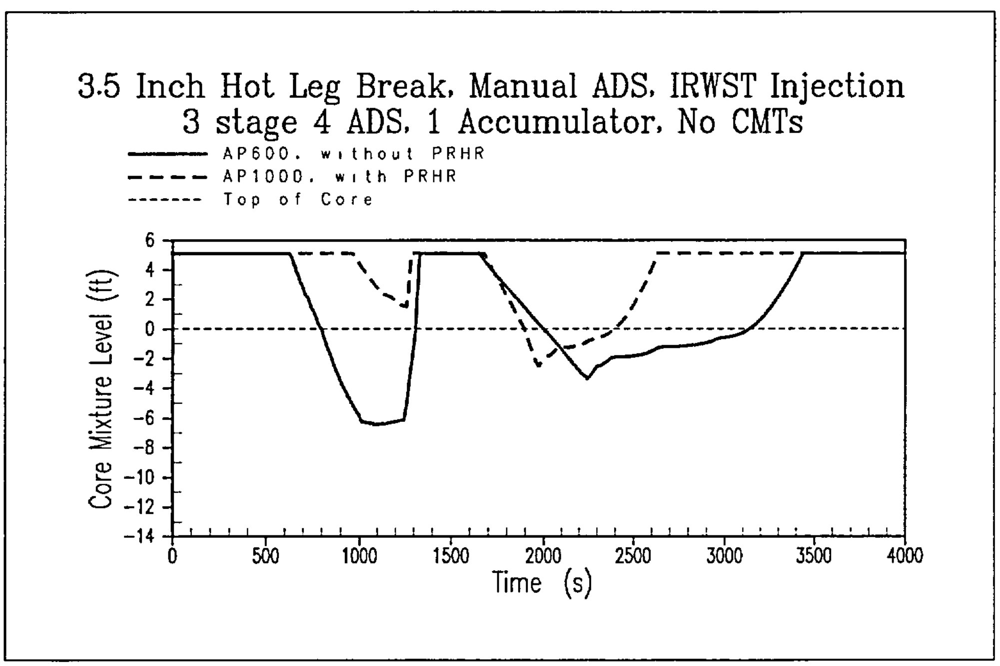


Figure A3.3-7

Core Mixture Level for 3.5 Inch Break with Manual ADS at 20 Minutes

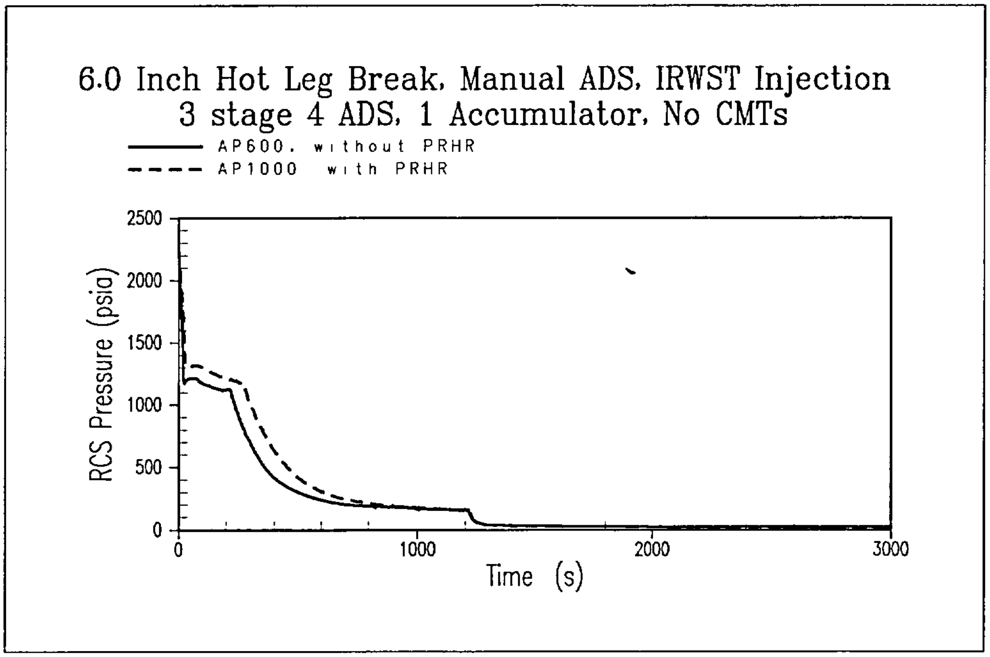


Figure A3.3-8

RCS Pressure for 6.0 Inch Break with Manual ADS at 20 Minutes

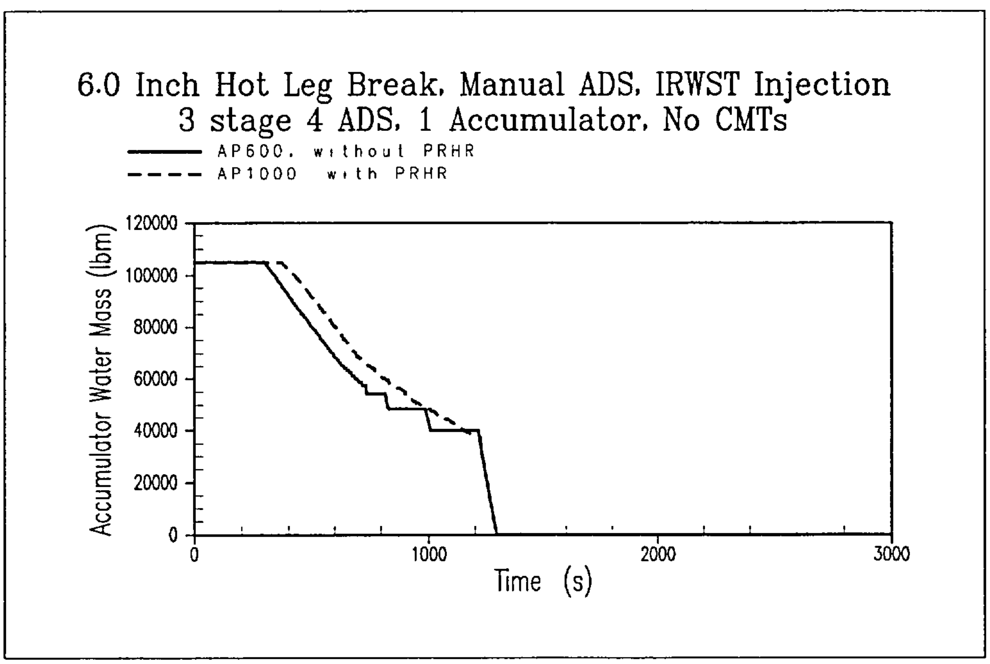


Figure A3.3-9

Accumulator Water Mass for 6.0 Inch Break with Manual ADS at 20 Minutes

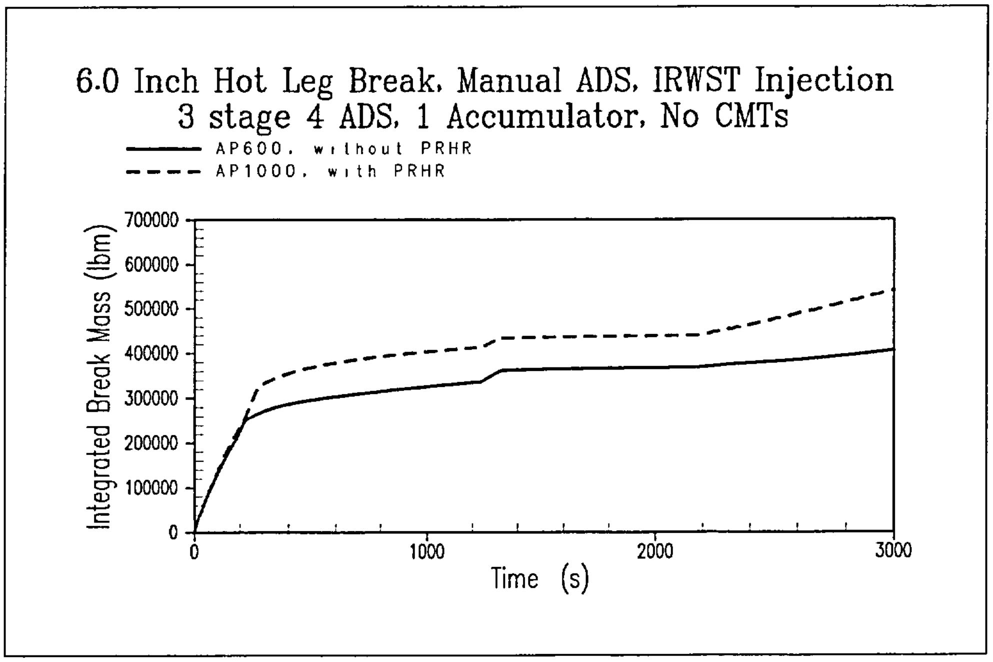


Figure A3.3-10

Integrated Break Flow for 6.0 Inch Break with Manual ADS at 20 Minutes

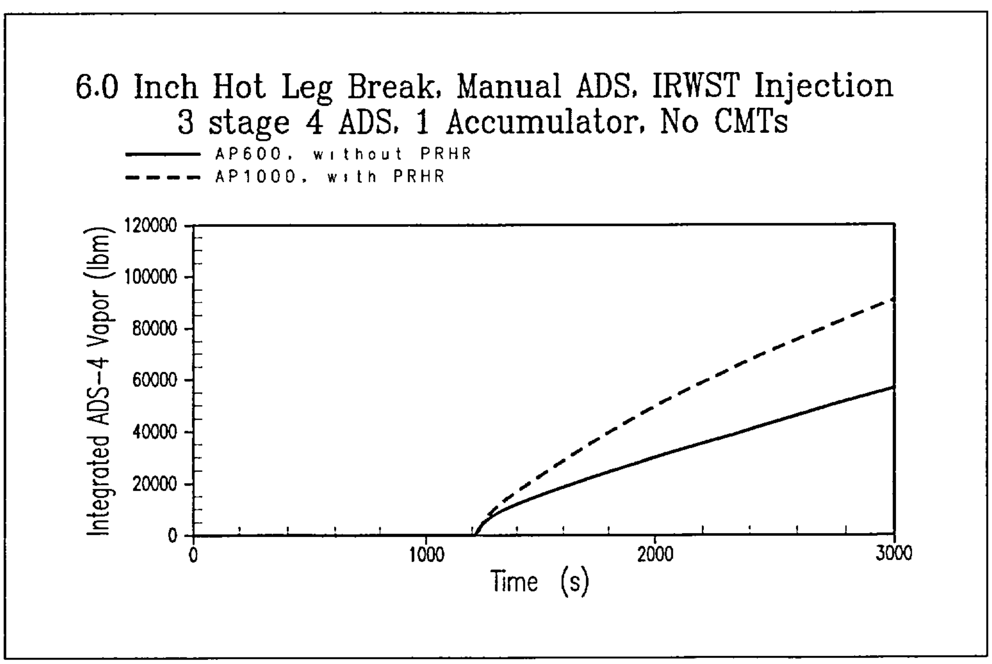


Figure A3.3-11

Integrated Stage 4 ADS Vapor for 6.0 Inch Break with Manual ADS at 20 Minutes

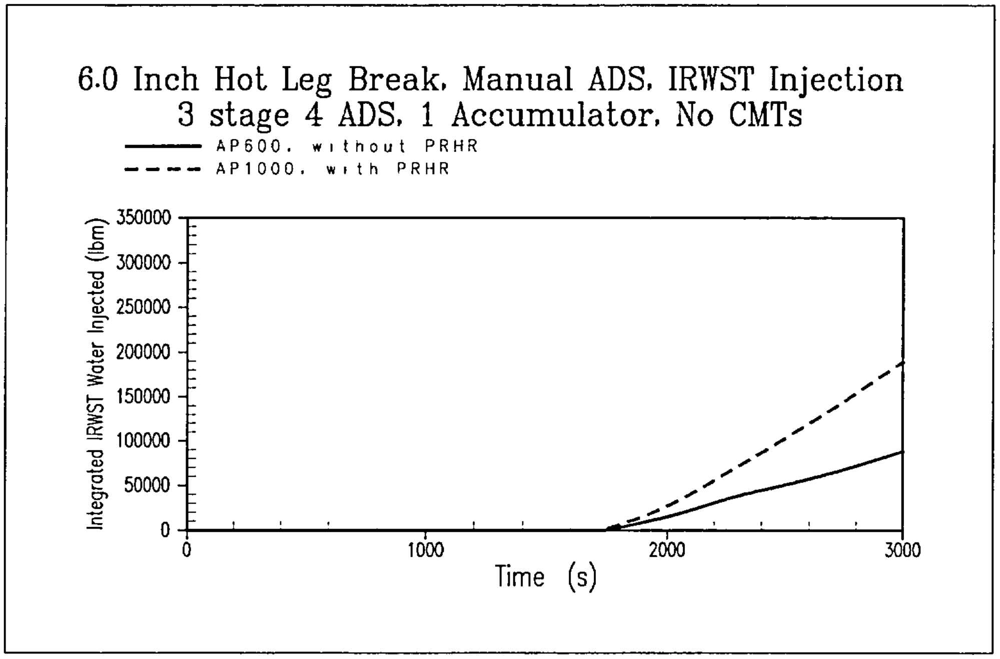


Figure A3.3-12

Integrated IRWST Water Injection for 6.0 Inch Break with Manual ADS at 20 Minutes

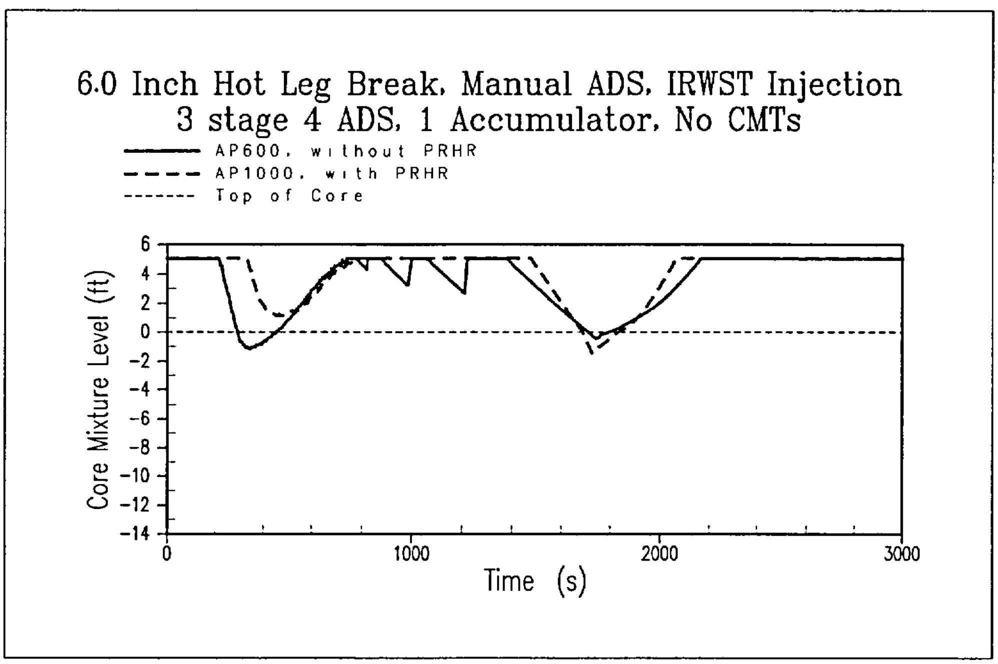


Figure A3.3-13

Core Mixture Level for 6.0 Inch Break with Manual ADS at 20 Minutes

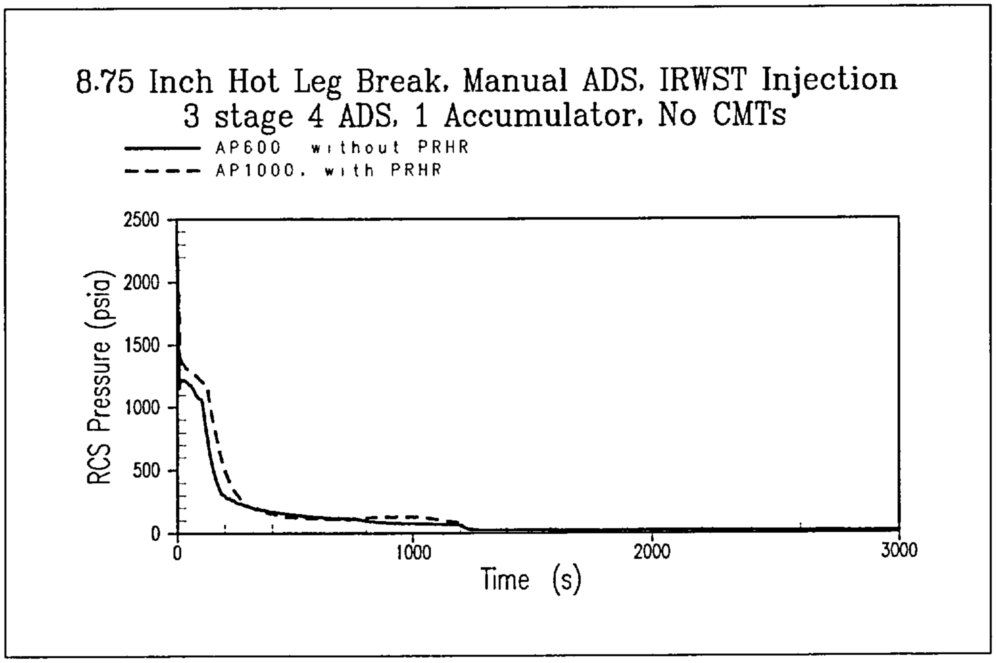


Figure A3.3-14

RCS Pressure for 8.75 Inch Break with Manual ADS at 20 Minutes

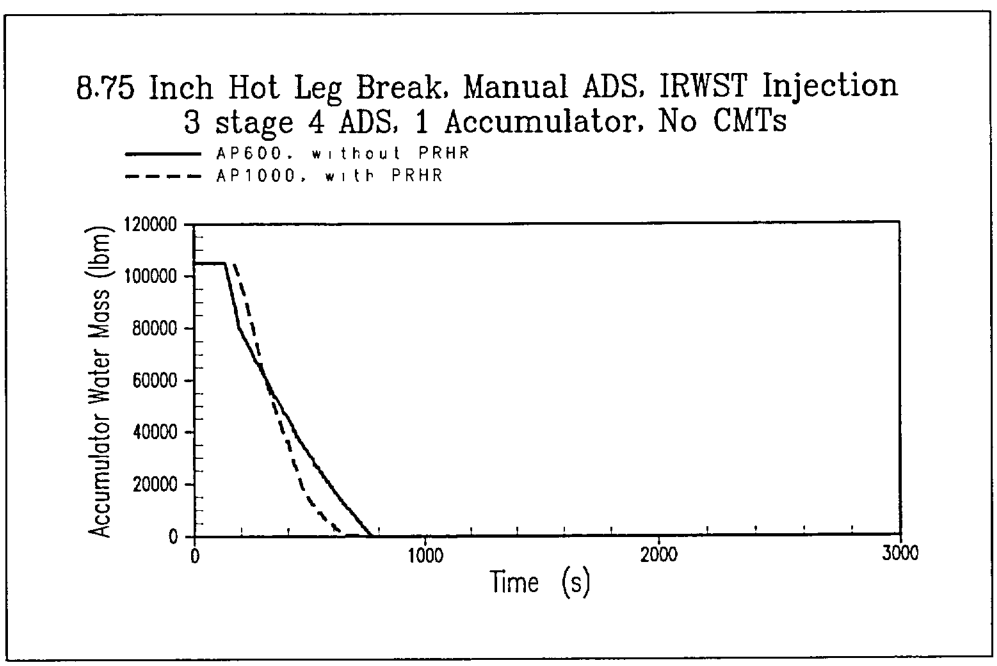


Figure A3.3-15

Accumulator Water Mass for 8.75 Inch Break with Manual ADS at 20 Minutes

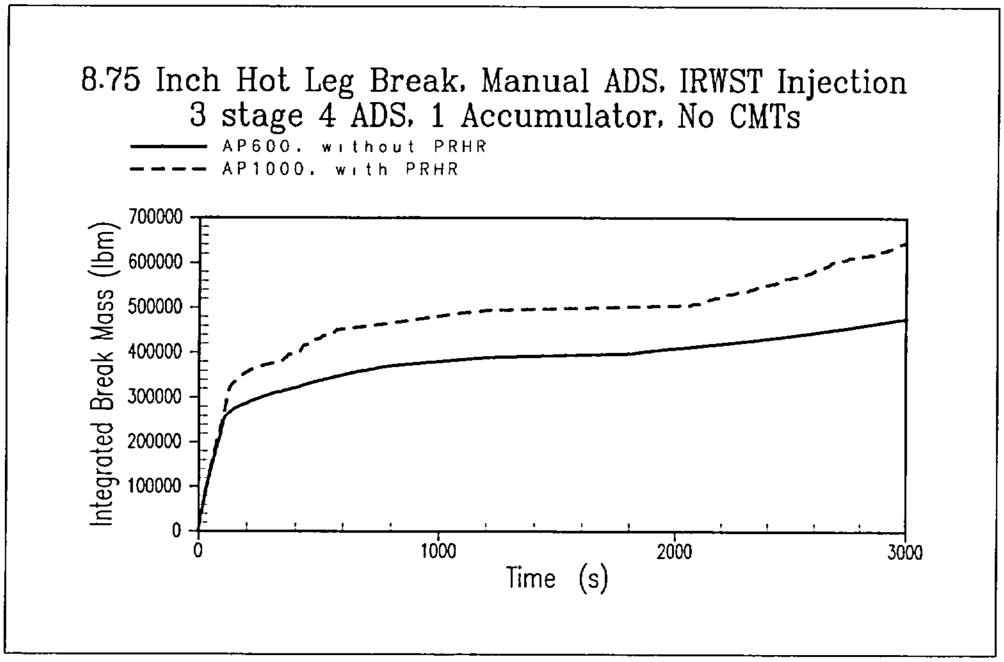


Figure A3.3-16

Integrated Break Flow for 8.75 Inch Break with Manual ADS at 20 Minutes

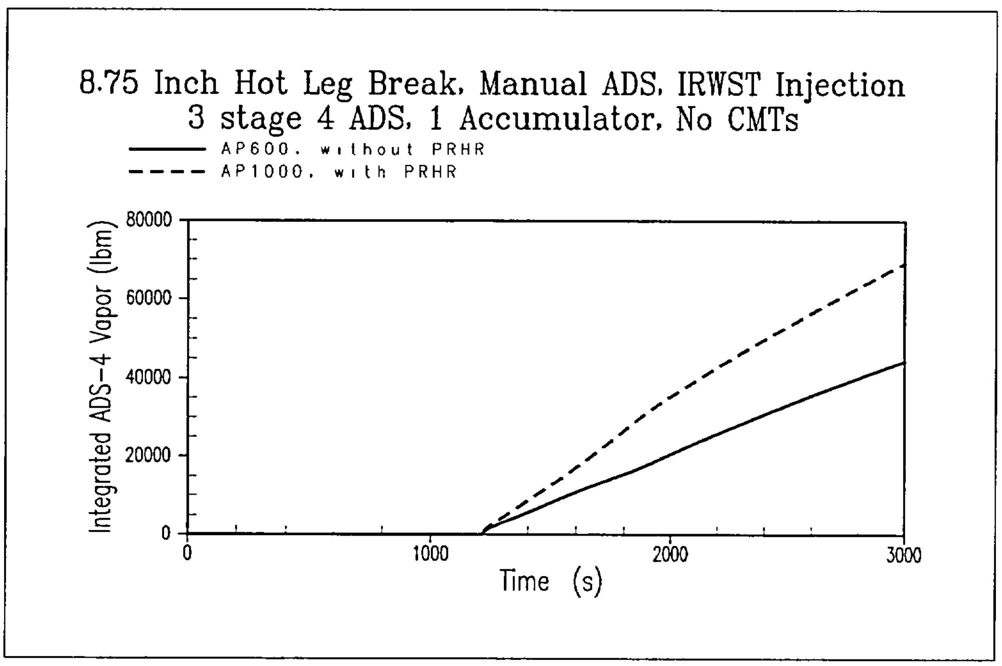


Figure A3.3-17

Integrated Stage 4 ADS Vapor for 8.75 Inch Break with Manual ADS at 20 Minutes

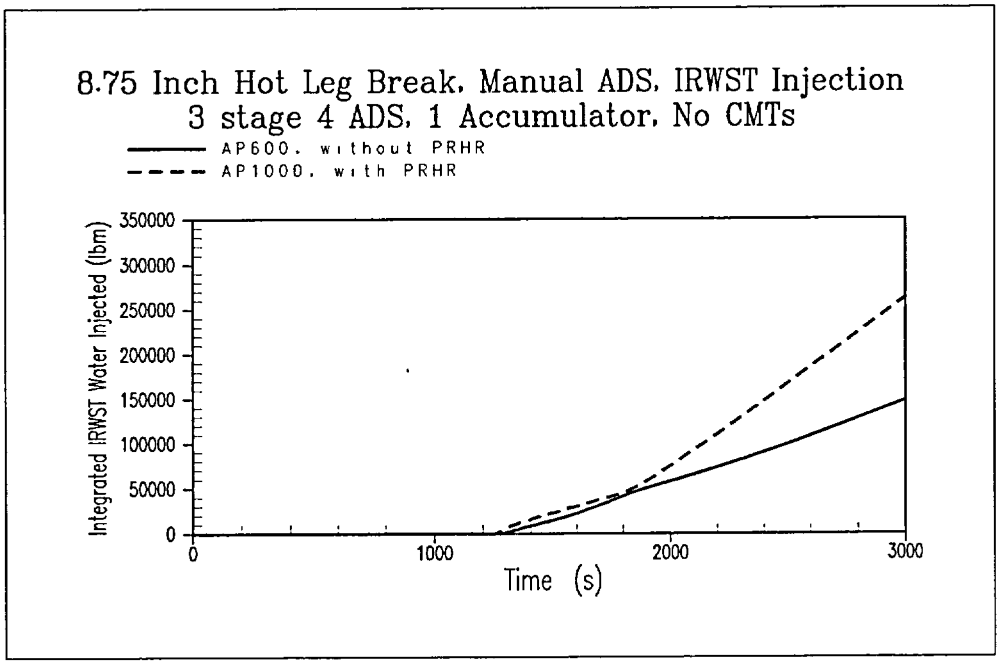


Figure A3.3-18

Integrated IRWST Water Injection for 8.75 Inch Break with Manual ADS at 20 Minutes

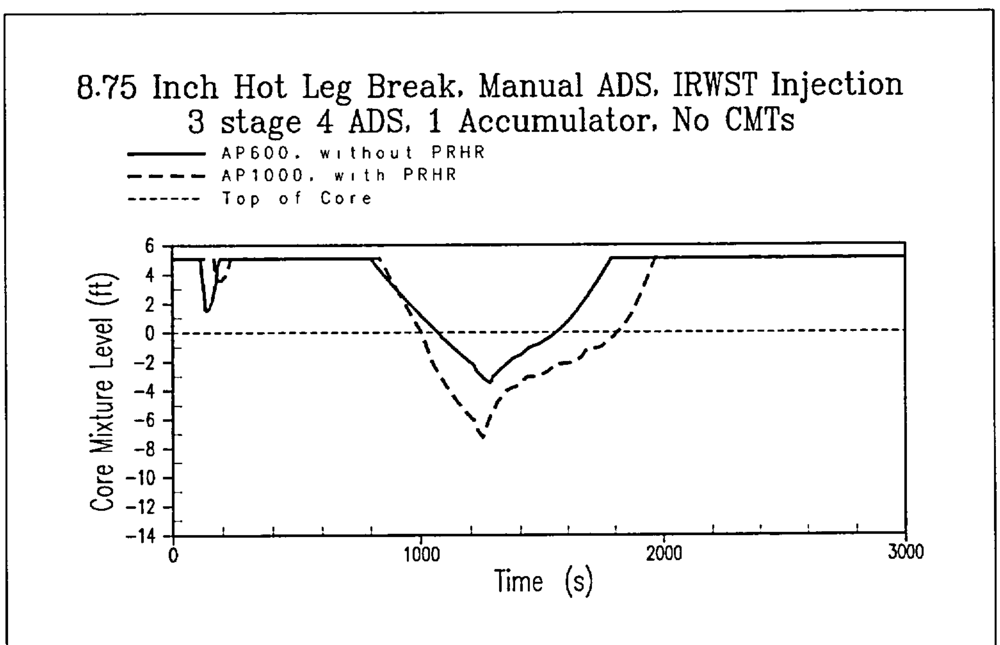


Figure A3.3-19

Core Mixture Level for 8.75 Inch Break with Manual ADS at 20 Minutes

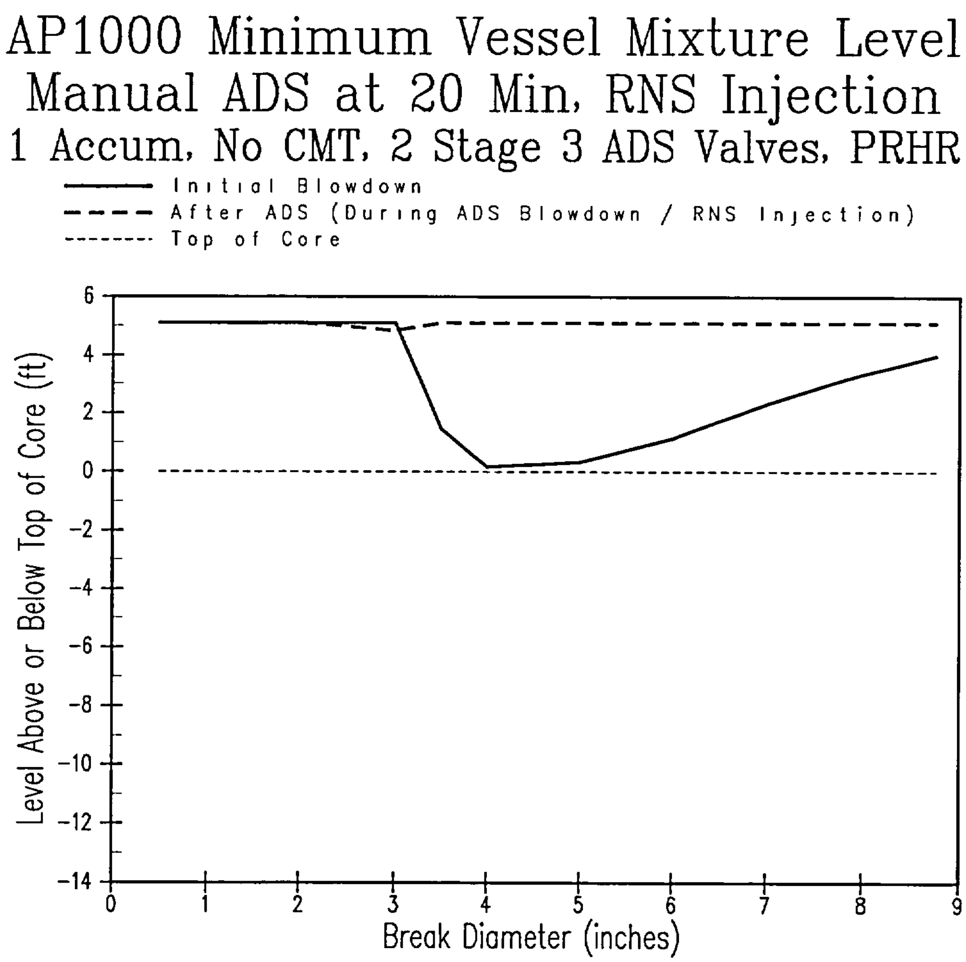


Figure A3.3-20

Minimum Core Mixture Level for Spectrum of Break Sizes (Manual ADS, RNS Injection)

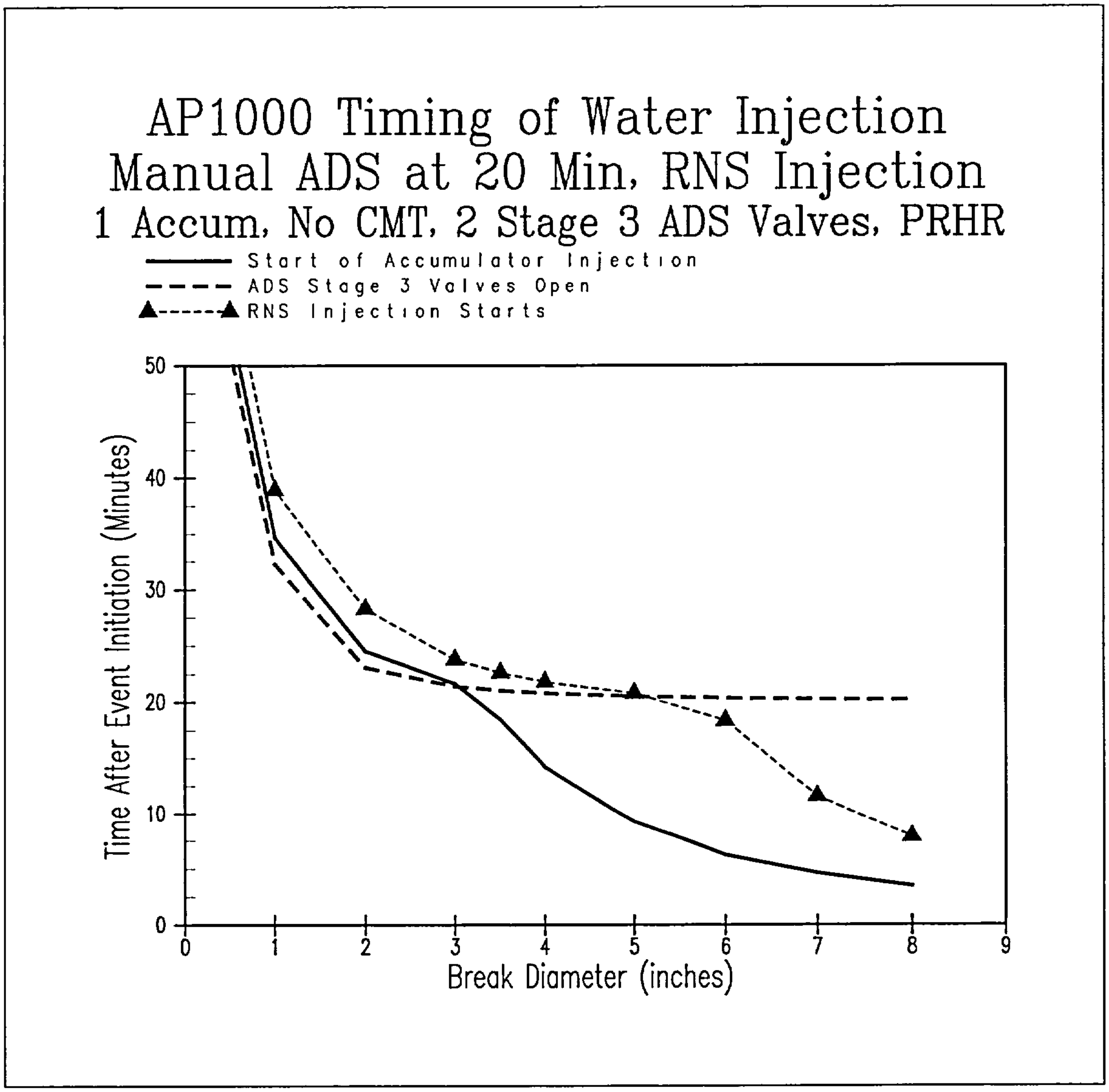


Figure A3.3-21

Timing of IRWST Injection for Spectrum of Break Sizes

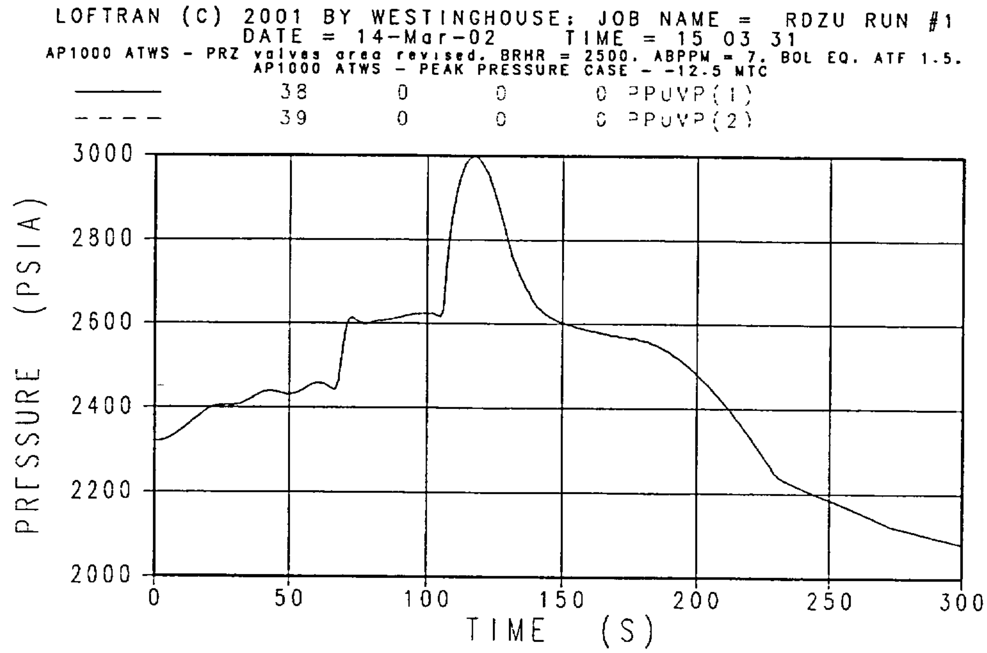
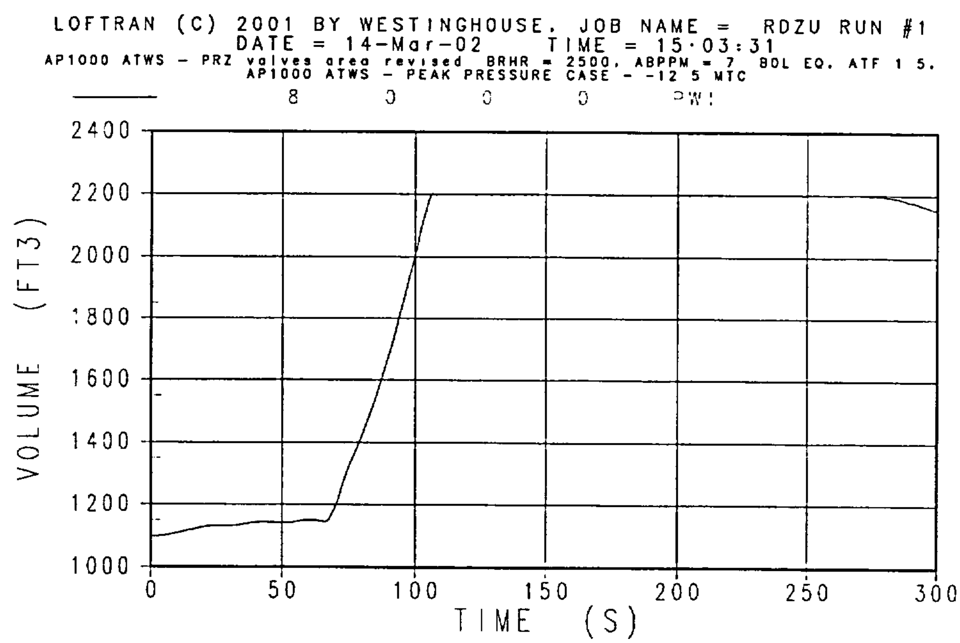


Figure A4.2.1-1

ATWS Eq. Cycle – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line.

Figure A4.2.1-2

ATWS Eq. Cycle – PRZ Volume

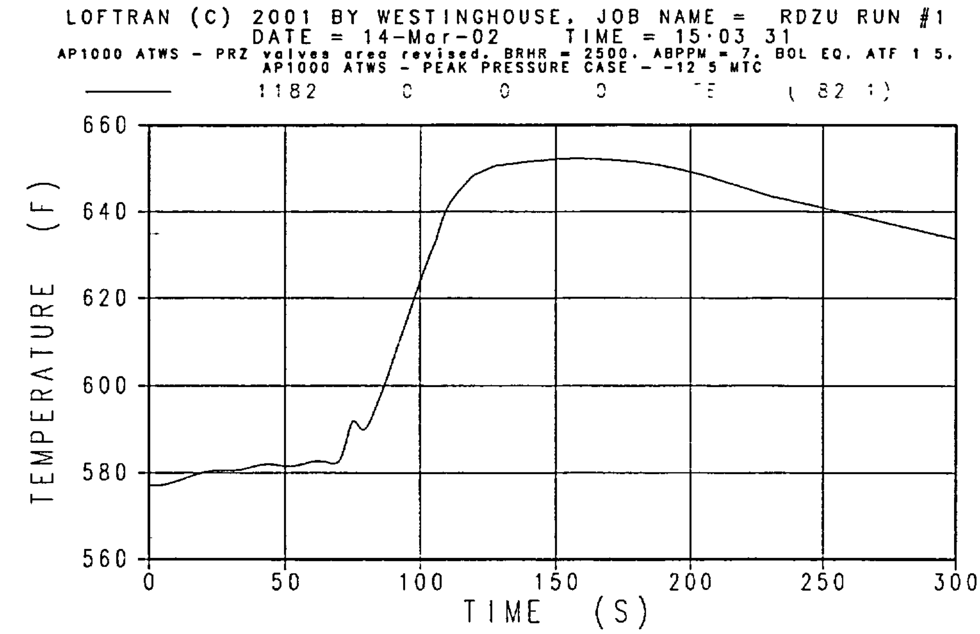


Figure A4.2.1-3

ATWS Eq. Cycle – Core Average Temperature

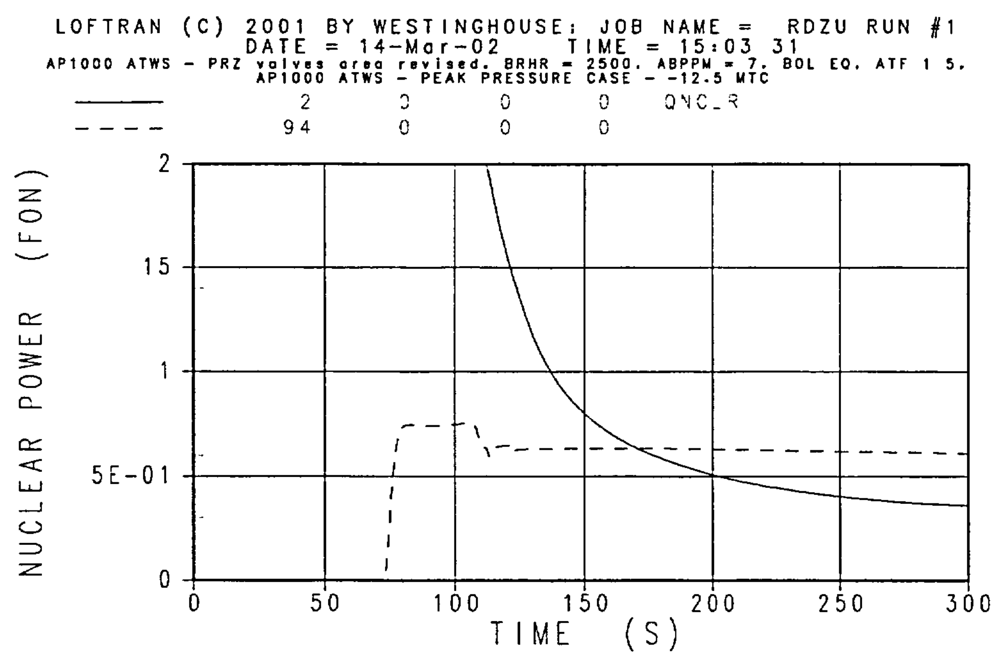


Figure A4.2.1-4

ATWS Eq. Cycle – Nuclear Power and PRHR Heat Flux

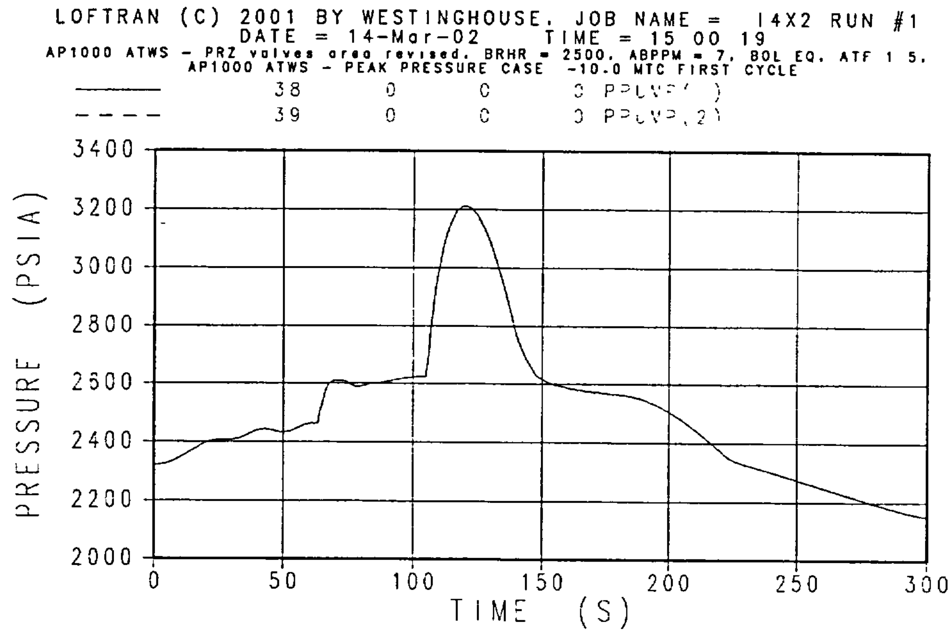
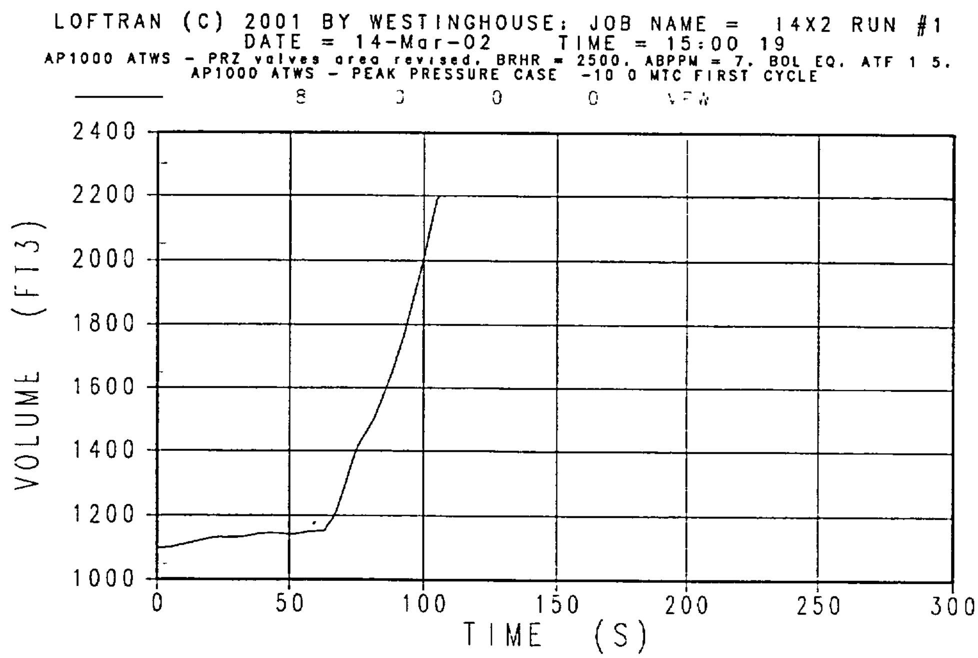


Figure A4.2.2-1

ATWS First Cycle – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line.

Figure A4.2.2-2

ATWS First. Cycle – PRZ Volume

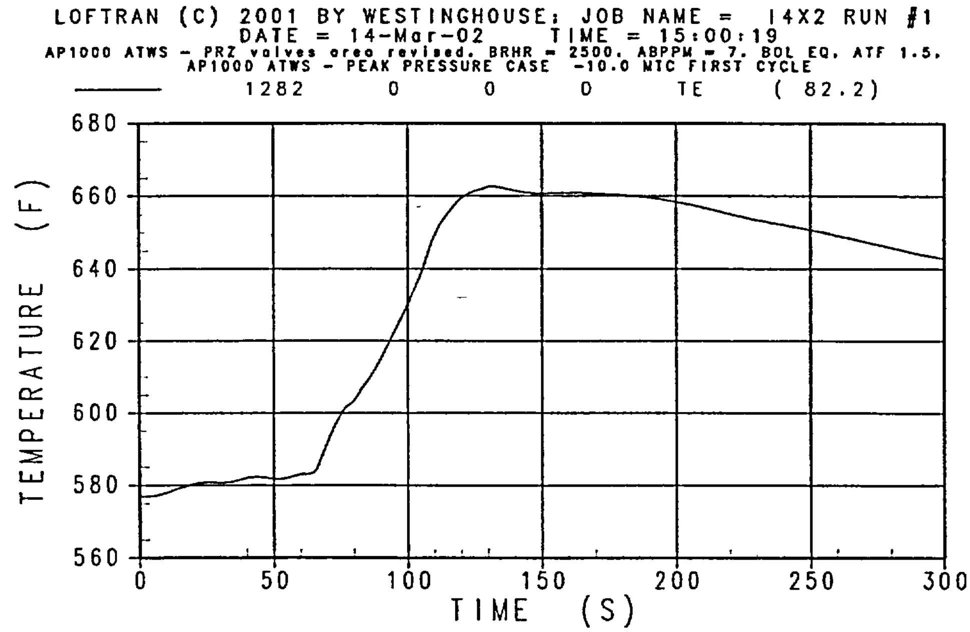


Figure A4.2.2-3

ATWS First. Cycle – Core Average Temperature

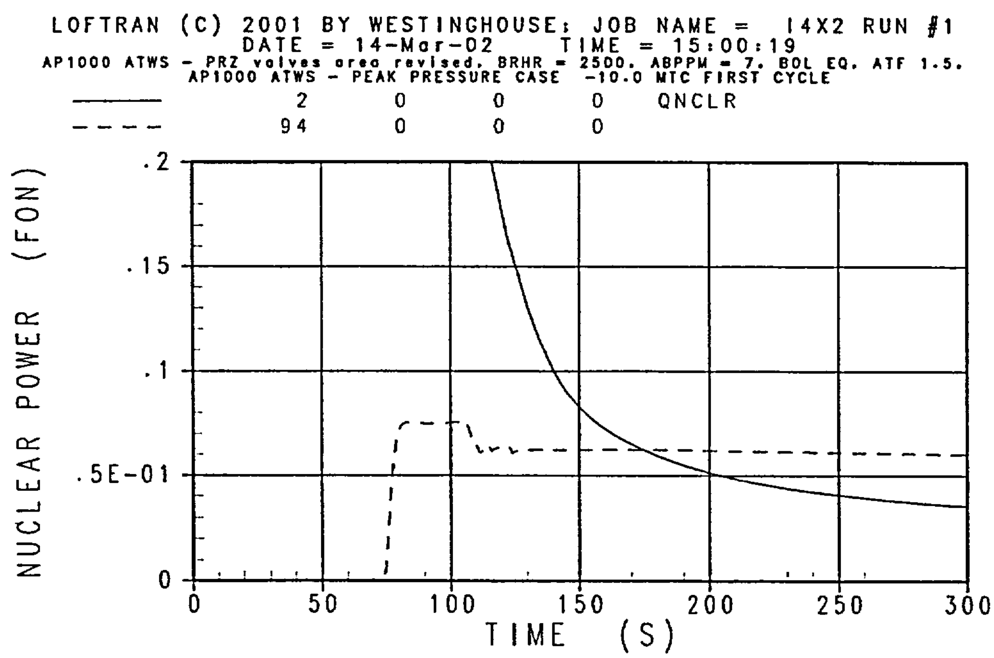


Figure A4.2.2-4

ATWS First. Cycle – Nuclear Power and PRHR Heat Flux

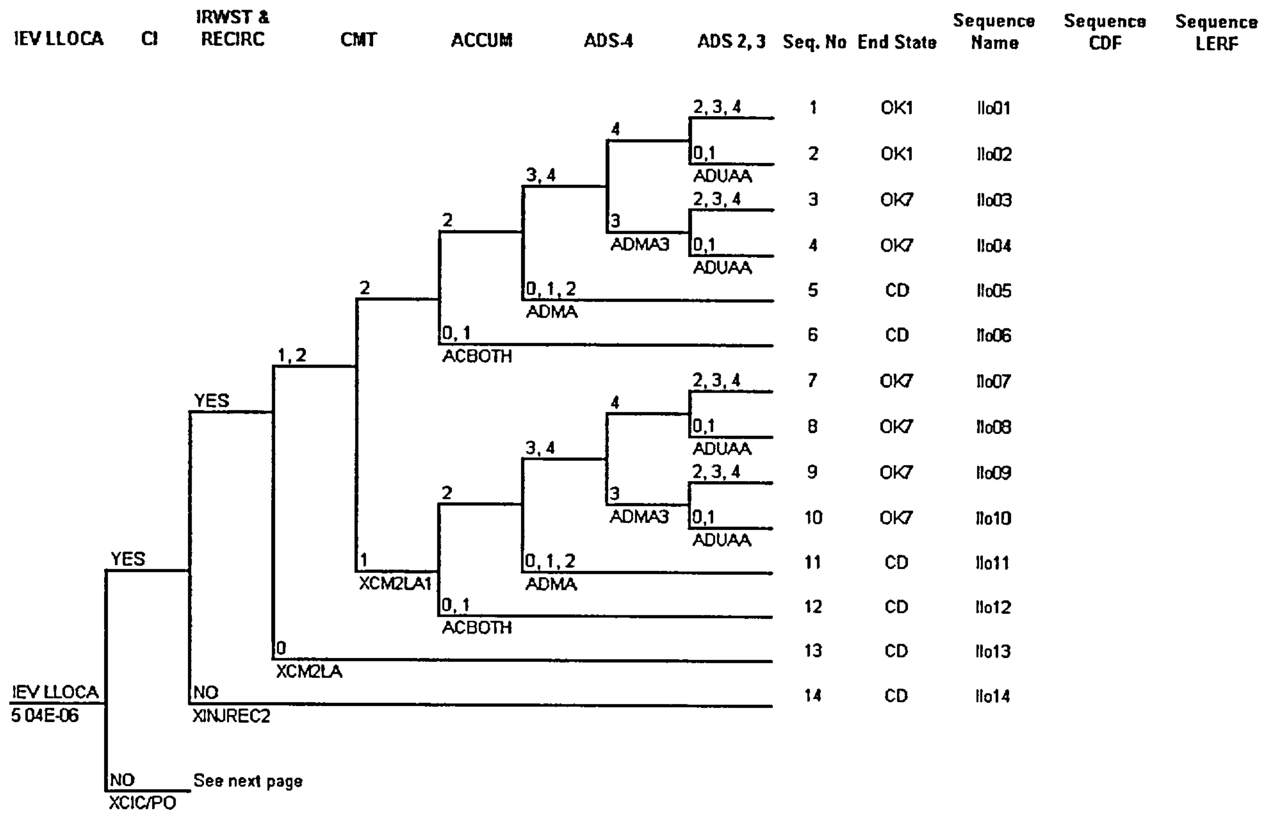


Figure A5.1-1 (Sheet 1 of 2)

Expanded LLOCA Initiating Event Tree

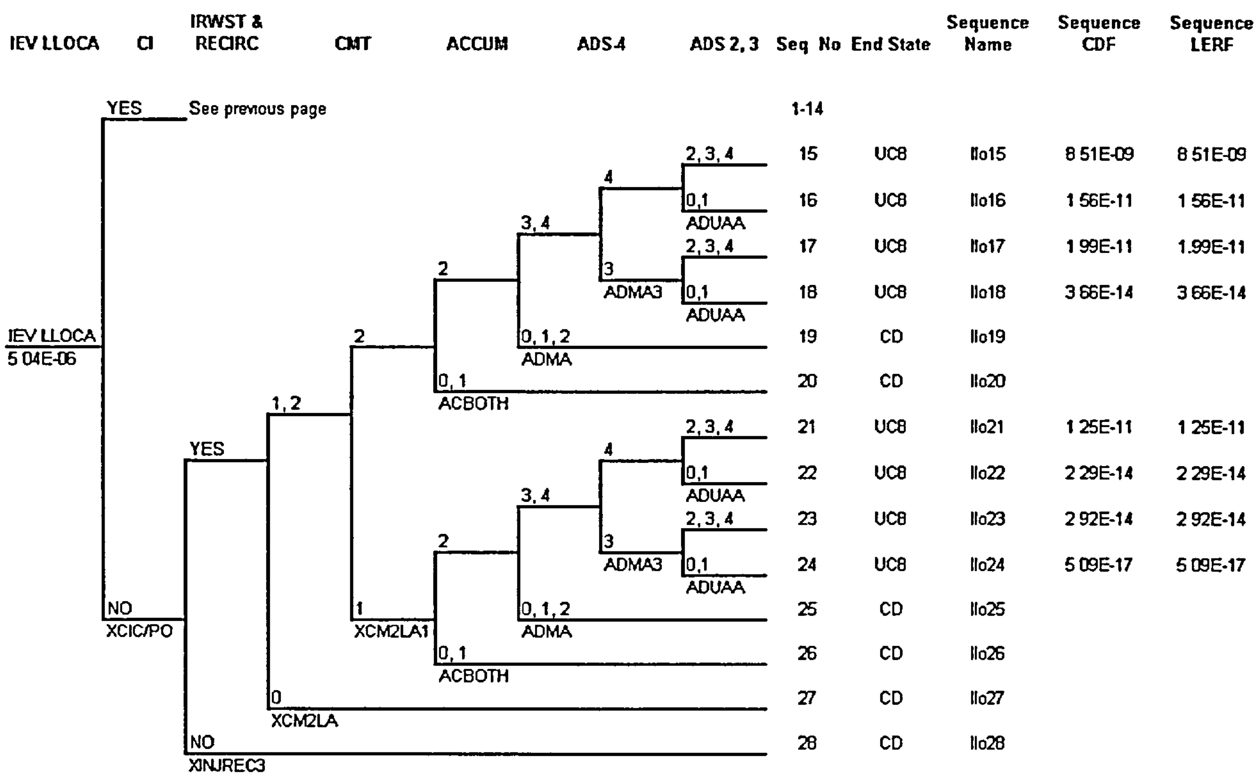
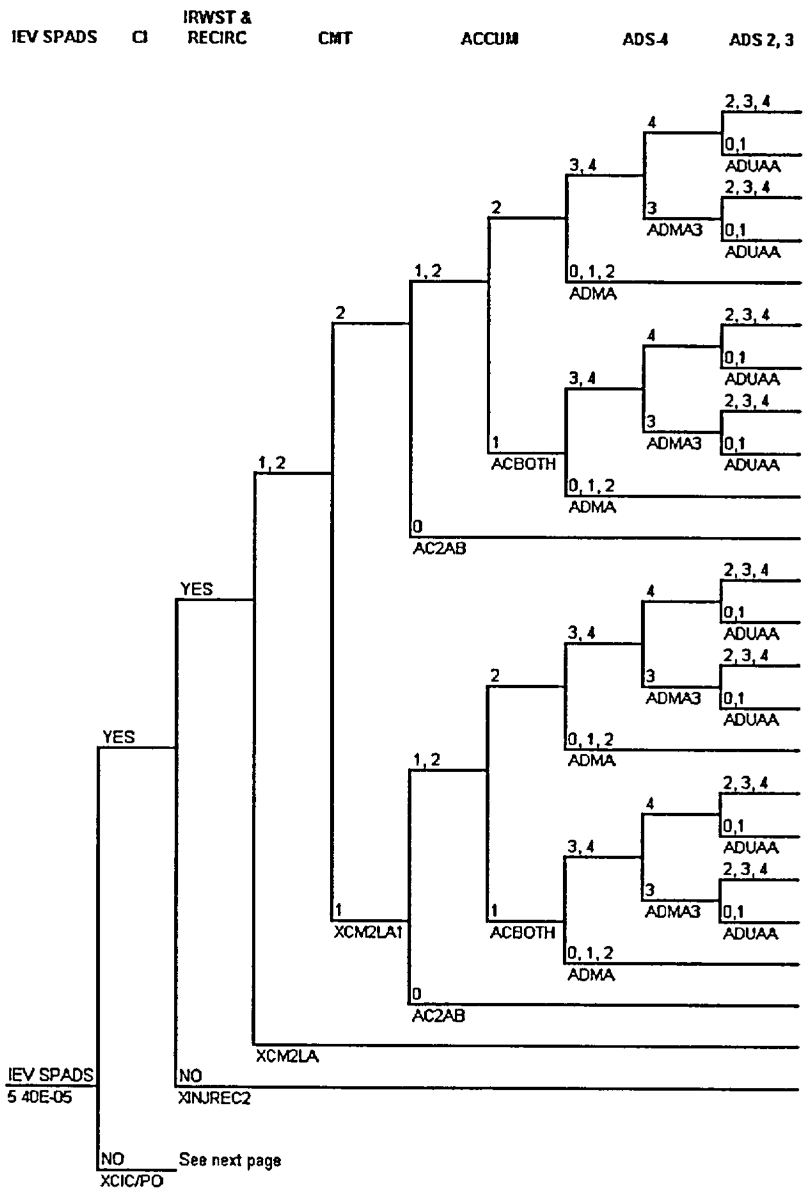


Figure A5.1-1 (Sheet 2 of 2)

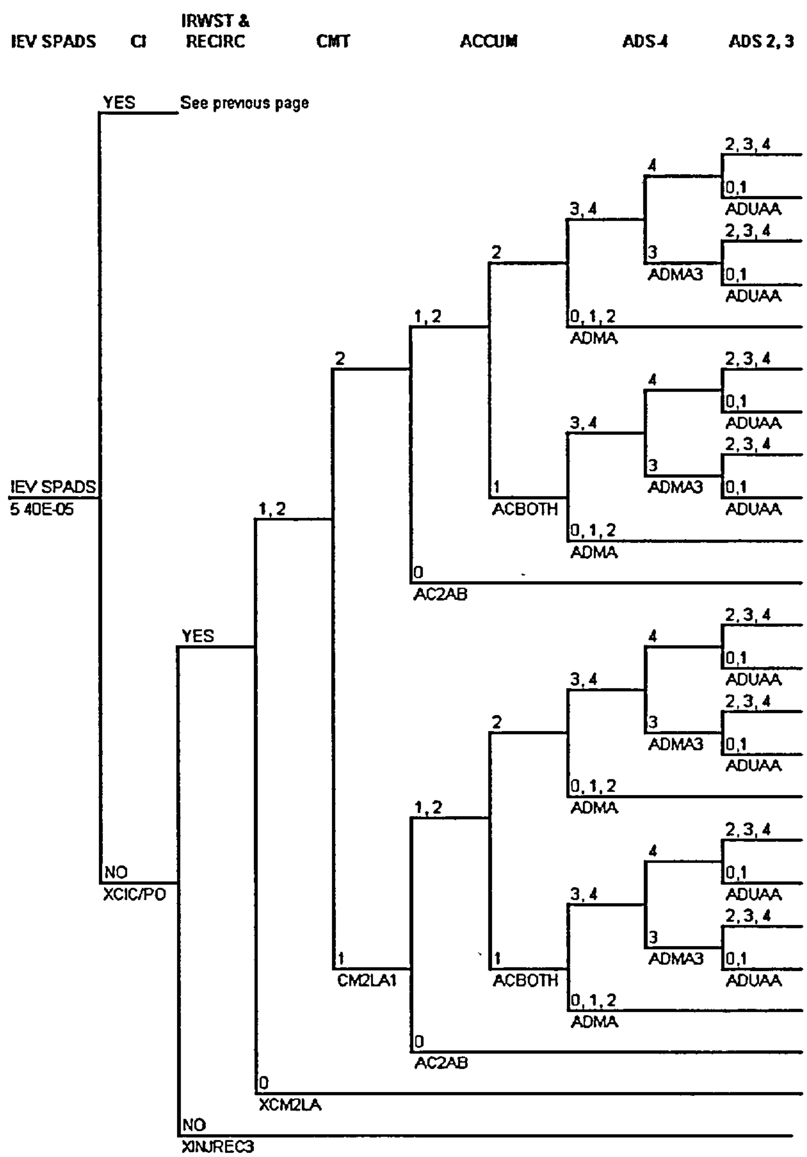
Expanded LLOCA Initiating Event Tree



Seq. No	End State	Sequence Name	Sequence CDF	Sequence LERF
1	OK1	sad01		
2	OK1	sad02		
3	OK7	sad03		
4	OK7	sad04		
5	CD	sad05		
6	UC4	sad06	4 58E-07	2 75E-08
7	UC4	sad07	8 40E-10	5 04E-11
8	UC4	sad08	1 07E-09	6 42E-11
9	UC4	sad09	1 97E-12	1 18E-13
10	CD	sad10		
11	CD	sad11		
12	OK7	sad12		
13	OK7	sad13		
14	OK7	sad14		
15	OK7	sad15		
16	CD	sad16		
17	UC4	sad17	6 76E-10	4 05E-11
18	UC4	sad18	1 23E-12	7 40E-14
19	UC4	sad19	1 57E-12	9 44E-14
20	UC4	sad20	2 78E-15	1 67E-16
21	CD	sad21		
22	CD	sad22		
23	CD	sad23		
24	CD	sad24		

Figure A5.1-2 (Sheet 1 of 2)

Expanded Spurious ADS Initiating Event Tree



Seq. No	End State	Sequence Name	Sequence CDF	Sequence LERF
1-24				
25	UC8	sad25	9 12E-08	5 47E-09
26	UC8	sad26	1 67E-10	1 00E-11
27	UC8	sad27	2 13E-10	1.28E-11
28	UC8	sad28	3 91E-13	2 35E-14
29	CD	sad29		
30	UC4	sad30	7 77E-10	4 66E-11
31	UC4	sad31	1 42E-12	8 51E-14
32	UC4	sad32	1 81E-12	1 09E-13
33	UC4	sad33	3 20E-15	1 92E-16
34	CD	sad34		
35	CD	sad35		
36	UC8	sad36	1 34E-10	8 04E-12
37	UC8	sad37	2 45E-13	1 47E-14
38	UC8	sad38	3 12E-13	1 87E-14
39	UC8	sad39	5 15E-16	3 09E-17
40	CD	sad40		
41	UC4	sad41	1 14E-12	6 85E-14
42	UC4	sad42	2 02E-15	1 21E-16
43	UC4	sad43	2 62E-15	1 57E-16
44	UC4	sad44	0 00E+00	0 00E+00
45	CD	sad45		
46	CD	sad46		
47	CD	sad47		
48	CD	sad48		

Figure A5.1-2 (Sheet 2 of 2)

Expanded Spurious ADS Initiating Event Tree

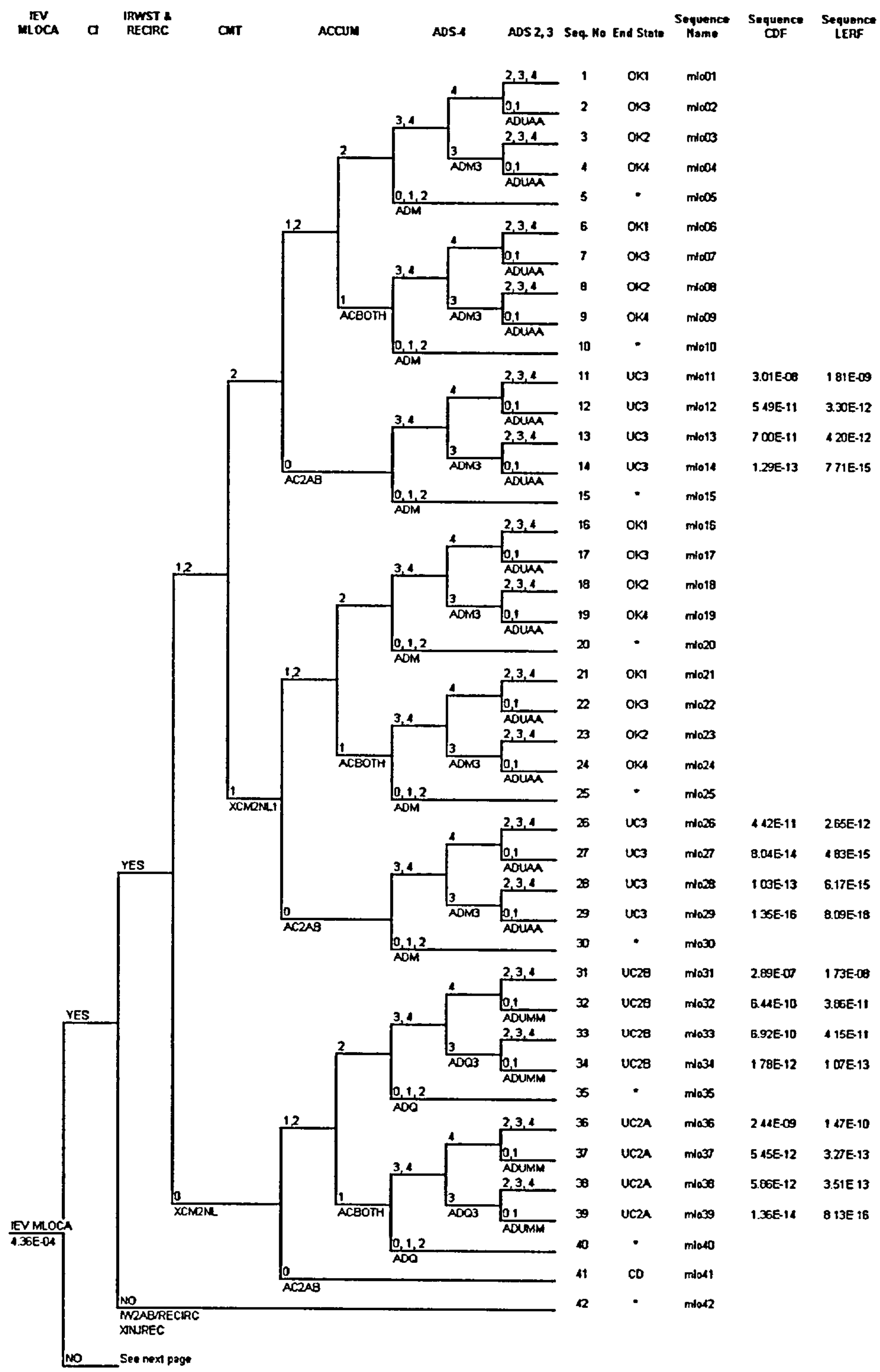
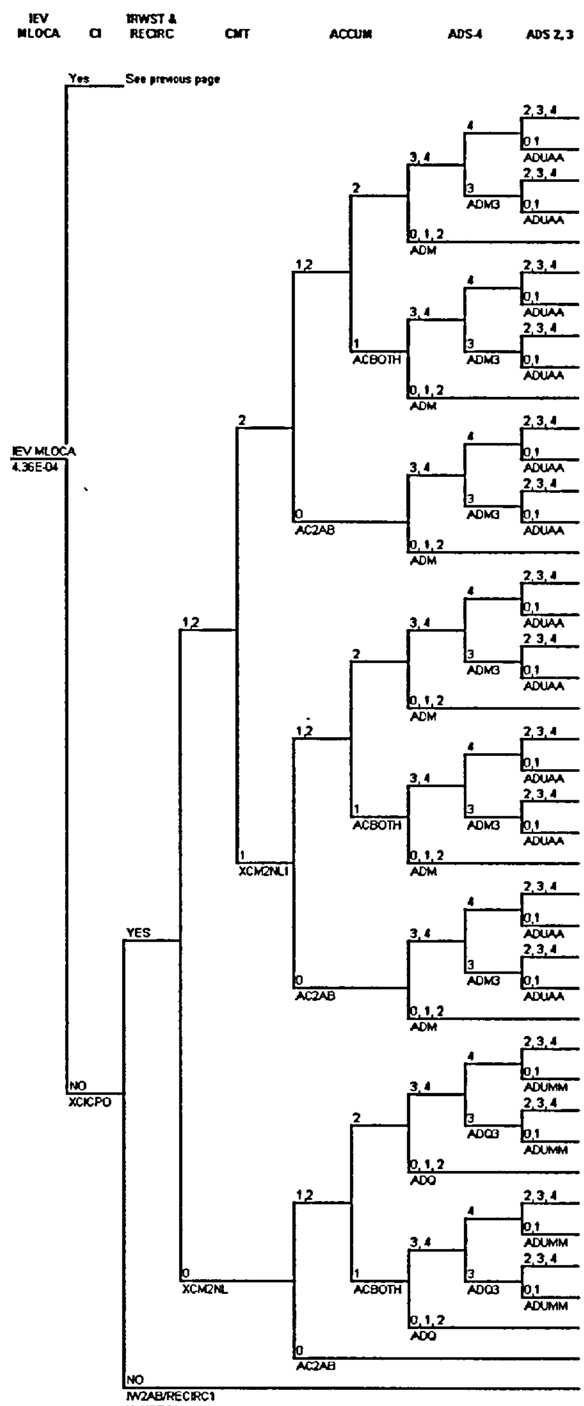


Figure A5.1-3 (Sheet 1 of 2)

Expanded MLOCA Initiating Event Tree

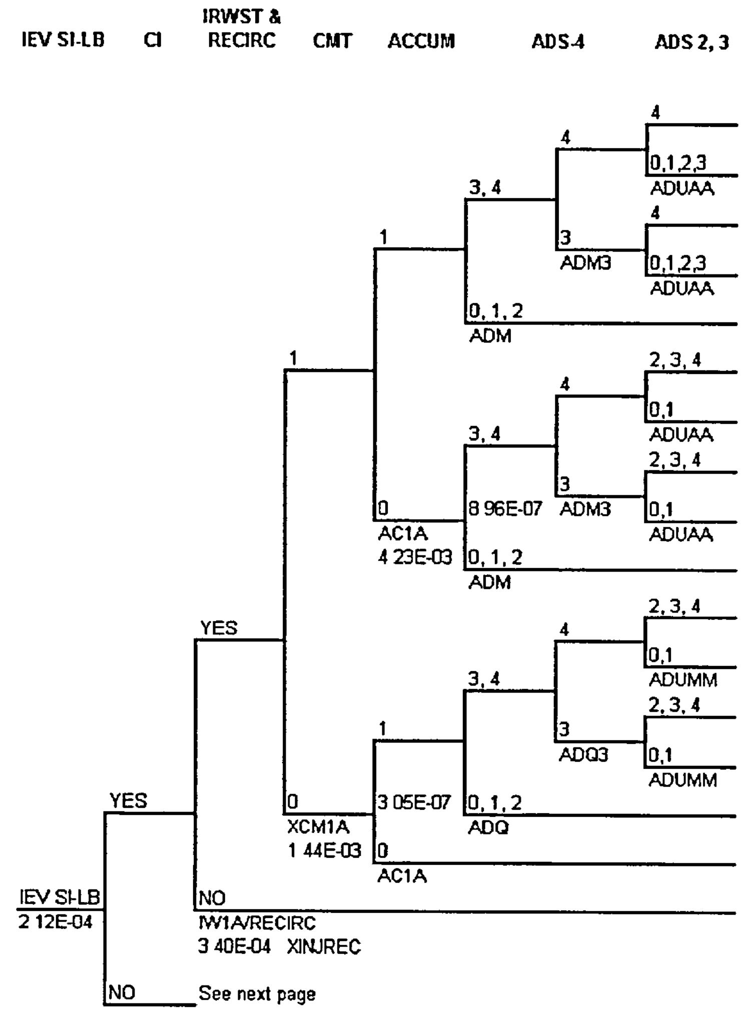


Seq. No	End State	Sequence Name	Sequence CDF	Sequence LERF
1-42				
43	OK1	mlo43		
44	OK3	mlo44		
45	OK2	mlo45		
46	OK4	mlo46		
47	*	mlo47		
48	OK1	mlo48		
49	OK3	mlo49		
50	OK2	mlo50		
51	OK4	mlo51		
52	*	mlo52		
53	UC3	mlo53	5.08E-11	5.08E 11
54	UC3	mlo54	9.24E-14	9 24E 14
55	UC3	mlo55	1 18E-13	1 18E 13
56	UC3	mlo56	1.51E-16	1 51E 16
57	*	mlo57		
58	OK1	mlo58		
59	OK3	mlo59		
60	OK2	mlo60		
61	OK4	mlo61		
62	*	mlo62		
63	OK1	mlo63		
64	OK3	mlo64		
65	OK2	mlo65		
66	OK4	mlo66		
67	*	mlo67		
68	UC3	mlo68	7 46E 14	7 46E-14
69	UC3	mlo69	9.54E 17	9.54E-17
70	UC3	mlo70	1.25E 16	1 25E-16
71	UC3	mlo71	0.00E+00	0.00E+00
72	*	mlo72		
73	UC2B	mlo73	4.88E 10	4 88E 10
74	UC2B	mlo74	1.09E-12	1.09E-12
75	UC2B	mlo75	1 17E-12	1 17E-12
76	UC2B	mlo76	2.30E 15	2.30E-15
77	*	mlo77		
78	UC2A	mlo78	4 13E 12	4 13E-12
79	UC2A	mlo79	8.10E 15	8 10E-15
80	UC2A	mlo80	9.01E-15	9.01E-15
81	UC2A	mlo81	0.00E+00	0.00E+00
82	*	mlo82		
83	CD	mlo83		
84	*	mlo84		

Notes: * These paths go to CD if full and partial ADS fails
If partial ADS is successful and RNS is successful they go to OK; otherwise they go to CD
The success paths that involve RNS are not further analyzed in this report as explained in Section 2.3.

Figure A5.1-3 (Sheet 2 of 2)

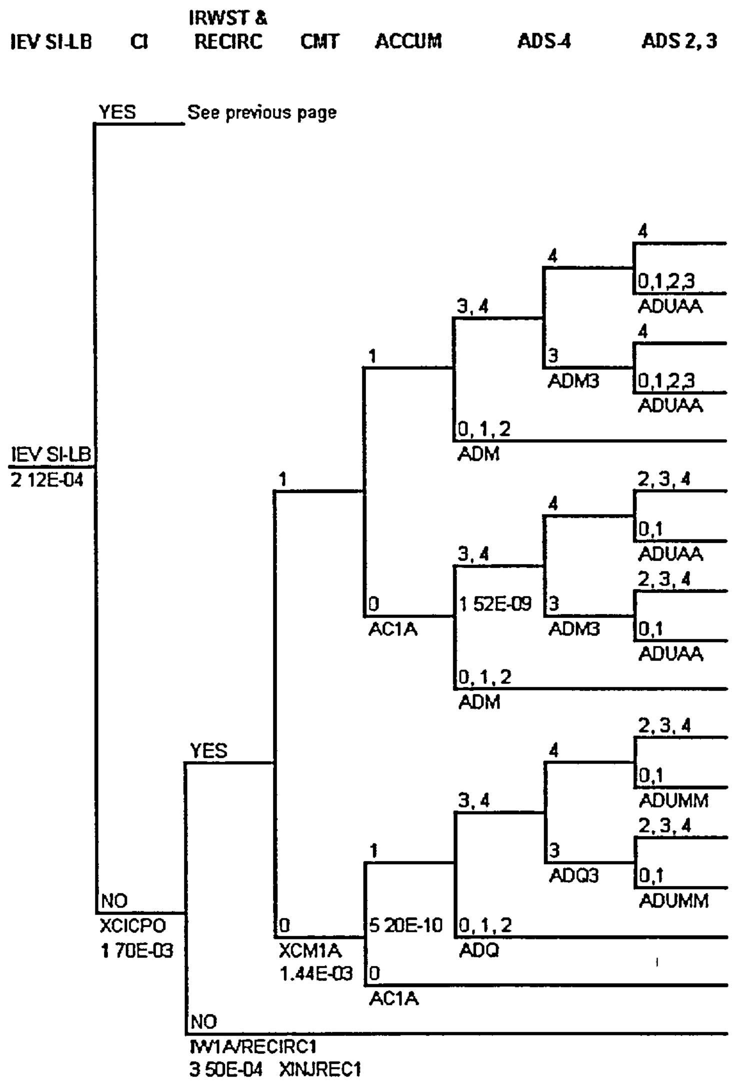
Expanded MLOCA Initiating Event Tree



Seq. No	End State	Sequence Name	Sequence CDF	Sequence LERF
1	OK1	silb01		
2	OK3	silb02		
3	OK2	silb03		
4	OK4	silb04		
5	CD	silb05		
6	UC5	silb06	8 96E-07	5 37E-08
7	UC5	silb07	1 64E-09	9.83E-11
8	UC5	silb08	2 09E-09	1 25E-10
9	UC5	silb09	3 85E-12	2.31E-13
10	CD	silb10		
11	UC1	silb11	3 05E-07	1 83E-08
12	UC1	silb12	6 15E-10	3 69E-11
13	UC1	silb13	7 21E-10	4 32E-11
14	UC1	silb14	1 57E-12	9 45E-14
15	CD	silb15		
16	CD	silb16		
17	CD	silb17		

Figure A5.1-4 (Page 1 of 2)

Expanded SI-LB Initiating Event Tree



Seq. No	End State	Sequence Name	Sequence CDF	Sequence LRF
1-17				
18	OK5A	silb18		
19	OK5B	silb19		
20	OK6	silb20		
21	OK6	silb21		
22	CD	silb22		
23	UC5	silb23	1 52E-09	1.52E-09
24	UC5	silb24	2 77E-12	2.77E-12
25	UC5	silb25	3 53E-12	3 53E-12
26	UC5	silb26	6 28E-15	6 28E-15
27	CD	silb27		
28	UC1	silb28	5 16E-10	5 16E-10
29	UC1	silb29	1 04E-12	1 04E-12
30	UC1	silb30	1.22E-12	1.22E-12
31	UC1	silb31	2.26E-15	2.26E-15
32	CD	silb32		
33	CD	silb33		
34	CD	silb34		

Figure A5.1-4 (Page 2 of 2)

Expanded SI-LB Initiating Event Tree

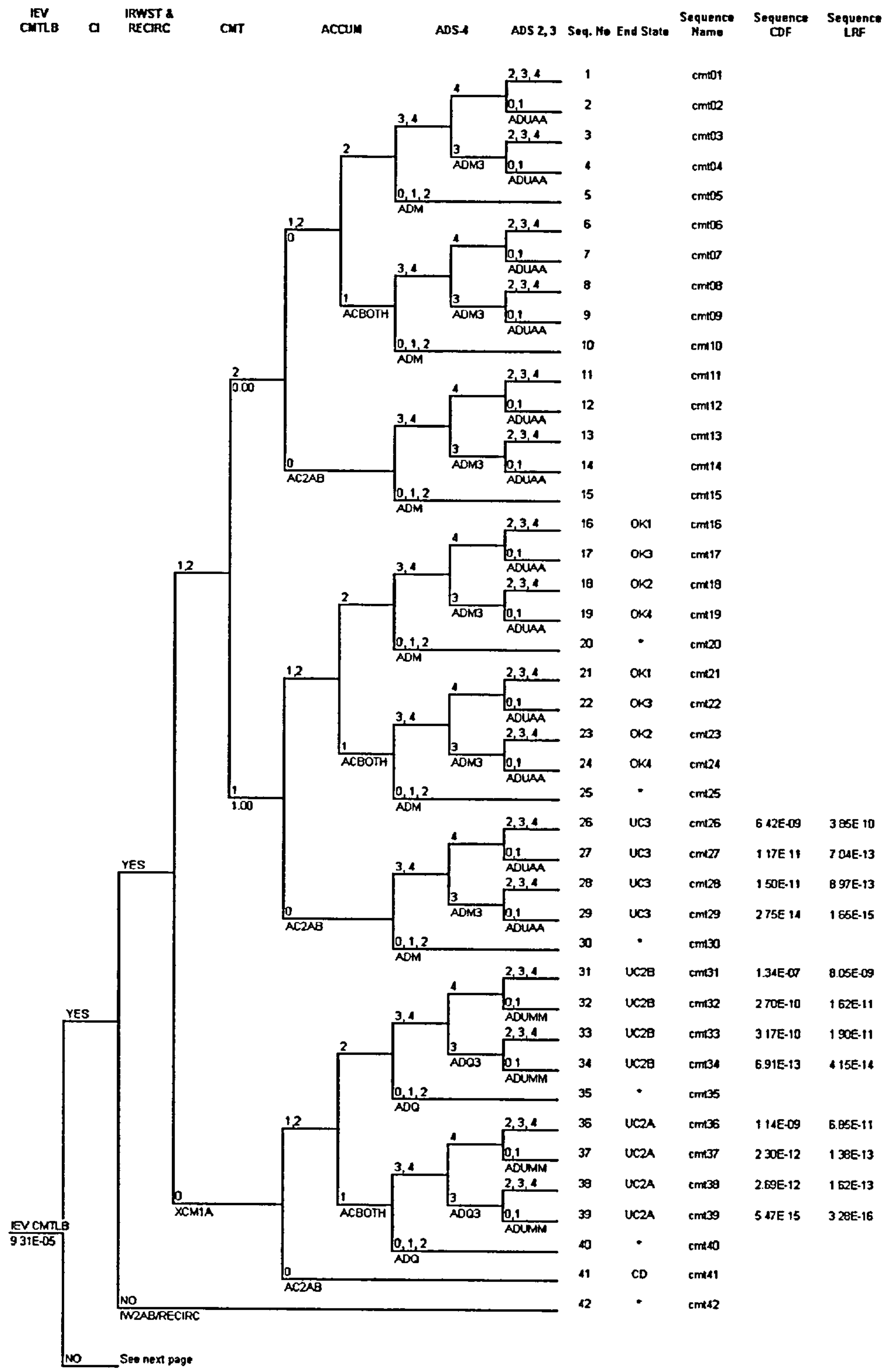
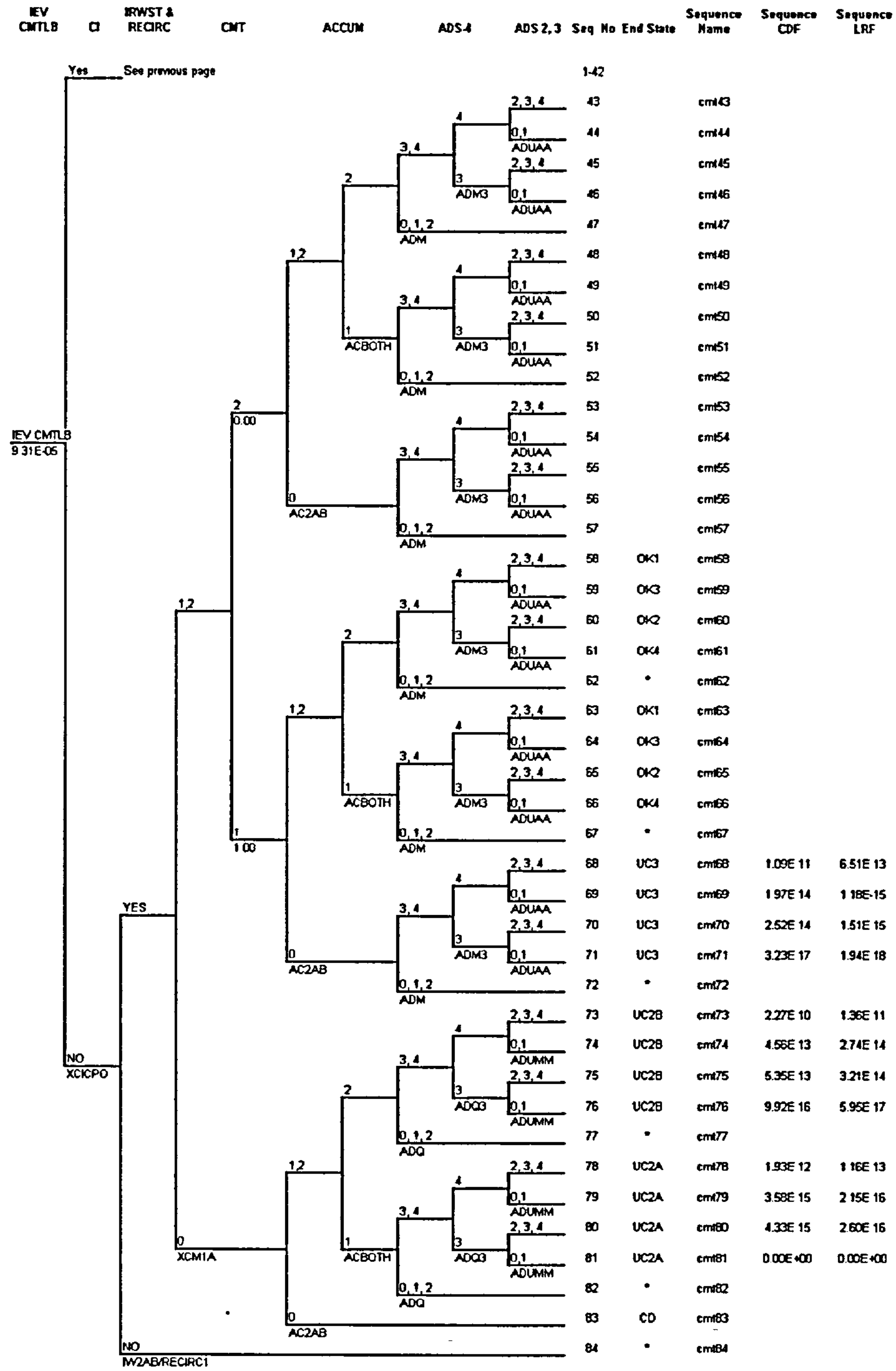


Figure A5.1-5 (Sheet 1 of 2)

Expanded CMTLB Initiating Event Tree



Notes *

These paths go to CD if full and partial ADS fails
If partial ADS is successful and RNS is successful, they go to OK, otherwise they go to CD
The success paths that involve RNS are not further analyzed in this report as explained in Section 2.3

Figure A5.1-5 (Sheet 2 of 2)

Expanded CMTLB Initiating Event Tree

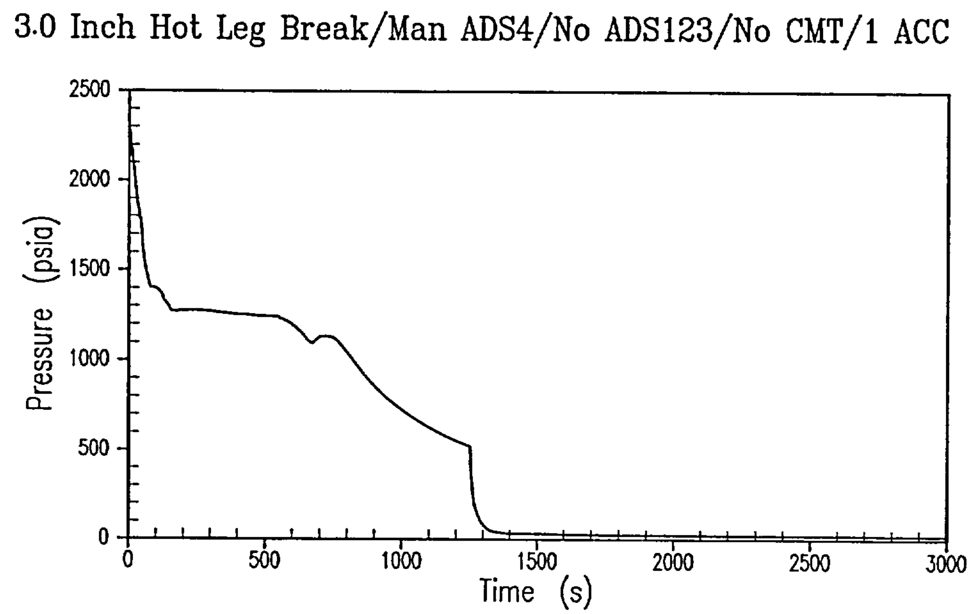


Figure A5.2-1

Case A – Pressurizer Pressure

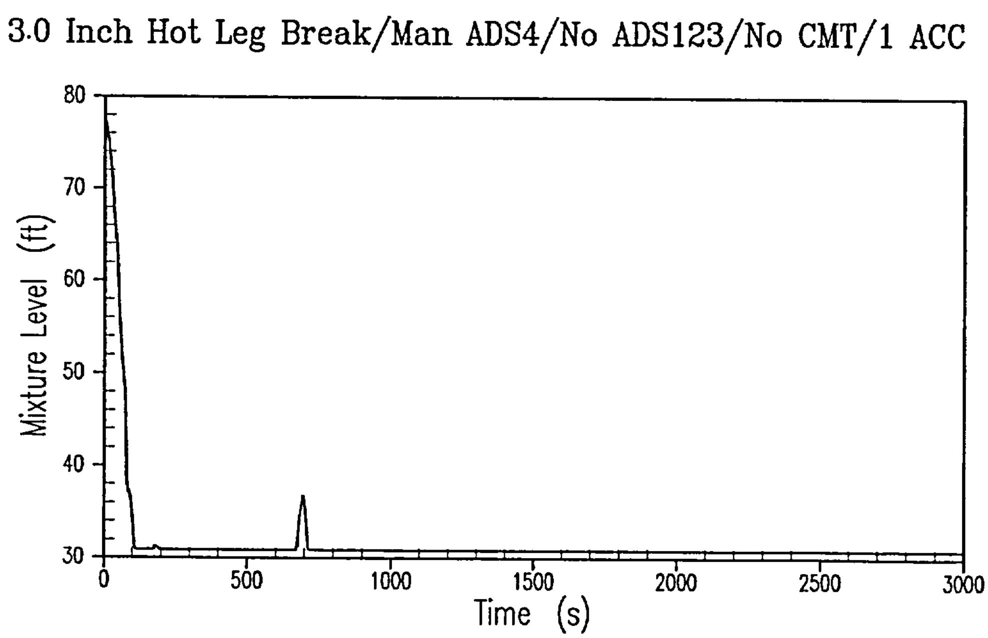


Figure A5.2-2

Case A – Pressurizer Level

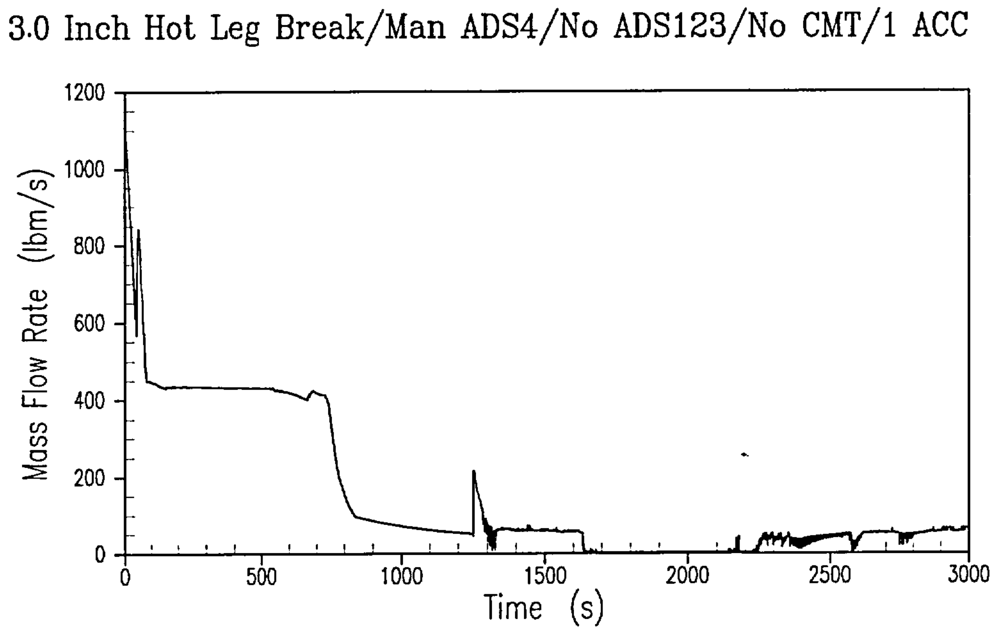


Figure A5.2-3

Case A – Break Liquid Flow

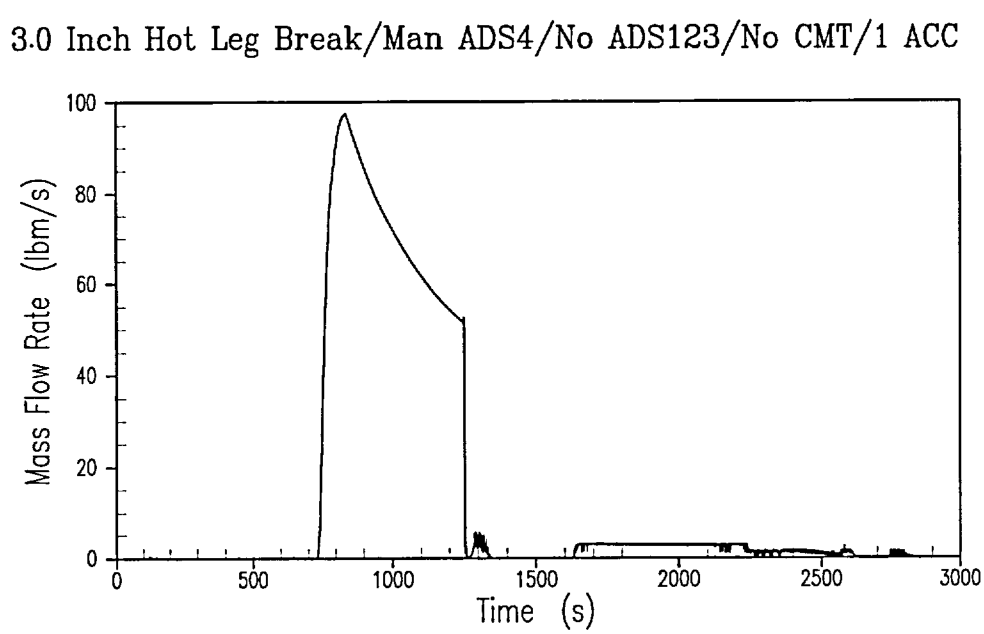


Figure A5.2-4

Case A – Break Vapor Flow

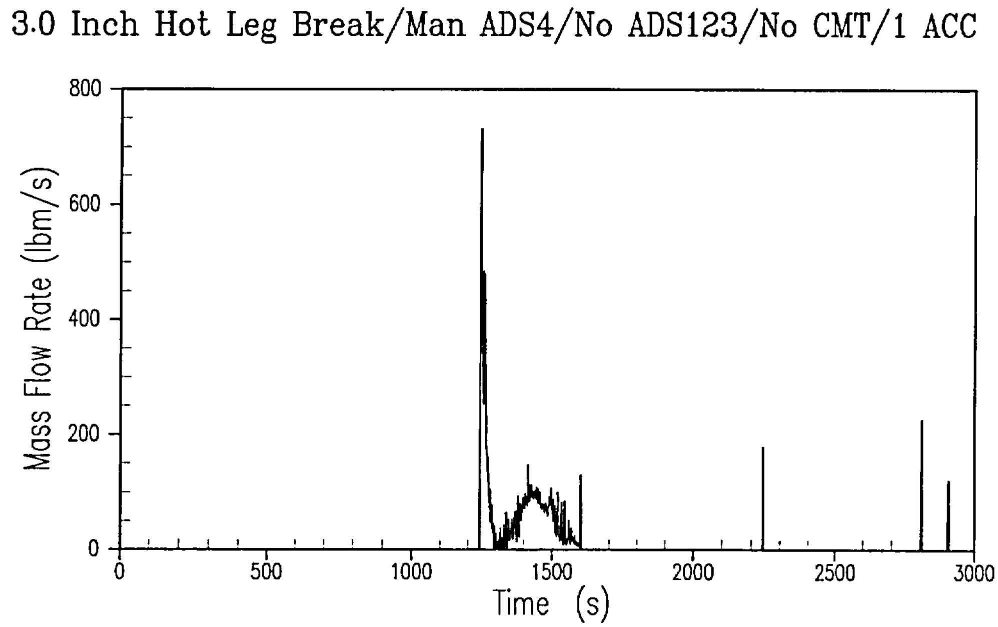


Figure A5.2-5

Case A – 4th Stage ADS Liquid Flow Through All Open Paths

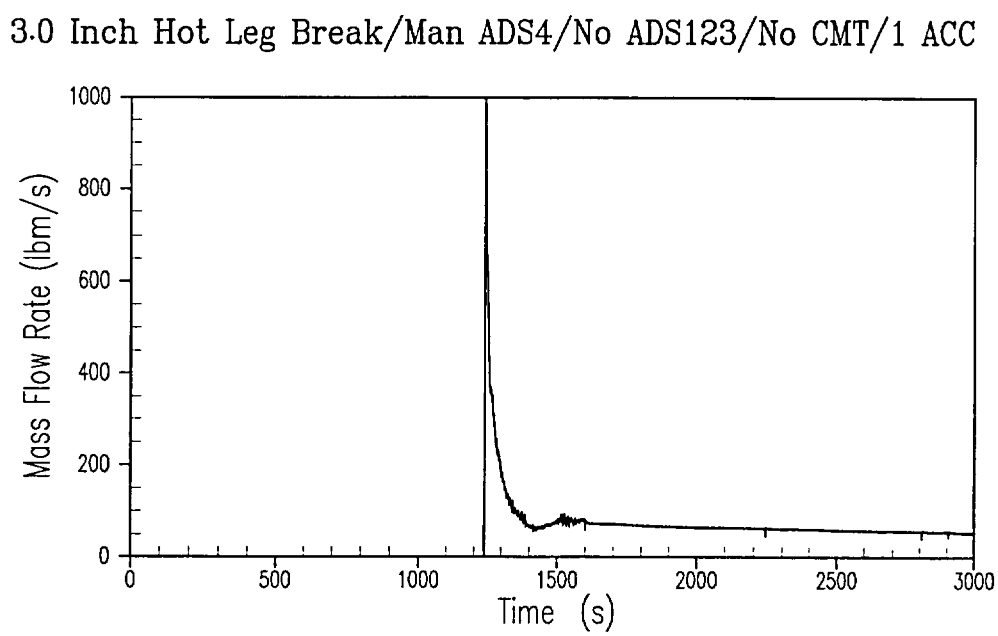


Figure A5.2-6

Case A – 4th Stage ADS Vapor Flow

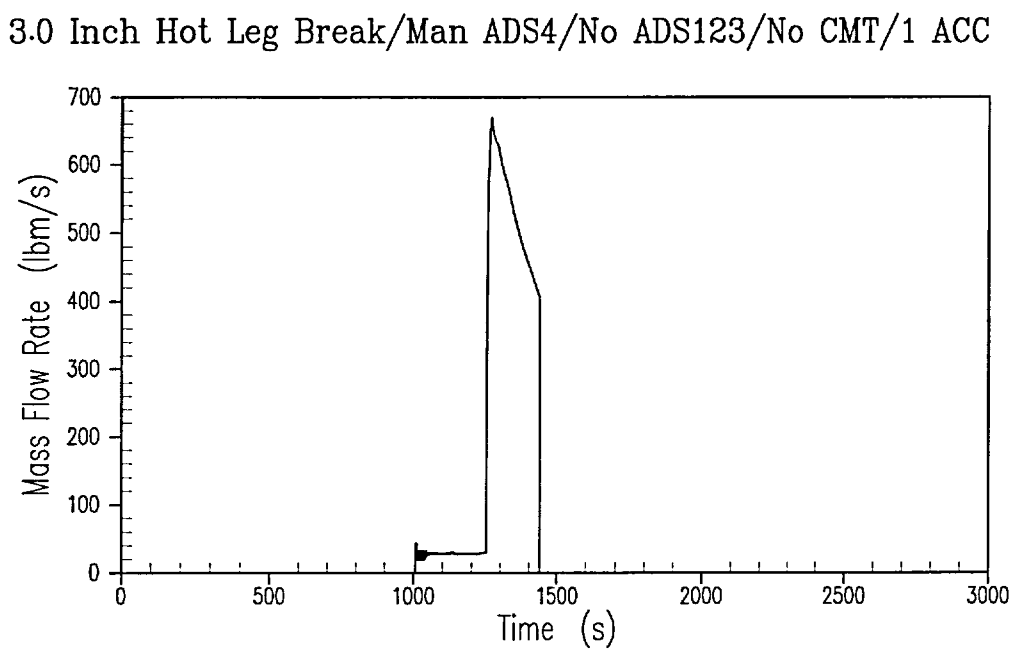


Figure A5.2-7

Case A – Accumulator Injection Flow

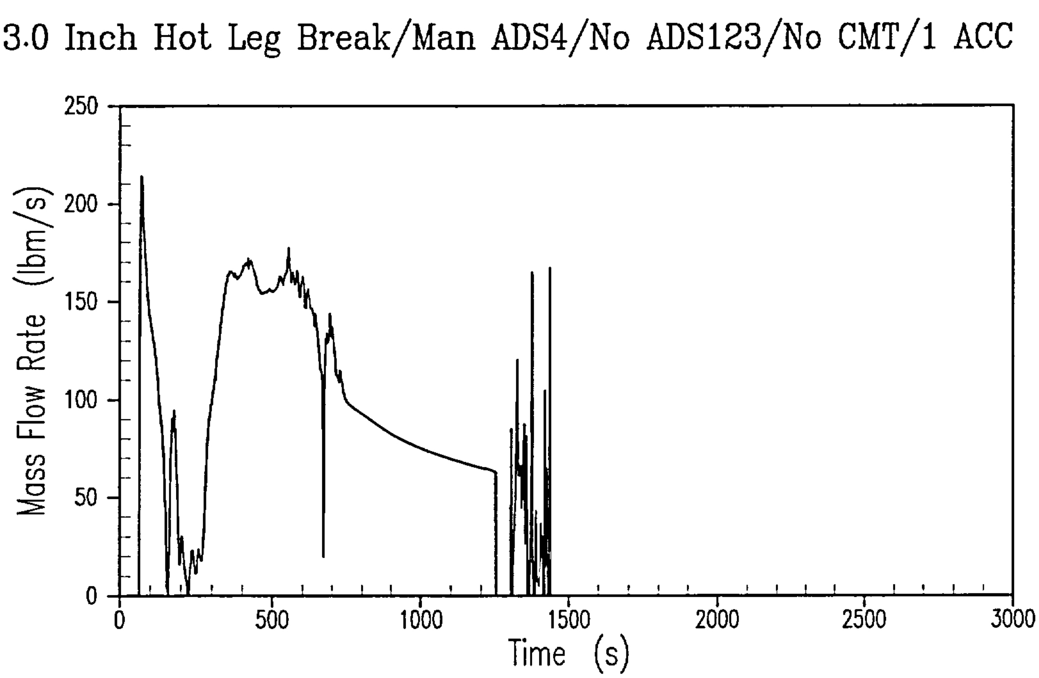


Figure A5.2-8

Case A – PRHR Discharge Flow

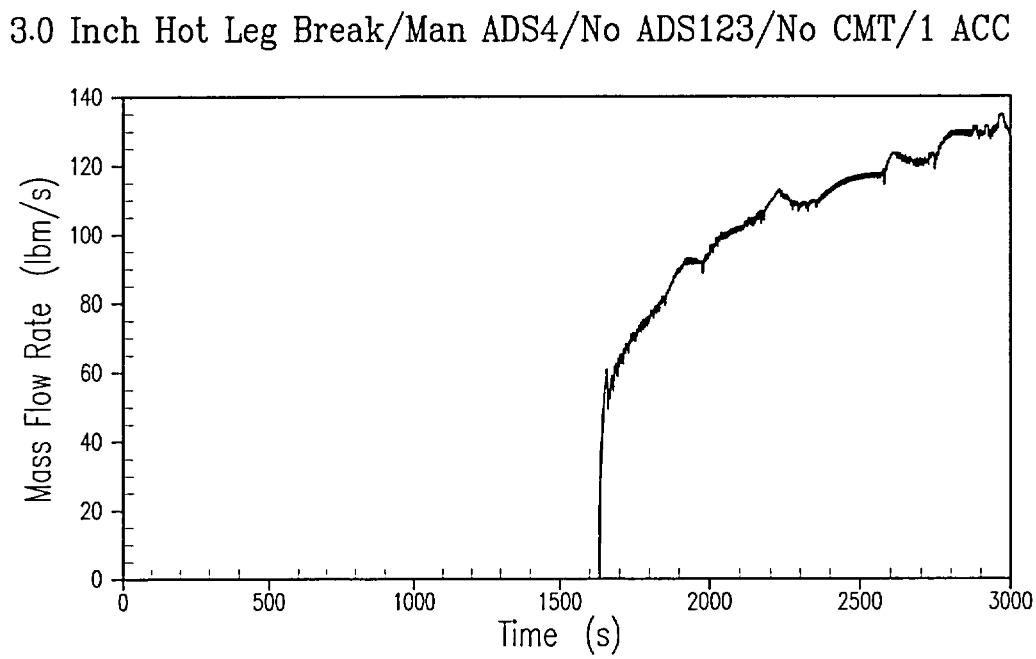


Figure A5.2-9

Case A – IRWST Injection Flow

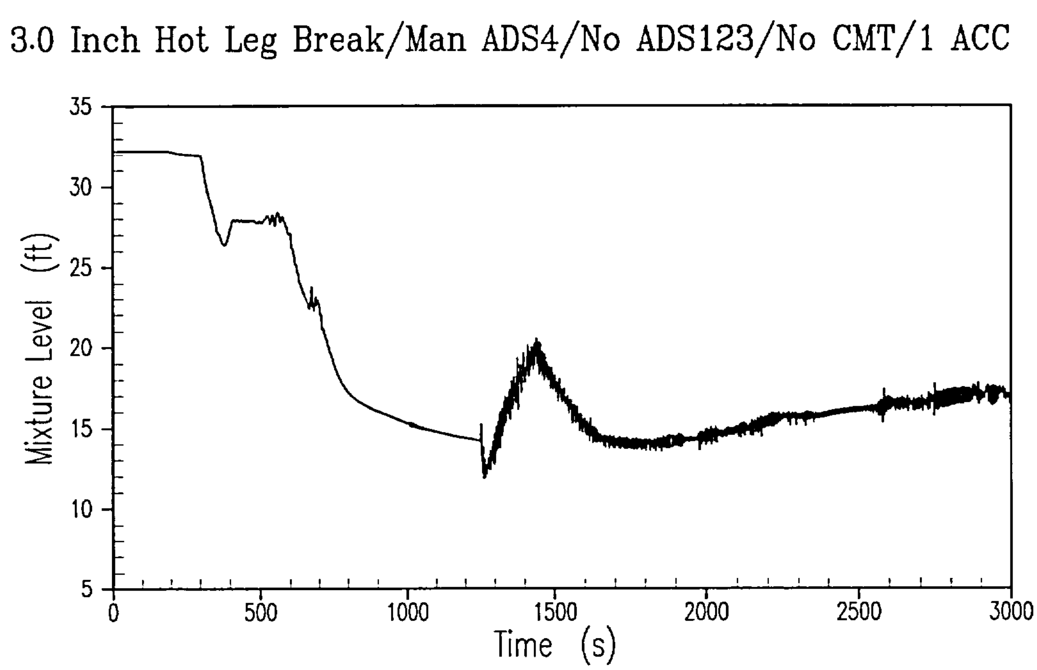


Figure A5.2-10

Case A – Downcomer Mixture Level

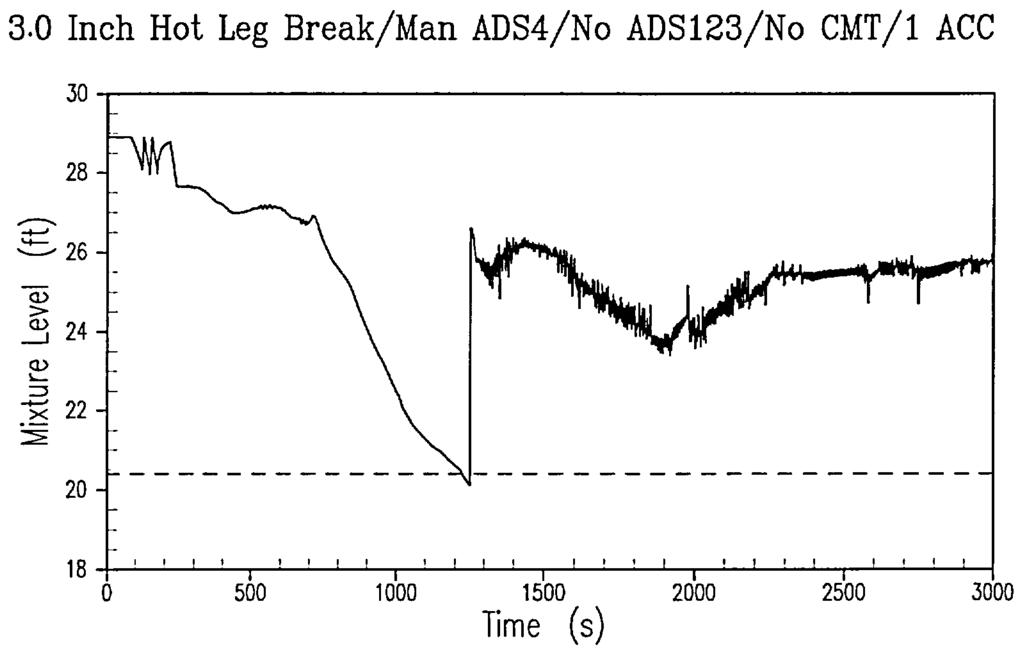


Figure A5.2-11

Case A – Upper Plenum and Core Mixture Level

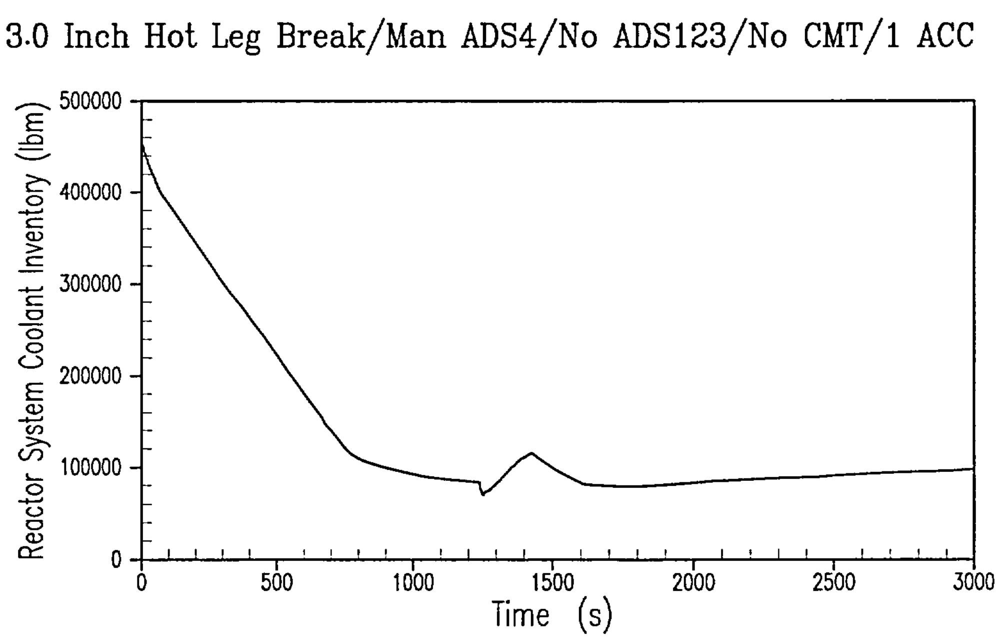


Figure A5.2-12

Case A – Reactor System Coolant Inventory

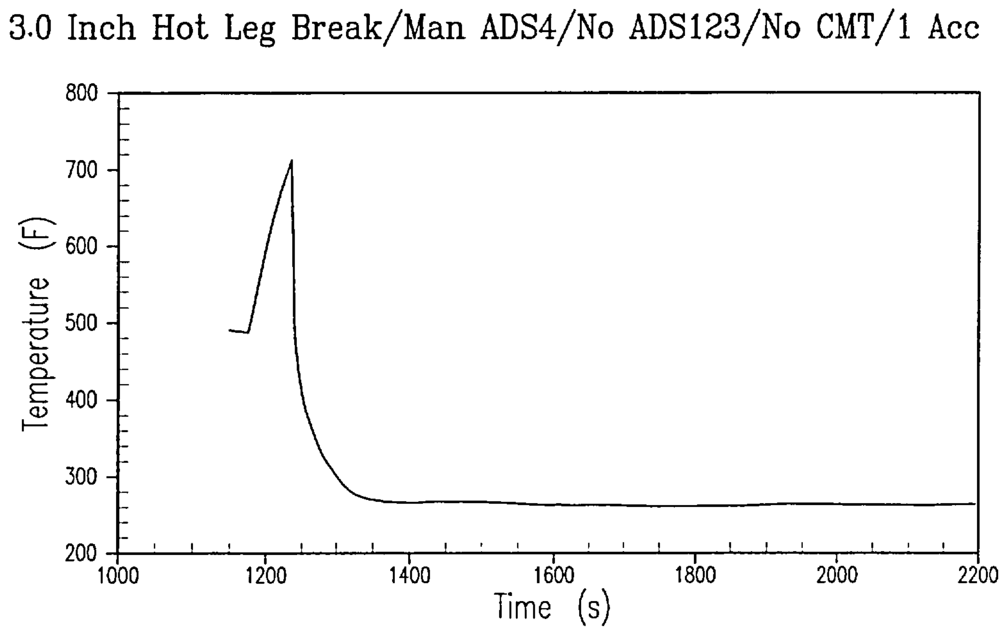


Figure A5.2-13

Case A – Peaking Cladding Temperature

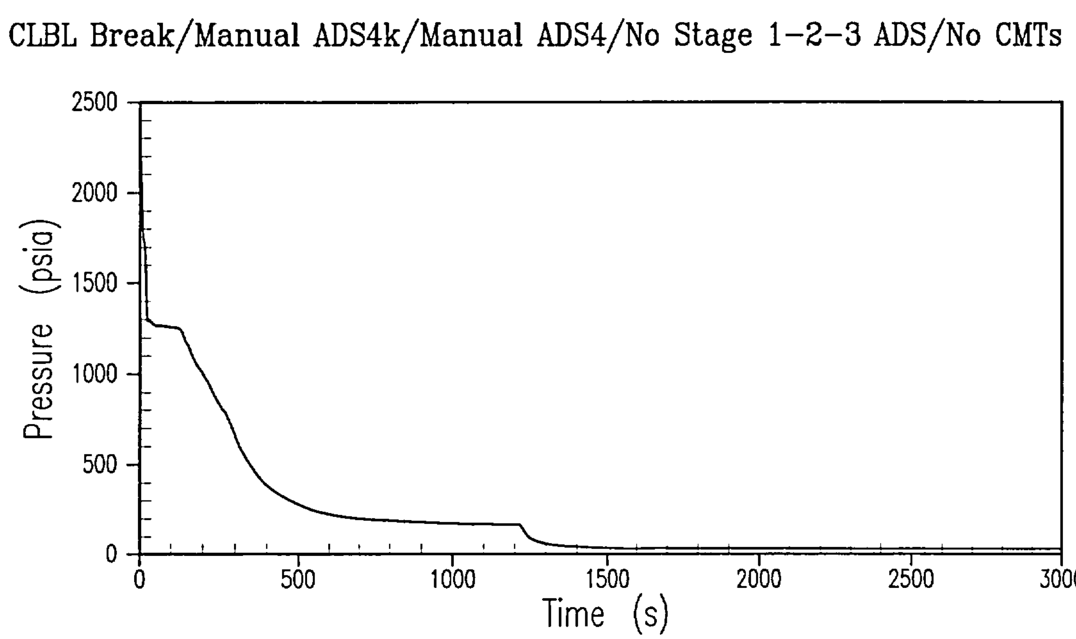


Figure A5.2-14

Case B – Pressurizer Pressure

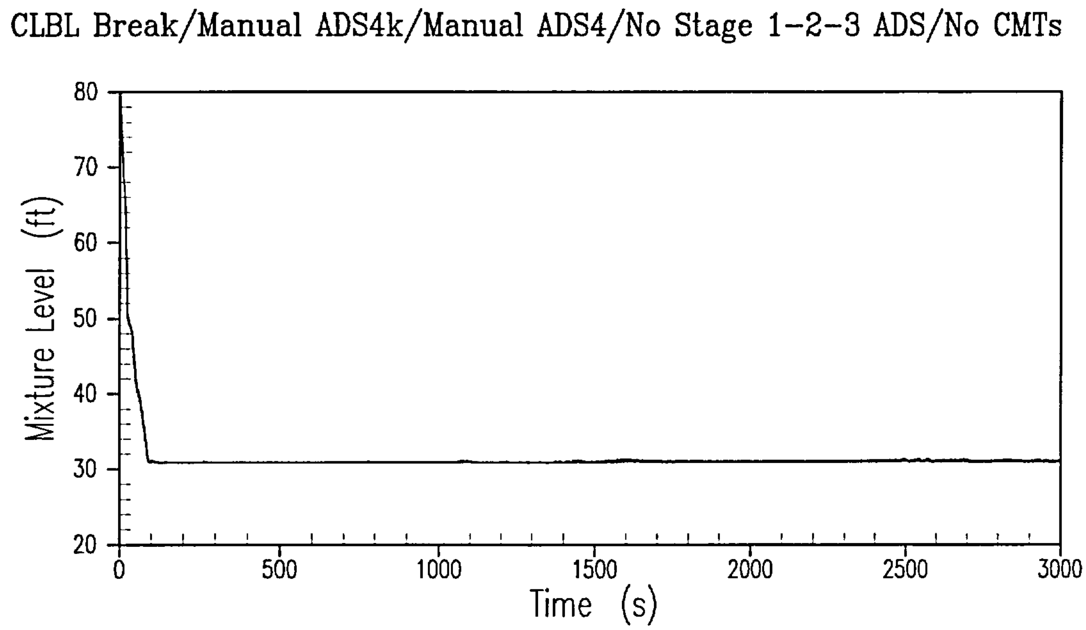


Figure A5.2-15

Case B – Pressurizer Level

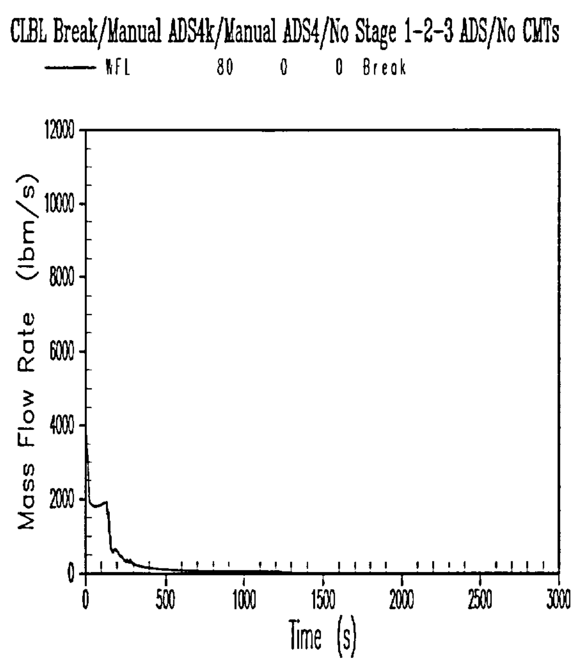


Figure A5.2-16

Case B – Break Liquid Flow

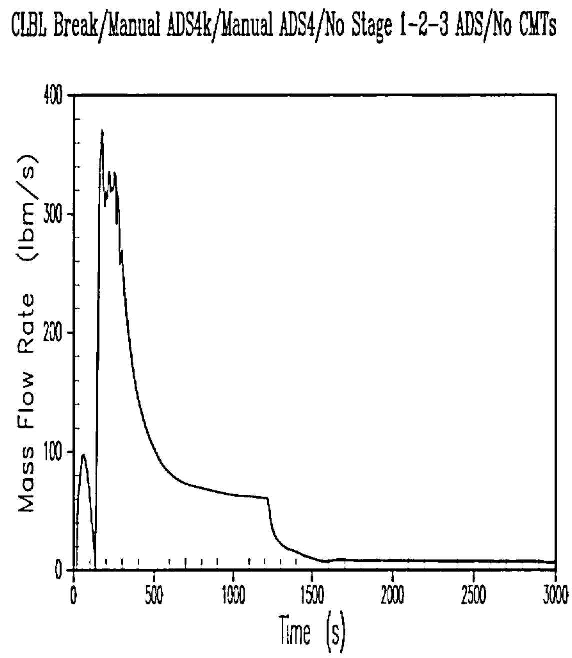


Figure A5.2-17

Case B – Break Vapor Flow

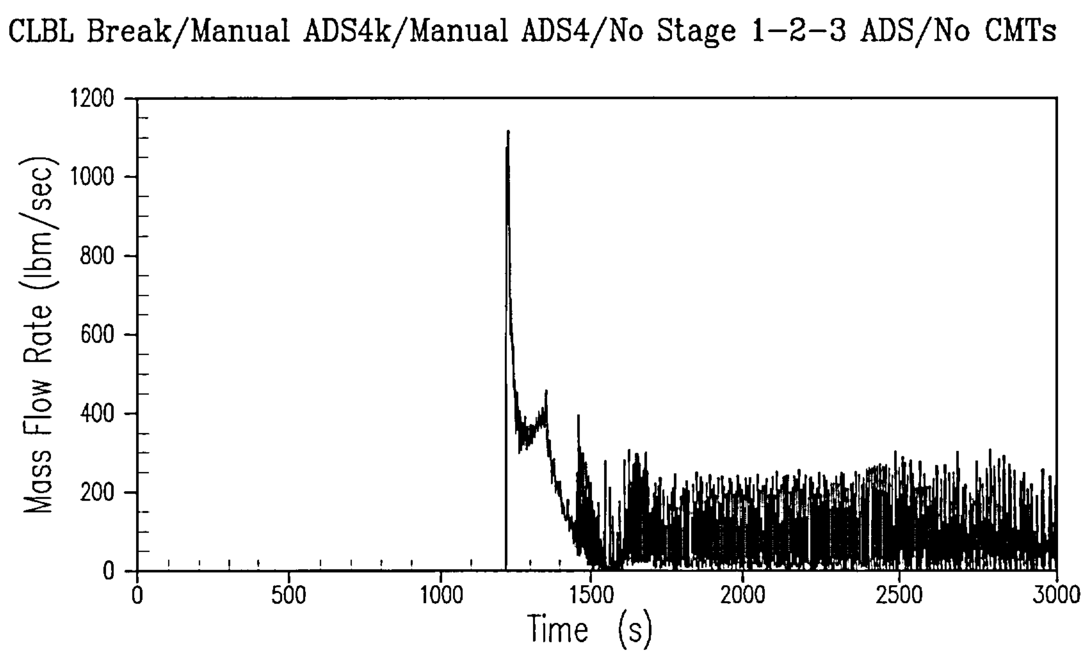


Figure A5.2-18

Case B – 4th Stage ADS Liquid Flow Through All Open Paths

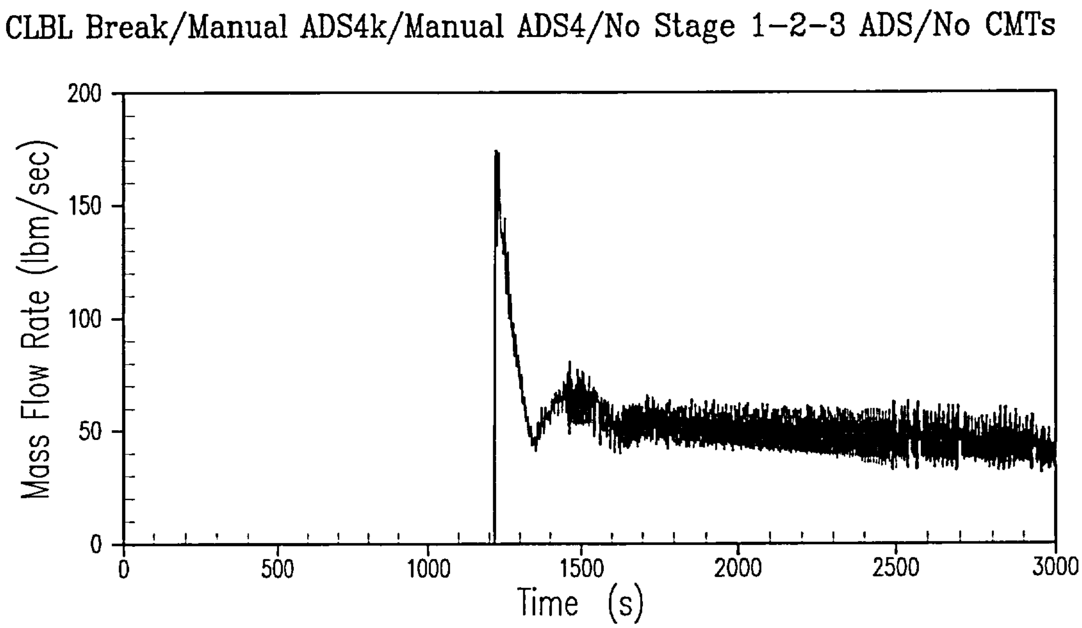


Figure A5.2-19

Case B – 4th Stage ADS Vapor Flow

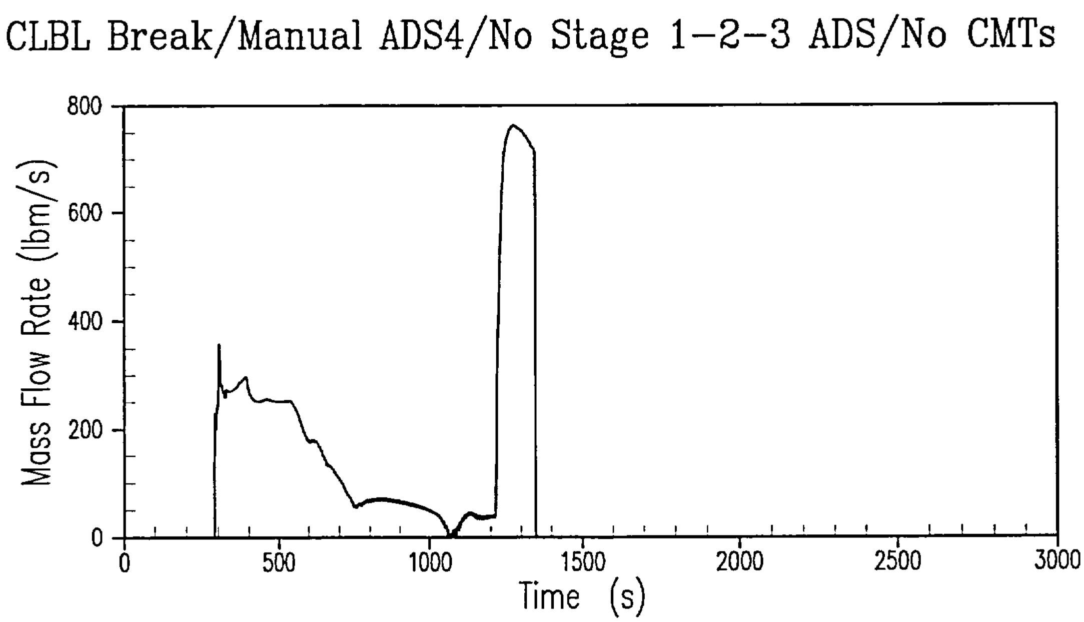


Figure A5.2-20

Case B – Accumulator Injection Flow

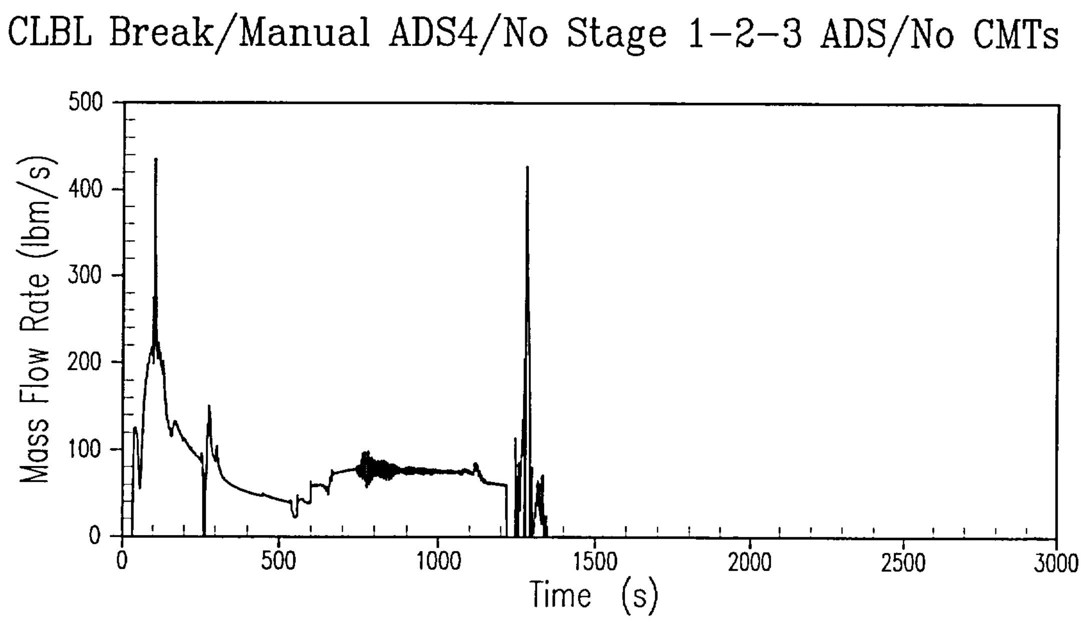


Figure A5.2-21

Case B – PRHR Discharge Flow

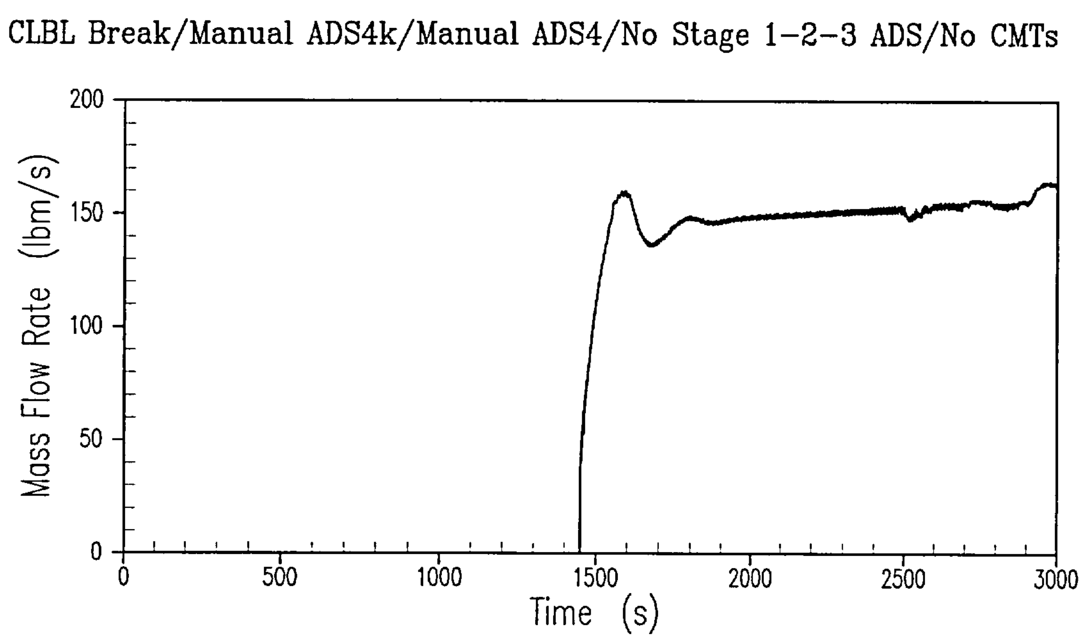


Figure A5.2-22

Case B – IRWST Injection Flow

CLBL Break/Manual ADS4k/Manual ADS4/No Stage 1-2-3 ADS/No CMTs

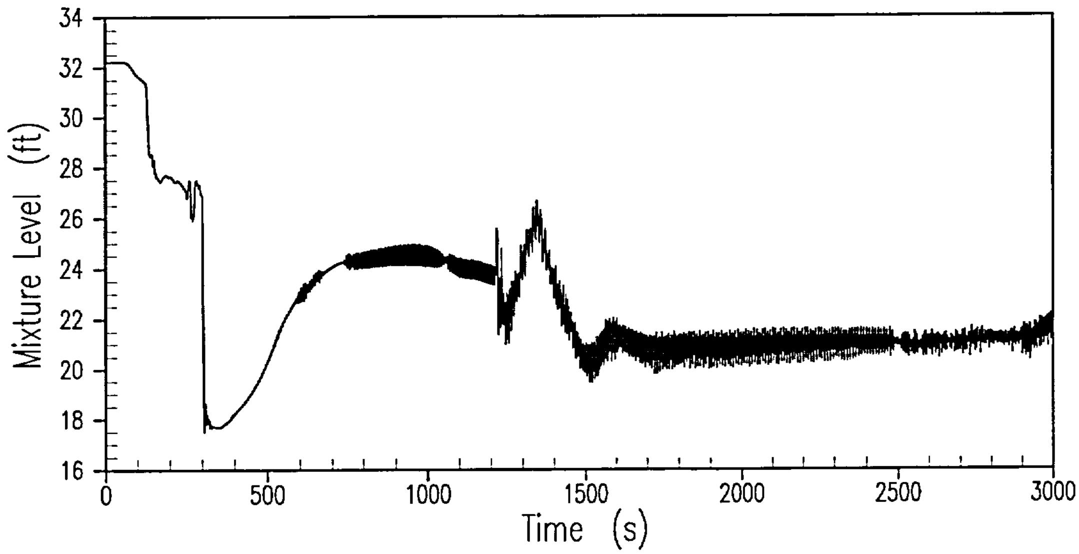


Figure A5.2-23

Case B – Downcomer Mixture Level

CLBL Break/Manual ADS4k/Manual ADS4/No Stage 1-2-3 ADS/No CMTs

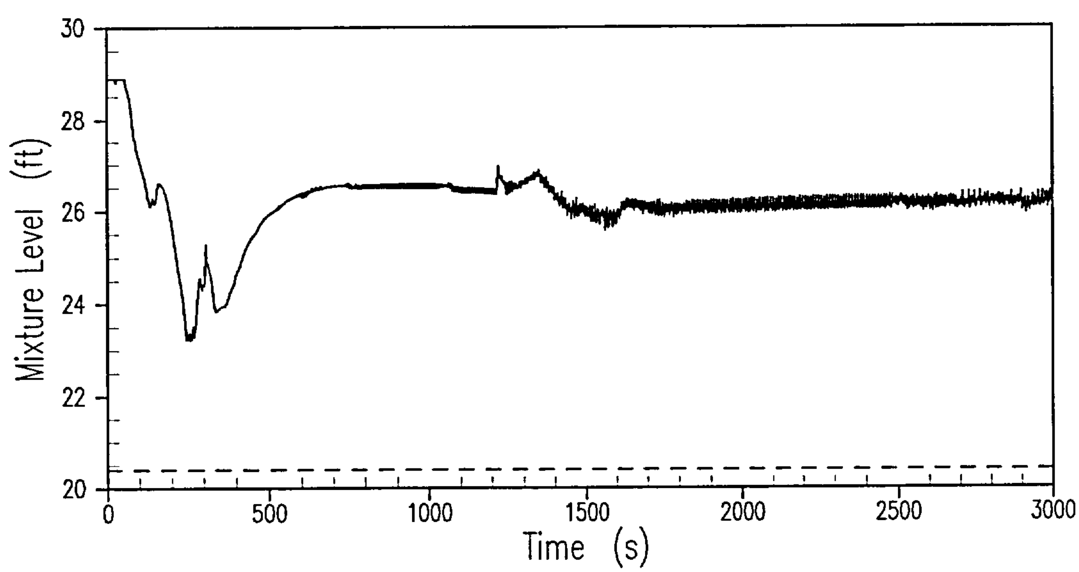


Figure A5.2-24

Case B – Upper Plenum and Core Mixture Level

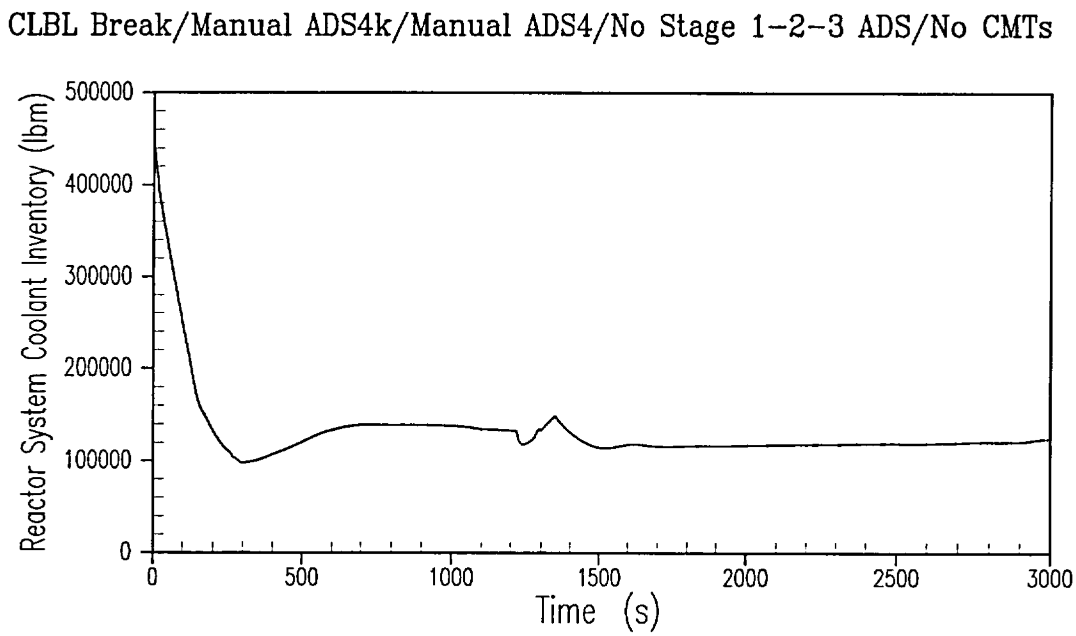


Figure A5.2-25

Case B – Reactor System Coolant Inventory

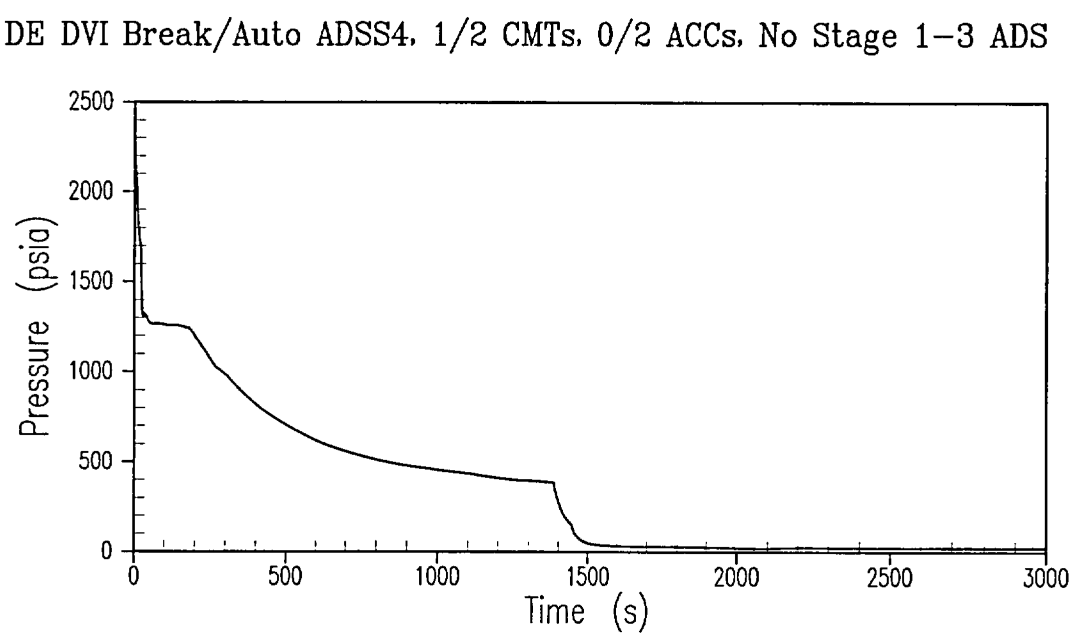


Figure A5.2-26

Case C – Pressurizer Pressure

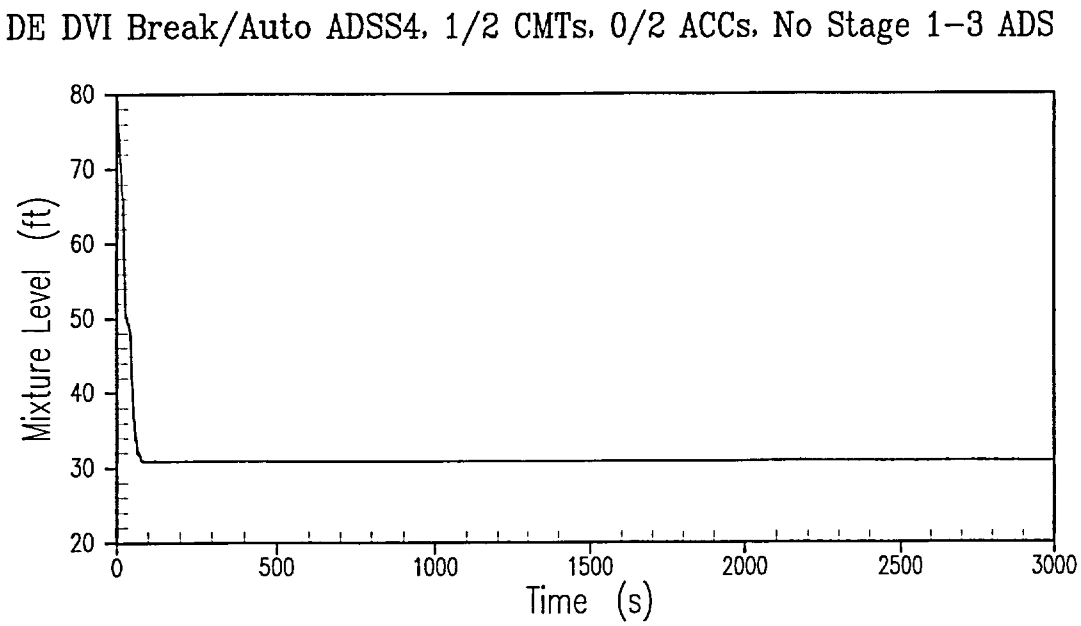


Figure A5.2-27

Case C – Pressurizer Level

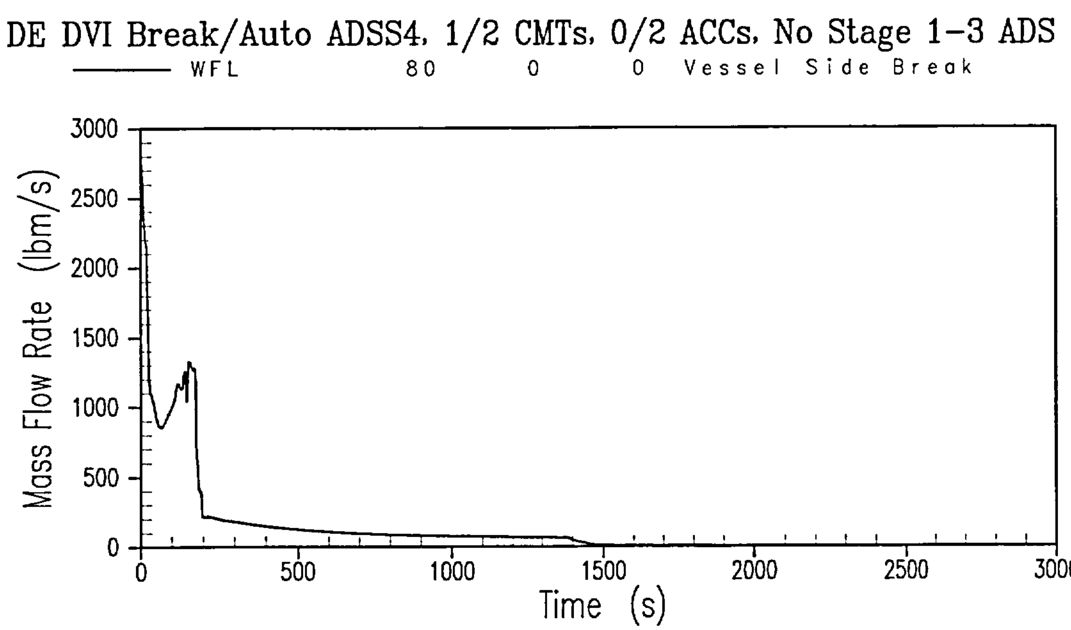


Figure A5.2-28

Case C – Break Liquid Flow

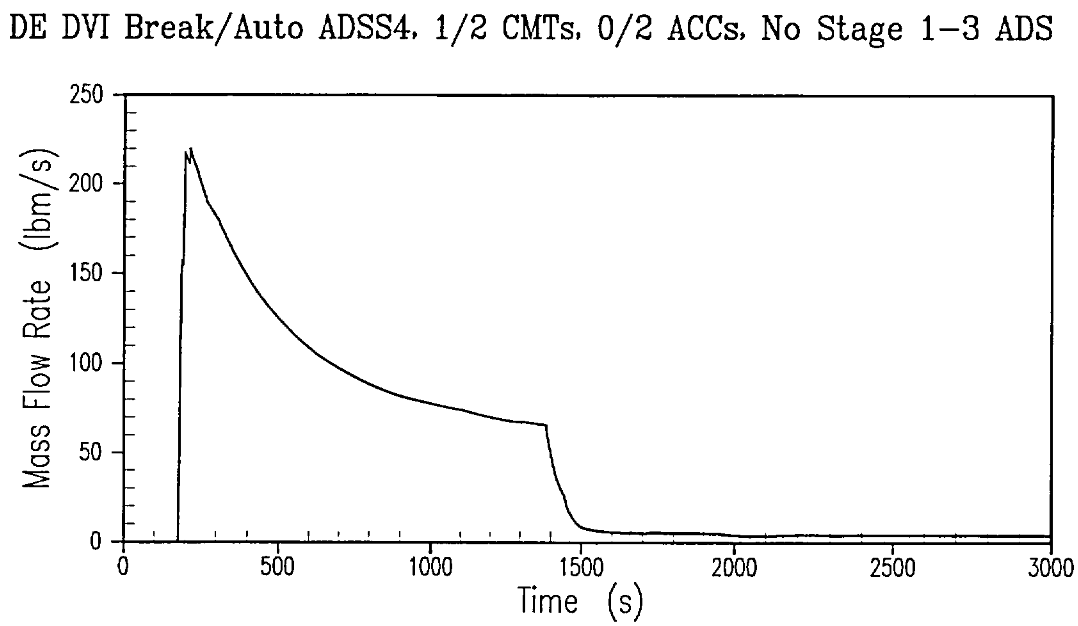


Figure A5.2-29

Case C – Break Vapor Flow

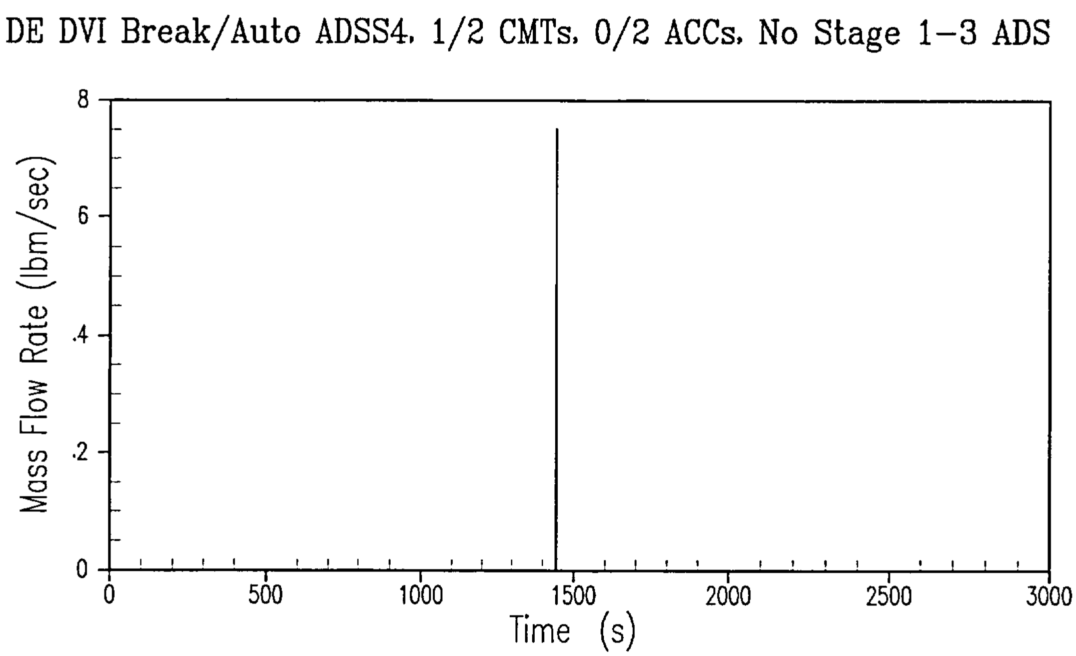


Figure A5.2-30

Case C – 4th Stage ADS Liquid Flow Through All Open Paths

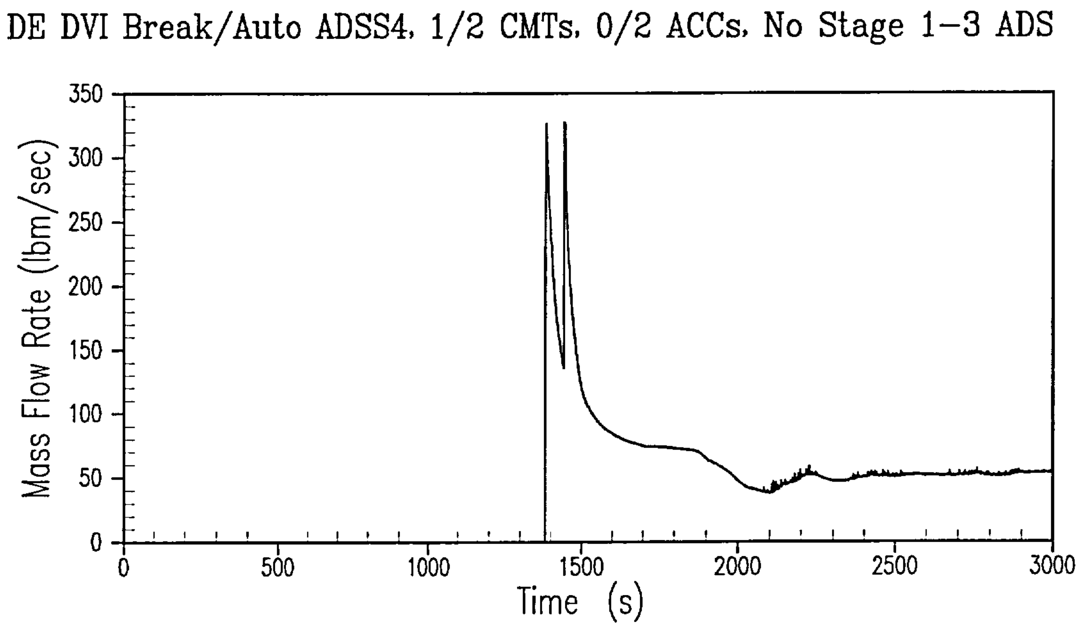


Figure A5.2-31

Case C – 4th Stage ADS Vapor Flow

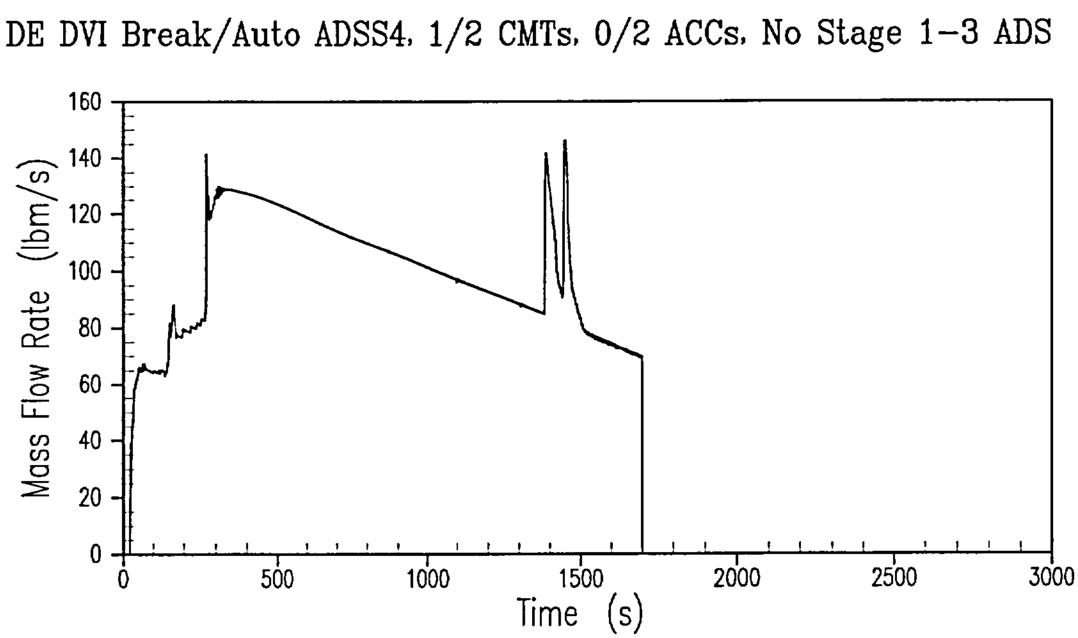


Figure A5.2-32

Case C – Core Makeup Tank Injection Flow

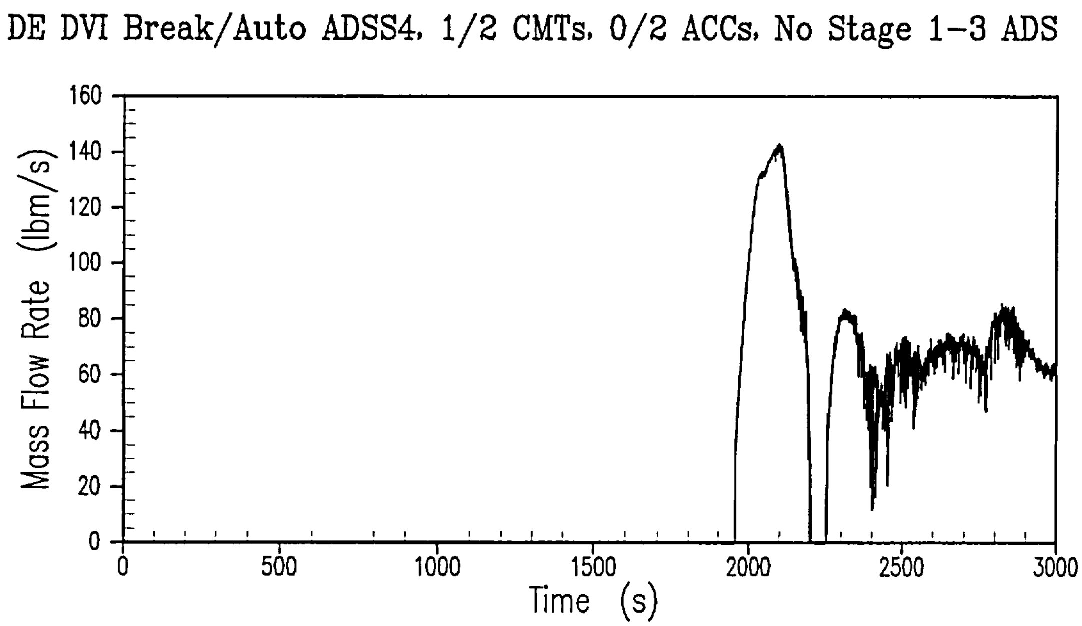


Figure A5.2-33

Case C – IRWST Injection Flow

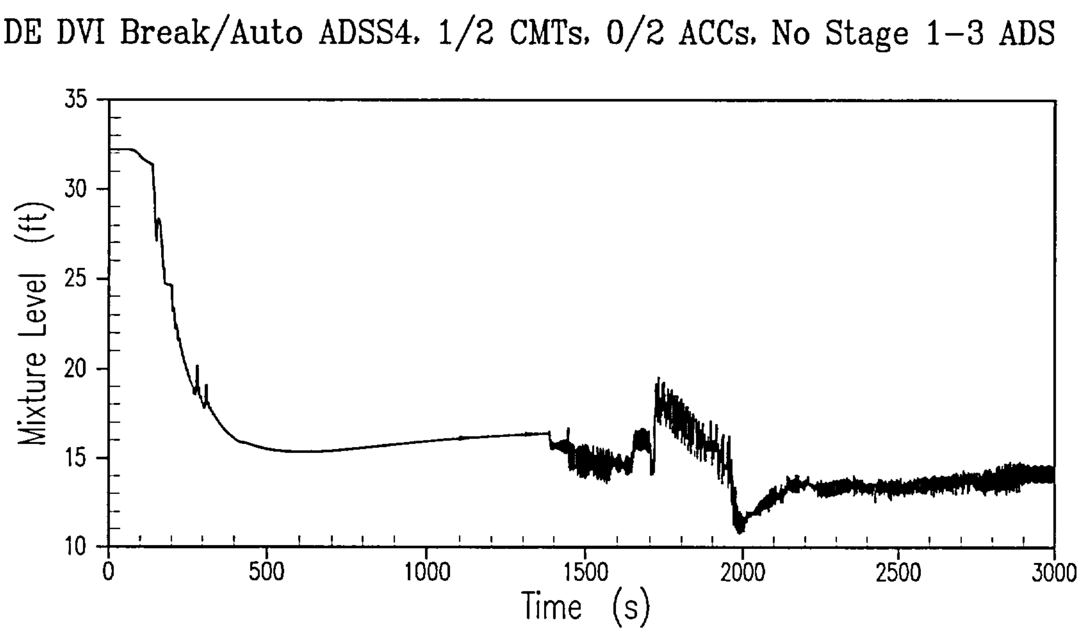


Figure A5.2-34

Case C – Downcomer Mixture Level

DE DVI Break/Auto ADSS4, 1/2 CMTs, 0/2 ACCs, No Stage 1-3 ADS

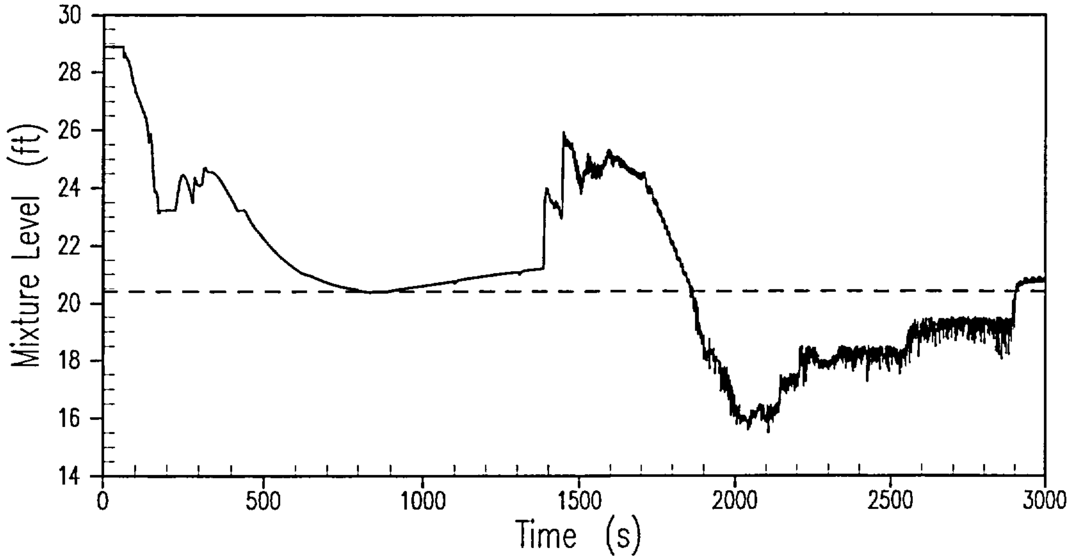


Figure A5.2-35

Case C – Upper Plenum and Core Mixture Level

DE DVI Break/Auto ADSS4, 1/2 CMTs, 0/2 ACCs, No Stage 1-3 ADS

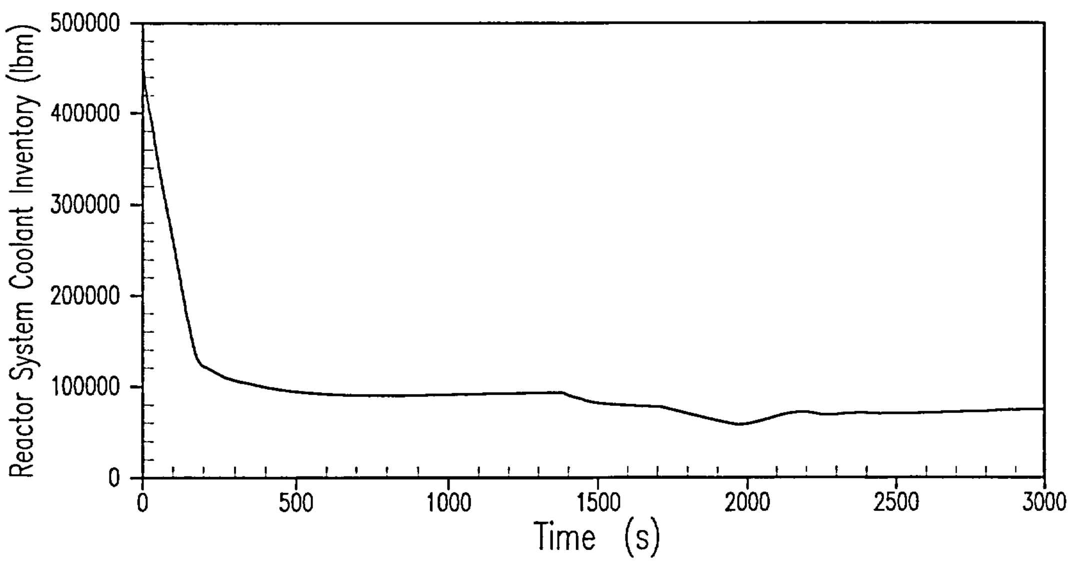


Figure A5.2-36

Case C – Reactor System Coolant Inventory

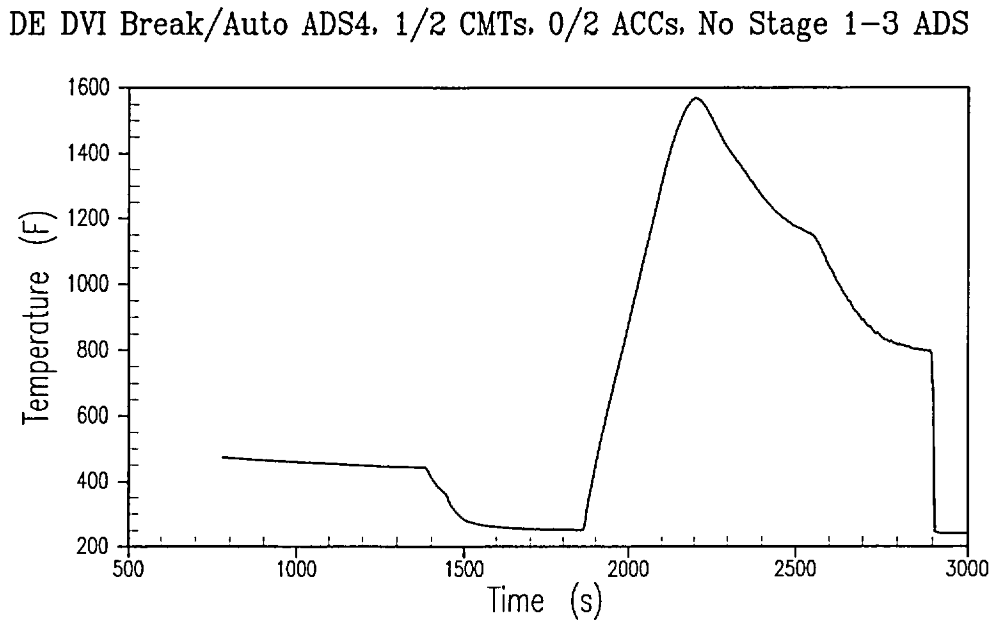


Figure A5.2-37

Case C – Peak Clad Temperature

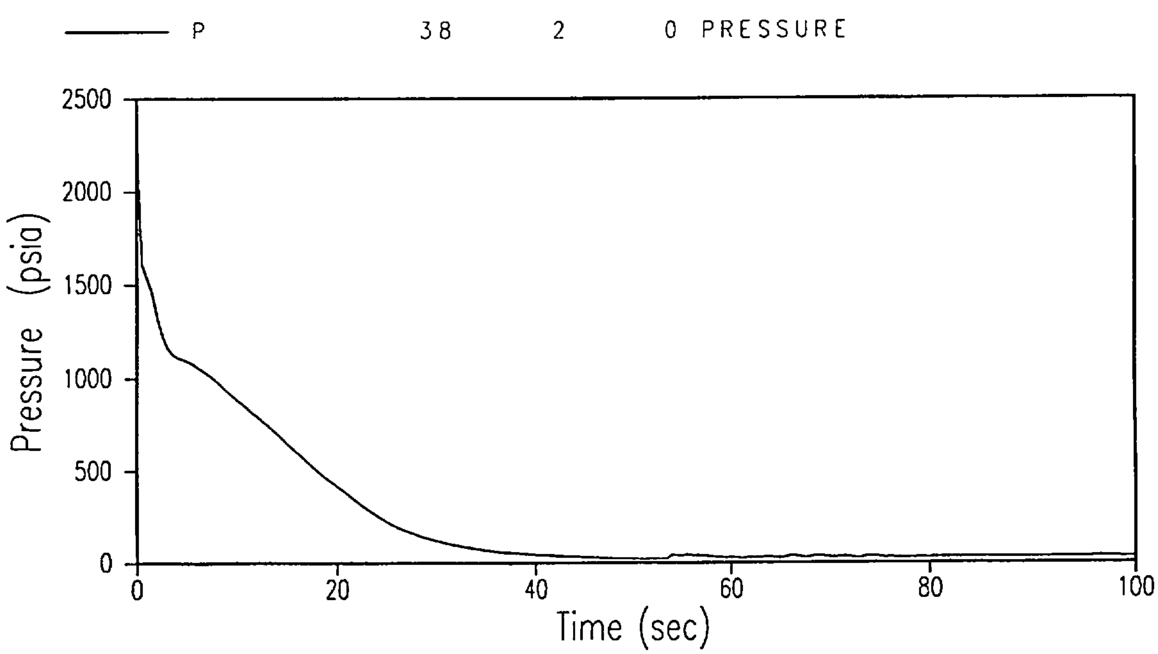


Figure A5.2-38

Case D – Reactor Vessel Upper Plenum Pressure

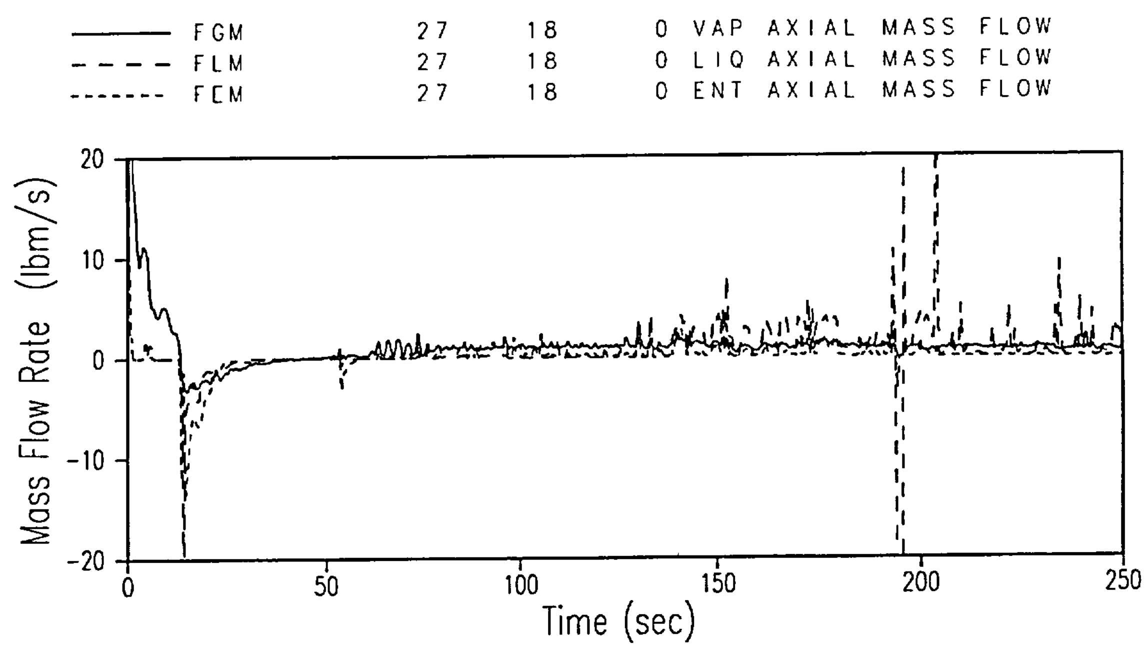


Figure A5.2-39

Case D – Hot Assembly Exit Vapor and Liquid Flows

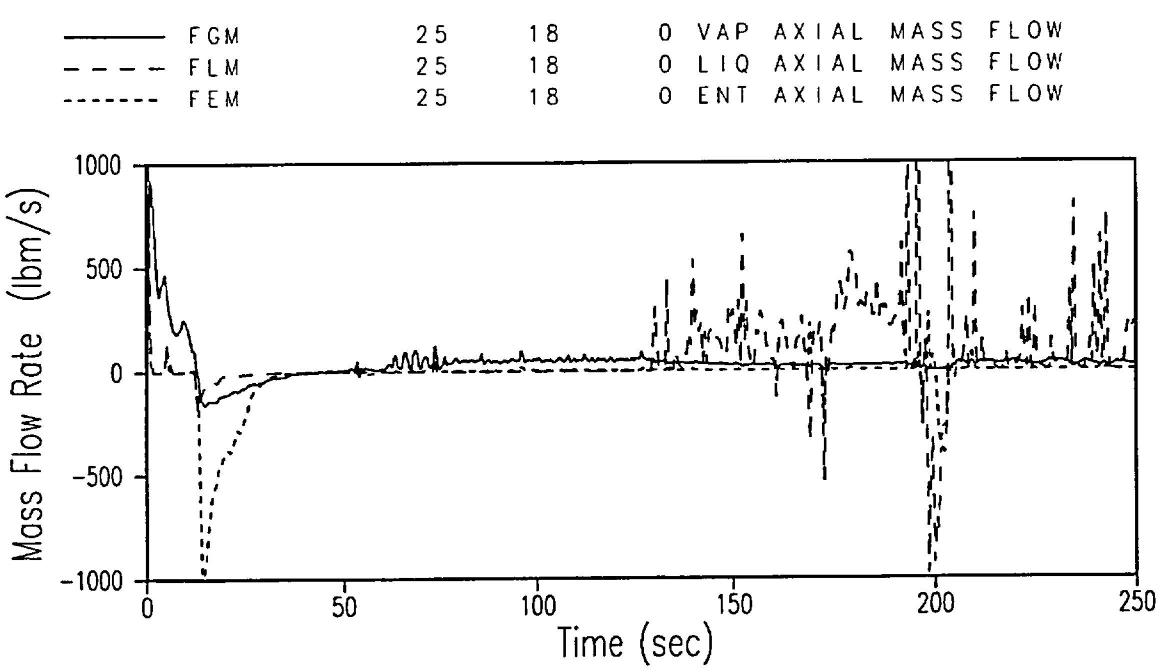


Figure A5.2-40

Case D – Open Hole Assembly Exit Vapor and Liquid Flows

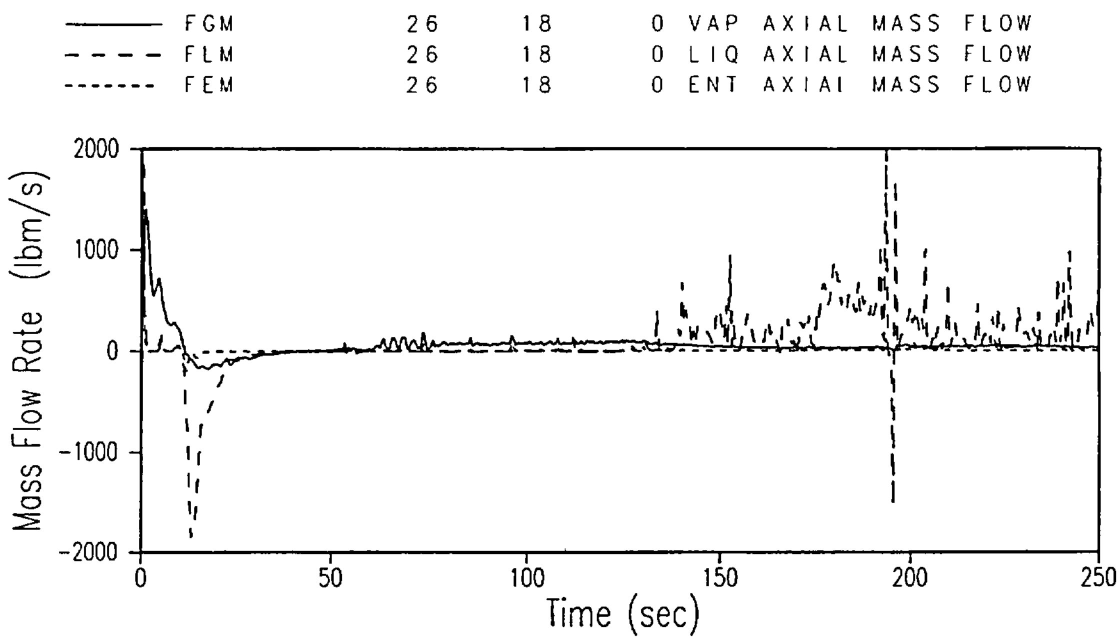


Figure A5.2-41

Case D – Guide Tube Assembly Exit Vapor and Liquid Flows

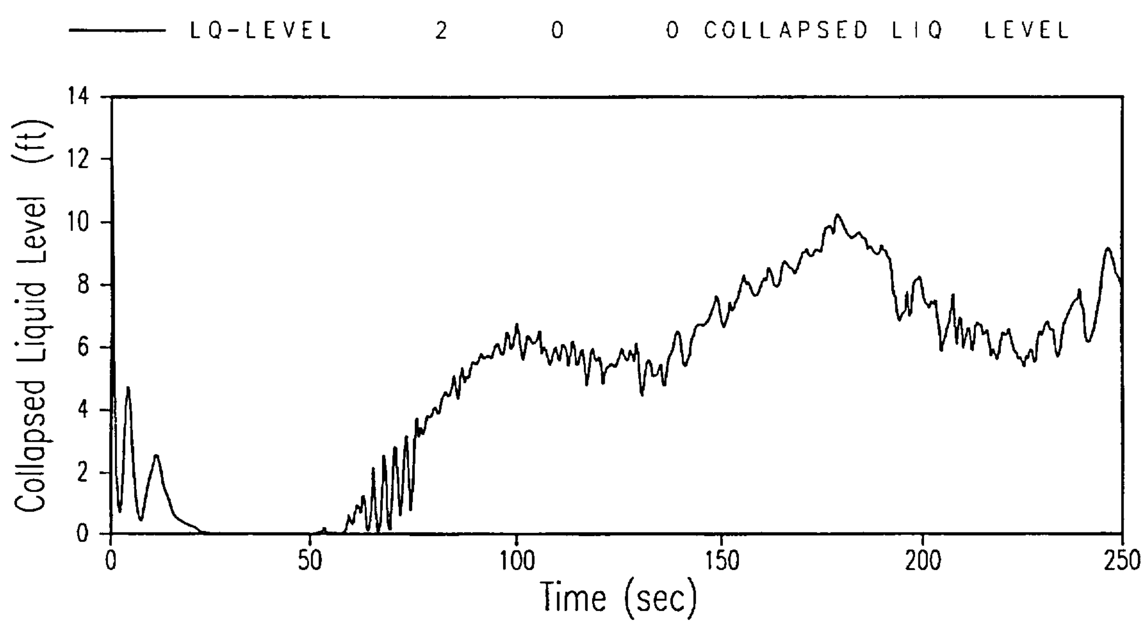


Figure A5.2-42

Case D – Core Collapsed Liquid Level

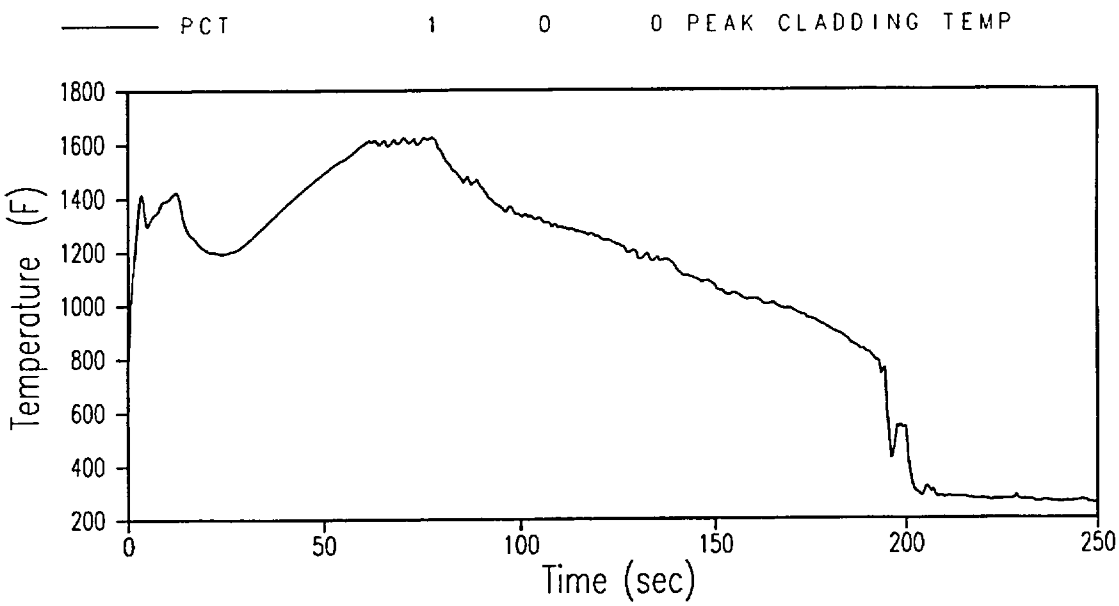


Figure A5.2-43

Case D – Peak Cladding Temperature

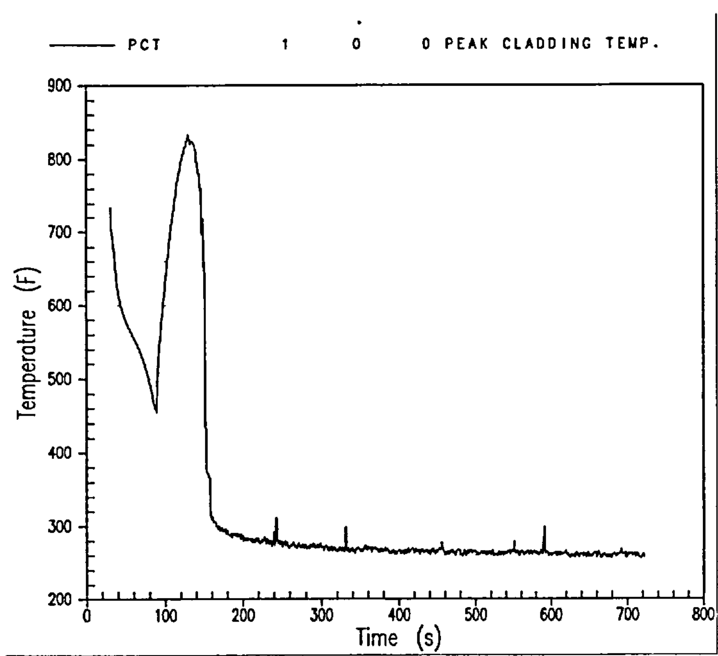


Figure A5.2-44

Case E –Peak Cladding Temperature

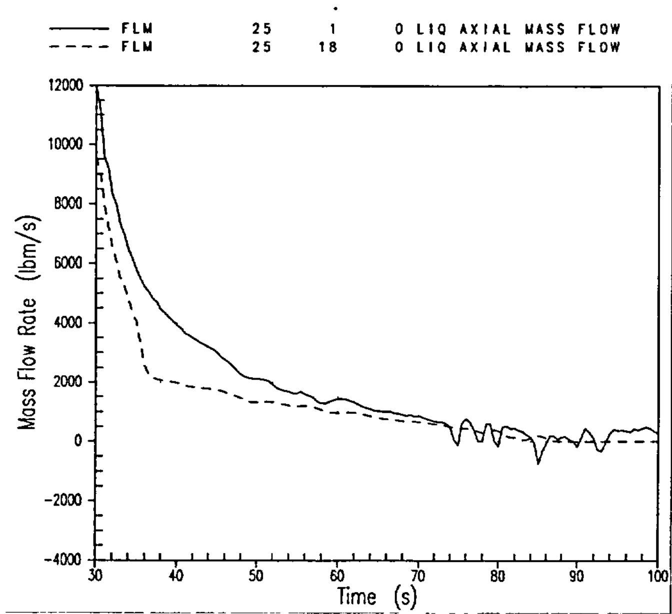


Figure A5.2-45

Case E – Core Liquid Entry/Exit Flow Rates

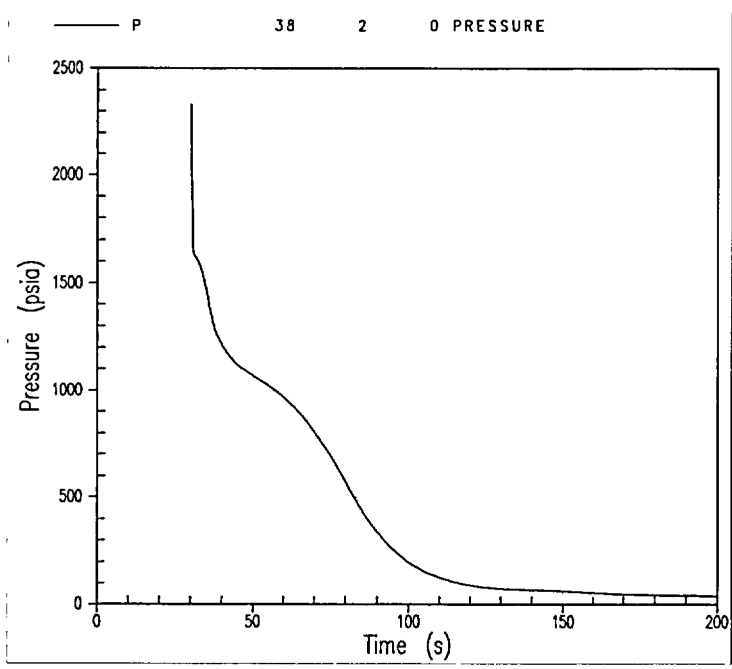


Figure A5.2-46

Case E – Core Pressure

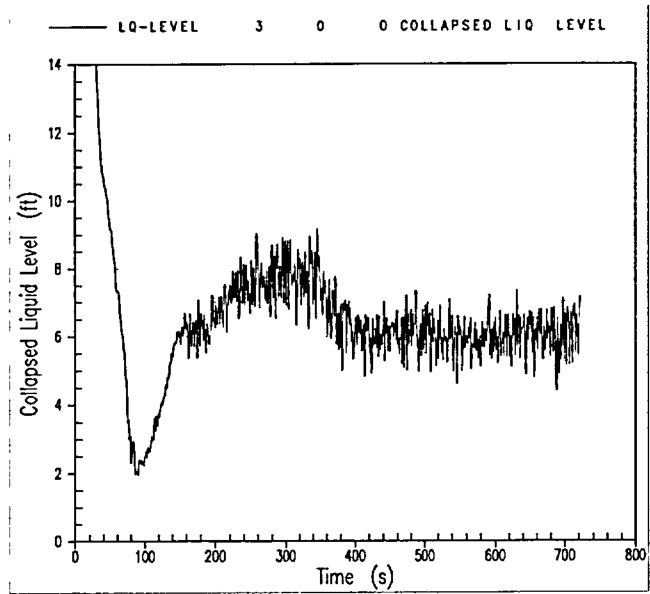


Figure A5.2-47

Case E – Core Liquid Level

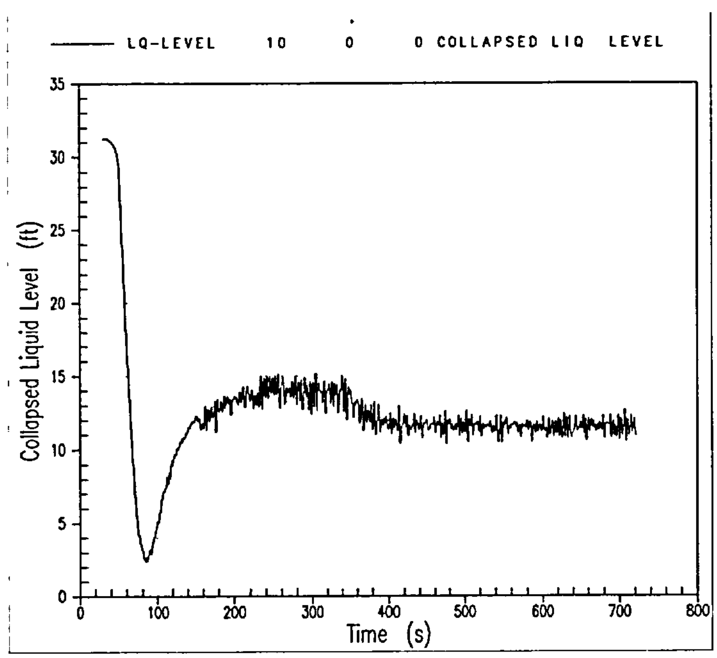


Figure A5.2-48

Case E – Downcomer Liquid Level

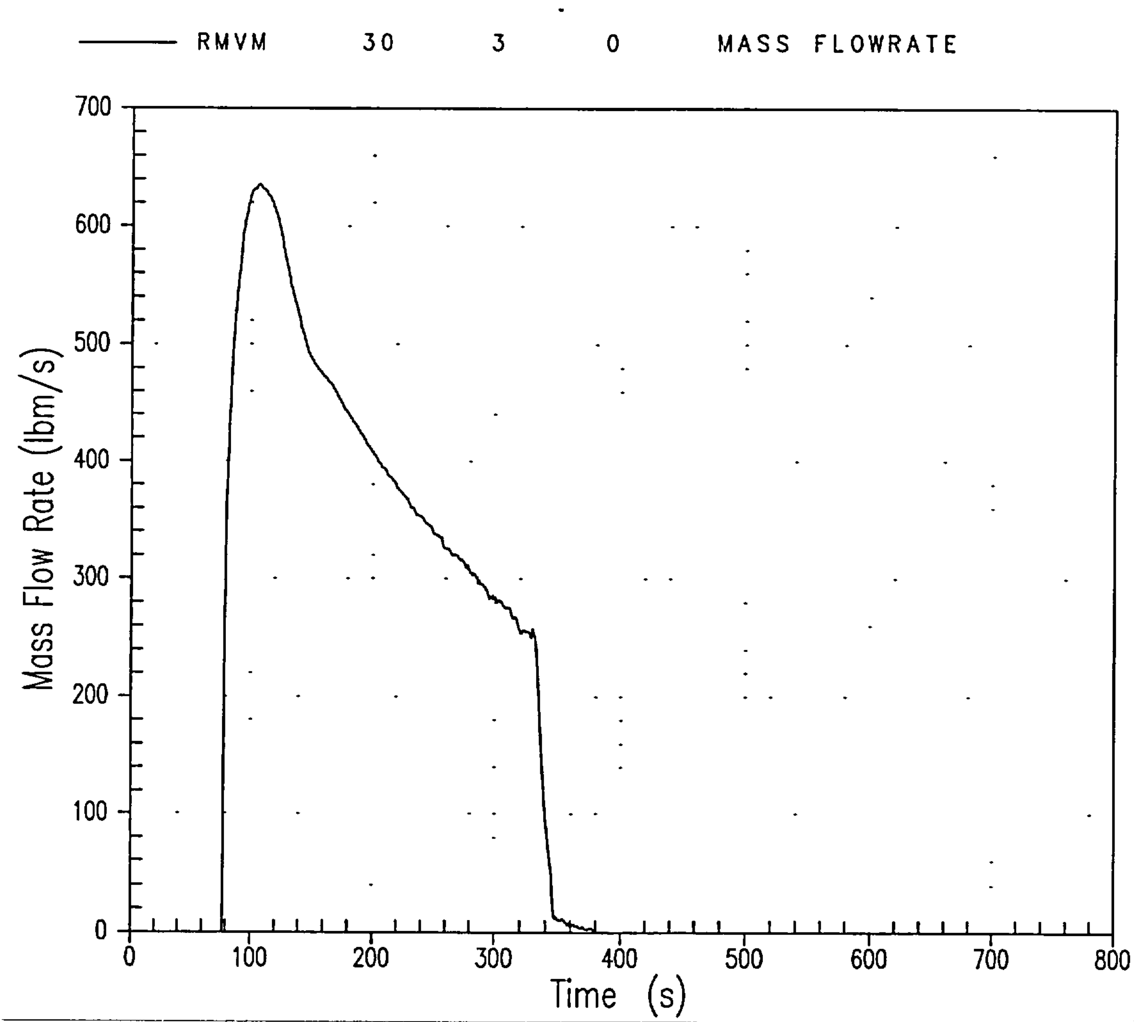


Figure A5.2-49

Case E – Accumulator Injection Flow Rate

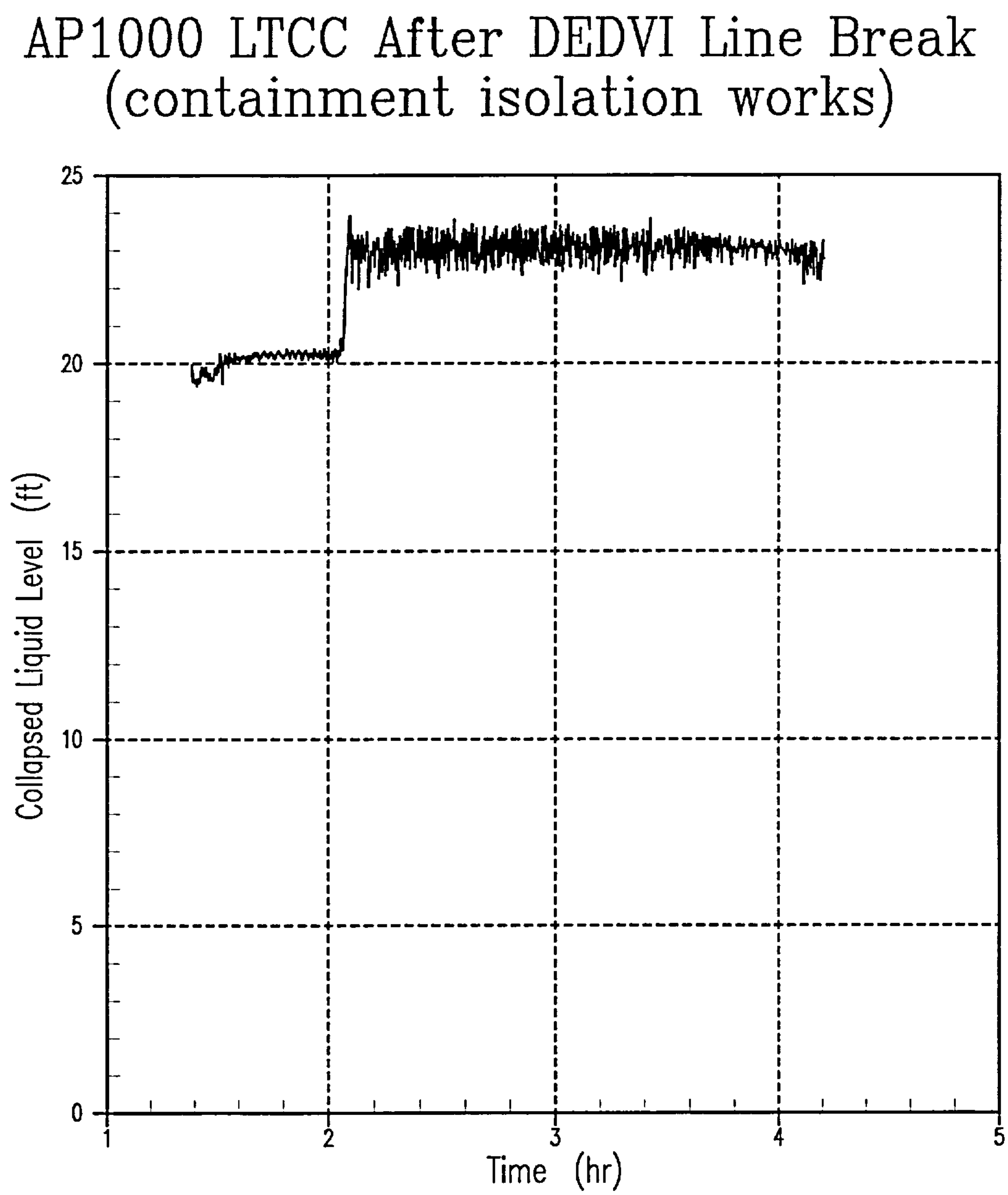


Figure A5.3-1

Case F – Collapsed Level of Liquid in the Downcomer

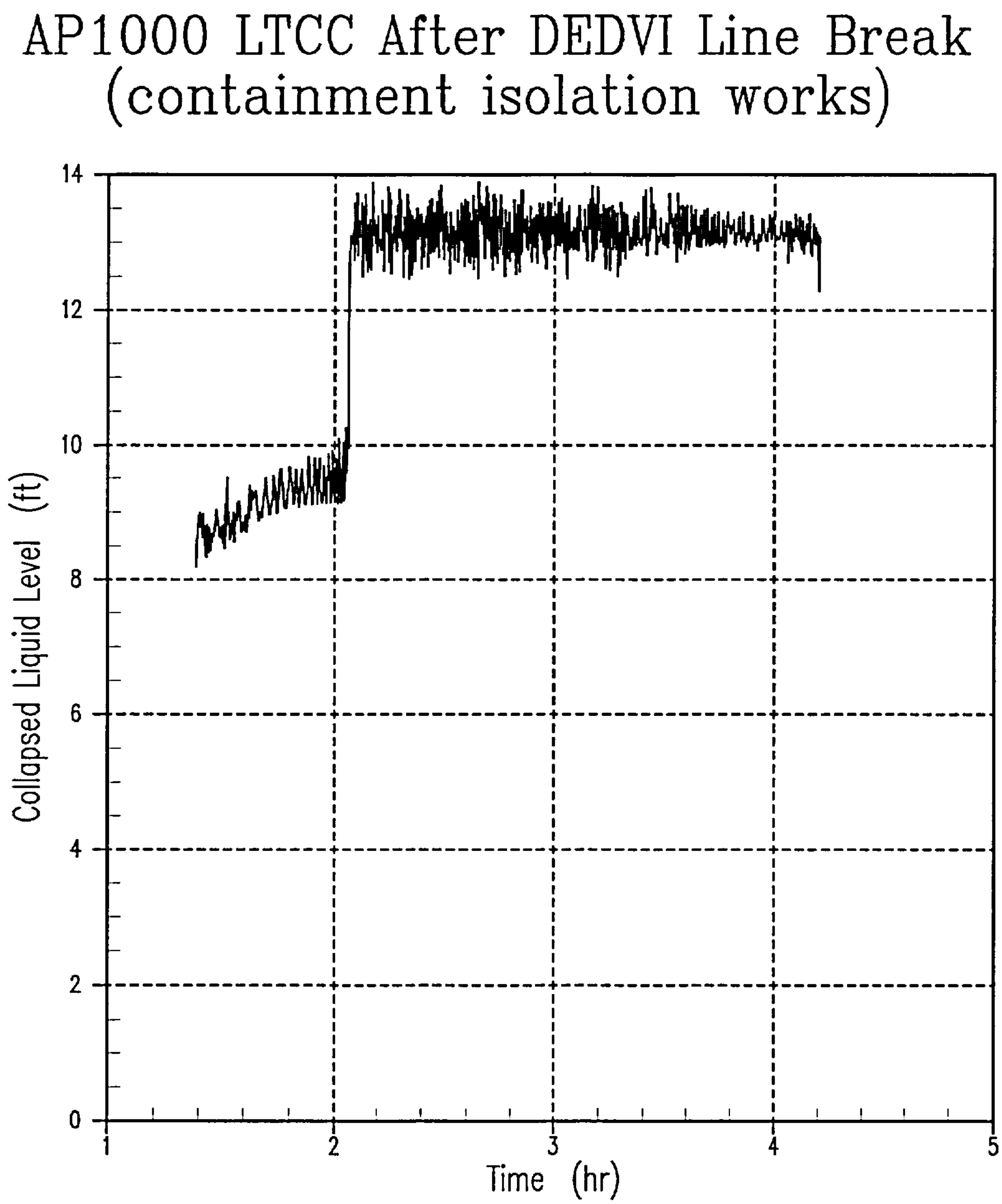


Figure A5.3-2

Case F – Collapsed Level of Liquid Over the Heated Length of the Fuel

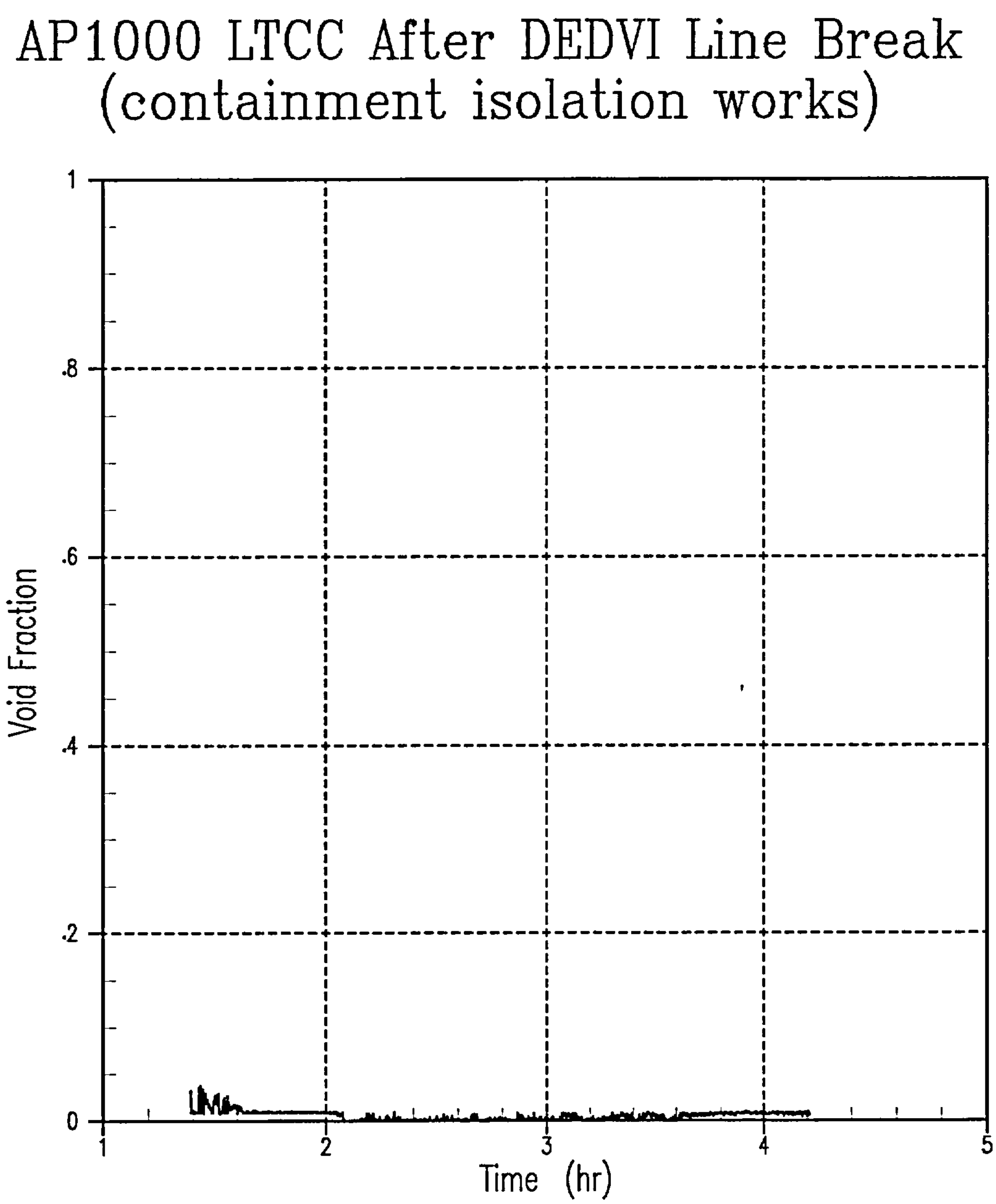


Figure A5.3-3

Case F – Void Fraction in Core Cell Level 1 of 2

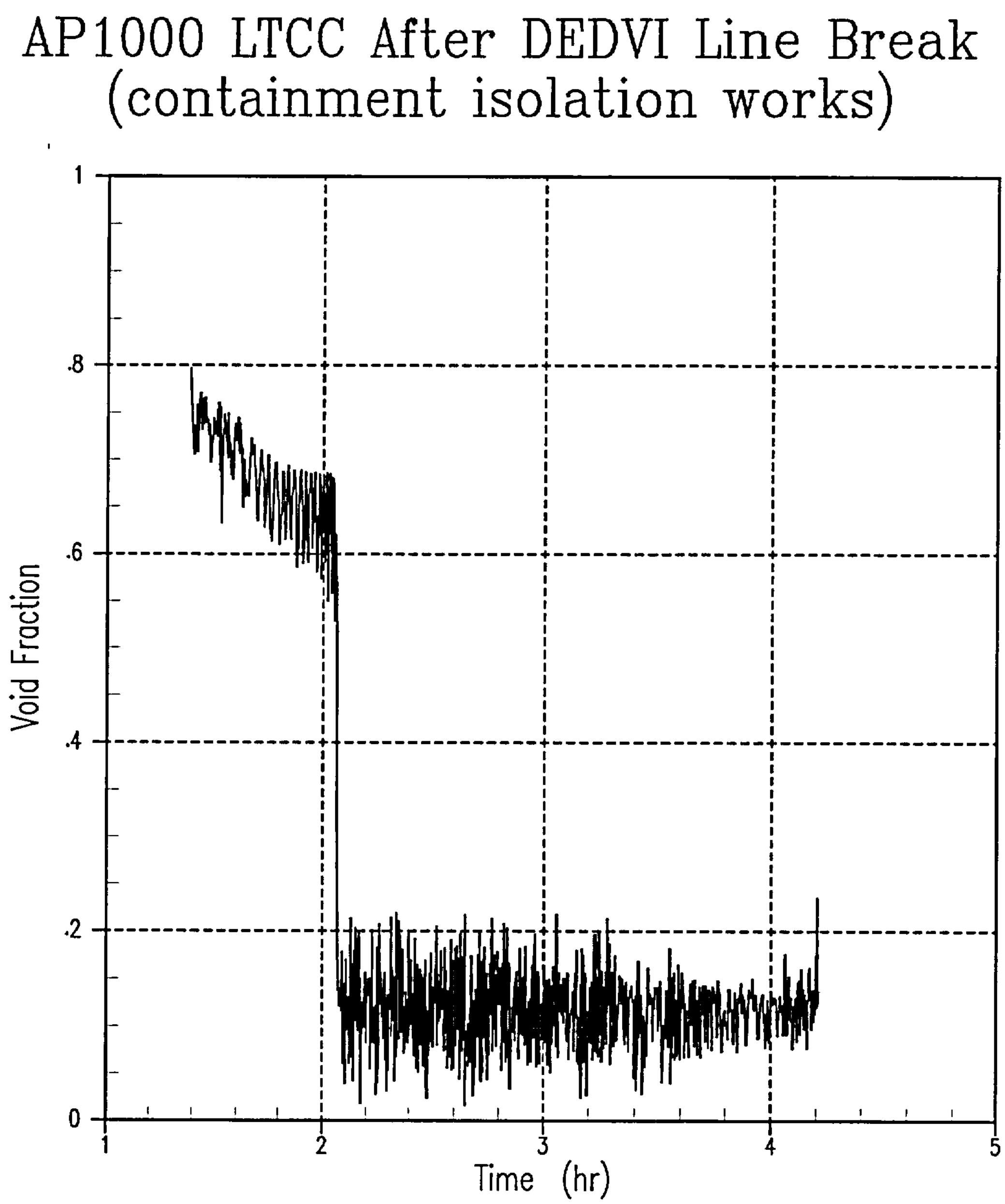


Figure A5.3-4

Case F – Void Fraction in Core Cell Level 2 of 2

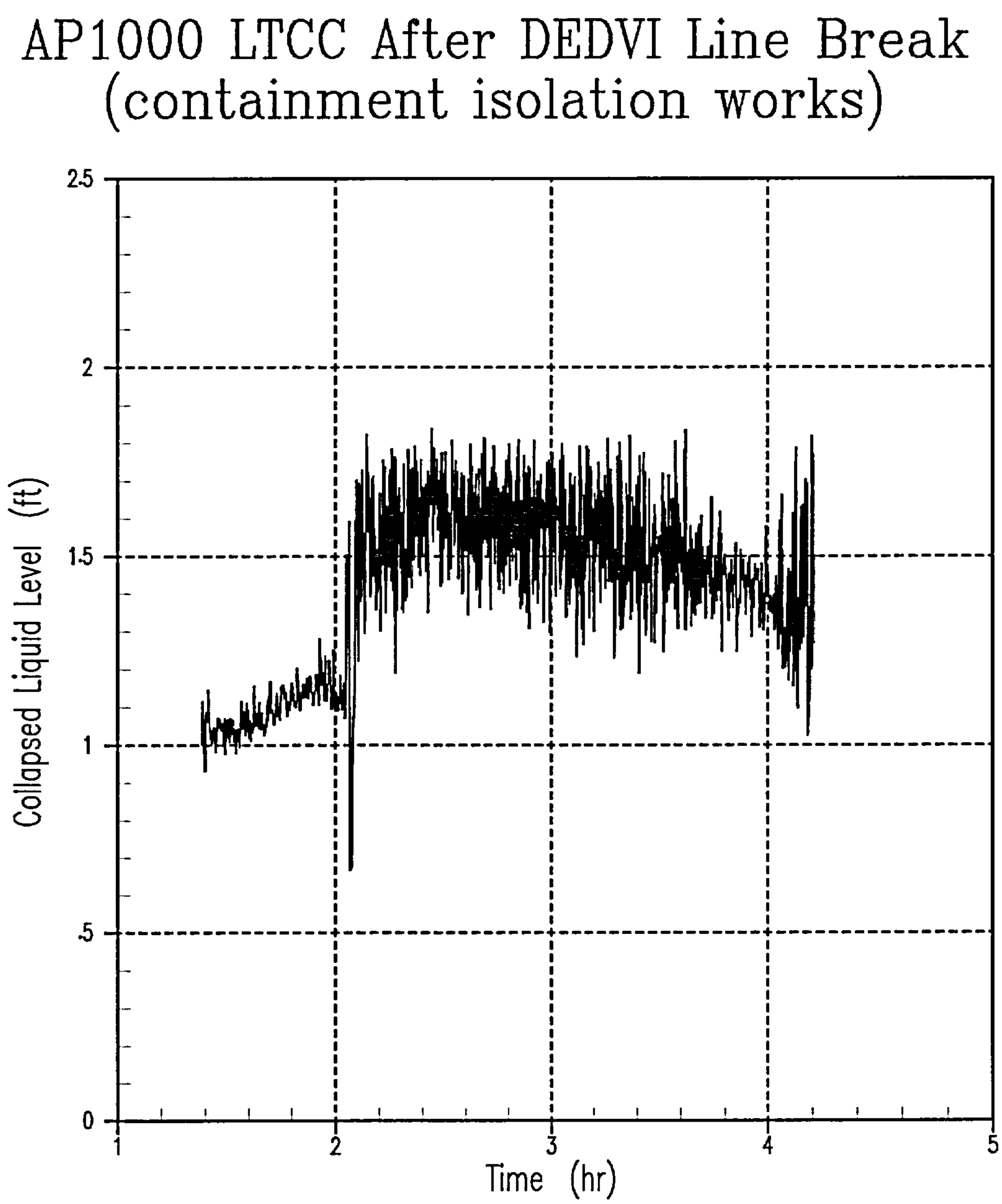


Figure A5.3-5

Case F – Collapsed Liquid Level in the Hot Leg of Pressurizer Loop

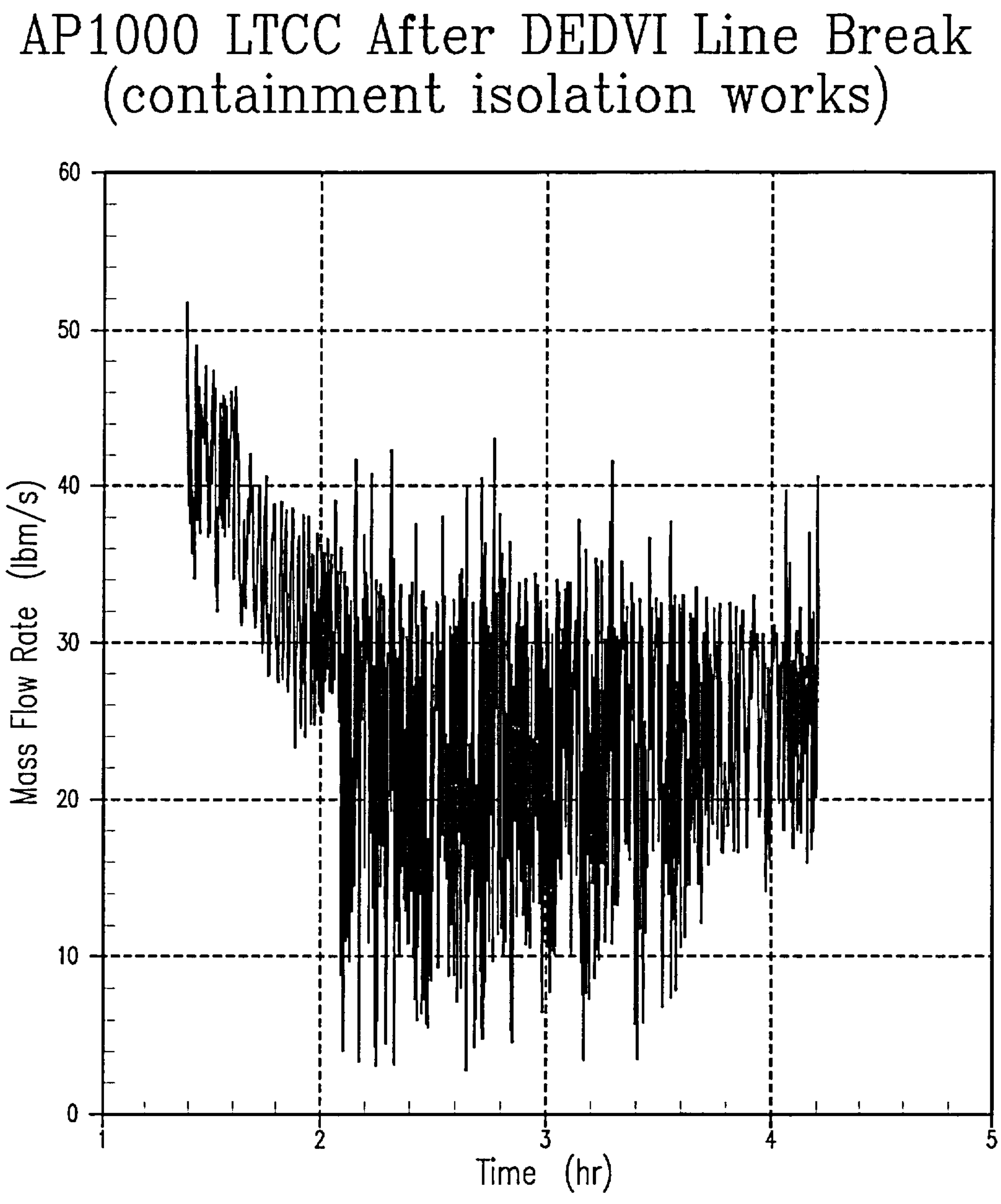


Figure A5.3-6

Case F – Vapor Rate out of the Core

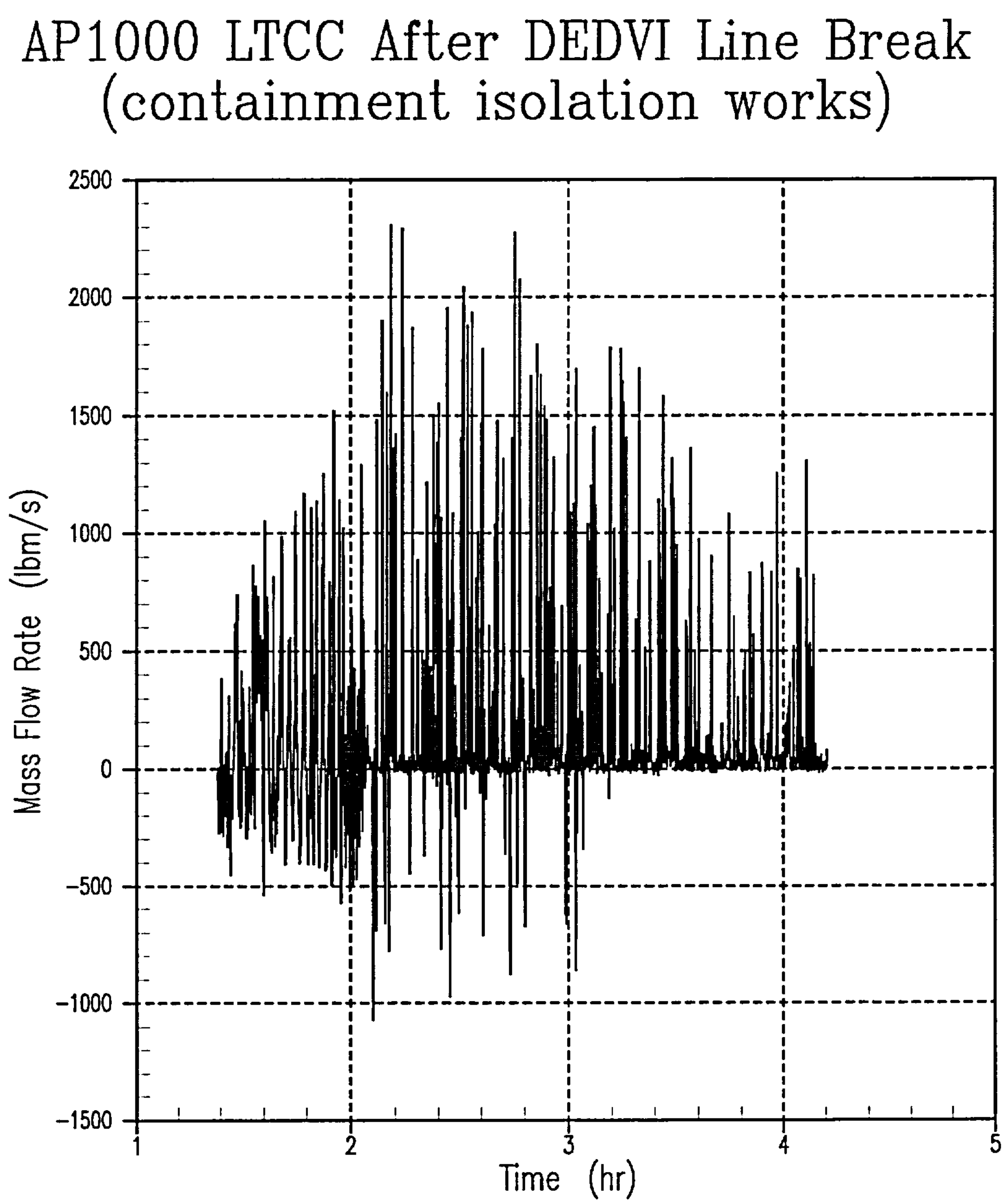


Figure A5.3-7

Case F – Liquid Flow Rate Out of the Core

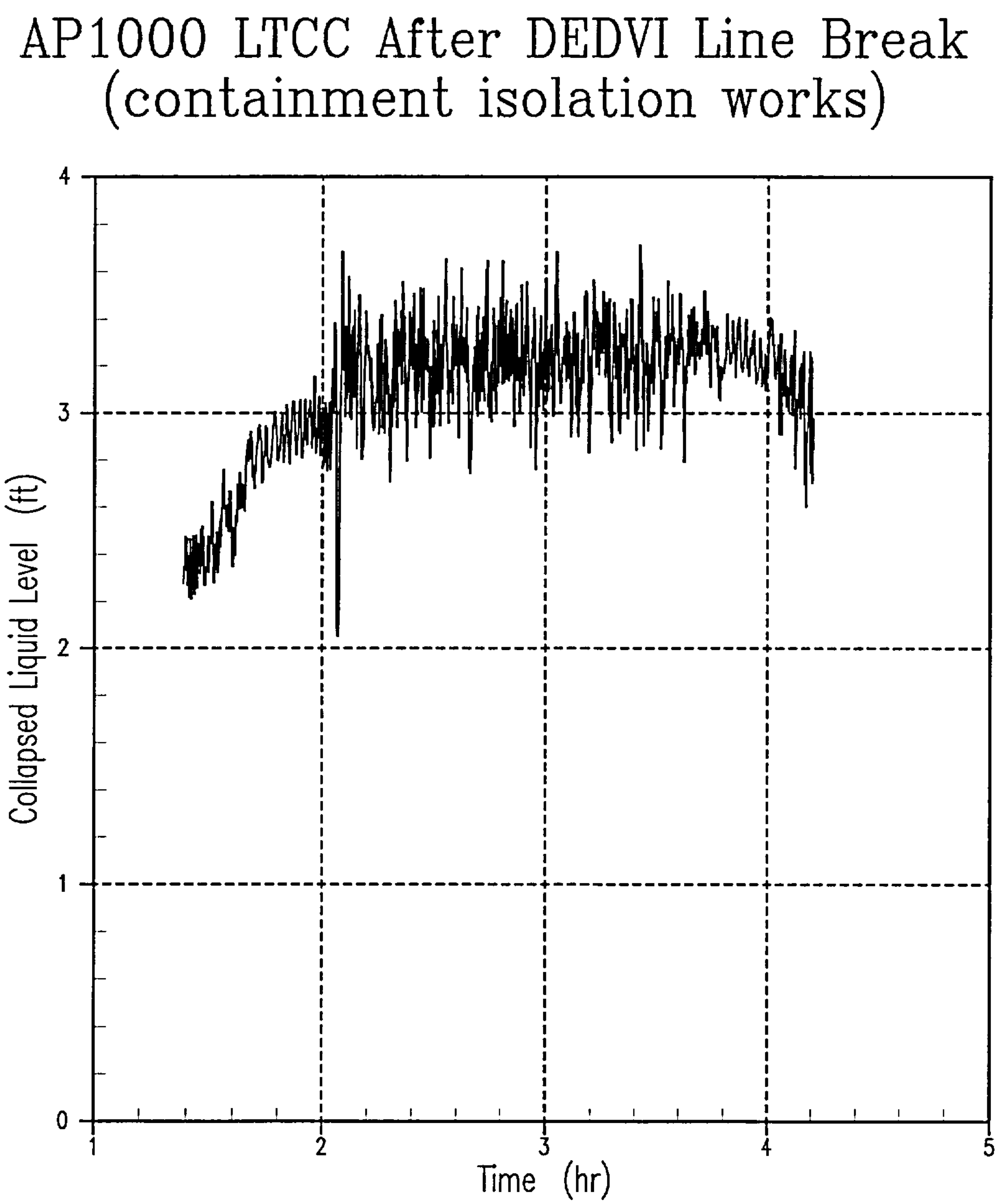


Figure A5.3-8

Case F – Collapsed Liquid Level in the Upper Plenum

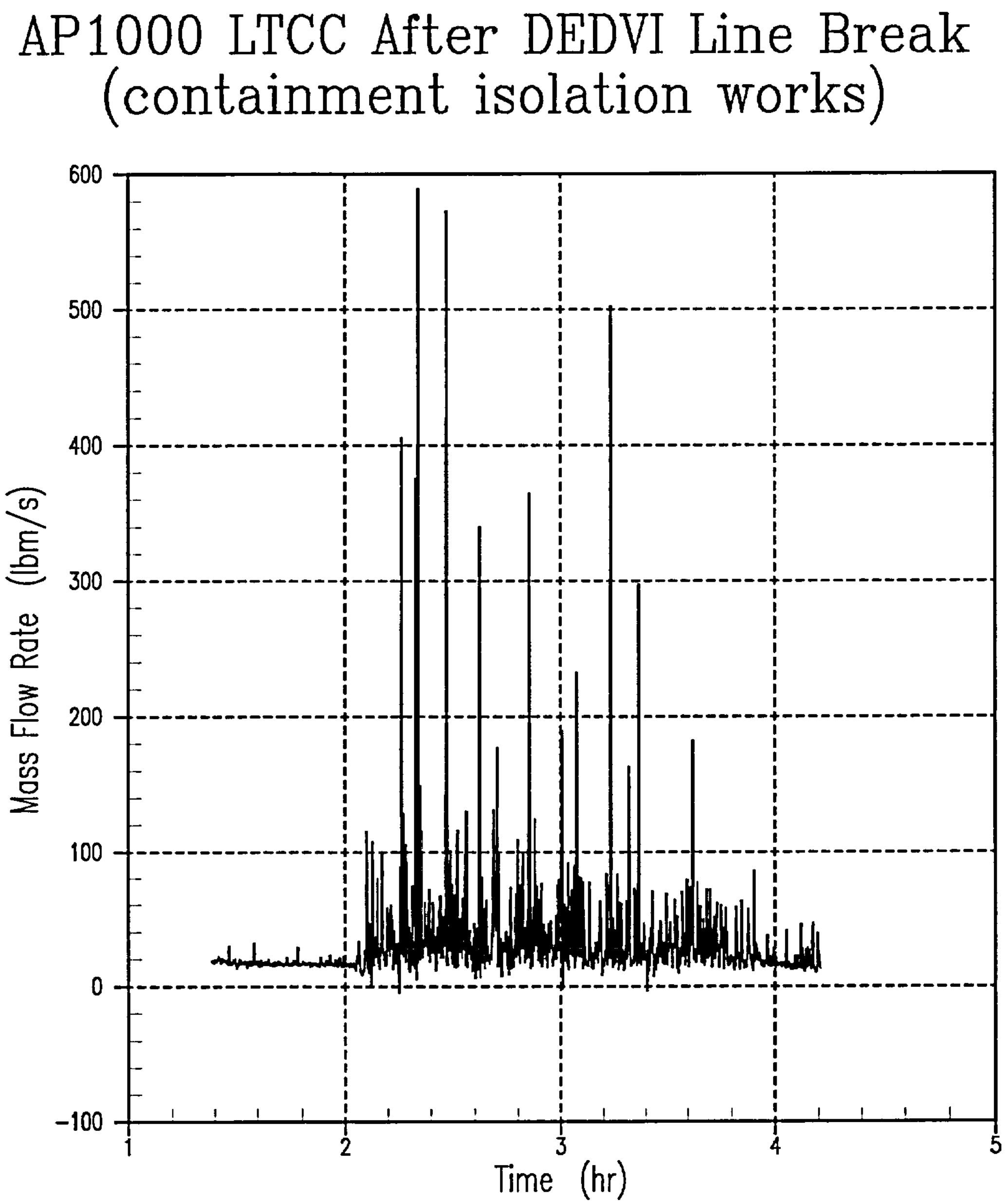


Figure A5.3-9

Case F – Mixture Flowrate Through ADS Stage 4A Valves

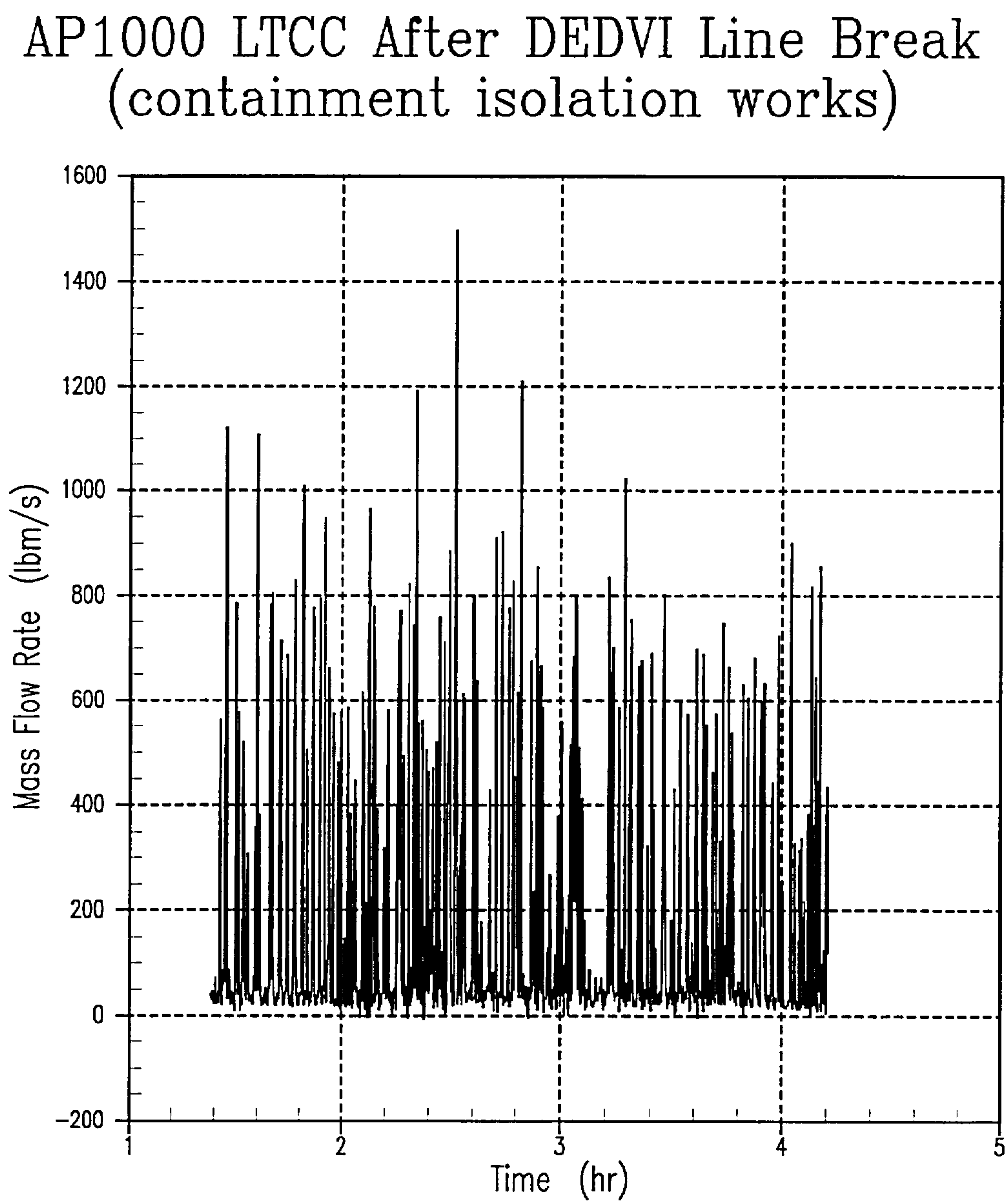


Figure A5.3-10

Case F – Mixture Flowrate Through ADS Stage 4B Valves

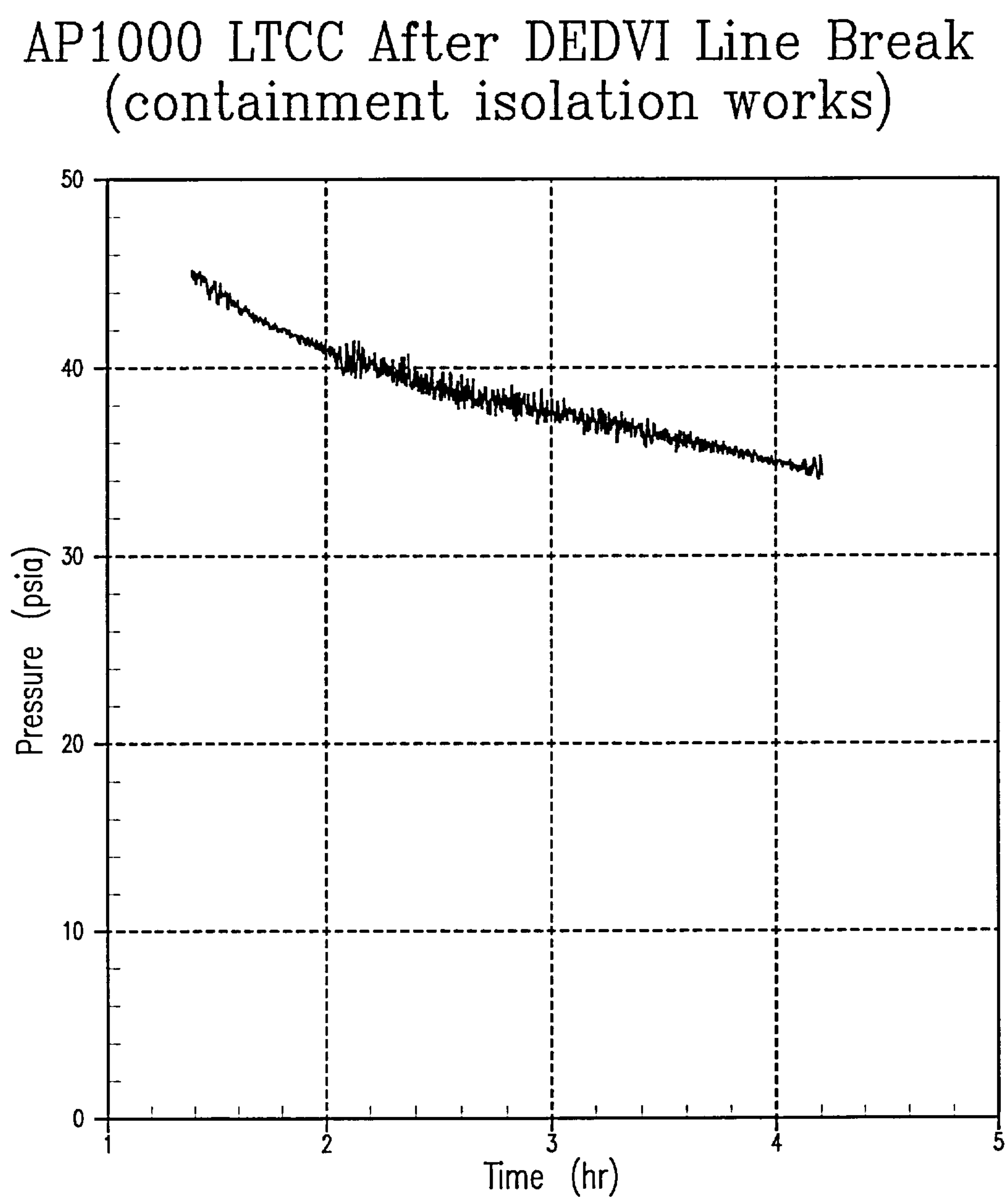


Figure A5.3-11

Case F – Upper Plenum Pressure

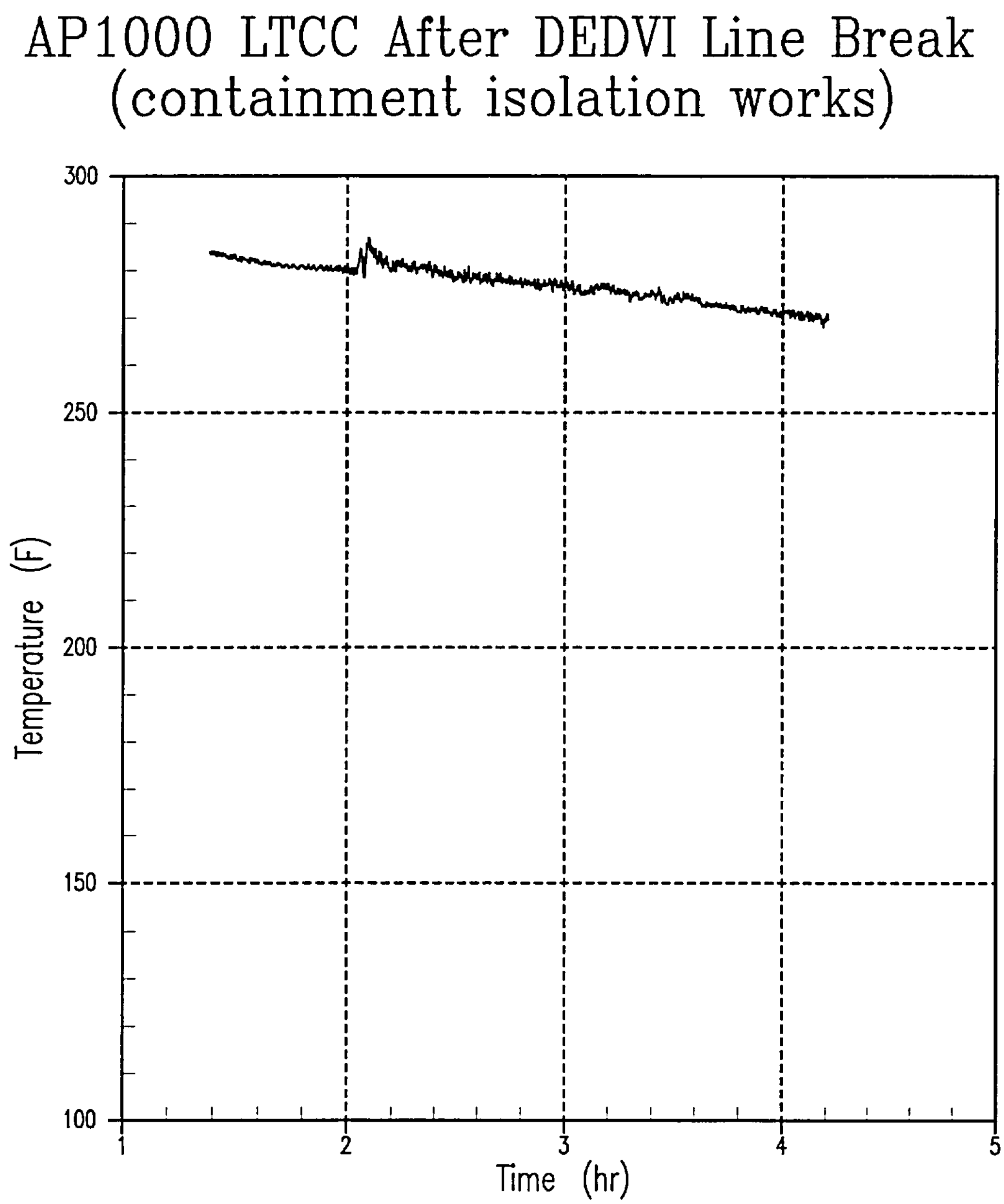


Figure A5.3-12

Case F – PCT of the Hot Rod

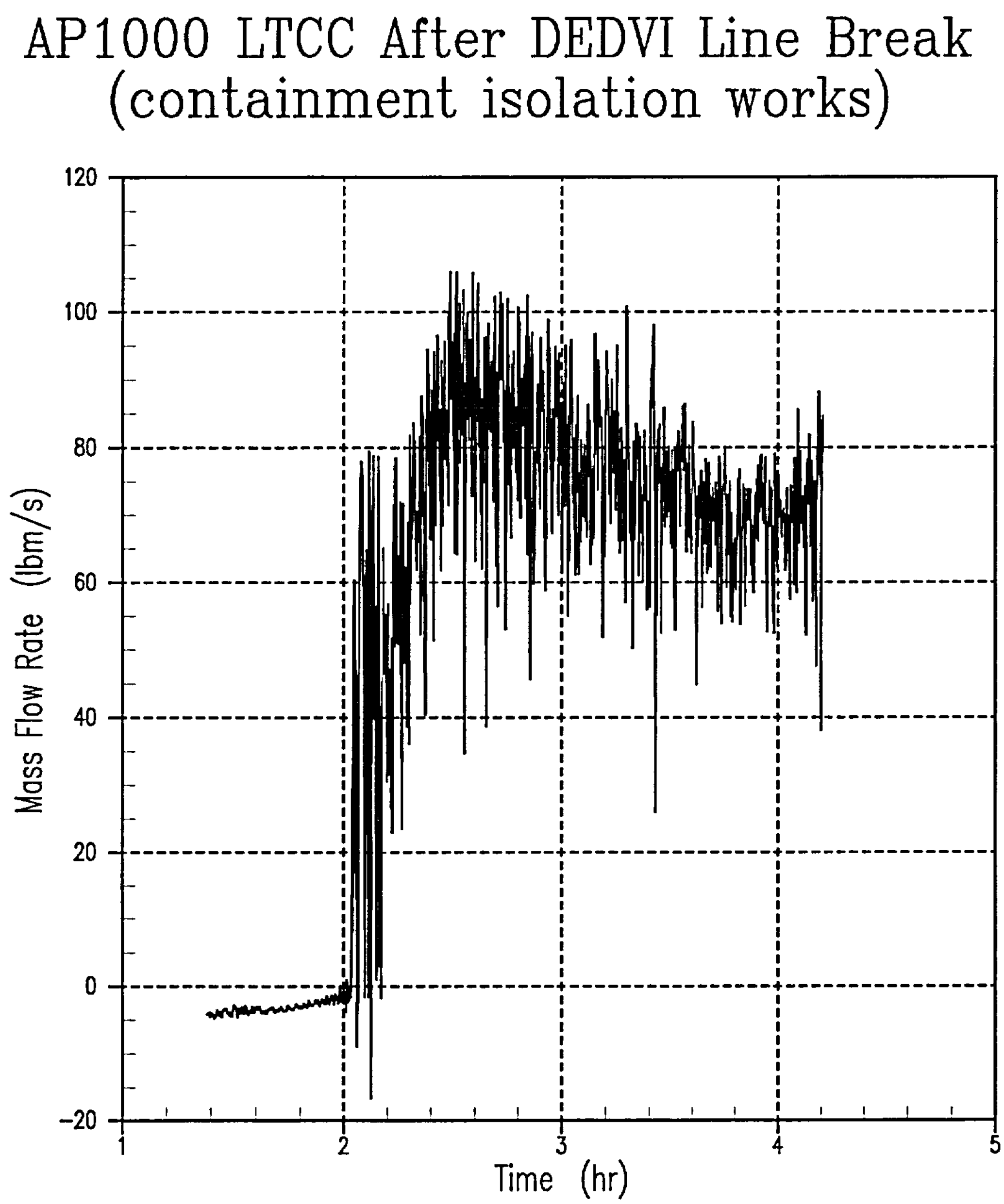


Figure A5.3-13

Case F – DVI-A Mixture Flow Rate

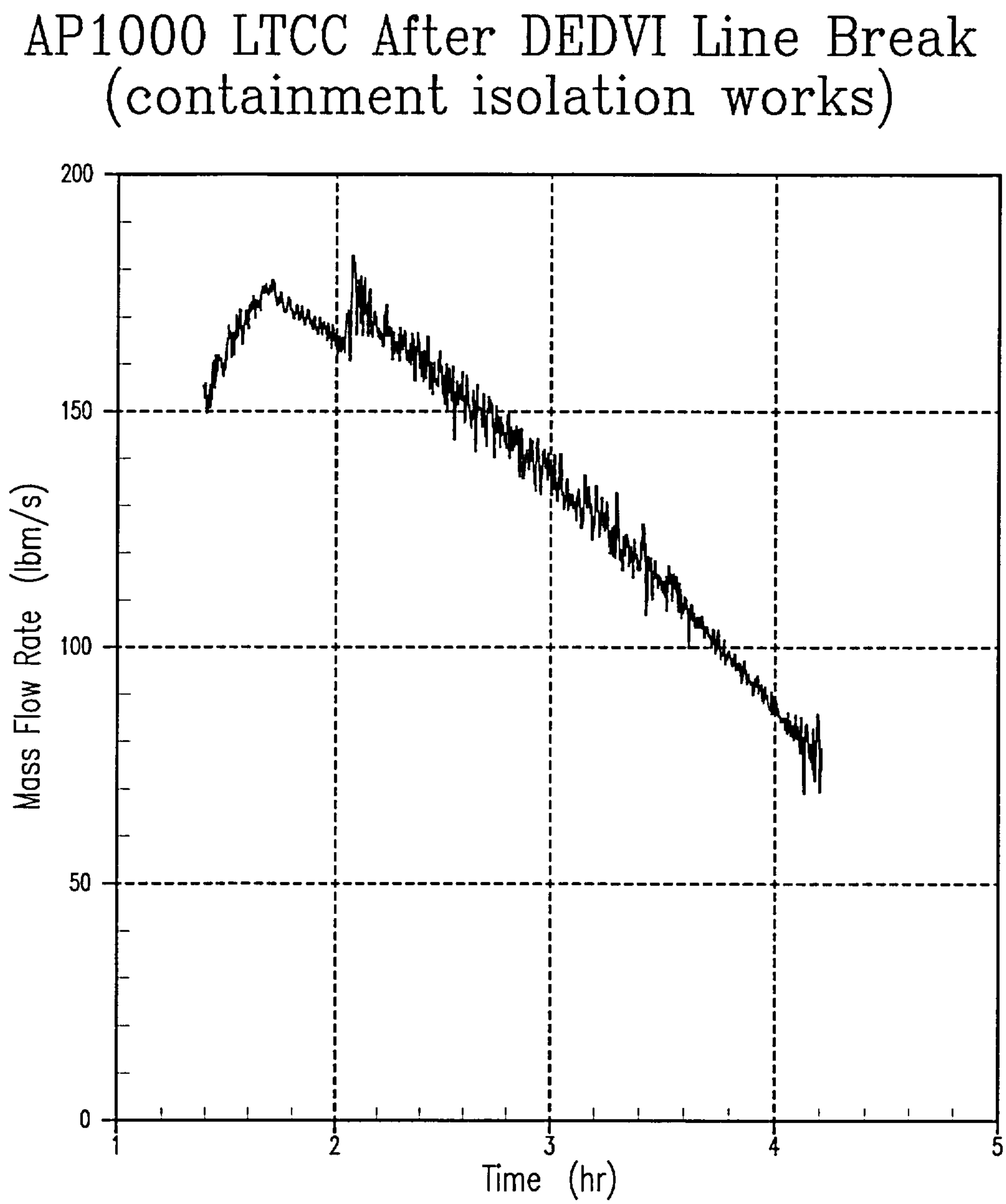


Figure A5.3-14

Case F – DVI-B Mixture Flow Rate

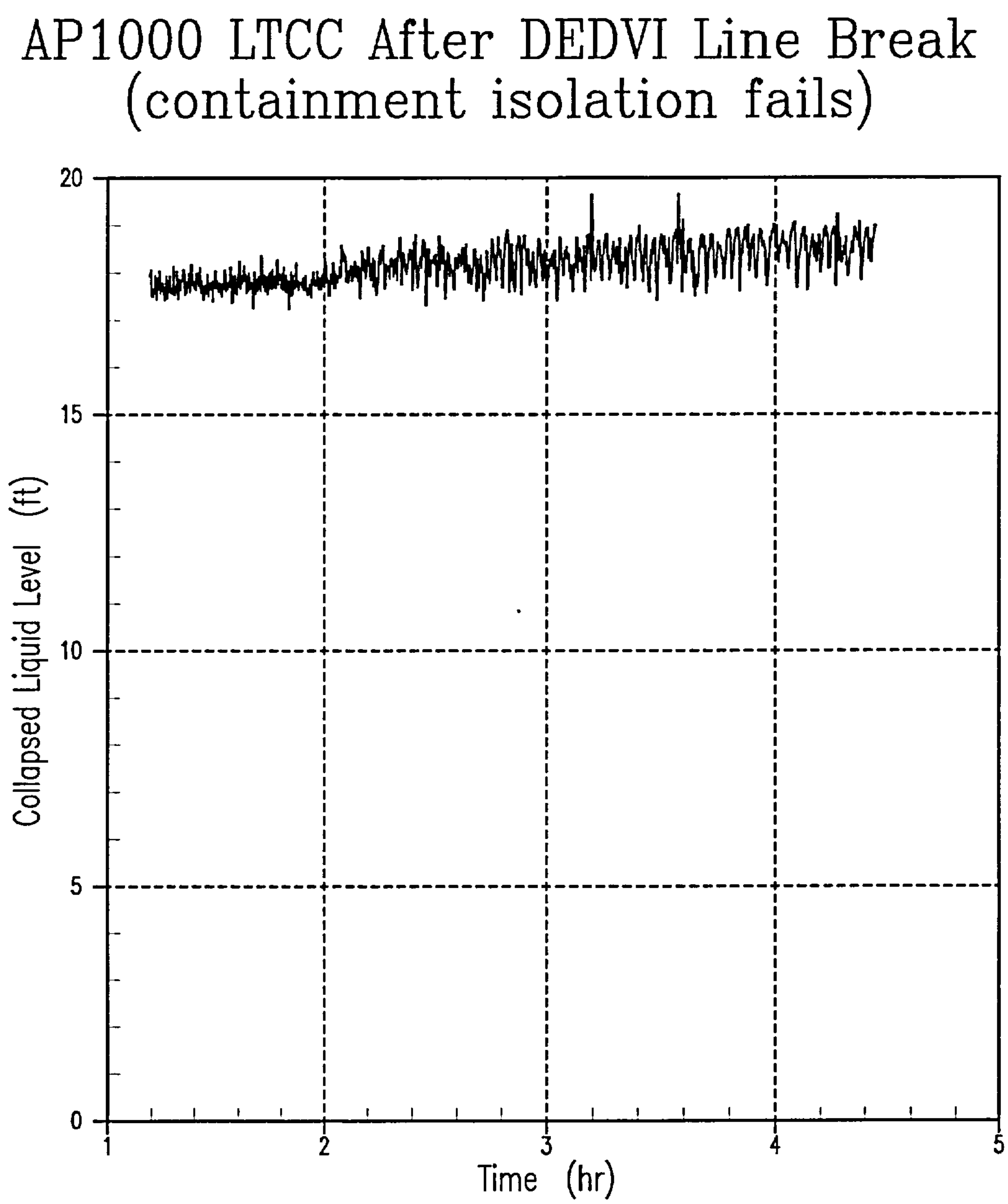


Figure A5.3-15

Case G – Collapsed Level of Liquid in the Downcomer

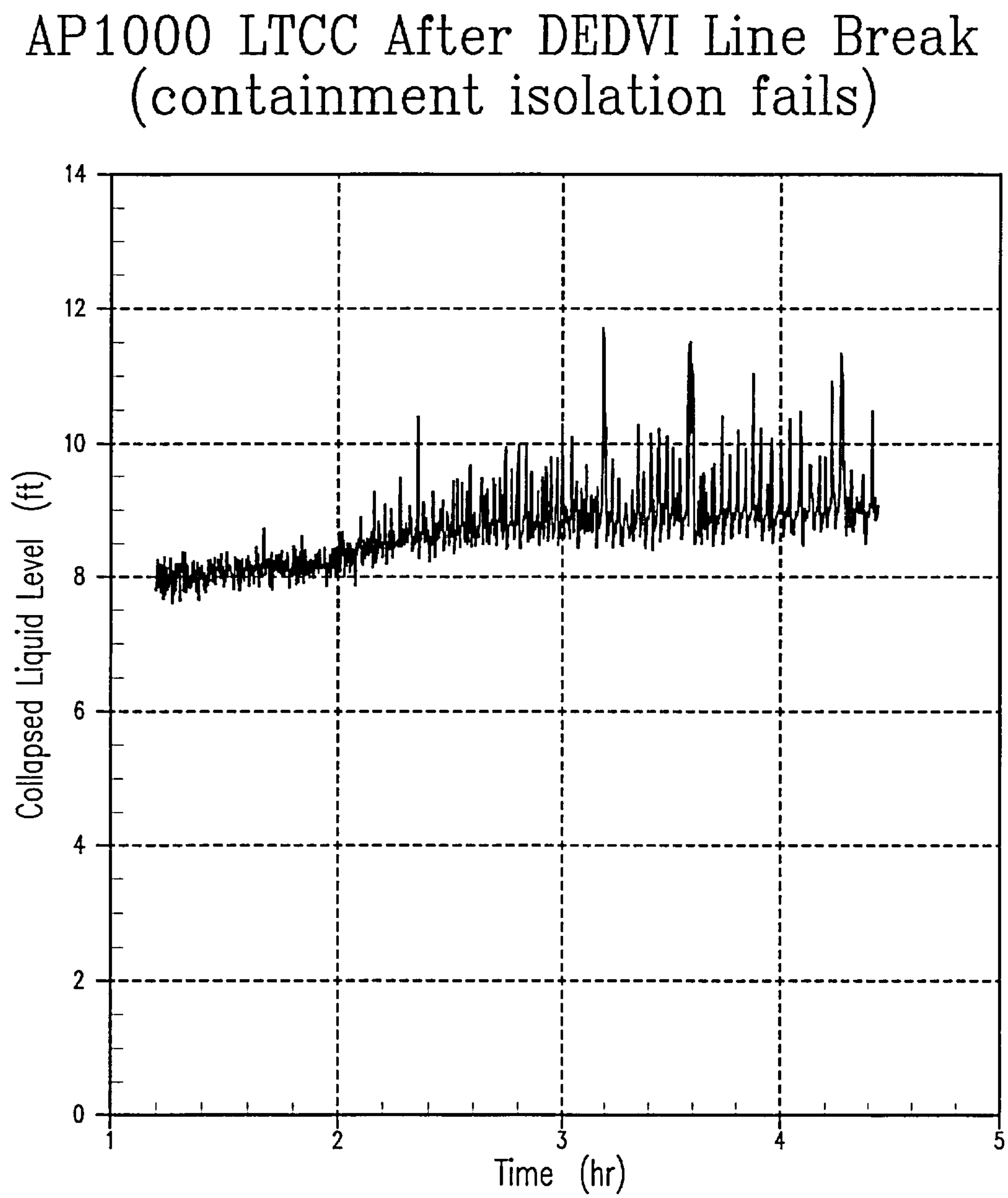


Figure A5.3-16

Case G – Collapsed Level of Liquid Over the Heated Length of the Fuel

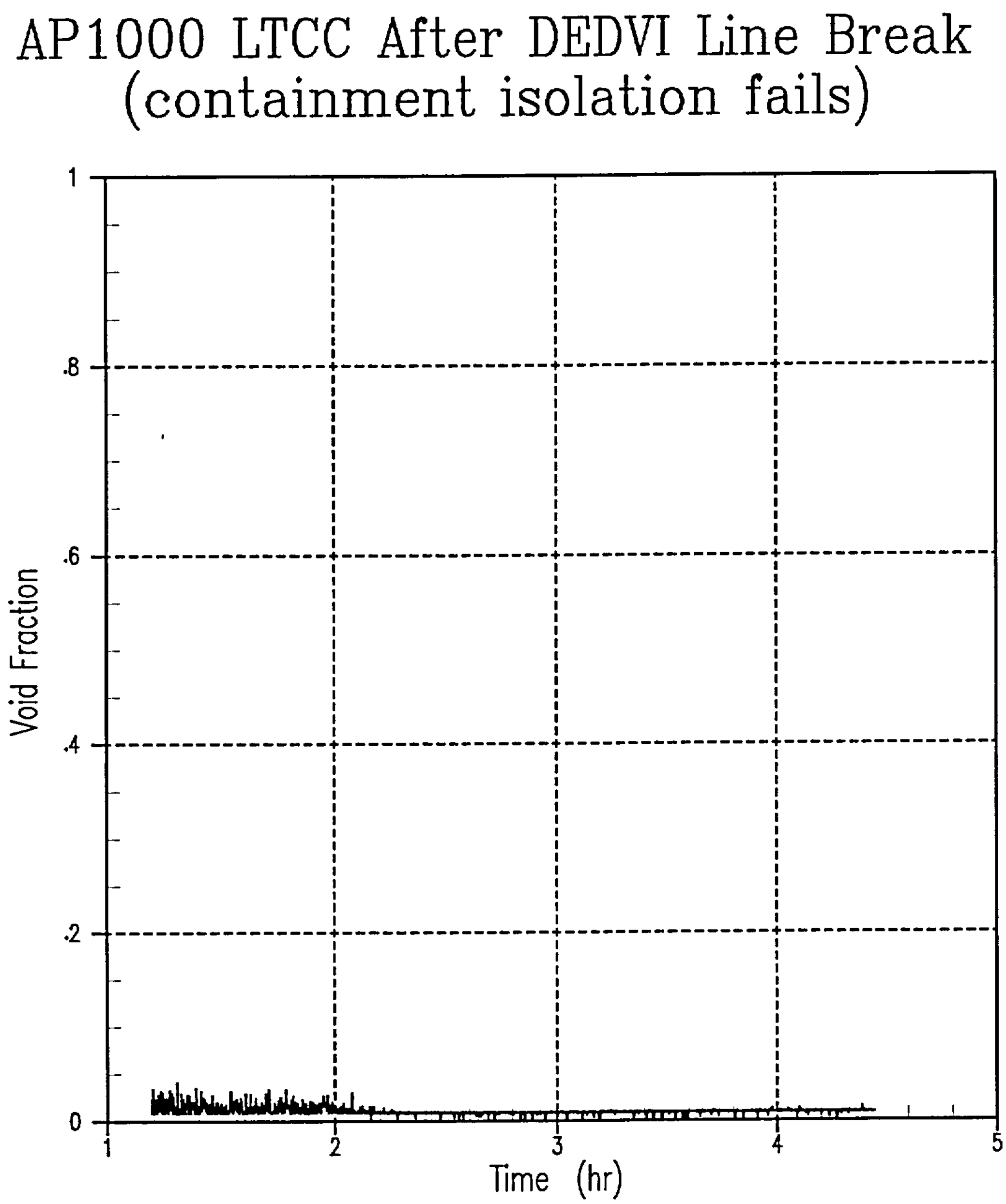


Figure A5.3-17

Case G – Void Fraction in Core Cell Level 1 of 2

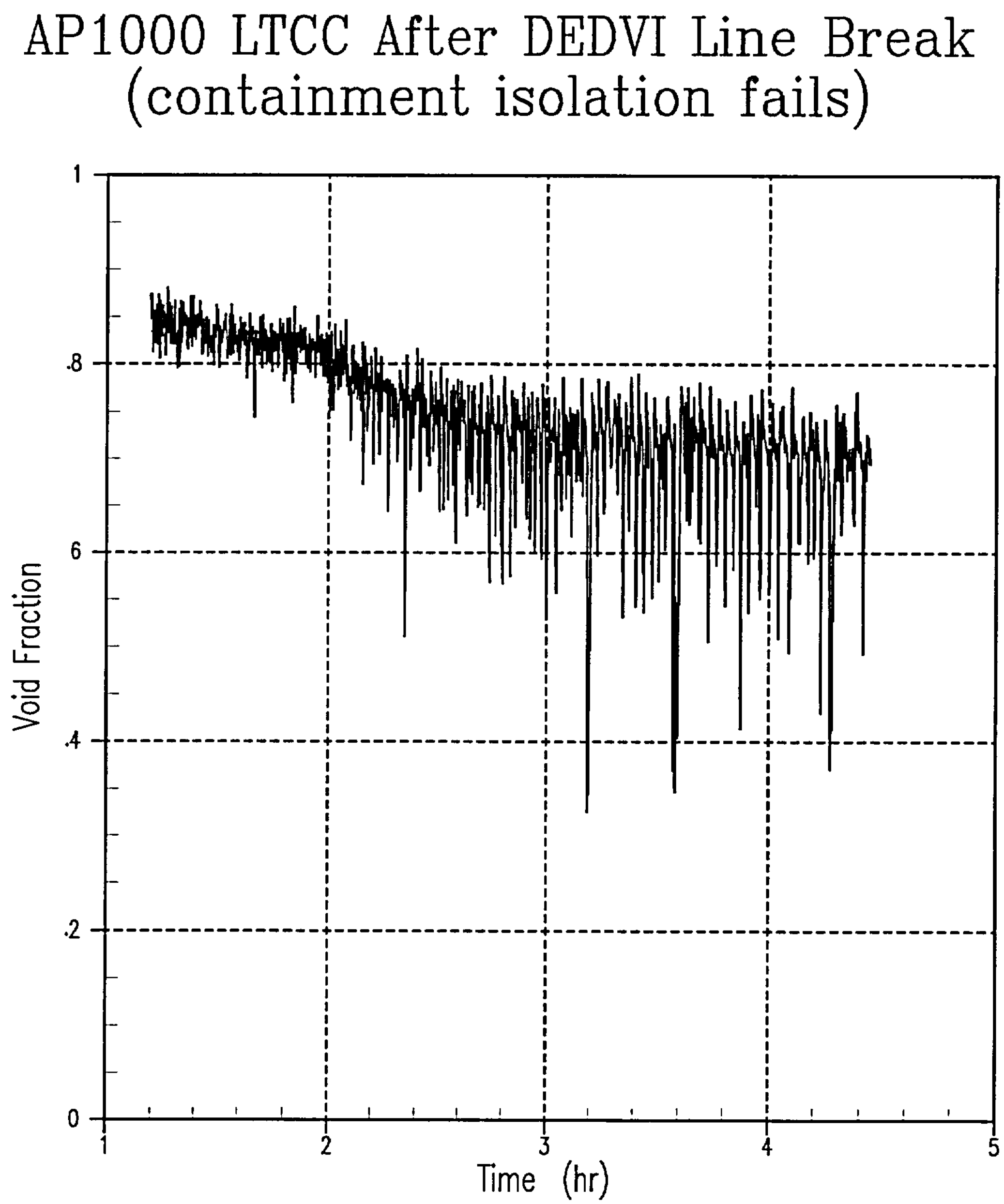


Figure A5.3-18

Case G – Void Fraction in Core Cell Level 2 of 2

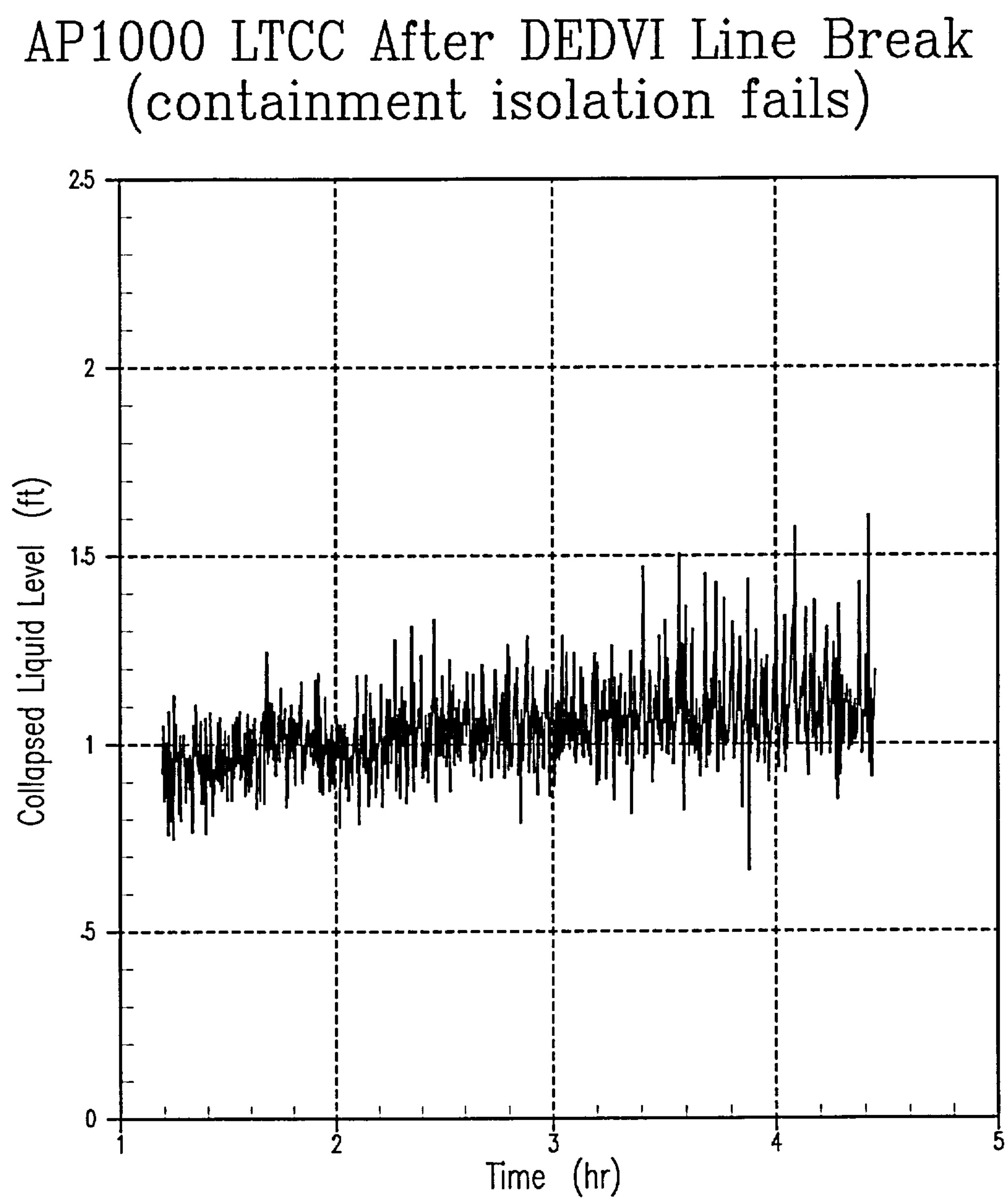


Figure A5.3-19

Case G – Collapsed Liquid Level in the Hot Leg of Pressurizer Loop

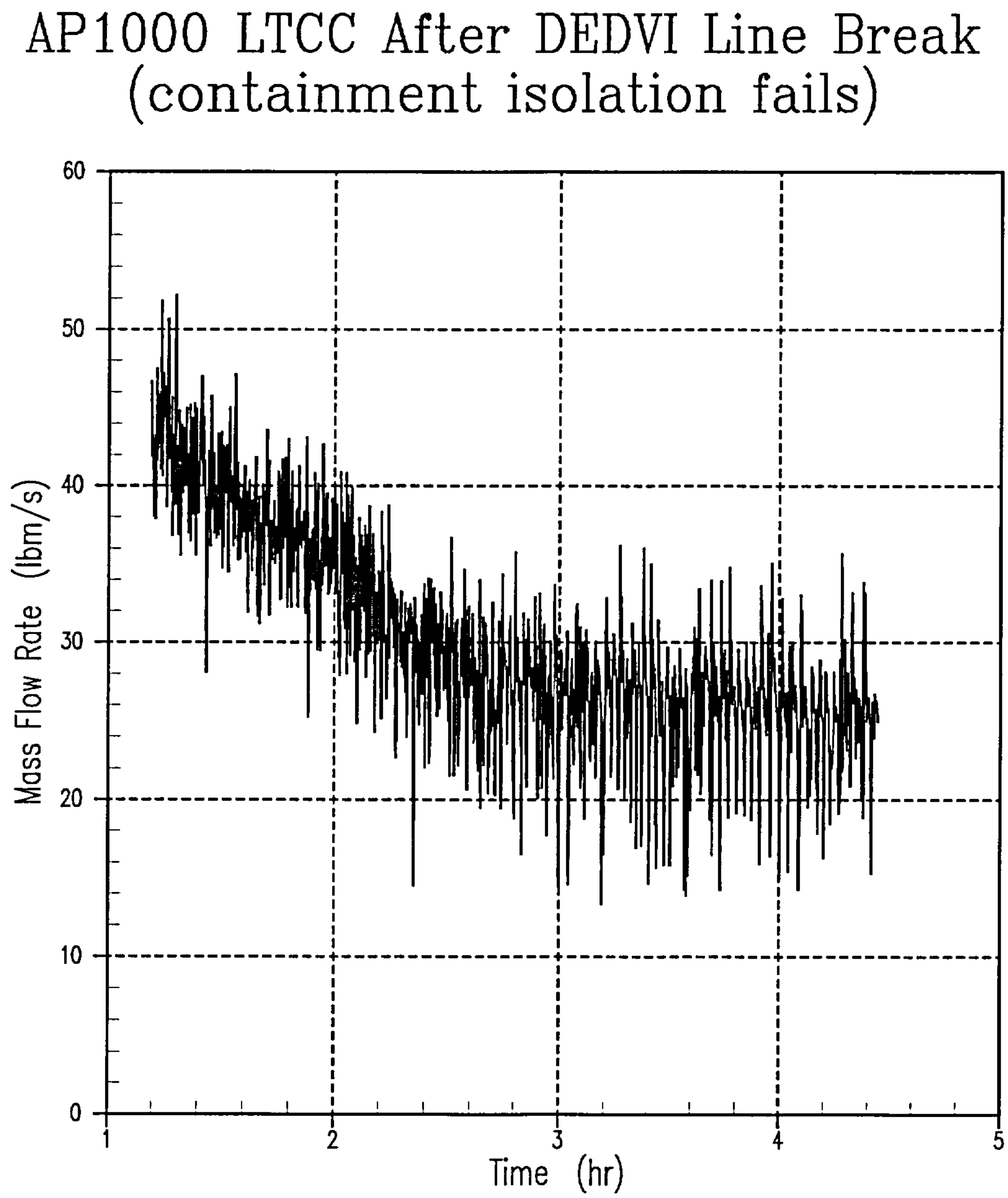


Figure A5.3-20

Case G – Vapor Rate out of the Core

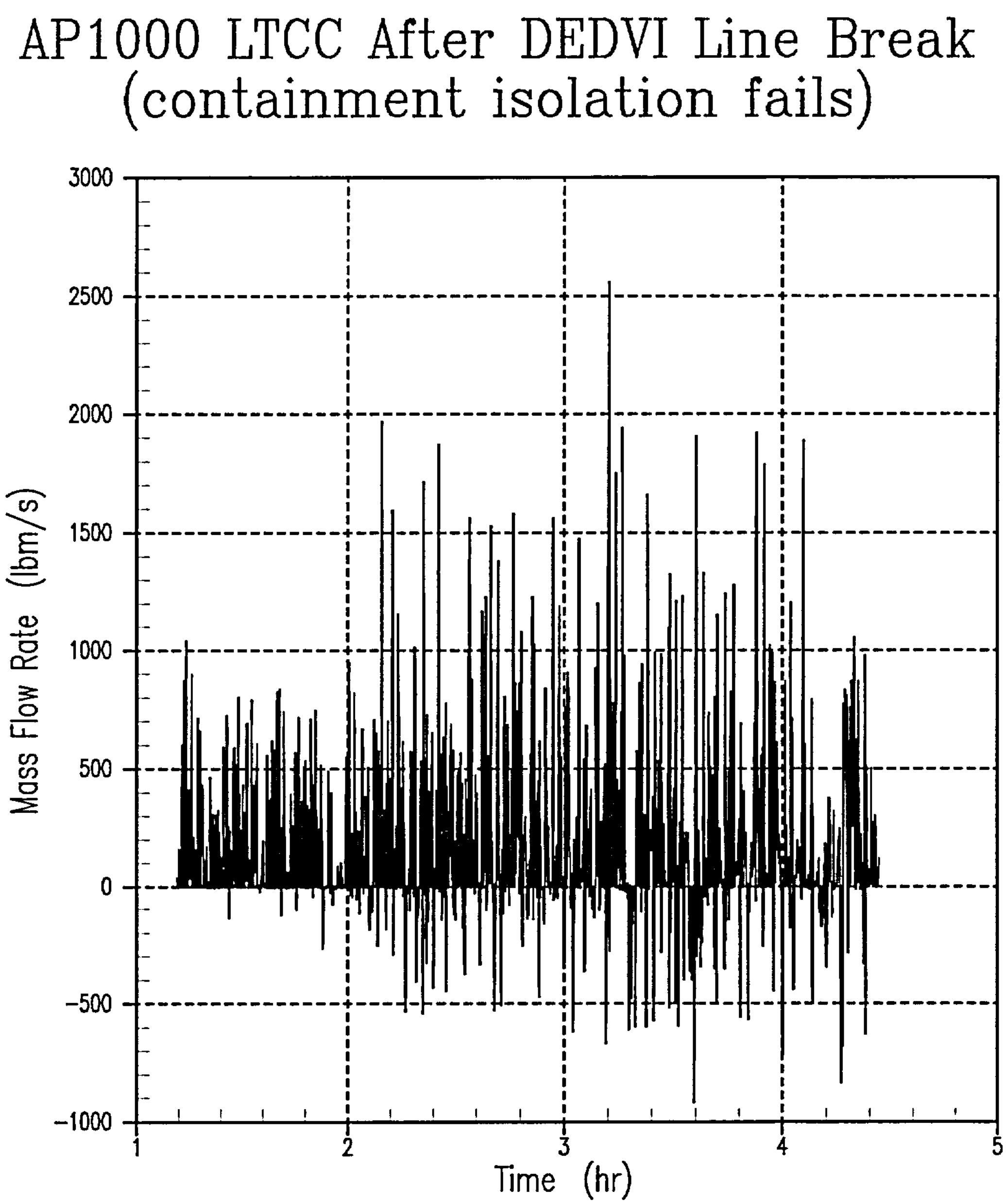


Figure A5.3-21

Case G – Liquid Flow Rate Out of the Core

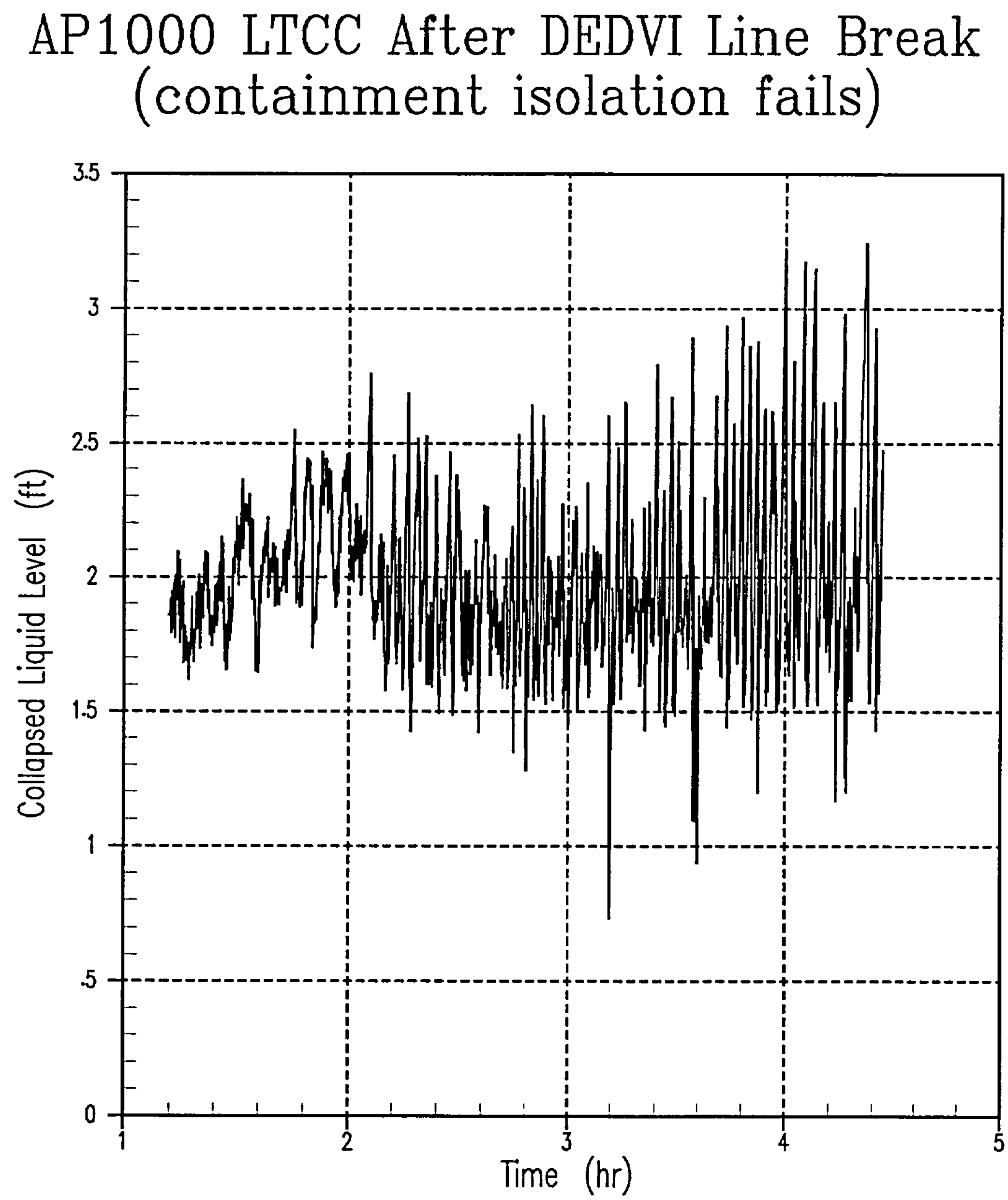


Figure A5.3-22

Case G – Collapsed Liquid Level in the Upper Plenum

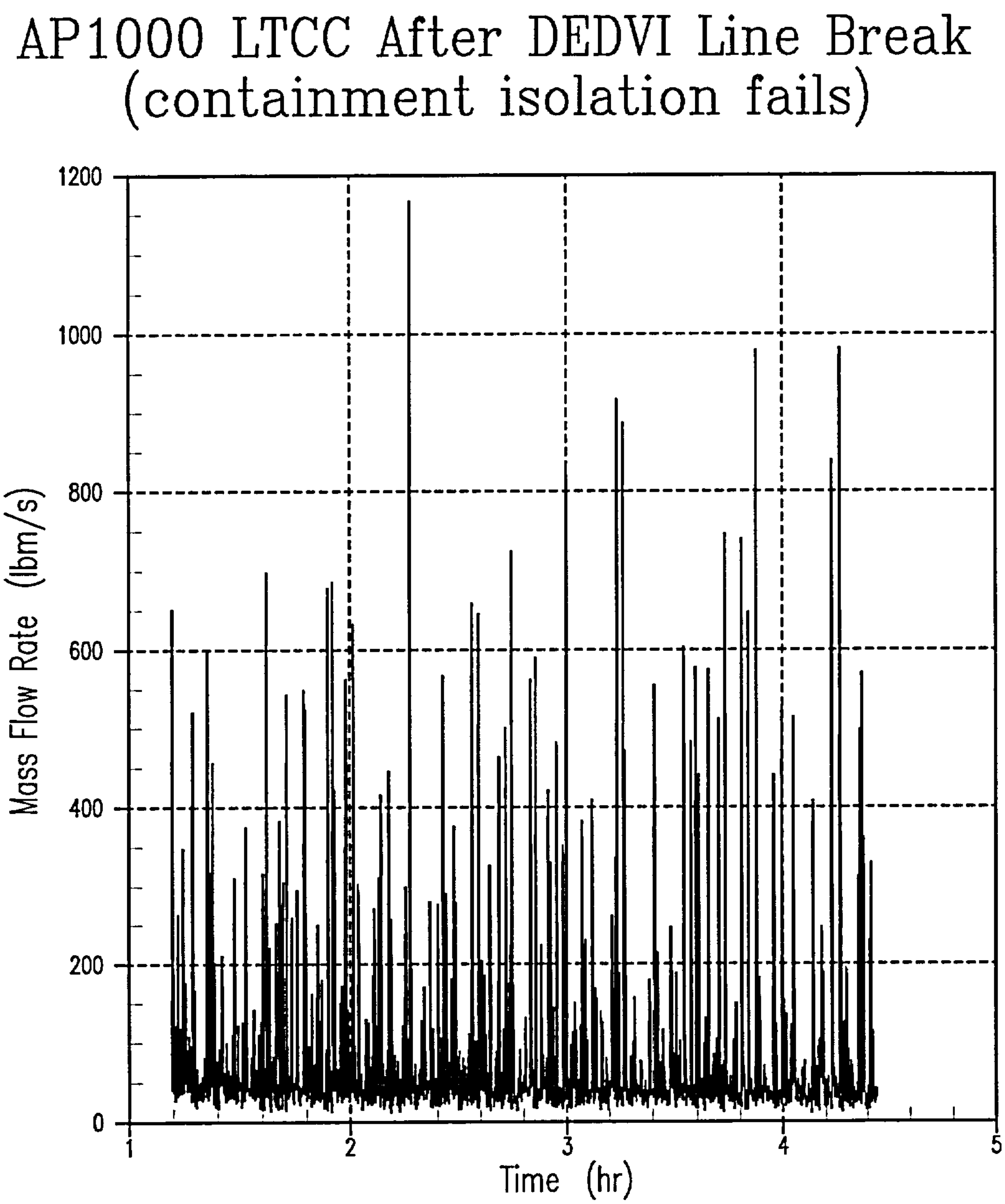


Figure A5.3-23

Case G – Mixture Flowrate Through ADS Stage 4A Valves

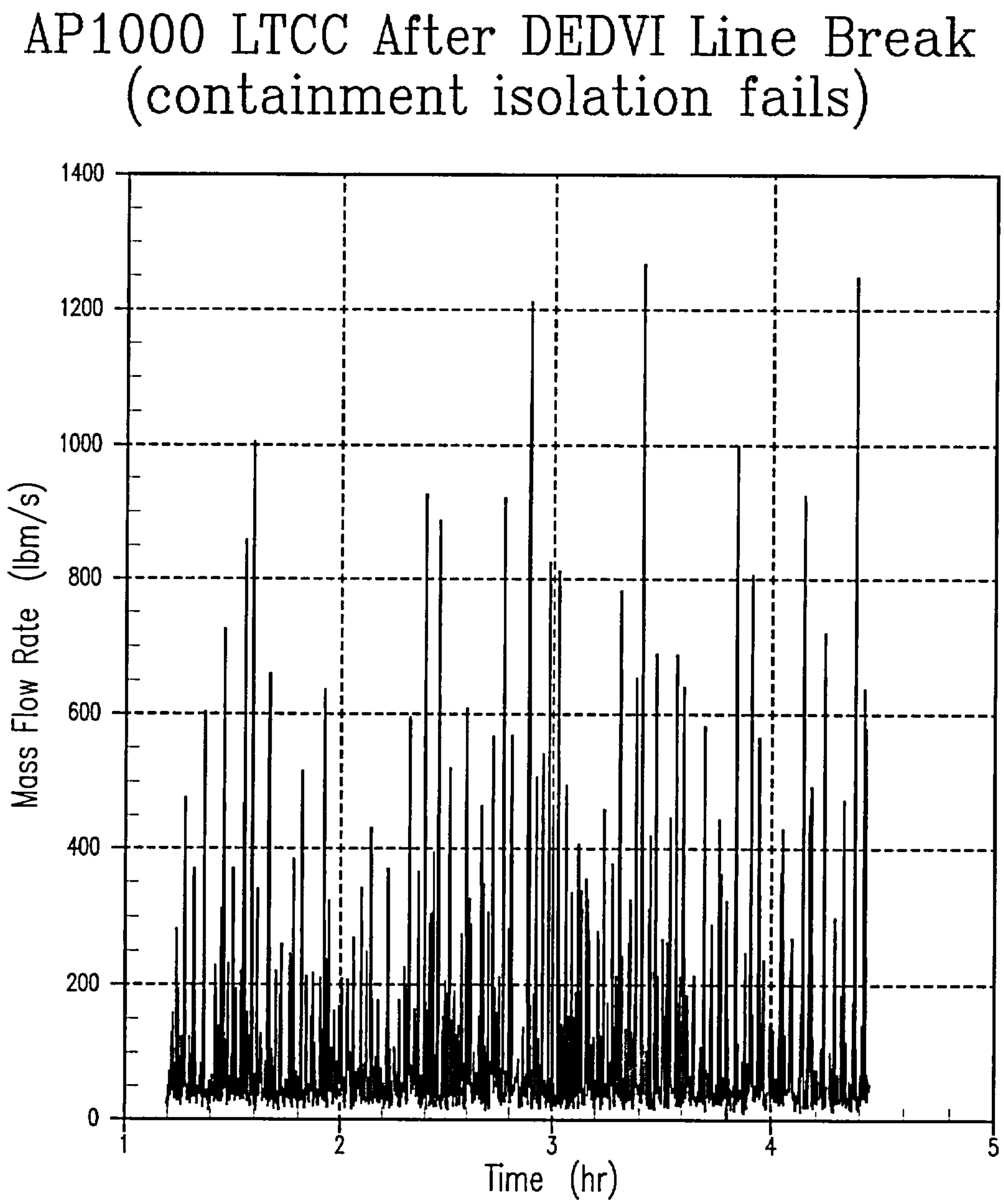


Figure A5.3-24

Case G – Mixture Flowrate Through ADS Stage 4B Valves

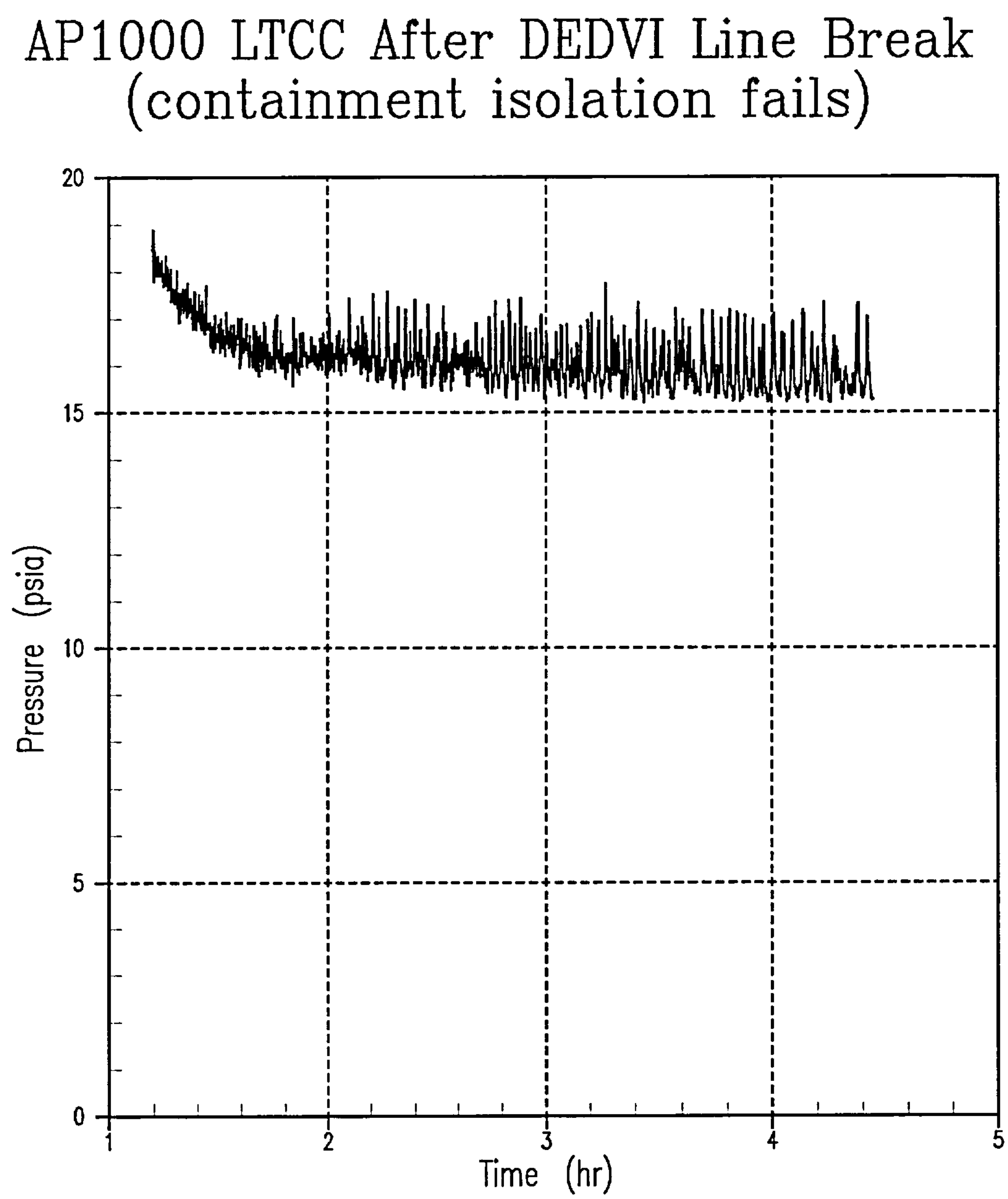


Figure A5.3-25

Case G – Upper Plenum Pressure

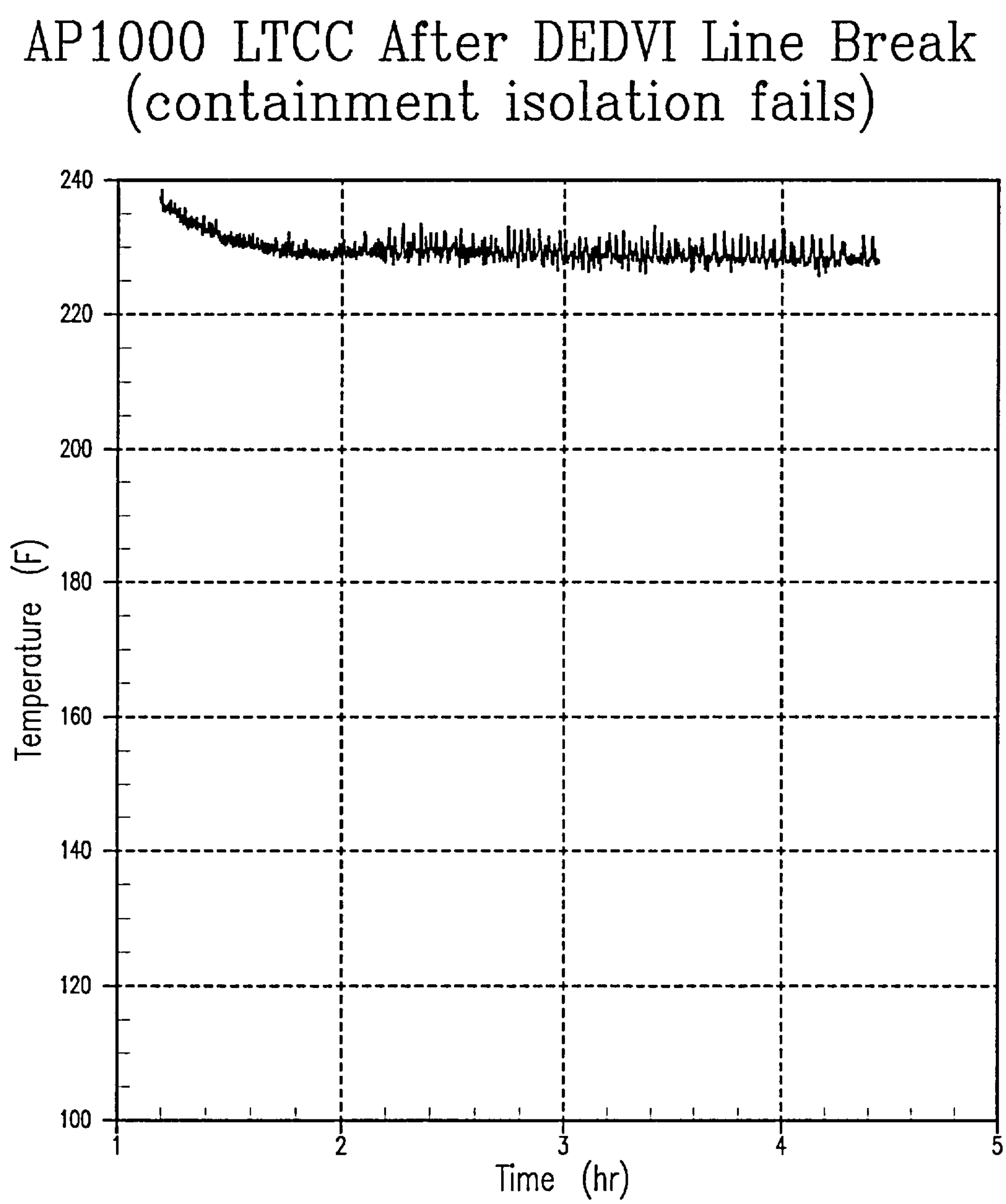


Figure A5.3-26

Case G – PCT of the Hot Rod

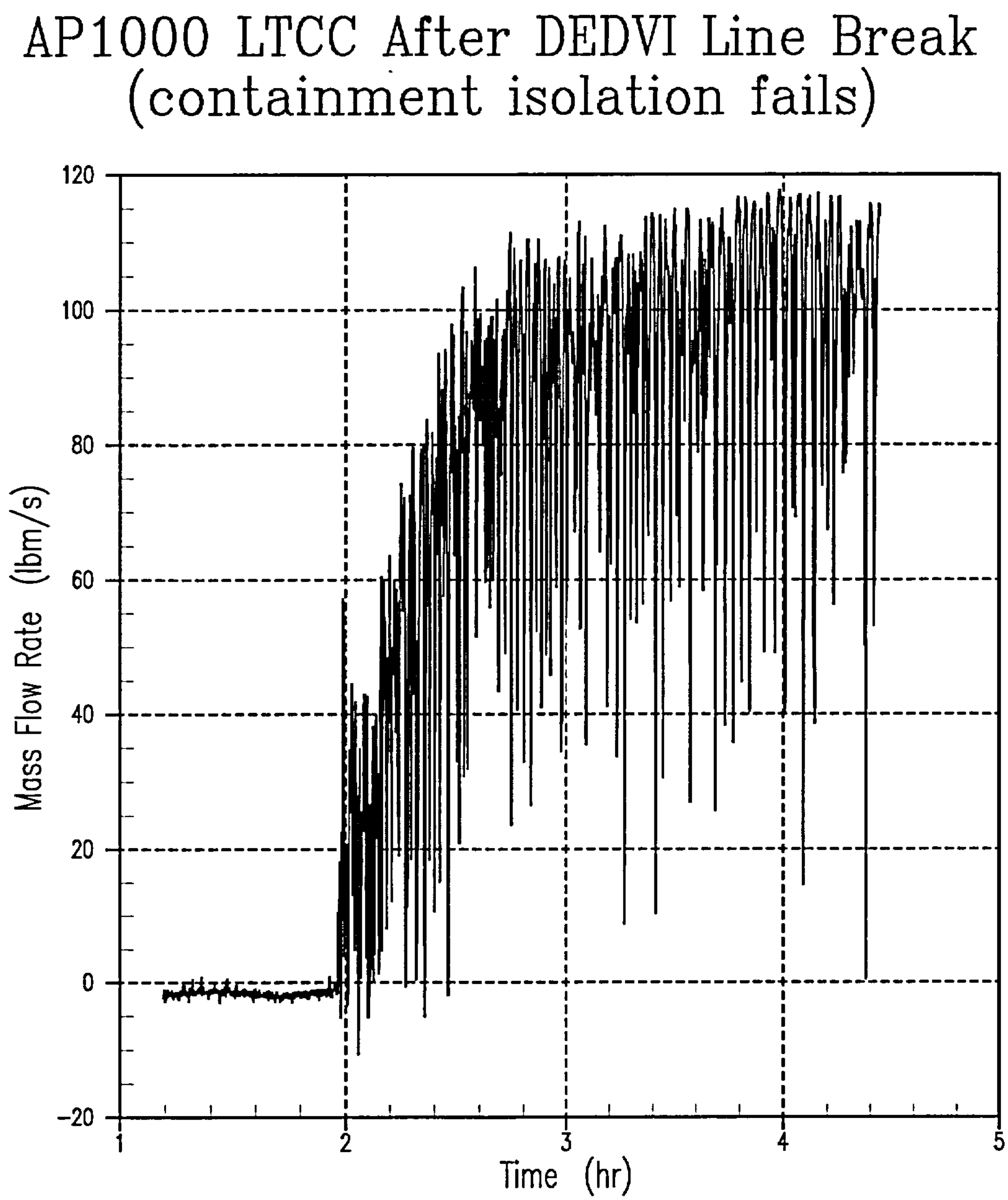


Figure A5.3-27

Case G – DVI-A Mixture Flow Rate

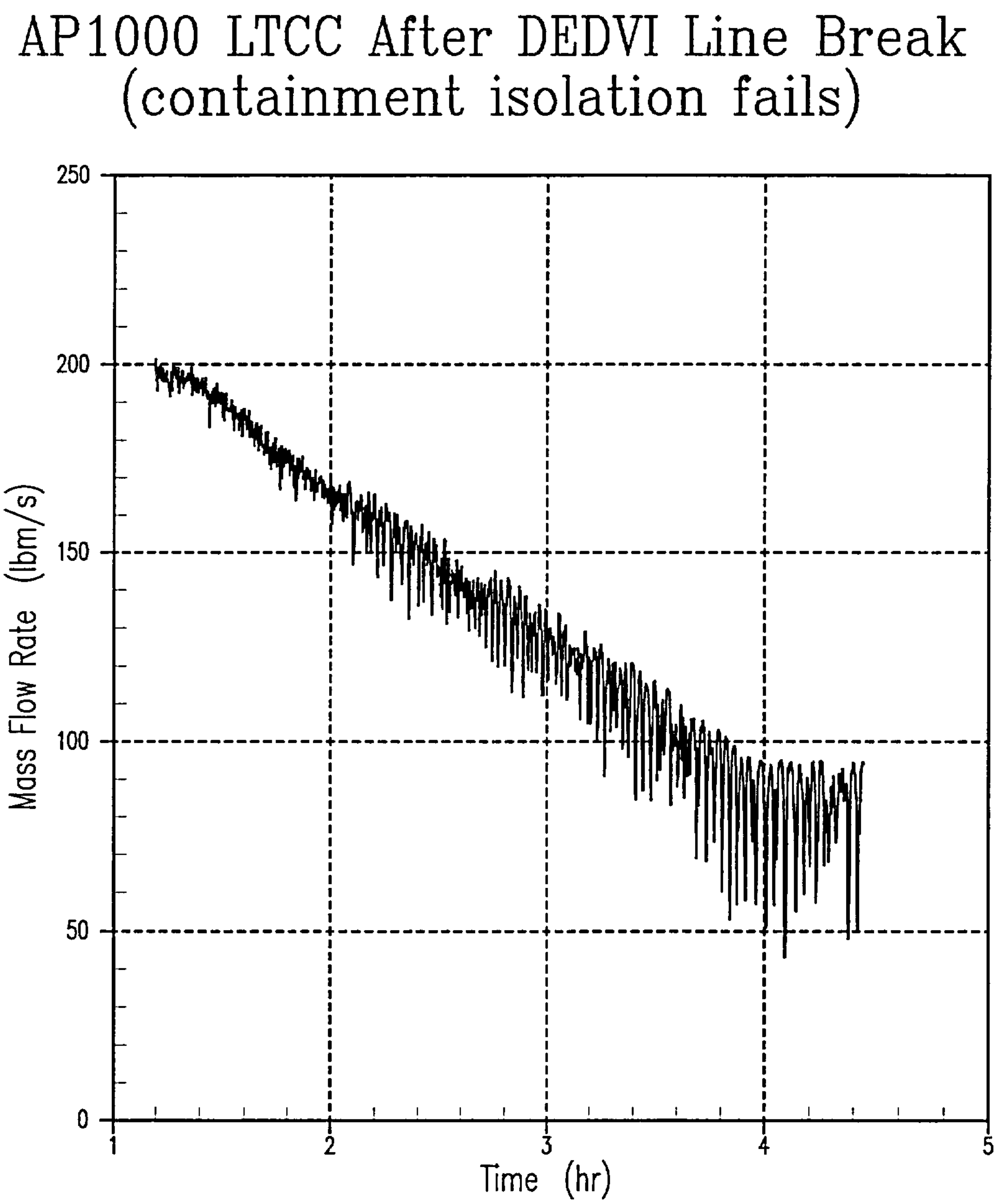


Figure A5.3-28

Case G – DVI-B Mixture Flow Rate

APPENDIX B

EX-VESSEL SEVERE ACCIDENT PHENOMENA

One of the key AP1000 severe accident design features is the capability to retain the core debris within the reactor vessel for a large number of severe accident sequences by flooding the reactor cavity and submerging the outer surface of the reactor vessel. The heat removal capability of the water on the external surface of the reactor vessel prevents the reactor vessel wall from reaching temperatures where failure of the reactor vessel could occur. This has been termed in-vessel retention (IVR) and is described in detail in Chapter 39 of the AP1000 Level 2 PRA. The primary benefit of in-vessel retention of the core is that ex-vessel severe accident phenomena associated with relocation of core debris to the containment, which can be a dominant containment failure mechanism, are physically prevented. Thus, retention of the core within the reactor vessel results in a significant reduction in the potential for large fission product releases to the environment for core damage accidents.

The probability of various levels of fission product releases (release categories) has been determined in the AP1000 Level 2 PRA, using a containment event tree which describes the various severe accident phenomena that can impact the fission product release quantities and probability of release. In the quantification of the AP1000 Level 2 PRA it was conservatively assumed that the containment would fail at the time of reactor vessel failure for all core damage sequences in which the core debris could not be retained within the reactor vessel. The two principle ways identified in the Level 2 PRA of retaining the core within the reactor vessel are reflooding the core with water before the core begins to relocate within the reactor vessel and submerging the outer surface of the reactor vessel to the reactor coolant loop nozzles. Using this approach, the regulatory and industry severe accident performance targets for the AP1000 design criteria were met. Therefore, it was considered unnecessary to investigate the consequences of reactor vessel failure on a realistic basis, including quantification of uncertainties.

The AP1000 design includes features to enhance the likelihood of retaining the core within the reactor vessel for severe accident sequences. These features include:

- Depressurization of the reactor coolant system (RCS) in the event of an accident by either automatic or manual actuation of the highly reliable automatic depressurization system (ADS)
- A containment layout wherein the water relieved from the reactor coolant system (either from the ADS discharge or a break in the RCS) accumulates in the reactor cavity region
- The capability to manually initiate flooding of the reactor cavity by gravity draining the in-containment refueling water storage tank (IRWST) into the reactor cavity
- The absence of in-core penetrations in the reactor vessel bottom head eliminates a possible reactor vessel failure mode
- The reactor cavity layout provides for rapid flooding of the reactor vessel to the reactor coolant loop nozzle elevation

- The reactor vessel insulation design promotes the two-phase natural circulation in the vessel cooling annulus

- The external reactor vessel surface treatment promotes wetability of the vessel

Some of the AP1000 design features to reduce the probability of a core damage accident and to enhance the likelihood of in-vessel retention of core debris in the event of a core damage accident are counter to the design philosophy that would be used to mitigate the consequences of ex-vessel severe accident phenomena. In particular, two of the design features are mutually exclusive between preventing ex-vessel phenomena and mitigating the consequences of ex-vessel phenomena. On balance, the AP1000 severe accident risk profile is substantially reduced by the features that prevent ex-vessel severe accident phenomena. Two of the more noteworthy features are:

- The large mass of the AP1000 core provides for a slower accident progression, which enhances the capability to prevent a core damage accident (i.e., a reduced core damage frequency). The larger mass of core materials may result in more severe consequences from some of the potential ex-vessel phenomena such as core debris coolability and core concrete interactions.

- The small reactor cavity floor area reduces the amount of water required to completely submerge the reactor vessel. The small cavity floor area also provides for a more rapid flooding of the cavity if manual initiation of IRWST draining to the reactor cavity is required to submerge the reactor vessel. The small reactor cavity floor area may result in more severe consequences from some of the severe accident ex-vessel phenomena such as core debris coolability and core concrete interactions.

The purpose of this section is to provide the results of a limited number of deterministic investigations of the consequences of ex-vessel severe accident phenomena for the AP1000 design. The results of these deterministic investigations show that the challenges to the integrity of the containment posed by ex-vessel severe accident phenomena are generally within the structural capability of the containment. From these investigations, the conclusion is the capability to prevent large fission product releases to the environment does not depend on the ability to retain the core within the reactor vessel for core damage accident sequences.

The limited deterministic investigations of ex-vessel severe accident phenomena described in this section includes: ex-vessel steam explosions, direct containment heating and core concrete interactions. These ex-vessel phenomena are strongly dependent on the assumptions made concerning the mode of reactor vessel failure for the AP1000 design. Therefore, the reactor vessel failure mode is described first, followed by a description of the ex-vessel phenomena investigations.

B.1 Reactor Vessel Failure

The AP1000 reactor vessel diameter and hemispherical bottom head configuration are identical to the AP600 reactor vessel. The AP1000 reactor vessel has a main cylindrical section approximately 4 meters in diameter and a hemispherical bottom head. The bottom head is approximately 15 cm (6 inches) thick and is made of carbon steel with an inner cladding of stainless steel to prevent contact between reactor coolant and carbon steel during normal plant operations. The bottom head of the reactor vessel does not contain any discontinuities or penetrations that could impact the mode of reactor vessel failure as the molten core material relocates to the bottom head.

Based on the identical vessel configurations of AP600 and AP1000, the possible failure modes for the AP600 reactor vessel, as documented in Reference B-1, are extended to the AP1000. The most likely failure mode is creep failure of the vessel wall due to heating of the vessel wall by the core debris that has relocated to the reactor vessel bottom head. Since creep failure is a strongly temperature-dependent phenomena, the location of the failure is predicted to be at the upper surface of the core debris pool that has relocated to the reactor vessel bottom head. For most severe accident sequences, this location is near the junction of the hemispherical bottom head and the cylindrical portion of the vessel.

As described in Reference B-2, the presence of water on the external surface of the reactor vessel, as in the case of a flooded reactor cavity, does not alter the conclusion that the highest heat fluxes to the reactor vessel walls will be at a point near the top of the in-vessel molten core pool. This would correspond to the region of the reactor vessel most susceptible to creep failure. However, as described in Chapter 39, reactor vessel failure will not occur for the case in which the reactor coolant system is depressurized and the reactor cavity is filled with water to the reactor coolant loop elevation.

For the case in which the outside of the reactor vessel is initially submerged but a sufficient in-flow of water to the reactor cavity cannot be maintained, the reactor vessel wall location experiencing the highest heat fluxes would uncover and lose its external cooling before other locations on the reactor vessel lower head. Thus, creep failure of the vessel would be expected to occur at the same location as the case with no water in the reactor cavity.

Two reactor vessel failure cases, as described below, are carried through the deterministic analyses of ex-vessel steam explosions and core concrete interactions. For the consideration of ex-vessel steam explosions and core concrete interactions, it is assumed that the reactor vessel is initially submerged in water but that gravity draining of water from the IRWST does not occur. In this case, the heat removed from the reactor vessel by the water in the reactor cavity results in a boil-down of the water in the cavity. As the water in the reactor cavity boils down, the outside of the reactor vessel at the elevation at the top of the in-vessel core pool will dry out and begin to heat up. As the vessel wall heats up, it undergoes thinning due to dissolution and melting until failure occurs. The manner in which the reactor vessel fails is treated in two separate scenarios described below.

In the first scenario, the formation of a localized opening occurs due to asymmetric heating around the circumference followed by the vessel tearing around nearly all of its circumference. This would result in the bottom part of the reactor vessel and the bottom head hinging such that the lower head swings downward and comes to rest on the cavity floor. This behavior is illustrated in Figure B-1. A hinging type of failure would result in an immediate pouring of core debris onto the cavity floor with metal flowing ahead of oxide. The relationship between the height of the reactor vessel above the floor is such that all but a minor part of the oxide melt would be free to flow immediately out of the head.

In the second scenario, the head and bottom part of the vessel do not hinge downward. In this scenario, the formation of a localized opening permits molten core debris to drain into the cavity lowering the in-vessel core debris depth and thereby decreasing the thermal load on the vessel wall formerly adjacent to the melt. This type of failure is illustrated in Figure B-2. In this case, the continued boildown of water level is followed by the release of the core debris located above the water level after a delay interval during which heatup, thinning, and localized failure of the wall will occur. Over time, the elevation of the failure location moves downward over the vessel wall and lower head. This type of failure gives rise to a very slow release rate with the core debris first relocating downward through the water before collecting and spreading on the cavity floor.

For both mechanistic reactor vessel failure scenarios, the condition of the core debris inside the reactor vessel at the time of vessel failure is defined in Table B-1.

B.2 Direct Containment Heating

Direct containment heating (DCH) is defined as the rapid energy addition to the containment atmosphere as a result of several physical and chemical processes that can occur if the core debris is forcibly ejected from the reactor vessel. The prerequisites for direct containment heating are vessel failure occurs at a location where a substantial portion of the core debris that has relocated to the lower head is ejected into the reactor cavity before the RCS gases are discharged from the RCS and the RCS is at a high pressure (sometimes called high pressure melt ejection or HPME).

To preclude the potential for high-pressure core melt ejection leading to containment failure via DCH, SECY-93-087 (Reference B-5) directs passive light water reactor (LWR) designs to:

- Provide a reliable depressurization system

- Provide cavity design features to decrease the amount of ejected core debris that reaches the upper compartment

The AP1000 design incorporated design features that prevent high-pressure core melt. These features include the passive residual heat removal (PRHR) system and the ADS, both subsystems of the passive core cooling system (PXS). Depressurization of the AP1000 RCS in the event of an accident is provided by automatic or manual actuation of the ADS. Redundancy and diversity are included within the ADS design to ensure a highly reliable depressurization system. The ADS consists of four different valve stages that open sequentially to reduce reactor coolant system pressure in a controlled fashion. All four-valve stages are arranged into two identical groups. Different valve types/sizes are utilized within the ADS stages to provide diversity. Based on these ADS design features, a highly reliable depressurization system is provided which precludes the potential for high-pressure core melt ejection in the AP1000 design. The AP1000 PRHR and ADS subsystems are described in additional detail in Chapters 8 and 11 of the PRA and in Section 6.3 of the *AP1000 Design Control Document* (DCD).

Even though high-pressure core melt ejection is not a likely scenario for the AP1000, SECY-93-087 directs passive LWR designs to include cavity design features to decrease the amount of ejected core debris from reaching the upper compartment. The AP1000 design includes design features to retain and quench the core debris within the reactor cavity in the unlikely event of core debris relocation outside the reactor vessel. These features include:

- A containment layout wherein the water accumulates in the reactor cavity region
- The capability to manually initiate flooding of the reactor cavity by gravity draining the IRWST into the reactor cavity
- The reactor cavity geometry is arranged to provide a torturous pathway from the reactor cavity to the loop compartment and no direct pathway for the impingement of debris on the containment shell

B.3 Ex-Vessel Steam Explosions

The first level of defense for ex-vessel steam explosion is the in-vessel retention of the molten core debris. If molten debris does not relocate from the vessel to the containment, there are no conditions for ex-vessel steam explosion. In the event that the reactor cavity is not flooded and the vessel fails, the PRA containment event tree assumes that the containment fails in the early time frame.

An analysis of the structural response of the reactor cavity was performed for the AP600 (Reference B-3). As in the in-vessel steam explosion analysis, the results of this AP600 ex-vessel steam explosion analysis are extended to the AP1000. The vessel failure modes for AP600 and AP1000 are the same. The initial debris mass participating in the interaction, superheat and composition are assumed to be the same as for AP600. The mass assumption is conservative since the AP1000 reactor vessel lower head is closer to the cavity floor resulting in less debris mass participating in the interaction. The reactor cavity geometry and water depth prior to vessel failure are the same as AP600. Therefore, the results of the AP600 ex-vessel steam explosion analysis are considered to be appropriate for the AP1000.

B.4 Core Concrete Interactions

If the reactor vessel fails when the RCS is at a low pressure, the molten core debris will pour from the reactor vessel onto the reactor cavity floor. If a steam explosion does not occur, the pour will spread over the cavity floor and begin to transfer heat to the concrete floor of the reactor cavity. Due to the predicted mode of reactor vessel failure and the shape of the AP600 reactor cavity, analyses of the possible spreading of the core debris over the cavity floor were conducted using the MELTSPREAD code (Reference B-4). The AP1000 cavity geometry is the same as AP600. In addition, the AP1000 initial debris location from the vessel to the cavity is similar to AP600 in terms of mass flowrate and superheat, and therefore, the MELTSPREAD analyses performed for AP600 can be extended to AP1000. The results of the MELTSPREAD analyses were used as input to the MAAP4 code for analysis of core concrete interactions for AP1000.

The AP1000 reactor cavity is at containment elevation 71′ 6″ and consists of two interconnected volumes. The volume, which includes the reactor vessel, is octagonal in shape. The other volume is rectangular in shape and houses the reactor coolant drain tank (RCDT) and also contains the reactor cavity sump. The two volumes are connected by a 5-foot wide tunnel whose floor is also at elevation 71′ 6″ and a 3-foot wide ventilation duct whose bottom is 4 inches above the cavity floor. The cavity sump is situated between the tunnel and the ventilation duct at the side of the reactor coolant drain tank room closest to the reactor vessel. There is a 3-foot thick wall that separates the reactor cavity drain tank region from the reactor vessel region of the cavity. The floor of the cavity sump is at elevation 69′ 6″ and is completely encompassed by a curb whose top is at elevation 73′ 6″ (24-inch high curb). The tunnel between the reactor vessel and reactor coolant drain tank portions of the cavity is protected by a door and shielding material to minimize radiation exposure to persons working in the reactor coolant drain tank area of the cavity. The door and shielding are not important to the analyses of core debris spreading in the reactor cavity due to the dynamic forces of the fuel coolant interactions that will occur at reactor vessel failure. Since the door and shielding are not designed to withstand "blast loading," they are expected to be destroyed prior to the arrival of core debris at their pre-vessel failure location. As added assurance that the door and shielding will not remain in their pre-vessel failure location, the high temperature of the core debris will quickly ablate and/or physically move any door and/or shielding components that might remain in place after the fuel coolant interaction loading. A schematic layout of the cavity region is provided in Figure B-3.

The embedded steel containment shell beneath the reactor cavity region is ellipsoidal in shape. The minimum distance from the reactor cavity floor to the steel shell occurs at the end of the reactor coolant drain tank room furthermost from the reactor vessel and is 2.78 feet (0.847 meters). The minimum distance from the cavity sump to the steel shell is 2.69 feet, which is just slightly less than the minimum distance from the cavity floor. The minimum distance is used in the following analyses, as opposed to the minimum vertical distances which are slightly greater than the minimum values presented, to simplify the analyses to a two-dimensional problem. The minimum distance from the reactor cavity floor to the bottom of the basemat, which is parallel to the cavity floor, is 11 feet (3.35 meters). This volume is filled entirely with reinforced concrete and containment vessel shell.

The reactor cavity sump is covered with a stainless steel plate that supports the reactor cavity drain pumps. While there are a number of penetrations in the steel plate (e.g., manway, piping, etc. they will not represent a significant flow path for viscous core debris to enter the cavity. The steel plate is expected to remain intact unless thermally attacked by core debris that accumulates on top of the plate. There is no piping buried in the concrete beneath the reactor cavity. The sump drain lines are not enclosed in either the reactor cavity floor or reactor cavity sump concrete. The sump has a number of sleeved (1/2 inch Schedule 80S piping) drain holes through the curbing at floor level to permit water to drain into the sump, but which will quench molten core material that drains into the reactor cavity if the reactor vessel fails during a severe accident.

An investigation of the spreading of core debris that pours into the reactor cavity was conducted for reactor vessel failure that occurs at low RCS pressure. The investigation considered the vessel failure mode and location, as well as the recognition that the oxide and metal components of the in-vessel core debris are predicted to be separated. Since the oxide and metal components of the core debris have very different physical characteristics (e.g., viscosity or heat capacity), the separated in-vessel layers influence the spreading of the core debris in the reactor cavity. The melt spreading analysis was conducted for both reactor vessel failure modes described in Section B.1 (hinged and localized failures).

For the hinged vessel failure mode, the entire in-vessel core debris mass was deposited on the cavity floor at a constant rate over a time scale of 10 seconds. The ten-second time release period used in this analysis is assumed to be representative of a rapid release in which the metal phase is released distinct from and ahead of the oxide phase. This is roughly equivalent to the assumption that the angle by which the lower head hinges downward increases linearly with time until the head contacts the cavity floor. Because of the assumed rapid hinging failure that conveys the melt largely inside the lower head, no effects of metal water interactions are modeled distinct from normal heat transfer from the melt to the water in the MELTSPREAD code. For the localized failure, a model was developed to calculate the boildown of cavity water level, time dependent melt release rate and melt superheat. This model treated the reactor vessel failure elevation as a function of the cavity water depth (i.e., the failure elevation was maintained just above the cavity water level) by calculating the cavity water boiloff rate as a function of the amount of core debris released to the reactor cavity and the amount of superheat in the core debris. The THIRMAL code was then used to investigate the effects of metal-water interactions upon arrival of materials at the bottom of the pool as the initial portion of the metallic melt relocates downward through the pool. Specifically, the combination of the slow initial release rate and the large water depth would be expected to result in breakup and freezing of the melt as it falls through the water pool, thereby collecting on the cavity floor as a debris bed of solidified particulate. The melt arrival conditions for the MELTSPREAD analysis were thus based on both calculated release conditions and the THIRMAL results. None of the structures and equipment (e.g., doors, shielding, reactor coolant drain tank and supports) in the reactor cavity was included in the MELTSPREAD analysis. This is justified since the mass of these materials is small compared to the mass of the core debris coming from the reactor vessel. Also, it was assumed that the forces acting on the structures and equipment from fuel coolant interactions that occur upon initial entry of core debris into the cavity, as well as the forces imposed by the movement of the core debris, would easily dislodge or melt the structures and equipment, or both.

The results of the THIRMAL analyses show that most of the core debris (~94 percent) is expected to reach the floor of the cavity in a partially frozen state while the remainder is expected to be fully frozen. None of the initial release of core material from the vessel is expected to be in a fully molten state. However, as the core debris accumulates on the cavity floor, the decreased water depth will result in an increasing fraction of the core debris arriving in a molten state. The debris arriving in a molten state can fill the interstices and may erode some of the previously solidified debris. Thus, while the initial formation of a porous debris bed cannot be ruled out, the continued addition of molten core material will likely result in a partially frozen debris layer that can further spread over the cavity floor. The results of the THIRMAL analysis were used as the initial conditions for the MELTSPREAD analysis of the localized reactor vessel failure case.

The MELTSPREAD analyses were performed using a reactor cavity model shown in Figure B-3. The model used in the MELTSPREAD analysis, which is based on an AP600-like cavity, accurately represents the AP1000 reactor cavity configuration in terms of spreading area and geometry.

For the hinged vessel failure case, the analysis results show that the core debris is spread relatively uniformly over the reactor cavity floor, as shown in Figure B-4. However, the distribution of the metal and oxide components of the core debris are not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component (e.g., 85 to 90 percent oxide). At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel (e.g., 75 to 85 percent metal). The core debris is still almost totally molten at the end of the spreading analysis. The steel liner over the cavity floor is completely eroded away and the core debris has begun to penetrate into the concrete floor. The penetration depth at the end of the MELTSPREAD analysis was approximately 1.2 inches (3 cm) under the reactor vessel and about 2.75 inches (7 cm) at the opposite end of the reactor cavity.

A different behavior is predicted for the localized reactor vessel failure case. The MELTSPREAD analysis predicts that the core debris will accumulate at the reactor vessel end of the reactor cavity as shown in Figure B-5. The distribution of the metal and oxide components of the core debris are not uniformly distributed over the reactor cavity floor. In the region directly under the reactor vessel, the core debris consists primarily of the oxide component (e.g., 70 to 80 percent oxide). At the opposite end of the reactor cavity, the core debris consists mainly of the metal component of the core debris released from the reactor vessel (e.g., 80 to 90 percent metal). The core debris is almost totally frozen at the end of the spreading analysis. The steel liner over the cavity floor is not eroded (except for one node of the cavity model under the reactor vessel) and the core debris has only begun to penetrate the concrete in one node.

The results of the MELTSPREAD analyses were used to assess the effectiveness of the curbing around the reactor cavity sump to prevent the accumulation of significant amounts of core debris in the sump. In particular, the height of the core debris adjacent to the reactor cavity sump curbing (cells 12 through 15 in Figure B-3) was examined to determine the potential for core debris to enter the cavity sump. For the hinged reactor vessel failure case, the maximum height of core debris adjacent to the sump curbing during the initial flow of core debris from the reactor vessel region to the reactor coolant drain tank region is about 32 inches. This occurs during a very brief time interval (at about 10 seconds after reactor vessel failure) when the flow is parallel to the curbing. Based on the high viscosity of the core debris, little of the core debris is expected to spill over onto the cavity sump cover plate. Following the initial wave, the analyses predict that the core debris is reflected off of the back wall of the reactor coolant drain tank portion of the cavity. The maximum height of the reflected wave in the area adjacent to the reactor cavity sump is 24.7 inches. This occurs at about 25 seconds after reactor vessel failure. At this time, core debris is expected to flow onto the reactor cavity sump cover. After the passage of the core debris reflected wave, the equilibrium height of core debris in the region of the cavity sump is about 22 inches. Since this is higher than the cavity sump curb, the continual presence of core debris on top of the sump cover will result in thermal failure of the cover and the subsequent flow of core debris into the cavity sump. For the localized reactor vessel failure case, the maximum depth of core debris in the cavity sump at any time in the transient is about 10 inches. Due to the characteristics of the core debris flow into the reactor cavity for the localized failure mode, there is no transient behavior with large amplitude waves of core debris transiting the cavity. Thus, for the localized reactor vessel failure mode, the reactor cavity curbing prevents the accumulation of core debris in the cavity sump.

The results of the MELTSPREAD analyses were used to establish initial conditions for assessment of core concrete interactions using the MAAP4 code models. Since MAAP4 can only treat the core debris that is uniformly spread over a cavity floor, two parallel MAAP4 analyses were done for each vessel failure mode. The first analysis for each vessel failure mode treats the core debris under the reactor vessel while the second analysis treats the core debris that is in the RCDT end of the cavity. In all cases, the results of the MELTSPREAD analysis were used to define the initial conditions for the core concrete interactions using the MAAP4 models. Since one portion of the reactor cavity initially contains oxide-rich core debris and the other end contains metal-rich core debris, the rate of concrete decomposition is controlled by different factors in each analysis. The oxide-rich debris contains most of the fission product decay heat and this controls the concrete decomposition. The metal-rich debris does not contain decay heat but is subject to exothermic heat of reaction of steam with the unoxidized metal and this controls the concrete decomposition.

The core concrete interactions for the AP1000 design were analyzed for two concrete types: basaltic concrete and common limestone-sand concrete. The common limestone-sand concrete has a significantly higher noncondensable gas generation rate, compared to basaltic concrete and should therefore present a more severe containment pressurization transient. On the other hand, the basaltic concrete suffers higher ablation rate, due to its physical properties (mainly, its lower decomposition energy), and should therefore present a more severe basemat penetration failure mode, compared to common limestone-sand concrete. The comparison of the results of the containment failure modes for the two types of concrete was used to determine the need for a specification of a concrete type for the containment basemat. In all cases, a 3.5 m deep water pool is initially present in the cavity while debris is being released into it.

For the hinged reactor vessel failure case, where release of the entire core debris into the cavity water pool occurs in 10 seconds, the core concrete interaction analysis shows (see Figures B-6a and B-6b) that only the concrete in the region of the cavity under the reactor vessel is eroded. On the RCDT side, the debris is quenched to the extent that concrete heating never reaches its melting point. As documented earlier, core debris may enter the cavity sump for the hinged reactor vessel failure case. Close examination of the MELTSPREAD analyses indicates that the core debris in the cavity sump would consist primarily of the metal component of the core debris, similar to the reactor coolant drain tank side of the reactor cavity. Since the MAAP core concrete interaction results show that the core concrete penetration on the reactor coolant drain tank side of the cavity is minimal, compared to the oxide melt penetration on the reactor vessel side of the cavity, the penetration of the debris in the cavity sump would not be controlling. Thus, based on the core debris penetration in the reactor vessel portion of the cavity and conservatively using a 2.78 foot distance to the containment liner, key results for the hinged reactor vessel failure scenario are:

Core Concrete Interaction Results Hinged Vessel Failure Case		
Parameter	Basaltic Concrete	Common Limestone-Sand Concrete
Time of Melt-Through of Embedded Shell	11 Hours	12.5 Hours
Time of Melt-Through of Basemat	3.3 Days	4.5 Days

Two indicators of potential challenge to containment integrity are shown in the above table of results: penetration of the steel vessel shell and penetration of the entire basemat. Detailed structural analyses were not performed to determine the containment fragility for various depths of basemat penetration as a function of containment pressure. Thus, the two values are termed "indicators of a potential challenge." It is highly unlikely that the containment fission product boundary would have been lost when the core first penetrates the basemat since there is still at least 8 feet of concrete layer below the shell. On the other hand, it is also highly unlikely that the containment integrity will still be intact just prior to the time that the core debris penetrates the entire basemat depth. Containment basemat failure and the subsequent release of fission products from the containment is likely to occur at some point in time between the two "indicators."

At 24 hours into the accident, the downward erosion on the reactor vessel side of the cavity is 5.7 feet (1.74 m) for basaltic concrete, and 4.9 feet (1.5 m) for common limestone-sand concrete. There is no erosion on the RCDT side.

For the case of the localized reactor vessel failure, where the entire core debris slowly drains to the initially flooded cavity for a period of 3.3 hours, the analysis shows (see Figures B-7a and B-7b) that the concrete in the region of the cavity under the reactor vessel is eroded by about a factor of 2 more rapidly than the region of the RCDT. In this vessel failure mode, no core debris is predicted to enter the reactor cavity sump, but concrete erosion on the RCDT side is substantial. Key results for the localized reactor vessel failure scenario are:

Core Concrete Interaction Results Localized Vessel Failure Case		
Parameter	Basaltic Concrete	Common Limestone-Sand Concrete
Time of Melt-Through of Embedded Shell	8.8 Hours	10.5 Hours
Time of Melt-Through of Basemat	2.8 Days	3.9 Days

For the basaltic concrete case, at 24 hours into the accident, the downward erosion on the reactor vessel side of the cavity is 6.2 ft (1.9 m) while the erosion is 3 ft (0.9 meters) on the RCDT side. For the common limestone-sand concrete case, the erosion depth at 24 hours is 5.5 ft (1.66 m) for the cavity side and 2.5 ft (0.77 m) for the RCDT side.

Based on these analyses, it can be concluded that: a) the goal of protecting the containment fission product boundary during the first 24 hours of a core melt accident is met, b) it is not necessary to specify a concrete type for the containment basemat since credible containment basemat failure that could lead to fission product releases to the atmosphere are likely to occur at times well beyond 24 hours, and c) the reactor cavity sump is adequately protected such that it is not a weakness in containment basemat integrity during postulated accidents that lead to core concrete interactions.

B.4.1 Containment Pressurization due to Core Concrete Interactions

The containment pressurization due to steam and noncondensable gas generation during the episodes of core concrete interactions described above was assessed to determine the effect of core concrete interactions on the containment integrity.

To estimate the effect of steam and noncondensable gas generation on the containment pressure and temperature, the AP1000 containment was included in the separate effects MAAP4 analysis of core concrete interactions described above. Since the core concrete interaction assessment with MAAP4 was a "separate effects" analysis, the initial containment conditions were specified to be 21.8 psia (0.15 MPa) at saturation conditions represented by a steam-air mixture.

The indicator of a challenge to containment integrity for the containment pressurization due to the noncondensable gases produced from core concrete interactions is the Service Level "C" pressure, which is 91 psig (0.73 MPa). This is well below the 50 percent containment failure probability value of 135 psig (1.03 MPa). The MAAP4 analysis results are:

Containment Pressurization Due to Core Concrete Interactions		
Parameter	Basaltic Concrete	Common Limestone-Sand Concrete
Hinged Vessel Failure Scenario		
Time to Pressurize Containment to Service Level "C"	Not Applicable (Basemat fails first)	Not Applicable (Basemat fails first)
Containment Pressure at Melt-Through of the Basemat	35 psig	51 psig
Containment Pressure at 24 hr	20 psig	24 psig
Localized Vessel Failure Scenario		
Time to Pressurize Containment to Service Level "C"	Not Applicable (Basemat fails first)	Not Applicable (Basemat fails first)
Containment Pressure at Melt-Through of the Basemat	53 psig	80 psig
Containment Pressure at 24 hr	30 psig	39 psig

From these results (see Figures B-8a and B-8b), it is evident that the containment conditions at 24 hours after reactor vessel failure are sensitive to the assumed mode of reactor vessel failure. This is the direct result of the pool quenching effect, which appears to vary with the vessel failure mode. The localized vessel failure mode with a slow release of the core debris tends to attack concrete with a shorter delay time, and therefore tends to pressurize the containment faster than the hinged failure mode. The highest containment pressurization to 39 psig in 24 hours occurs in the localized vessel failure case with common limestone-sand concrete due to the steam and noncondensable gases generated during core concrete interactions.

The results also show that, in all cases the containment does not pressurize to Service Level "C" containment challenge indicator value prior to the time that the core debris completely penetrates the containment basemat. Thus, for these cases there is no potential challenge to containment integrity due to overpressurization since: a) there is no longer a source of mass and energy input to the containment after the core debris penetrates the entire basemat, and b) basemat penetration assures that the containment will be depressurized through the basemat failure.

These results indicate that the containment pressure is still well below the point where the integrity of the containment may be challenged before the containment basemat fails and the containment is depressurized by basemat penetration.

Based on these analyses, it can be concluded that it is not necessary to specify a concrete type for the containment basemat since containment overpressure failure due to non-condensable gas generation from core concrete interactions is not likely for any credible severe accident scenarios.

B.5 Conclusions

The results of the limited deterministic analyses of ex-vessel severe accident phenomena presented in this section show that early containment failure is not a certainty if the reactor vessel fails. Based on the deterministic analyses, direct containment heating that might ensue from a high pressure melt ejection would not challenge the integrity of the containment. Ex-vessel steam explosions, assessed on a very conservative basis would not produce impulse loads that would challenge the integrity of the containment due to localized failures of the reactor cavity floor and walls. In addition, these analyses indicate that the ex-vessel steam explosion loads are not strong enough to displace the reactor vessel from its location inside the biological shield. Thus, there is no challenge to any containment penetrations connected to the reactor vessel or to the reactor coolant loops. In the case of a vessel failure at a low RCS pressure, the core concrete interactions analyses indicate that the containment integrity would not be challenged in the first 24 hours of the event and thus no significant releases of fission products are predicted in that time frame.

Thus, it is concluded that prevention of large fission product releases to the environment is not dependent on the integrity of the reactor vessel. If reactor vessel failure occurs, there may be challenges to the containment integrity, but these challenges are highly uncertain and the most likely challenge (containment failure by core penetration of the cavity basemat) would not occur in the first 24 hours of the accident. Thus, the AP1000 assumption that reactor vessel failure always leads to containment failure is a conservatism in the AP1000 risk profile.

B.6 References

B-1 "AP600 Phenomenological Evaluation Summaries," WCAP-13388 (Proprietary) Rev. 0, June 1992 and WCAP-13389 (Nonproprietary), Rev. 1, 1994.

B-2 Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

B-3 "AP600 Probabilistic Risk Assessment," GW-GL-022, August 1998.

B-4 Pilch, M. M., "Adiabatic Equilibrium Models For Direct Containment Heating," SAND-91-2407C, December 1992.

B-5 "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design," SECY-93-087, dated April 2, 1993.

<table>
<tr><th colspan="5">Table B-1

INITIAL CONDITIONS FOR
MOLTEN CORE CONCRETE INTERACTION ANALYSIS</th></tr>
<tr><th>Parameter</th><th colspan="2">Scenario I (hinged failure)</th><th colspan="2">Scenario II (localized failure)</th></tr>
<tr><td>Time of Vessel Failure</td><td colspan="2">7,200 seconds</td><td colspan="2">7,200 seconds</td></tr>
<tr><td>Duration of Debris Release</td><td colspan="2">10 seconds</td><td colspan="2">12,029 seconds</td></tr>
<tr><td>Mass of Release
Metal
Oxide</td><td colspan="2">
65,755 kg
107,226 kg</td><td colspan="2">
54,526 kg
122,162 kg</td></tr>
<tr><td>Composition of Core Debris in Cavity
Fe
Cr
Ni
Zr
FeO
Cr_2O_3
UO_2
ZrO_2
Concrete Slag</td><td colspan="2">
36,720 kg
9,690 kg
4,590 kg
14,755 kg
0 kg
0 kg
96,500 kg
10,726 kg
0 kg</td><td colspan="2">
33,048 kg
8,721 kg
4,131 kg
8,626 kg
5,246 kg
1,416 kg
96,500 kg
19,000 kg
0 kg</td></tr>
<tr><td>Decay Heat in Debris at Time of Vessel Failure</td><td colspan="2">2.06 MW/m^3</td><td colspan="2">2.06 MW/m^3</td></tr>
<tr><td>Core Debris Configuration</td><td>RV Side
Homogeneous</td><td>RCDT Side
Homogeneous</td><td>RV Side
Layered Oxide over Metal[(1)]</td><td>RCDT Side
Layered Oxide over Metal[(1)]</td></tr>
<tr><td>Temperature of Core Debris
(inside lower plenum)
Metal
Oxide</td><td colspan="2">

1750°K
3251°K</td><td colspan="2">

1750°K
3251°K</td></tr>
<tr><td>Water Depth at Time of Vessel Failure</td><td colspan="2">3.5 m</td><td colspan="2">3.5 m</td></tr>
<tr><td>Water Temperature</td><td colspan="2">Saturated</td><td colspan="2">Saturated</td></tr>
<tr><td>Core Debris Fraction in Cavity
Metal
Oxide</td><td>RV Side
20%
68%</td><td>RCDT Side
80%
32%</td><td>RV Side
46%
70%</td><td>RCDT Side
54%
30%</td></tr>
<tr><td>Cavity Floor Area</td><td>22 m^2</td><td>26 m^2</td><td>22 m^2</td><td>26 m^2</td></tr>
<tr><td>Cavity Floor Concrete</td><td colspan="2">Limestone-Sand and Basaltic</td><td colspan="2">Limestone-Sand and Basaltic</td></tr>
<tr><td>Containment Pressure
Temperature</td><td colspan="2">0.15 MPa
385°K</td><td colspan="2">0.15 MPa
385°K</td></tr>
</table>

Note:

1. The molten core debris once released to the cavity cannot be modeled by MAAP4 as separate layers of oxides and metal, but is treated as a homogeneous mixture. The MCCI analysis conservatively assumes that the decay heat-generating oxide is first released. The release of metal begins only when all the oxides have been released. Hence, at least initially there is a layer of oxide alone on the concrete floor.

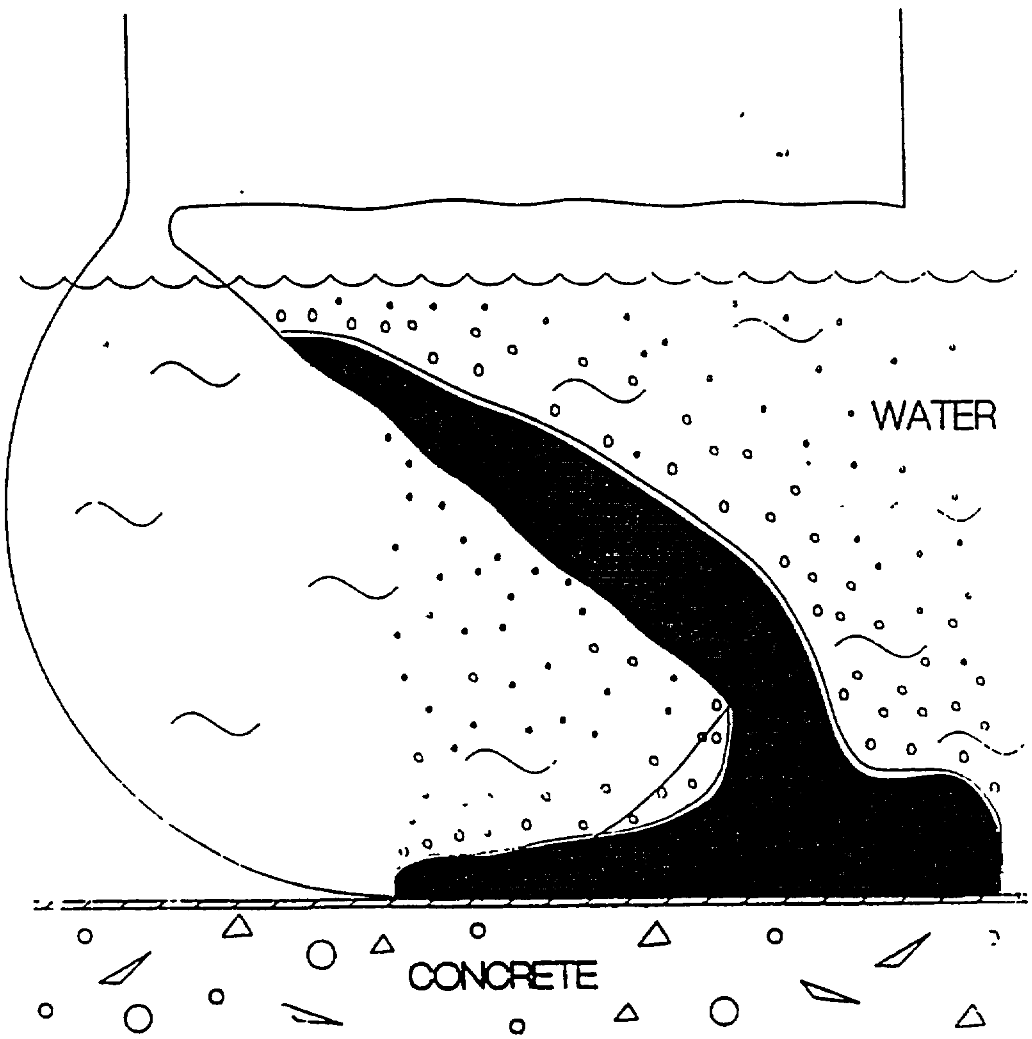


Figure B-1

Illustration of Hinging Type of Failure Resulting in Rapid Melt Release (from Reference B-4)

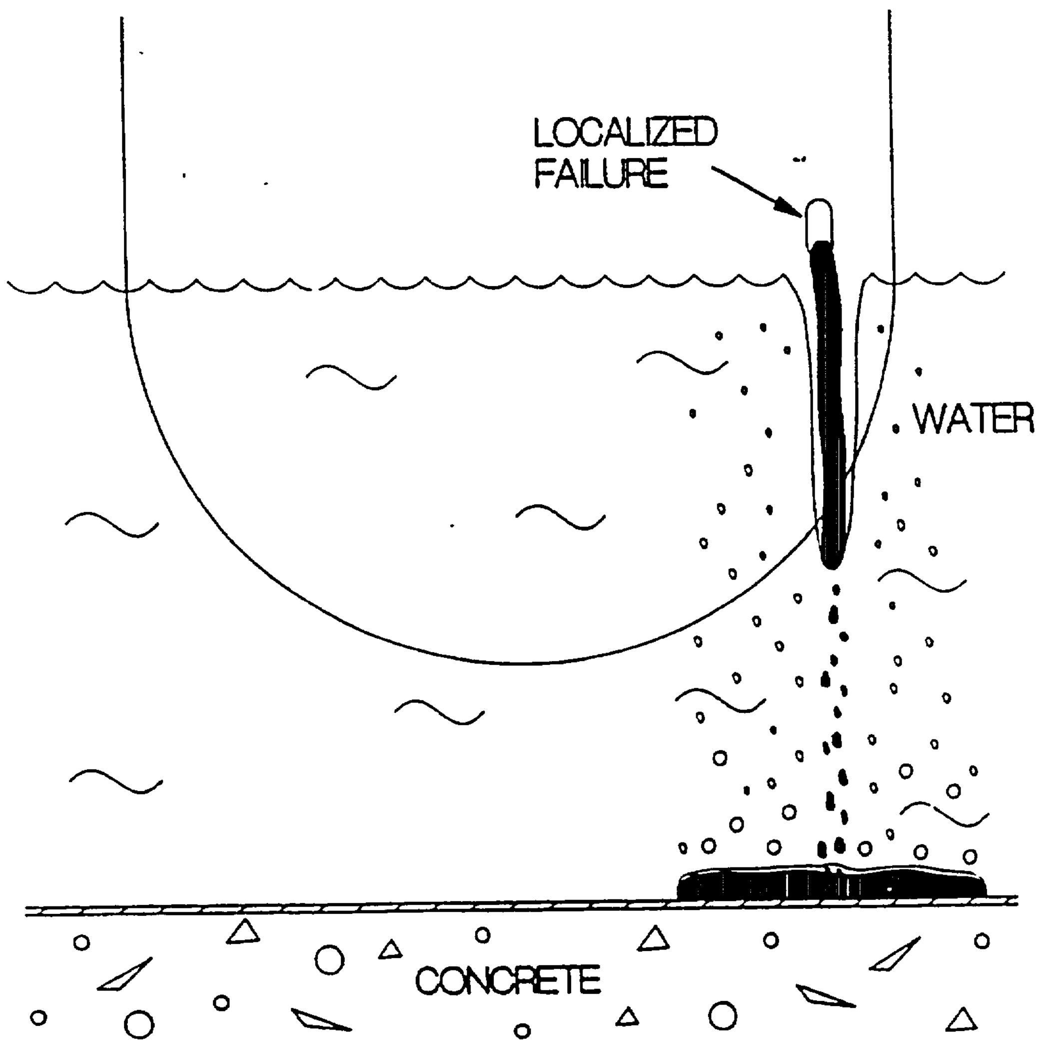


Figure B-2

Illustration of Localized Type of Failure Resulting in Slow Melt Release (from Reference B-4)

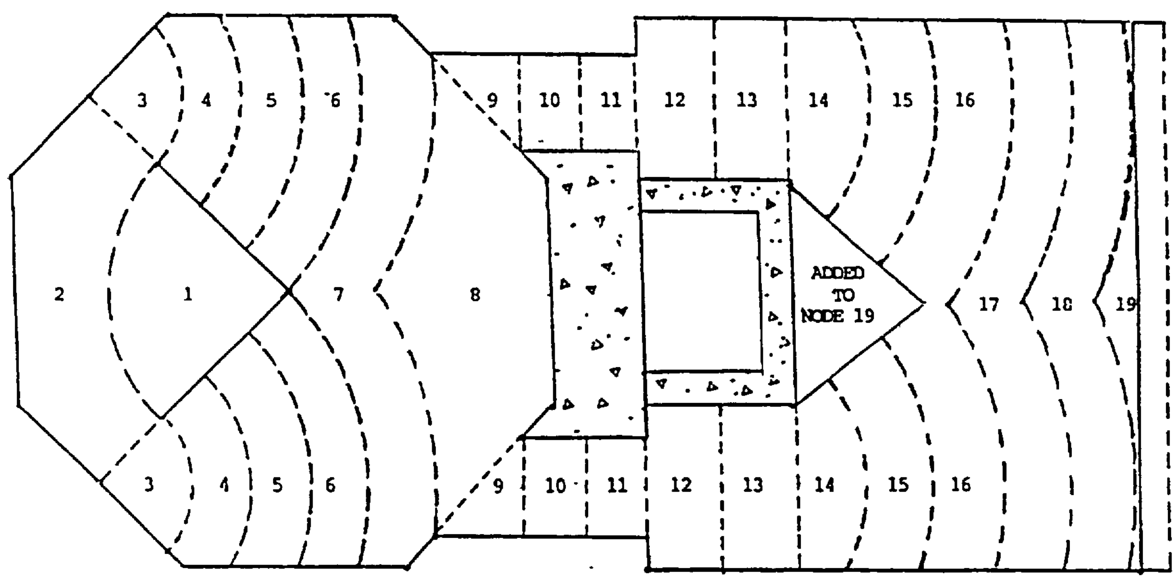


Figure B-3

Nodalization of Cavity, Personnel Access, Ventilation Duct, and Reactor Coolant Drain Tank (RCDT) (from Reference B-4)

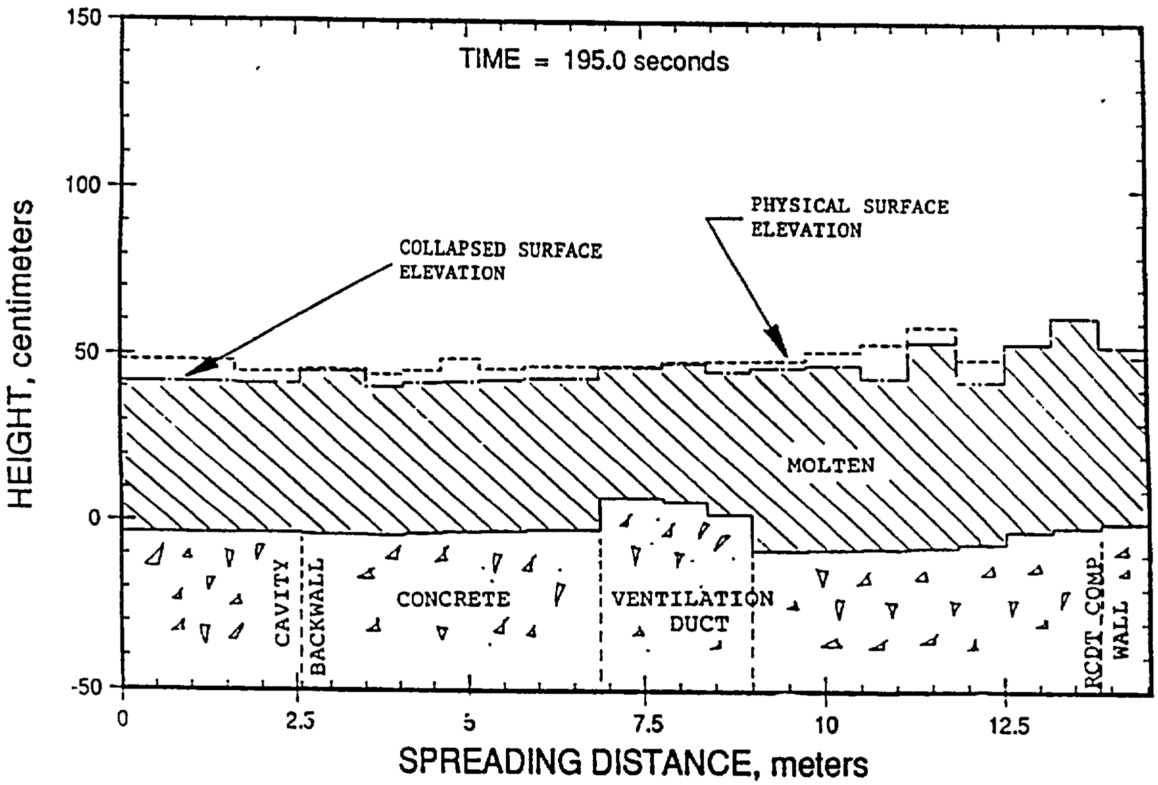


Figure B-4

Scenario I Melt Spreading and Floor Surface Elevation Distribution at 195 Seconds (from Reference B-4)

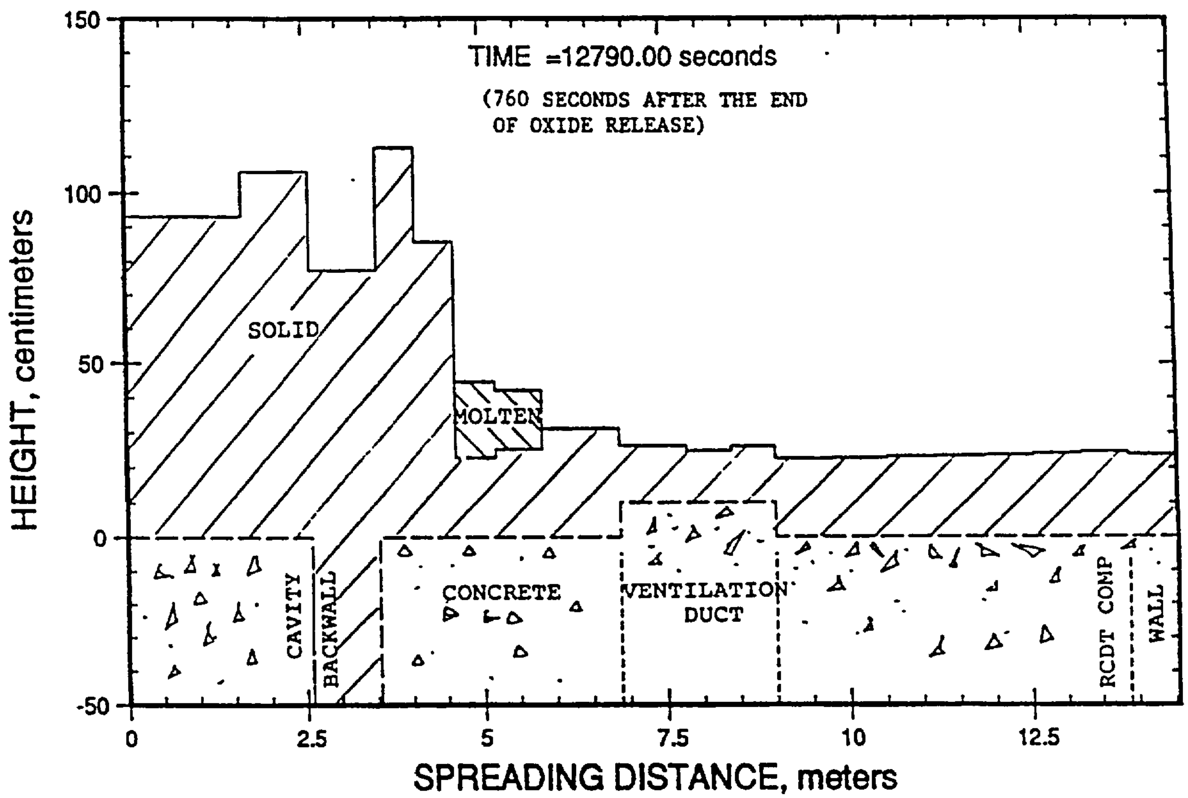


Figure B-5

Scenario II Melt Spreading and Floor Surface Elevation Profiles at 12790 Seconds (from Reference B-4)

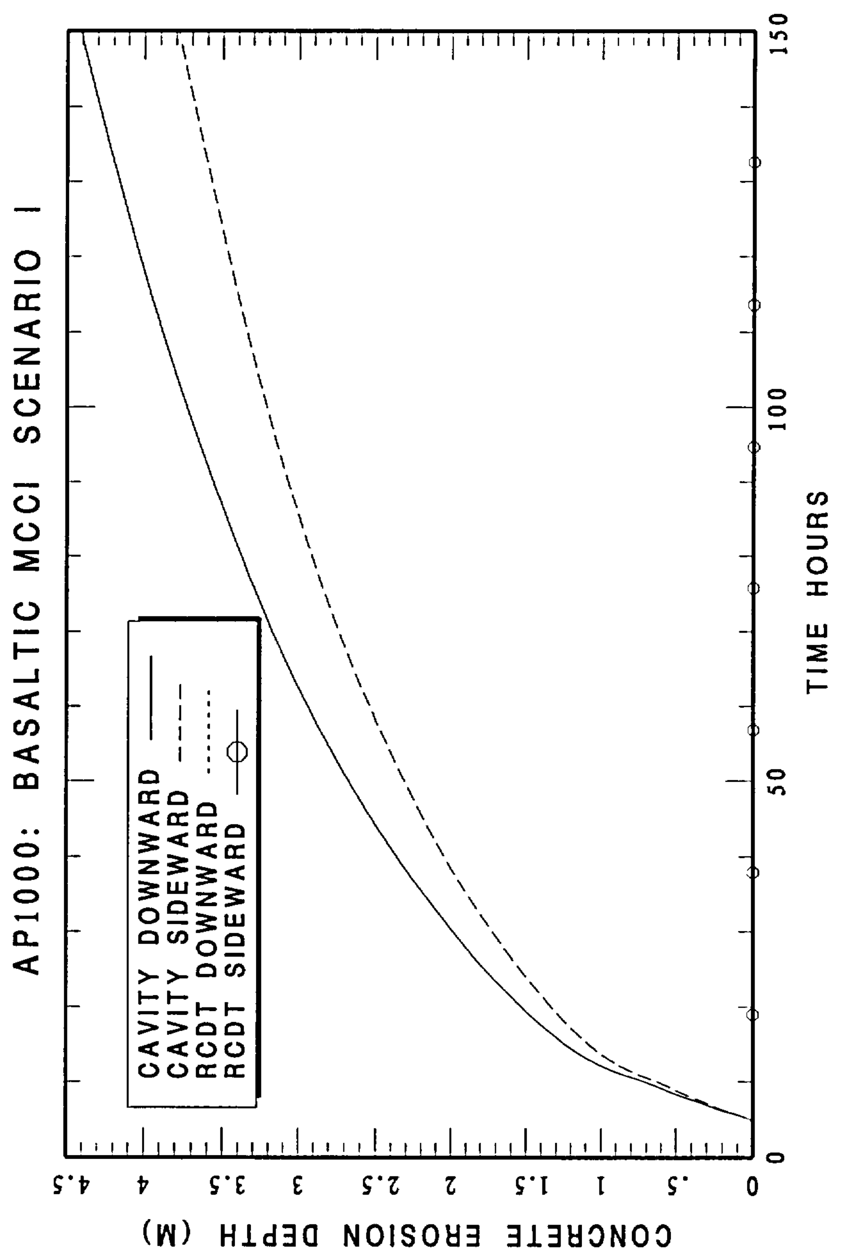


Figure B-6a

Scenario I (hinged failure): Basaltic Concrete Ablation

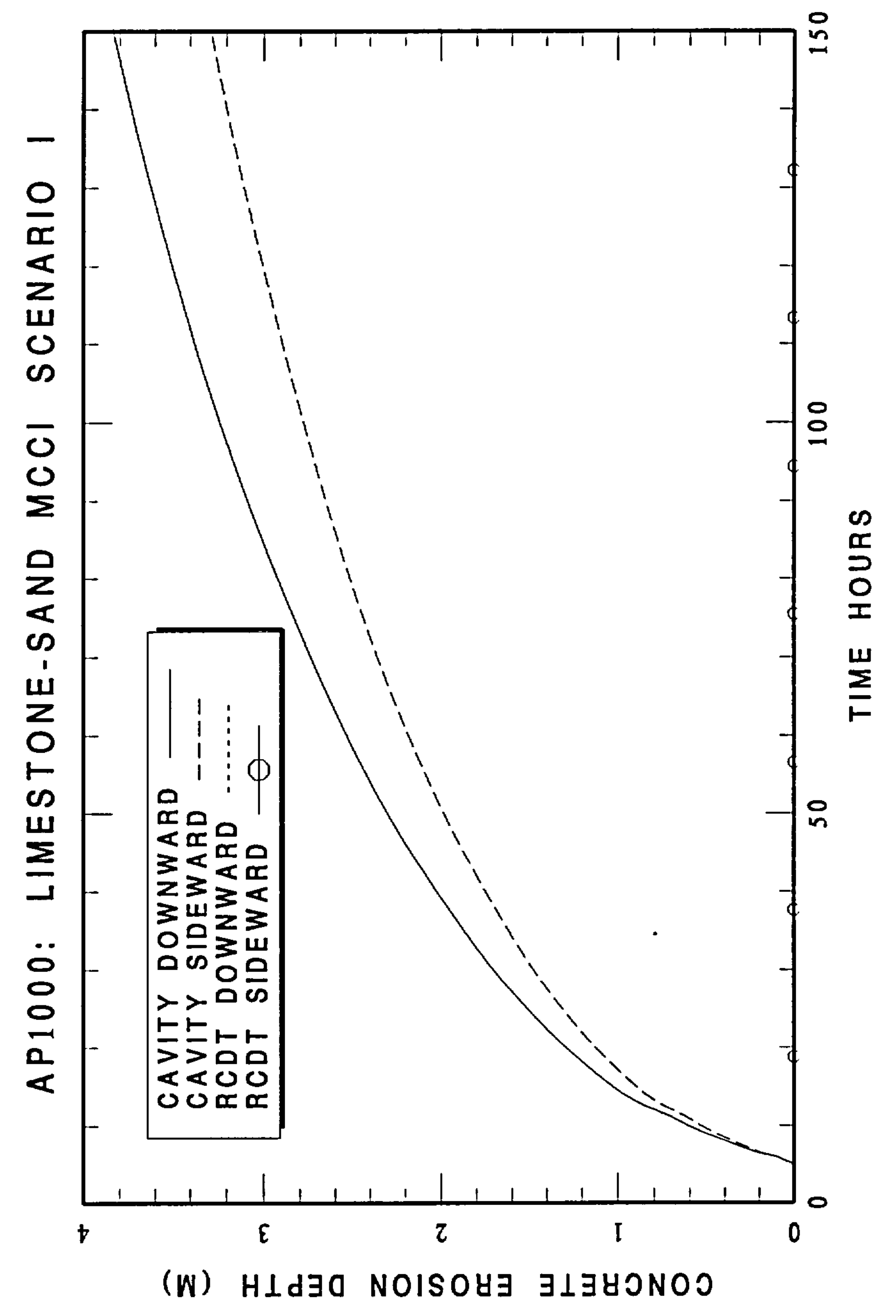


Figure B-6b

Scenario I (hinged failure): Common Limestone-Sand Concrete Ablation

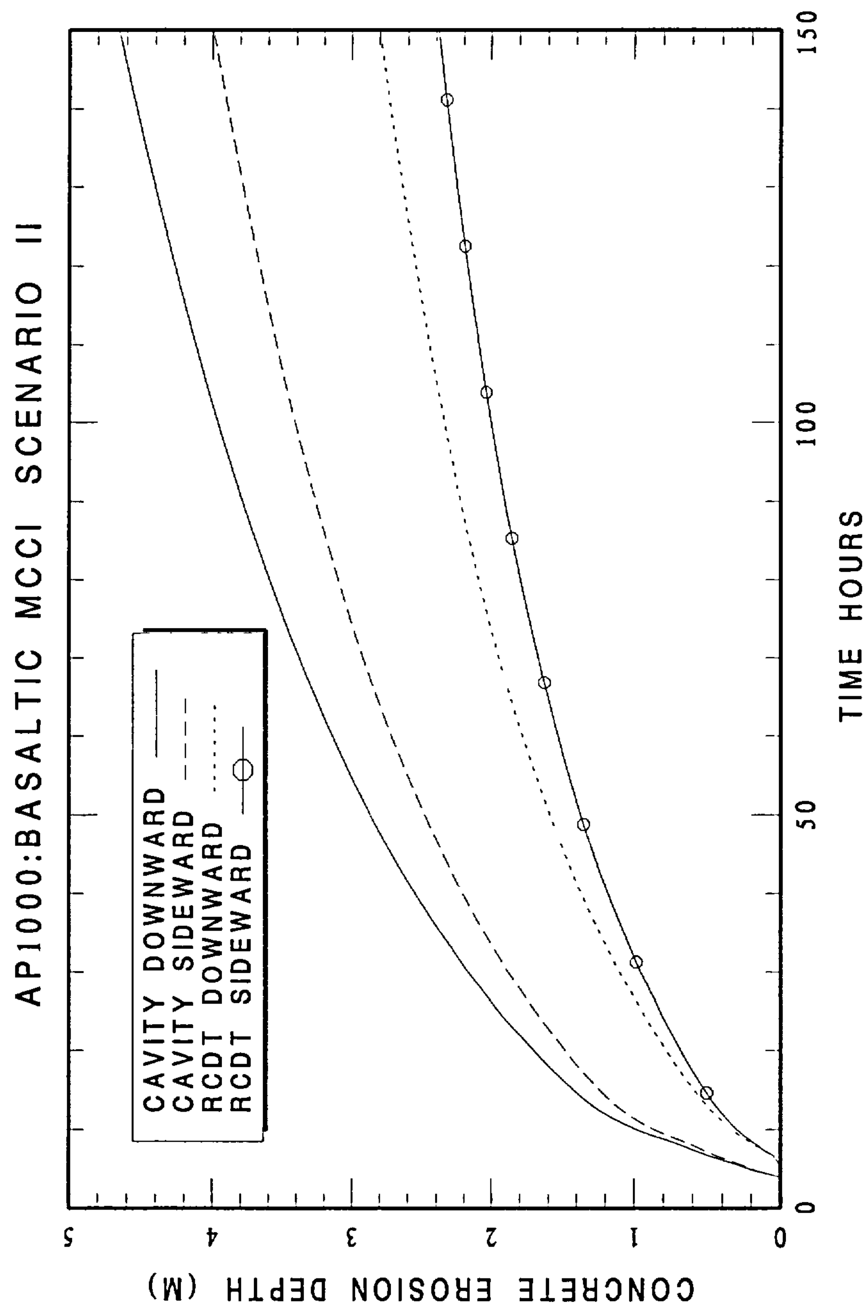


Figure B-7a

Scenario II (localized failure): Basaltic Concrete Ablation

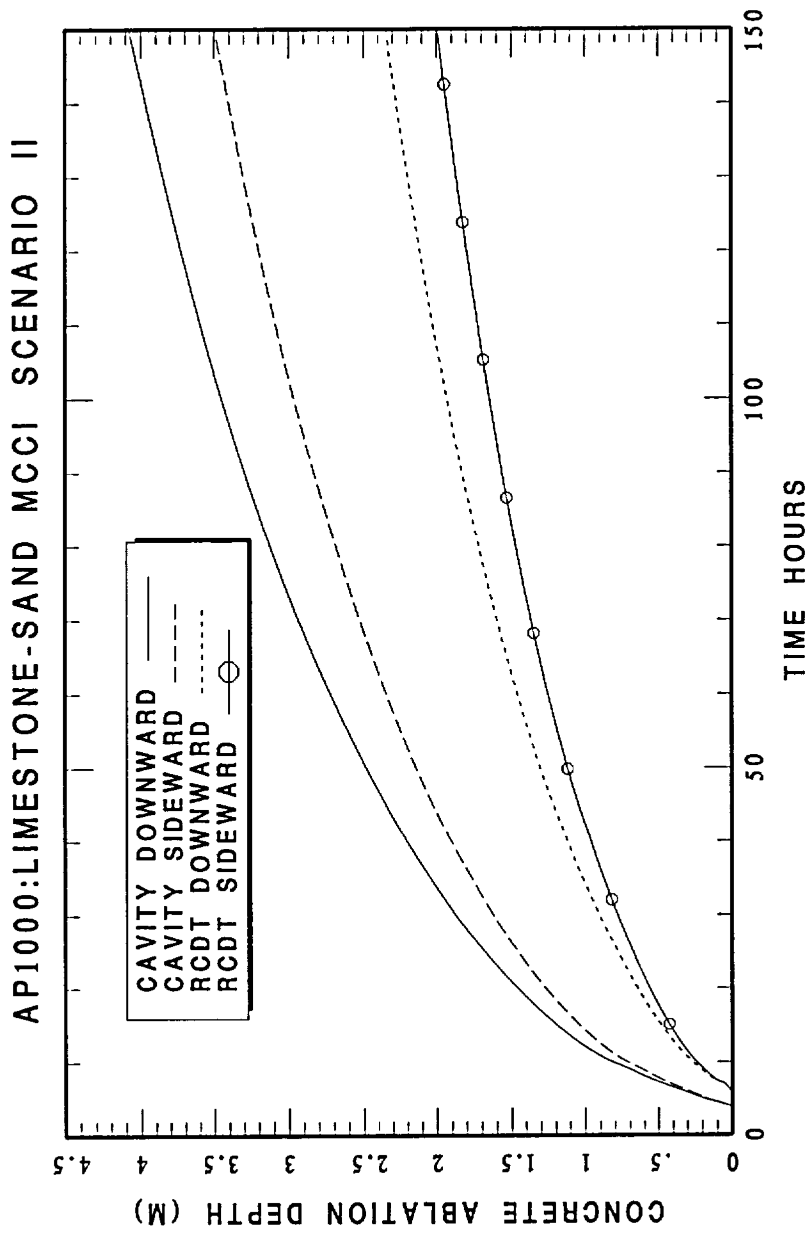


Figure B-7b

Scenario II (localized failure): Common Limestone-Sand Concrete Ablation

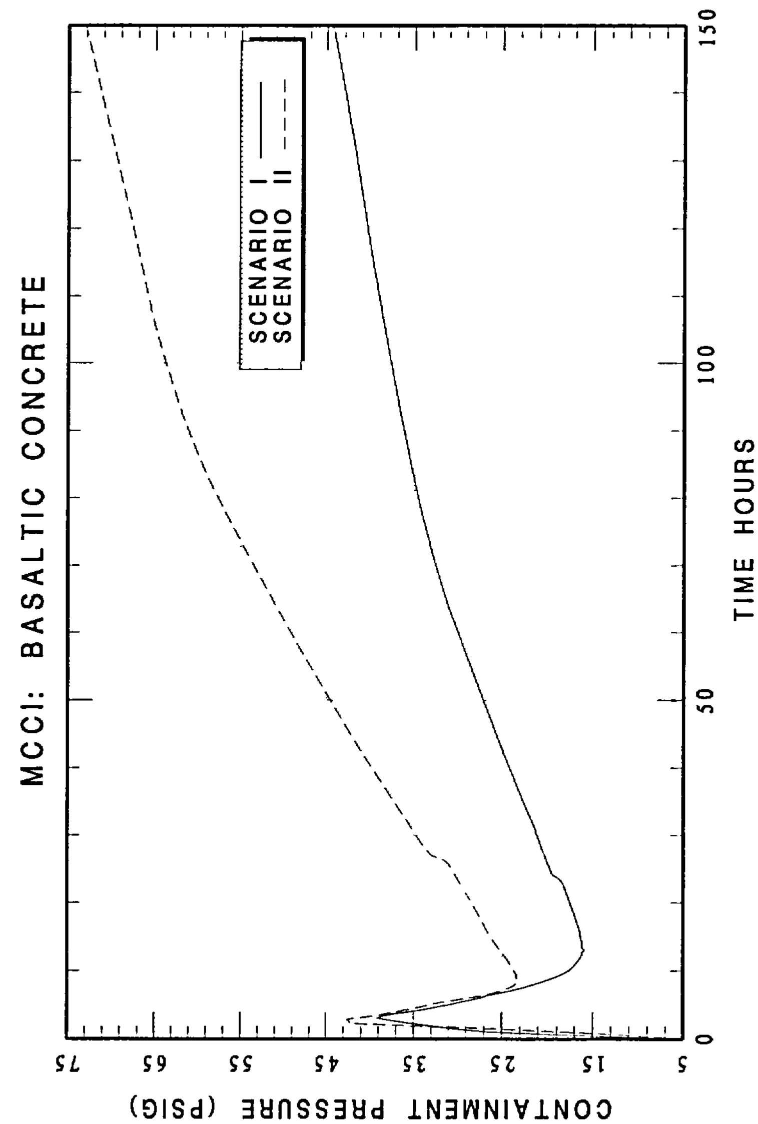


Figure B-8a

Containment Pressure During MCCI Scenarios with Basaltic Concrete

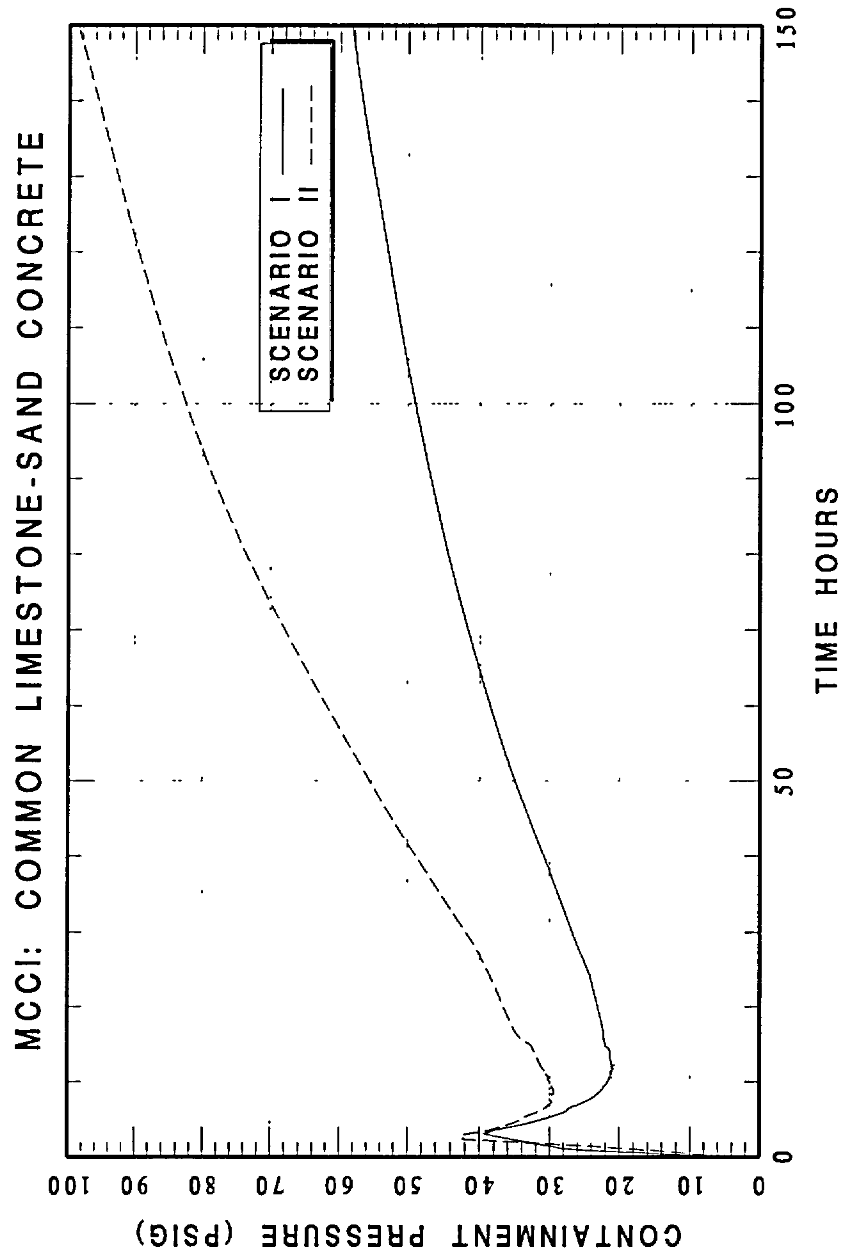


Figure B-8b

Containment Pressure During MCCI Scenarios with Common Limestone-Sand Ablation

APPENDIX D

EQUIPMENT SURVIVABILITY ASSESSMENT

D.1 Introduction

The purpose of the equipment survivability assessment is to evaluate the availability of equipment and instrumentation used during a severe accident to achieve a controlled, stable state after core damage under the unique containment environments. Severe accident phenomena may create harsh, high temperature and pressure containment environments with a significant concentration of combustible gases. Local or global burning of the gases may occur, presenting additional challenges to the equipment. Analyses demonstrate that there is reasonable assurance that equipment used to mitigate and monitor severe accident progression is available at the time it is called upon to perform.

The methodology used to demonstrate equipment survivability is:

- Identify the high level actions used to achieve a controlled, stable state
- Define the accident time frames for each high level action
- Determine the equipment and instruments used to diagnose, perform and verify high level actions in each time frame
- Determine the bounding environment within each time frame
- Demonstrate reasonable assurance that the equipment will survive to perform its function within the severe environment.

D.2 Applicable Regulations and Criteria

Equipment that is classified as safety-related must perform its function within the environmental conditions associated with design-bases accidents. The level of assurance provided by equipment required for design-bases events is "equipment qualification."

The environmental conditions resulting from beyond design basis events may be more limiting than conditions from design-bases events. The NRC has established criteria to provide a reasonable level of assurance that necessary equipment will function in the severe accident environment within the time span it is required. This criterion is referred to as "equipment survivability."

The applicable criteria for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents are provided in 10 CFR 50.34(f):

- Part 50.34(f)(2)(ix)(c) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

- Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluent.
- Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.
- Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in Section III.F, "Equipment Survivability" of SECY-90-016. The NRC recommends in SECY-93-087 that equipment provided only for severe accident protection need not be subject to 10 CFR 50.49 equipment qualification requirements, the 10 CFR 50 Appendix B quality assurance requirements, or 10 CFR 50 Appendix A redundancy/diversity requirements. However, mitigation features must be designed to provide reasonable assurance they will operate in the severe accident environment for which they are intended and over the time span for which they are needed.

D.3 Definition of Controlled, Stable State

The goal of accident management is to achieve a controlled, stable state following a beyond design basis accident. Establishment of a controlled, stable state protects the integrity of the containment pressure boundary. The conditions for a controlled, stable state are defined by WCAP-13914, the Framework for AP600 Severe Accident Management Guidance (SAMG) (Reference D-1) which is considered valid for AP1000.

For a controlled, stable core state:

- A process must be in place for transferring the energy being generated in the core to a long-term heat sink
- The core temperature must be well below the point where chemical or physical changes might occur

For a controlled, stable containment state:

- A process must be in place for transferring the energy that is released to the containment to a long-term heat sink
- The containment boundary must be protected

- The containment and reactor coolant system conditions must be well below the point where chemical or physical processes (severe accident phenomena) might result in a dynamic change in containment conditions or a failure of the containment boundary.

D.4 Definition of Equipment Survivability Time Frames

The purpose of the equipment survivability time frames is to identify the time span in the severe accident in which specific equipment is required to perform its function. The phenomena and environment associated with that phase of the severe accident defines the environment which challenges the equipment survivability. The equipment survivability time frame definitions are summarized in Table D-1.

D.4.1 Time Frame 0 – Pre-Core Uncovery

Time Frame 0 is defined as the period of time in the accident sequence after the accident initiation and prior to core uncovery. The fuel rods are cooled by the water/steam mixture in the reactor vessel. The accident has not yet progressed beyond the design basis of the plant, and hydrogen generation and the release of fission products from the core is negligible. Emergency response guidelines (ERGs) are designed to maintain or recover the borated water inventory and heat removal in the reactor coolant system to prevent core uncovery and establish a safe, stable state. Recovery within Time Frame 0 prevents the accident from becoming a severe accident. Equipment survivability in Time Frame 0 is covered under the design basis equipment qualification program.

D.4.2 Time Frame 1 – Core Heatup

Time Frame 1 is defined as the period of time after core uncovery and prior to the onset of significant core damage as evidenced by the rapid oxidation of the core. This is the transition period from design basis to severe accident environment. The overall core geometry is intact and the uncovered portion of the core is overheating due to the lack of decay heat removal. Hydrogen releases are limited to relatively minor cladding oxidation and some noble gas and volatile fission products may be released from the fuel-clad gap. As the core-exit gas temperature increases, the ERGs transition to a red path indicating inadequate core cooling. The operators attempt to reduce the core temperature by depressurizing the RCS and re-establish the borated water inventory in the reactor coolant system. If these actions do not result in a decrease in core-exit temperature, the control room staff initiate actions to mitigate a severe accident by turning on the hydrogen igniters for hydrogen control and flooding the reactor cavity to prevent reactor pressure vessel failure. Recovery in Time Frame 1 prevents the accident from becoming a core melt. Equipment survivability in Time Frame 1 is evaluated to demonstrate it is within the equipment qualification envelope.

D.4.3 Time Frame 2 – In-Vessel Severe Accident Phase

Time Frame 2 is the period of time in the severe accident after the accident progresses beyond the design basis of the plant and prior to the establishment of a controlled, stable state (end of in-vessel core relocation), or prior to reactor vessel failure. The onset of rapid oxidation of the fuel rod cladding and hydrogen generation defines the beginning of Time Frame 2. The heat of the exothermic reaction accelerates the degradation, melting and relocation of the core. Fission products are released from the fuel-clad gap as the cladding bursts and from the fuel matrix as the UO_2 pellets melt. Over the period of Time Frame 2, the initial, intact geometry of the core is lost as it melts and relocates downward. Severe accident management strategies exercised during Time Frame 2 are designed to recover reactor coolant system inventory and heat removal, to maintain reactor vessel integrity and to maintain containment integrity. Recovery actions in Time Frame 2 may create environmental challenges by increasing the rate of hydrogen and steam generation.

D.4.4 Time Frame 3 – Ex-Vessel Severe Accident Phase

Time Frame 3 is defined as the period of time after the reactor vessel fails until the establishment of a controlled, stable state. The AP1000 reliably provides the capability to flood the reactor vessel and prevent the vessel failure in a severe accident. This severe accident time phase 3 is of such low frequency, it is considered to be remote and speculative. Molten core debris is relocated from the reactor vessel onto the containment cavity floor which creates the potential for rapid steam generation, core-concrete interaction and non-condensible gas generation. Severe accident management strategies implemented in Time Frame 3 are designed to monitor the accident progression, maintain containment integrity and mitigate fission product releases to the environment.

D.5 Definition of Active Operation Time

Equipment only needs to survive long enough to perform its function to protect the containment fission product boundary. In the case of some items, such as valves or motor-operators, once the equipment performs its function, it changes state and the function is completed. For other items, such as pumps, the equipment must operate continuously to perform its function. The time of active operation is the time during which the equipment must change state or receive power to perform its function.

D.6 Equipment and Instrumentation for Severe Accident Management

The AP600 emergency response guidelines (Reference D-2) and severe accident management guidance (SAMG) framework (Reference D-1), which are considered valid for AP1000, define actions that accomplish the goals for achieving a controlled, stable state and terminating fission product releases in a severe accident. The high level actions from the accident management framework are summarized in Table D-2 and provide the basis for the actions considered for identifying equipment. The purpose of this section is to review ERG and SAMG actions within each of the time frames of the severe accident to determine the equipment and instrumentation and the active operation time in which they are needed to provide reasonable assurance of achieving a controlled, stable state. The AP600-specific accident management framework is used to identify the equipment for performing the high level actions. These high level actions are applicable to AP1000.

The Westinghouse Owners Group (WOG) SAMG (Reference D-3) provides the primary input to the selection of the instrumentation used for monitoring the actions. The instrument used to diagnose the need for the action and monitor the response are listed. Instruments to evaluate potential negative impacts are covered under other high level actions in the framework and therefore are also considered for survivability.

The equipment and instrumentation used in each time frame are summarized in Tables D-3 through D-5.

D.6.1 Time Frames 0 and 1 – Accident Initiation, Core Uncovery and Heatup

Time Frame 0 represents the accident time prior to core uncovery. Time Frame 1 represents the time following core uncovery, prior to the rapid oxidation of the core. Aside from potential ballooning of the cladding, the core has not lost its initial intact geometry and is coolable.

During Time Frames 0 and 1, most of the equipment that is automatically actuated will receive a signal to start. However, given a severe accident sequence, some critical equipment does not actuate. From accident initiation until the time of core uncovery (Time Frame 0) the conditions are bounded by the design basis and covered under equipment qualification. During Time Frame 1, the environment is still within the design basis of the plant and the control room is operating within the Emergency Response Guidelines, but the conditions have the potential to degrade. To achieve a controlled, stable state, accident management, via the ERGs, is geared toward recovering the core cooling before the coolable geometry is lost. Failing that, the plant is configured to keep the core debris in the vessel, and mitigate the containment hydrogen that will be generated in Time Frame 2.

D.6.1.1 Injection into the RCS

Failure of RCS injection is likely to be the reason the accident has proceeded to core uncovery. Successful injection into the RCS removes the sensible and decay heat from the core. Prior to the rapid oxidation of the cladding, successful RCS injection essentially recovers the accident before it progresses to substantial core melting and relocation and establishes a controlled, stable state. Failure to inject into the RCS at a sufficient rate allows the accident to proceed into Time Frame 2 and the SAMG.

The equipment and systems used to inject into the RCS are the core makeup tanks, accumulators and IRWST (which are part of the passive core cooling system (PXS)), the chemical and volume control system (CVS) pumps, and the normal residual heat removal (RNS) pumps. For non-LOCA and small LOCA sequences, depressurization of the RCS is required for successful injection.

The plant response is monitored using the system flowrates, RCS pressure, core exit temperature, or RCS temperature.

D.6.1.2 Injection into Containment

The operator is instructed via the ERGs to inject water into the containment to submerge the reactor vessel and cool the external surface if injection to the RCS cannot be established. This action is performed at the end of Time Frame 1, immediately prior to entry into the SAMG. Successful cavity flooding prevents vessel failure in the event of molten core relocation to the vessel lower head. Failure of cavity flooding may allow the accident to proceed to vessel failure and molten core relocation into the containment (Time Frame 3) if timely injection into the reactor vessel cannot be established to cool the core and prevent substantial core relocation to the lower head.

The PXS motor-operated and squib recirculation valves are opened manually to drain the IRWST water into the containment.

The plant response is monitored by containment water level or IRWST water level indication.

D.6.1.3 Decay Heat Removal

In the event of non-LOCA or small LOCA sequences, the RCS pressure is elevated above the secondary pressure. Failure of the PRHR may be the initiating event of such sequences. Recovery of the PRHR will provide decay heat removal. Failure of feedwater to the steam generators with the PRHR failed may also be the initiating event for such sequences and recovery of injection to the steam generators may be required. If the steam generators remain dry without PRHR recovery and the core is uncovered, the tube integrity or hot leg nozzle integrity will be threatened by creep rupture failure at the onset on rapid oxidation (entry into Time Frame 2). Injecting to the steam generators provides a heat sink to the RCS by boiling water on the secondary side, and protects the tubes by cooling them. Successful steam generator injection can establish a controlled, stable state if the losses from the RCS can be recovered and mitigated. Failure to inject to the steam generator requires depressurization of the RCS to prevent creep rupture failure of the tubes and loss of the containment integrity at the onset of rapid oxidation in Time Frame 2.

For accident sequences initiated by steam generator tube rupture, the procedures instruct the control room to isolate injection to the faulted steam generator, and to use injection to the intact steam generator in conjunction with steam generator depressurization and PRHR initiation to cooldown the reactor coolant system and isolate the break. In Time Frame 1, PRHR initiation or injection to the intact steam generators may be used to re-establish a primary heat sink to cooldown the RCS and a controlled, stable state if the losses from the RCS can be recovered and mitigated. Failure to recover the PRHR or to inject to the steam generator may lead to a continued loss of coolant to the faulted steam generator and progression to Time Frame 2.

The main feedwater and startup feedwater pumps are used to inject into a pressurized secondary system. If the secondary system can be depressurized sufficiently, condensate, fire water or service water can also be used to inject into the secondary side.

The plant response is monitored with the steam generator level and steamline pressure.

D.6.1.4 Depressurize Reactor Coolant System

D.6.1.4.1 Non-LOCA and Small LOCA Sequences

In the event of non-LOCA or a small LOCA sequences, the RCS pressure is above the secondary pressure. If the steam generators are dry and the core is uncovered, the hot leg nozzle or tube integrity is threatened by creep rupture failure at the onset of rapid cladding oxidation (beginning of Time Frame 2). Timely depressurization (prior to significant cladding oxidation) of the RCS mitigates the threat to the tubes, allows injection of the accumulators and IRWST water, and provides a long-term heat sink to establish a controlled, stable state. Failure to depressurize can result in the failure of the tubes and a loss of containment integrity when oxidation begins.

For steam generator tube rupture (SGTR) initiated sequences, depressurization of the RCS can be used to isolate the faulted steam generator, and re-establish core cooling via injection.

The automatic depressurization system (ADS) is required to fully depressurize the RCS to allow the PXS systems to inject. However, the recovery of passive residual heat removal (PRHR) or injection to the steam generators will provide a substantial heat sink to depressurize the RCS and mitigate the threat to the tubes. The auxiliary pressurizer sprays are not evaluated for survivability since the inclusion of several other safety-related systems which perform the same function provides reasonable assurance of RCS depressurization in the event of a non-LOCA or small LOCA severe accident.

The RCS pressure, core-exit temperature and RCS temperature can be used to monitor the plant response to the RCS depressurization.

D.6.1.4.2 LOCA Sequences

LOCA sequences (other than small LOCA sequences) by definition are depressurized below the secondary system pressure by the initiating event and therefore, are not a threat to steam generator tube integrity upon the onset of rapid oxidation. Depressurization may be required for injection to establish a long-term heat sink. Medium LOCAs require additional depressurization to allow the injection of RNS or PXS. Large LOCAs are fully depressurized by the initiating event.

In LOCA sequences, only the ADS is effective in providing depressurization capability to allow injection to the RCS. Steam generator cooldown and auxiliary pressurizer sprays are not effective.

The RCS pressure, core-exit temperature and RCS temperature can be used to monitor the plant response to the RCS depressurization.

D.6.1.4.3 Prevent Reactor Vessel Failure

Depressurization of the RCS, along with injecting into the containment is an accident management strategy to prevent vessel failure. The depressurization of the RCS reduces the stresses on the damaged vessel wall facilitating the in-vessel retention of core debris.

The ADS is used to depressurize the RCS to prevent reactor vessel failure.

The RCS pressure, core-exit temperature and RCS temperature can be used to monitor the plant response to the RCS depressurization.

D.6.1.5 Depressurize Steam Generators

The steam generators are depressurized to facilitate low-pressure injection into the secondary system and to depressurize the RCS in non-LOCA and small LOCA sequences. Injection to the steam generator must be available to depressurize the secondary system to prevent creep rupture failure of the tubes.

The steam generator PORV and steam dump valves are used for depressurizing the steam generators.

Depressurization of the steam generators is outlined in the ERGs as a means to facilitate injection into the steam generators.

The steamline pressure and RCS pressure can be used to monitor the plant response.

D.6.1.6 Containment Heat Removal

Containment heat removal is not explicitly listed as a high level action in the AP600 SAMG Framework, but it is implicit in the high level action "Depressurize Containment." Containment heat removal is provided by the passive containment cooling system (PCS). Water cooling of the shell is needed to establish a controlled, stable state with the containment depressurized. The actuation of PCS water is typically automatic in Time Frame 0.

PCS water is supplied to the external surface of the containment shell from the PCS water storage tank or the post-72 hour water tank. Alternative water sources can be provided via separate connections outside containment.

The containment heat removal can be monitored with the containment pressure and the PCS water flowrate or PCS water storage tank level.

D.6.1.7 Containment Isolation

Containment isolation is not explicitly listed as a high level action in the AP600 SAMG Framework, but it is implicit as a requirement to protect the fission product barrier.

Containment isolation is provided by an intact containment shell and the containment isolation system which closes the isolation valve in lines penetrating the containment shell.

The containment isolation can be monitored by the containment pressure and the containment isolation system valve positions.

D.6.1.8 Hydrogen Control

Maintaining the containment hydrogen concentration below a globally flammable limit is a requirement for a controlled, stable state. The containment can withstand the pressurization from a global deflagration, but potential flame acceleration can produce impulsive loads for which containment integrity is uncertain. While hydrogen is not generated in a significant quantity until Time Frame 2, provisions are provided in the ERGs within Time Frame 1 to turn on the igniters before hydrogen generation begins so that hydrogen can be burned as it is produced.

Severe accident hydrogen control in the AP1000 is provided by hydrogen igniters. The containment has passive auto-catalytic recombiners (PARs) as well, but they are not credited for severe accidents.

The igniters are manually actuated from the control room in the ERGs on high core-exit temperature. The intention is to actuate the igniters prior to the cladding oxidation (Time Frame 1). The containment hydrogen concentration is monitored prior to igniter actuation so that a globally flammable mixture is not unintentionally ignited by the hydrogen igniters.

The plant response to the igniter actuation can be monitored by containment hydrogen concentration using the hydrogen monitors or containment atmosphere sampling, which is part of the primary sampling system. The containment pressure response can also be used to indicate hydrogen burning.

D.6.1.9 Accident Monitoring

Accident monitoring is a post-TMI requirement as outlined in 10 CFR 50.34(f). Aside from the accident management purposes outlined above, monitoring the progression of the accident and radioactive releases provides input to emergency response and emergency action levels.

Accident monitoring is provided by the in-containment monitors for pressure, hydrogen concentration, water levels, and radiation.

D.6.2 Time Frame 2 – In-Vessel Core Melting and Relocation

Time Frame 2 represents the period of core melting and relocation and the entry into the SAMG. The intact and coolable in-vessel core geometry is lost, and relocation of core debris into the lower head is likely. The in-vessel hydrogen generation and fission product releases from the fuel matrix occur during this time frame.

D.6.2.1 Injection into the RCS

In Time Frame 2, the in-vessel core configuration loses its coolable geometry and it is likely that at least some of the core debris will migrate to the reactor vessel lower head. If the RCS is depressurized and the reactor vessel is submerged, the core debris will be retained in the reactor vessel. However, injection into the RCS to cover and cool the core debris is required to achieve a controlled, stable state. RCS injection is not required to protect the containment fission product boundary. Injection is successful if it is sufficient to quench the sensible heat from the core debris and maintained to remove decay heat.

RCS injection is outlined from SAMG (Reference D-3). Water can be injected into the RCS using the CVS or the RNS systems. The PXS is not credited in Time Frame 2 because automatic and manual activation of the system is attempted several times in Time Frame 1, and diverse pumped systems are credited to provide reasonable assurance of RCS injection survivability in this time frame. Post-core damage, the actions may be monitored with RCS pressure or temperature or containment pressure.

D.6.2.2 Injection into Containment

The objective of injection to the containment prior to reactor vessel failure (Time Frame 3) is to cool the external surface of the reactor vessel to maintain the core debris in the vessel. Reasonable assurance of injecting to the containment for in-vessel retention is achieved by instructing the operator to drain the IRWST in the ERGs within Time Frame 1. After relocation of core debris to the lower head in Time Frame 2, the success of this action becomes uncertain. If the vessel fails, the accident progresses to Time Frame 3. Active operation for injection to containment is completed prior to Time Frame 2.

D.6.2.3 Decay Heat Removal

In transients and small LOCAs, initiation of PRHR or injection into the steam generators is required to be recovered in Time Frame 1 to be successful. Steam generator tubes or the hot leg nozzles will fail when the cladding oxidation begins at the onset of Time Frame 2. Steam generator injection is not required for LOCAs which depressurize the RCS below the secondary system pressure.

Within Time Frame 2 SAMG, steam generator injection can be utilized in unisolated SGTR sequences to maintain the water level on the secondary side for mitigation of fission product releases. Injecting into the steam generators, along with depressurization of the RCS, is an accident management action to isolate containment or scrub fission products. Failure to inject to the faulted steam generator in Time Frame 2 can lead to continued breech of the containment fission product boundary and large offsite doses.

The main feedwater and startup feedwater pumps are used to inject into a pressurized secondary system. If the secondary system can be depressurized sufficiently, condensate, fire water or service water can also be used to inject into the secondary side.

Injection into the steam generators is covered in the WOG SAMG (Reference D-3). The plant response is monitored with the steam generator level and steamline pressure.

D.6.2.4 Depressurize RCS

RCS depressurization is required within Time Frame 1 for facilitating in-vessel retention of core debris and for successfully preventing steam generator tube failure in high pressure severe accident sequences. The steam generator tubes or hot leg nozzles will fail due to creep rupture after the onset of rapid oxidation at the beginning of Time Frame 2. This action facilitates in-vessel retention of core debris in conjunction with injection into the containment to give time to recover pumped injection sources to establish a controlled, stable state. Reasonable assurance of successful RCS depressurization is provided by instructing the operator to depressurize the system in the ERGs in Time Frame 1. Active operation of RCS depressurization is completed prior to Time Frame 2.

D.6.2.5 Depressurize Steam Generators

Active operation to depressurize the steam generators is used to cooldown the RCS prior to Time Frame 2. After the onset of core melting and relocation, depressurizing steam generators could threaten steam generator tube integrity. Depressurizing the steam generator in Time Frame 2 does not facilitate the establishment of a controlled, stable state.

D.6.2.6 Containment Heat Removal

Reasonable assurance of successful containment heat removal is provided since automatic actuation of PCS water occurs in Time Frame 0. PCS flowrate and level are monitored to determine if additional water is needed. Alternate water sources can be provided by connections to the external PCS water tank which is outside the containment pressure boundary and not subjected to the harsh environment.

D.6.2.7 Containment Isolation

Active operation of containment isolation valves is required in Time Frame 0 or 1 to establish the containment fission product barrier. Therefore, only the survivability of the containment pressure boundary, including penetrations, is required to maintain containment isolation after Time Frame 1.

D.6.2.8 Hydrogen Control

The operator action to actuate the igniters occurs prior to the hydrogen generation at the onset of Time Frame 2. The igniters need to survive and receive power throughout the hydrogen release to maintain the hydrogen concentration below the lower flammability limit during the hydrogen generation in Time Frame 2.

D.6.2.9 Mitigate Fission Product Releases

A nonsafety-related containment spray system is provided in AP1000 to wash aerosol fission products from the containment atmosphere. The spray system is manually actuated from the SAMG which is entered at the onset of Time Frame 2. Operating the spray involves opening an air-operated valve inside the containment and actuating valves and a pump outside the containment. Once open, the active operation of the valve inside the containment is completed.

D.6.2.10 Accident Monitoring

During the initial core melting and relocation, containment hydrogen and radiation monitors are used for core damage assessment and verification of the hydrogen igniter operation. Steam generator radiation monitoring is used to determine steam generator tube integrity. In the longer term, containment atmosphere sampling can be used to monitor hydrogen and radiation. Containment pressure and temperature need to be monitored throughout Time Frame 2.

D.6.3 Time Frame 3 – Ex-Vessel Core Relocation

Time Frame 3 represents the phase of the accident after vessel failure. The core debris is in the reactor cavity, and the IRWST water is not injected into the containment.

D.6.3.1 Injection into the RCS

The RCS is failed. Injection to the RCS is no longer needed in Time Frame 3.

D.6.3.2 Injection into Containment

Reasonable assurance of sufficient water coverage to the ex-vessel debris bed is passively provided by the containment design to drain water from the RCS, CMTs, and accumulators to the lower containment. Water condensing on the PCS shell is returned to the reactor cavity after filling the IRWST to the overflow. Without draining the IRWST water to the cavity, the CMT, accumulator and RCS water provides sufficient water return to the cavity to maintain water coverage over the ex-vessel debris bed.

D.6.3.3 Decay Heat Removal

The RCS is failed. PRHR activation or injection into the steam generators is no longer needed in Time Frame 3. Injection to the steam generator for SGTR fission product scrubbing is not required to maintain the water level.

D.6.3.4 Depressurize RCS

The RCS is depressurized by the vessel failure in Time Frame 3.

D.6.3.5 Depressurize Steam Generators

The RCS is failed. Steam generator depressurization is not needed in Time Frame 3.

D.6.3.6 Containment Heat Removal

Active initiation of PCS water is completed prior to Time Frame 3. PCS flowrate and level are monitored for post-72 hour activities.

D.6.3.7 Containment Isolation and Venting

Continued operation of the containment shell as a pressure boundary is needed to maintain containment isolation in Time Frame 3.

In the event of containment pressurization above design pressure due to core concrete interaction non-condensable gas generation, the containment can be vented. Venting protects containment isolation by preventing an uncontrolled containment failure airborne release pathway. The vent can be opened and closed as required to maintain pressure in the containment below service Level C. Containment venting does not prevent or mitigate containment basemat failure due to core concrete interaction.

D.6.3.8 Combustible Gas Control

The hydrogen igniters are used to control combustible gases. Active operation of igniters continues to control the release of combustible gases from the degradation of concrete in the reactor cavity.

D.6.3.9 Mitigate Fission Product Releases

The nonsafety-related sprays are actuated in Time Frame 2. The operation of the nonsafety fire pump which provide containment spray continues, possibly into Time Frame 3, until the water from the source tank is depleted.

D.6.3.10 Accident Monitoring

Containment pressure, temperature, and the containment atmosphere sampling function are sufficient to monitor the accident in the long-term.

D.6.4 Summary of Equipment and Instrumentation

The equipment and instrumentation used in achieving a controlled, stable state following a severe accident, and the time it operates are summarized in Tables D-3 through D-5.

D.7 Severe Accident Environments

D.7.1 Radiation Environment – Severe Accident

The radiation exposure inside the containment for a severe accident is conservatively estimated by considering the dose in the middle of the AP1000 containment with no credit for the shielding provided by internal structures.

Sources are based on the emergency safeguards system core thermal power rating and the following analytical assumptions:

- Power Level (including 2% power uncertainty) .... 3,468 MWt

- Fraction of total core inventory released to the containment atmosphere:

  Noble Gases (Xe, Kr) 1.0
  Halogens (I, Br) 0.40
  Alkali Metals (Cs, Rb) 0.30
  Tellurium Group (Te, Sb, Se) 0.05
  Barium, Strontium (Ba, Sr) 0.02
  Noble Metals (Ru, Rh, Pd, Mo, Tc, Co) 0.0025
  Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am) 0.0002
  Cerium Group (Ce, Pu, Np) 0.0005

The radionuclide groups and elemental release fractions listed above are consistent with the accident source term information presented in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants - Final Report."

The timing of the releases are based on NUREG-1465 assumptions. The release scenario assumed in the calculations is described below.

An initial release of activity from the gaps of a number of failed fuel rods at 10 minutes into the accident is considered. The release of 5 percent of the core inventory of the volatile species (defined as noble gases, halogens, and alkali metals) is assumed. The release period occurs over the next 30 minutes, that is, from 10 to 40 minutes into the accident. At this point, 5 percent of the total core inventory of volatile species has been considered to be released.

Over the next 1.3 hours, releases associated with an early in-vessel release period are assumed to occur, that is, from 40 minutes to 1.97 hours into the accident. This source term is a time-varying release in which the release rate is assumed to be constant during the duration time. Additional releases during the early in-vessel release period include 95 percent of the noble gases, 35 percent of the halogens, and 25 percent of the alkali metals, as well as the fractions of the tellurium group, barium and strontium, noble metals, lanthanides, and cerium group as listed above.

There is no additional release of activity to the containment atmosphere after the in-vessel release phase.

The above source terms are consistent with the guidance provided by the NRC in Regulatory Guide 1.183 for design basis accident (DBA) loss-of-coolant accident (LOCA) evaluations.

The resulting instantaneous gamma and beta dose rates are provided in Figures D-1 and D-2, respectively.

D.7.2 Thermal-Hydraulic Environments

Bounding severe accident environments are provided in this section. Five severe accident cases are analyzed with the MAAP4.04 code to generate the environment. The MAAP4 code input parameters are set to produce bounding cladding oxidation in each of the analyses.

The five cases are:

- IGN – DVI line break with vessel reflood, cavity flooding, and igniter
- IVR – DVI line break with cavity flooding and igniters, no vessel reflood
- NOIGN – 4-inch DVI line break with vessel reflood, cavity flooding, and no igniters
- CCI – Large LOCA with igniters, no vessel reflood and no cavity flooding
- GLOB – Global burning of hydrogen from 100-percent cladding reaction

The event timing for each case is presented in Table D-6. These key events relate directly to the equipment survivability time frames.

D.7.2.1 Case IGN – Large In-Vessel Hydrogen Release Burned at Igniters

Case IGN provides a containment environment with a high rate of hydrogen generation from vessel reflooding, sustained steaming from stage 4 ADS, and hydrogen burning at the igniters. The MAAP4 results are presented in Figures D-3 through D-11.

The accident sequence is initiated by a DVI line break into a PXS compartment. The compartment floods with water from the IRWST and fills above the break elevation, allowing the vessel to reflood. Reflooding the overheated core causes a large fraction of the zirconium cladding to oxidize; however, relocation of the core to the lower plenum of the reactor vessel is prevented.

The hydrogen produced in-vessel is released to the containment through the ADS and through the break. It burns at igniters placed throughout the containment.

D.7.2.2 Case IVR – In-Vessel Retention of Core Debris with Cavity Flooding and Igniters, No Vessel Reflood

Case IVR provides a containment environment from the bounding in-vessel retention case. The MAAP4 results are presented in Figures D-12 through 20.

The case is initiated by a DVI line break. The cavity is flooded, but the PXS compartment is not. The break is above the water level in the compartment. Water is unable to get back into the vessel and reflood the core. The core melts and collects in the lower plenum of the vessel, but is not quenched. The external surface of the vessel lower head is cooled with water from the IRWST, and the vessel remains intact. The amount of hydrogen generation in-vessel is low since the cladding oxidation reaction is water-limited.

Hydrogen that is generated in-vessel is released to the containment through the ADS, and burns at the igniters placed throughout the containment.

D.7.2.3 Case NOIGN – Igniter Failure

Case NOIGN provides a containment environment with a high rate of hydrogen generation from vessel reflooding, sustained steaming from stage 4 ADS, and large global hydrogen burn in the long term. The MAAP4 results are presented in Figures D-21 through 29.

The accident sequence is initiated by a DVI line break into a PXS compartment. The compartment floods with water from the IRWST and fills above the break elevation, allowing the vessel to reflood. Reflooding the overheated core causes a large fraction of the zirconium cladding to oxidize; however, relocation of the core to the lower plenum of the reactor vessel is prevented.

The hydrogen produced in-vessel is released to the containment through the ADS and mixes in the containment. It is assumed to be ignited randomly at 8 hours and produces a large global burn.

D.7.2.4 Case CCI – Vessel Failure and Core Concrete Interaction

Case CCI provides a post-vessel failure containment environment with an unquenched and non-coolable debris bed in the reactor cavity. The MAAP4 results are presented in Figures D-30 through D-38.

The accident sequence is initiated by a spurious opening of an ADS stage 4 valve. The cavity is not flooded. The core melt progresses to vessel failure and debris is released to the containment.

There is little hydrogen produced in-vessel since the oxidation reaction is water-limited. However, hydrogen is released from the debris during the core-concrete interaction. The hydrogen burns at the igniters until the containment becomes oxygen-starved.

D.7.2.5 Case GLOB – Global Combustion of Hydrogen Produced from 100-Percent Cladding Reaction

Case GLOB presents a bounding hydrogen combustion case, burning the mass of hydrogen produced from 100-percent oxidation of the active cladding in the core. The oxidation reaction produces 788 kg of hydrogen. The MAAP4 results are presented in Figures D-39 through D-46.

D.7.2.6 Sustained Burning Environments

Sustained burning of combustible gases can occur in the containment as a diffusion flame at the location where the gas plume encounters a continuous oxygen source. Equipment that is needed after core damage (Time Frames 2 and 3) should either be located well away or shielded from these locations or there should be other redundant equipment located outside the zone of influence to demonstrate reasonable assurance of the function survivability.

Burning at igniters away from combustible gas sources will be limited. The igniters will light off the plume, and the flame will flash back to the source of the combustible gas within a compartment supplied with air. The locations of the diffusion flames can be identified by identifying the combustible gas release points in the containment. Combustible gas generation begins as hydrogen is released from the RCS to the containment during the in-vessel phase of the accident (Time Frame 2). After vessel failure (Time Frame 3), hydrogen and carbon monoxide can be released by core-concrete interaction. A continuous oxygen source can be provided in the compartments, which form the natural circulation flow path in the containment, the steam generator and loop compartment rooms, the CMT room, and the upper compartment. Therefore, diffusion flame environments can be postulated in these compartments near the combustible gas source.

Except for the break, the source locations are pre-determined by ADS vent points and the geometry of the containment. During the in-vessel phase of the accident, hydrogen will be released from the break and from the ADS system. If the system is fully depressurized, the sustained hydrogen release will be from the stage 4 ADS valves and, depending on the size and location, the break. ADS stages 1 through 3 relieve to the IRWST. The hydrogen is preferentially released from the IRWST through the pipe vents along the steam generator doghouse wall. The stage 4 valves relieve to the steam generator loop compartments. Generally, the break location can be postulated to be in one of the steam generator loop compartments also, which is essentially lumped together with the hydrogen release from the ADS stage 4 valves, but piping connected to the RCS is also located in the accumulator rooms or valve vaults and the CVS compartment. For releases to these dead-ended compartments, the plume cannot encounter an oxygen supply until it reaches the CMT room.

Ex-vessel combustible generation occurs in the reactor cavity in Time Frame 3. The reactor cavity does not have a continuous air supply, so the first locations where oxygen is available along the flow pathways is where the sustained burning can be postulated. The flow paths from the reactor cavity to the containment air supply are through the RCDT room access into the vertical access tunnel, through the loop nozzle holes into the steam generator rooms, and past the reactor vessel flange through the seal ring (should the seal ring fail) into the refueling pool.

Therefore, sustained burning can be postulated in the following locations:

1. IRWST pipe vent exits along the SG doghouse wall (Time Frame 2)
2. Stage 4 valves outlet in the steam generator loop compartments at 112-ft elevation (Time Frame 2)
3. Vents from the accumulator rooms into the CMT room (Time Frame 2). Sustained burning will, however, exist from only one of the two accumulator rooms for a given break.
4. RCDT room access into the vertical access tunnel (Time Frame 3)
5. Loop nozzle holes into the steam generator loop compartment (Time Frame 3)
6. Seal ring into the refueling pool (Time Frame 3)

D.8 Assessment of Equipment Survivability

Since severe accidents are very low probability events, the NRC recommends in SECY-93-087, that equipment desired to be available following a severe accident need not be subject to the qualification requirements of 10CFR50.49, the quality assurance requirements of 10CFR50 Appendix B, or the redundancy/diversity requirements of 10CFR50 Appendix A. It is satisfactory to provide reasonable assurance that the designated equipment will operate following a severe accident by comparing the AP1000 severe accident environments to design basis event/severe accident testing or by design practices.

D.8.1 Approach to Equipment Survivability

The approach to survivability is by equipment type, equipment location, survival time required, and the use of design basis event qualification requirements and severe environment experimental data.

D.8.1.1 Equipment Type

The various types of equipment needed to perform the activities discussed above are transmitters, thermocouples, resistance temperature detectors (RTDs), hydrogen and radiation monitors, valves, pumps, valve limit switches, containment penetration assemblies, igniters, and cables.

D.8.1.2 Equipment Location

Some of the in-containment equipment, i.e., transmitters, have been deliberately located to avoid the most severe calculated environments. Other equipment is located outside containment. The performance of the equipment was judged based on the most severe postulated event for that location.

D.8.1.3 Time Duration Required

Requirements are defined for each time frame, so the equipment evaluation only discusses performance during these periods. A limited amount of equipment has been designated for the long term (Time Frame 3) and these parameters can be monitored outside containment.

D.8.1.4 Severe Environment Experiments

The primary source for performance expectations of similar equipment in severe accident environments is EPRI NP-4354, "Large Scale Hydrogen Burn Equipment Experiments." This information is supplemented by NUREG/CR-5334, "Severe Accident Testing of Electrical Penetration Assemblies." These programs tested equipment types that had previously been qualified for design basis event environmental conditions. The temperature in the chamber for the first program was in the 700° – 800°F range for ten to twenty minutes during the continuous hydrogen injection tests. Although the conditions at the equipment would be somewhat less severe, the chamber conditions envelop all of the longer duration profiles indicated for the AP1000 events. The equipment in this program was also exposed to significant hydrogen burn spikes that are also postulated for the AP1000. The same equipment was exposed to and survived several events, both pre-mixed and continuous hydrogen injection which provides confidence in its ability to survive a postulated severe accident. The second program tested containment penetrations to high temperatures for long durations. A penetration was tested under severe accident conditions simulated with steam up to 400°F and 75 psia for ten days. The results indicated that the electrical performance of the penetration would not lead to degraded equipment performance for the first four days. The mechanical performance did not degrade (no leaks) during the entire test.

D.8.2 Equipment Located in Containment

The exposure to elevated temperatures as a direct result of the postulated severe accident or as a result of hydrogen burning is the primary parameter of interest. Pressure environments do not exceed the design basis event conditions for which the equipment has been qualified. Radiation environments also do not exceed the design basis event conditions throughout Time Frames 1 & 2.

D.8.2.1 Differential Pressure and Pressure Transmitters

The functions defined for severe accident management that utilize in-containment transmitters are IRWST water level, reactor coolant system pressure, steam generator wide range water level and containment pressure. Most of these transmitters that provide this information are located in rooms where the environment is limited to short duration temperature transients. These transients exceed ambient design basis temperature conditions but should not impact the transmitter performance since the internal transmitter temperature do not increase significantly above that experienced during design basis testing. EPRI NP-4354 documents transmitter performance during several temperature transients with acceptable results. The IRWST water level transmitters are located in the maintenance floor and are only required during Time Frames 1 & 2. The environment during Time Frames 1 & 2 does not exceed the design basis qualification parameters of the transmitters. Reactor system pressure and steam generator wide range water level are required through the second time frame. The only long term application is the containment pressure transmitter which may eventually be impacted by the severe accident radiation dose. Containment pressure could also be measured outside containment if necessary.

D.8.2.2 Thermocouples

The functions defined for severe accident management that utilize thermocouples are core exit temperature and containment water level. The core exit temperature is only required during Time Frame 1 and the containment water level is required through Time Frame 2. The temperatures to which the thermocouples are exposed during the defined time frames do not exceed the thermocouple design.

D.8.2.3 Resistance Temperature Detectors (RTDs)

Both hot and cold leg temperatures are defined as parameters for severe accident management in Time Frame 1. RTDs are utilized for these measurements and will perform until their temperature range is exceeded. The hot leg RTDs could fail as the temperature increases well above the design conditions of the RTDs but the cold leg RTDs should perform throughout Time Frame 1. RTDs are also utilized through Time Frame 3 for the containment temperature measurement and are exposed to temperature transients that exceed design basis qualification conditions. EPRI

NP-4354 documents RTD performance during several temperature transients with acceptable results.

D.8.2.4 Hydrogen Monitors

Containment hydrogen is defined as a parameter to be monitored throughout the severe accident scenarios. Early in the accident, the hydrogen is monitored by a device that operates on the basis of catalytic oxidation of hydrogen on a heated element. The hydrogen monitors are located in the main containment area. The design limits of this device may be exceeded after the first few hours of some of the postulated accidents and performance may be uncertain. If the device fails, hydrogen concentration is determined through the containment atmosphere sampling function.

D.8.2.5 Radiation Monitors

Containment radiation is defined as a parameter to be monitored throughout the severe accident scenarios. The containment radiation monitors are located in the main containment area. Early in the accident, the design basis event qualified containment radiation monitor provides the necessary information until the environment exceeds the design limits of the monitor. If the device fails, containment radiation is determined through the containment atmosphere sampling function.

D.8.2.6 Solenoid Valve

Qualified solenoid valves are used to vent air-operated valves (AOVs) to perform the function required. In Time Frame 1, the core makeup tank AOVs located in the accumulator room provide a path for RCS injection, the PRHR AOVs located in the maintenance floor provide a path for RCS heat removal and the containment is isolated by AOVs located in the maintenance floor and the PXS valve/accumulator room. The environment to which these solenoid valves may be exposed in Time Frame 1 is not significantly different than the design basis events to which the devices are qualified. In Time Frame 2, the RCS boundary AOV located in the maintenance floor is used for CVS injection into the RCS and the containment spray AOV located in the maintenance floor is used for control of fission product release. In addition, throughout Time Frame 3, access to the containment environment from the containment atmosphere sampling function is through solenoid valves located in the maintenance floor. During Time Frames 2 and 3, these valves may be exposed to transient conditions due to hydrogen burns that exceed design basis event qualification. Solenoid valves in an energized condition were included in the hydrogen burn experiments (EPRI NP-4354) and survived many transients. Shielding provided by the location of the valves limits the severe accident radiation dose to the typical design basis qualification dose for these valves.

D.8.2.7 Motor-Operated Valves

Motor-operated valves (MOVs) are utilized in several applications during the severe accident scenarios. MOVs in the accumulator and core makeup tank path are normally open and remain open. In Time Frame 1, the PXS recirculation MOVs located in the PXS valve/accumulator room are required for injection of water into the containment, MOVs for the first three stages of ADS located in a compartment above the pressurizer are required for RCS depressurization and the containment is isolated by MOVs located in the maintenance floor and the PXS valve/accumulator room. The environment to which these MOVs may be exposed in Time Frame 1 is not significantly different than the design basis events to which they are qualified. In Time Frame 2, the charging and injection MOV located in the maintenance floor provides a path from the CVS for RCS injection, an RNS MOV located in the PXS valve/accumulator room provides a path from the IRWST for RCS injection and an RNS MOV located in the WLS monitor tank room provides a path from the cask loading pit for RCS injection. In addition, throughout Time Frame 3, containment venting to the spent fuel pool is available through RNS hot leg suction line MOVs located in the RNS valve room. During Time Frames 2 and 3, these valves may be exposed to transient conditions due to hydrogen burns that exceed design basis event qualification. MOVs were included in the hydrogen burn experiments (EPRI NP-4354) and survived many transients. Shielding provided by the location of the valve limits the severe accident radiation dose to the typical design basis qualification dose for these valves.

D.8.2.8 Squib Valves

Squib valves are only required in Time Frame 1 when the severe accident environment is not significantly different than the design basis environment for which these valves are qualified. IRWST and PXS recirculation squib valves located in the accumulator room are used for injection into the RCS and containment, respectively. For RCS depressurization, the fourth stage ADS squib valves are located in steam generator compartments 1 and 2.

D.8.2.9 Position Sensors

Position sensors are required to monitor the position of containment isolation valves that could lead directly to an atmospheric release. These isolation valves actuate early in the transient, so verification is only required during Time Frame 1. The position sensors are located in the maintenance floor and the environment in this time frame does not exceed the design basis event qualification environment of the position sensors.

D.8.2.10 Hydrogen Igniters

The hydrogen igniters are distributed throughout the containment and are designed to perform in environments similar to those postulated for severe accidents. The igniters' transformers are located outside containment. The successful results of glow plug testing through several hydrogen burns is documented in EPRI NP-4354 and provides confidence in the performance of these devices.

D.8.2.11 Electrical Containment Penetration Assemblies

The electrical containment penetrations are located in the lower compartment and are required to perform both electrically and mechanically throughout the severe accident. The hydrogen burn equipment experiments documented by EPRI NP-4354 included penetrations qualified for nuclear plants. Electrical testing on the penetration cables after all the pre-mixed and continuous injection tests concluded that most of the cables passed the electrical tests while submerged in water. These tests consisted of ac (at rated voltage) and dc (at three times rated voltage) withstand tests and insulation resistance tests at 500 volts. The penetrations were also tested under simulated severe accident conditions at 400°F and 75 psia for about 10 days (NUREG/CR-5334). The results indicated that some degradation in instrumentation connected to the penetration may occur in four days under these severe conditions. The maintenance floor may experience short temperature transients above 400°F but stable temperatures are significantly less, so it is expected that the electrical performance would be maintained throughout the event. The only long term measurement utilizing these penetrations is containment pressure and this can be measured outside containment if necessary. There was no degradation of mechanical performance of the electrical penetrations (maintaining the seal) in either test program.

D.8.2.12 Cables

The hydrogen burn equipment experiments documented by EPRI NP-4354 included twenty-four different cable types qualified for nuclear plants. Electrical testing on these cables after all the pre-mixed and continuous injection tests concluded that all (fifty two samples) of the cables passed the electrical tests while submerged. These tests consisted of ac (at rated voltage) and dc (at three times rated voltage) withstand tests and insulation resistance tests at 500 volts. Due to the exposure to many events, some cable samples had extensive damage in the form of charring, cracking and bulging of the outer jackets and still performed satisfactorily. The cables tested are representative of cables specified for the AP1000 and are only exposed to short single temperature transients in their respective locations. Proper performance can be expected. The only long term measurement utilizing cables is containment pressure, which can be measured outside containment if necessary.

D.8.2.13 Assessment of Equipment for Sustained Burning

The equipment necessary for equipment survivability in sustained burning environments is defined in Tables D-3 through D-5. The equipment in Table D-3 includes equipment and instrumentation operation during Time Frame 1 - core uncovery and heatup, and is prior to the release of significant quantities of hydrogen. Therefore, it does not have to be qualified for sustained hydrogen burning. Table D-7 specifies the equipment and instrumentation used in Time Frames 2 and 3 to provide reasonable assurance of achieving a controlled stable state.

D.8.3 Equipment Located Outside Containment

Other functions defined for severe accident management are performed outside containment and the equipment is not subjected to the harsh environment of the event. This equipment includes:

- The steamline radiation monitor,
- Transmitters for monitoring steamline pressure,
- The passive containment cooling system flow and tank level,
- The containment atmosphere sampling function,
- The CVS pumps and flow measurement,
- The RNS pumps and flow measurement,
- SFS MOV for injection to the IRWST,
- RNS MOV for injection from cask loading pit to RCS
- MFW pumps and valves,
- SFW pumps and valves and condensate,
- Fire water and service water to feed steam generators
- Steam generator PORVs and steam dump valves for depressurization,
- PCS valves and fire water pumps and valves for containment heat removal,

- Containment isolation valves (outside containment),
- Auxiliary building radiation monitor,
- MOV and manual valve from RNS hot leg suction lines to the spent fuel pool and
- Fire water, fire pumps, valves and flow measurement used to provide containment spray and containment cooling.

D.9 Conclusions of Equipment Survivability Assessment

The equipment defined for severe accident management was reviewed for performance during the environments postulated for these events. Survivability of the equipment was evaluated based on design basis event qualification testing, severe accident testing, and the survival time required following the initiation of the severe accident. The equipment that is qualified for design basis events, has a high probability of surviving postulated severe accident events and performing satisfactorily for the time required.

AP1000 provides reasonable assurance that equipment, both electrical and mechanical, used to mitigate the consequences of severe accidents and achieve a controlled, stable state can perform over the time span for which they are needed.

D.10 References

D-1 "Framework for AP600 Severe Accident Management Guidance," WCAP-13914.

D-2 AP600 Emergency Response Guidelines.

D-3 Westinghouse Owner's Group Severe Accident Management Guidance, June 1994.

Table D-1

DEFINITION OF EQUIPMENT SURVIVABILITY TIME FRAMES

Time Frame	Beginning Time	Ending Time	Comments
0	Accident initiation	safe, stable state or core uncovery	• Bounded by design basis equipment qualification environment
1	Core uncovery	controlled, stable state or rapid cladding oxidation	• Core uncovery and heatup • Bounded by design basis equipment qualification environment
2	Rapid cladding oxidation	controlled, stable state or vessel failure	• In-vessel core melting and relocation • Entry into SAMG
3	Vessel failure	controlled, stable state or containment failure	• Ex-vessel core relocation

Table D-2

AP1000 HIGH LEVEL ACTIONS RELATIVE TO ACCIDENT MANAGEMENT GOALS
(taken from Table 5-1, Reference D-1)

Goal	Element	High Level Action*
Controlled, stable core	water inventory in RCS	• inject into RCS • depressurize RCS
	water inventory in containment	• inject into containment
	heat transfer to IRWST	• initiate PRHR
	heat transfer to SGs	• inject into RCS • inject into SGs • depressurize SGs
	heat transfer to containment	• inject into RCS • inject into containment • depressurize RCS • initiate PRHR
Controlled, stable containment	heat transfer from containment	• depressurize containment • vent containment • water on outside containment
	isolation of containment	• inject into SGs • depressurize RCS
	hydrogen prevention/control	• burn hydrogen • pressurize containment • depressurize RCS • inject into containment • vent containment • water on outside containment
	core concrete interaction prevention	• inject into containment
	high pressure melt ejection prevention	• inject into containment • depressurize RCS
	creep rupture prevention	• depressurize RCS • inject into SGs
	containment vacuum prevention	• pressurize containment
Terminate fission product release	isolation of containment	• inject into SGs • depressurize RCS
	reduce fission product inventory	• inject into containment • depressurize RCS
	reduce fission product driving force	• depressurize containment • water on outside containment

* See Tables D-3, D-4 and D-5

Table D-3 (Sheet 1 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION PRIOR TO END OF TIME FRAME 1 - CORE UNCOVERY AND HEATUP

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS	• PXS • CVS • RNS • IRWST	• core exit t/c's • RCS pressure • RCS RTDs • CVS flow • RNS flow • IRWST water level	• restore core cooling	• injection must often be recovered to be successful in severe accident
Inject Into Containment	• PXS recirc • SFS injection to refueling cavity • IRWST drains	• core-exit t/c's • containment water level • IRWST water level	• prevent vessel failure	• manual cavity flooding action in ERG
Decay heat removal	Initiate PRHR • High Pressure - MFW - SFW • Low Pressure - condensate - fire water - service water	• IRWST water level • SG WR water level • steamline pressure	• establish heat sink • make SGs available to depressurize RCS • prevent creep rupture	• injection source must often be recovered to be successful in severe accident
Depressurize RCS	• Pressurizer spray • ADS • PRHR HX • via SGs	• RCS pressure • core-exit t/c's • RCS RTDs • IRWST water level	• facilitate injection to RCS • long-term heat transfer path	• ADS often automatic
			• prevent creep rupture • containment integrity	• RCS depressurization required prior to cladding oxidation to prevent creep rupture
			• isolate break in SGTR	• uses intact SG or PRHR
			• prevent vessel failure	• requires injection to containment to be successful

Table D-3 (Sheet 2 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION PRIOR TO END OF TIME FRAME 1 - CORE UNCOVERY AND HEATUP

Action	Equipment	Instrumentation	Purpose	Comment
Depressurize SGs	• SG PORV • Steam dump	• steamline pressure • RCS pressure	• facilitate injection to SGs • depressurize RCS	• requires injection into SGs to prevent creep rupture
Containment Heat Removal	• PCS water • external water	• containment pressure • PCS flowrate • PCS tank level	• containment integrity • alleviate environmental challenge to equipment • long-term heat transfer path	• PCS water often automatic
Containment Isolation	• containment isolation system • containment shell • penetrations	• containment isolation system valve position • containment pressure	• containment integrity	• containment isolation system often automatic • manual action in ERG
Control Hydrogen	• igniters	• containment hydrogen concentration • containment pressure	• containment integrity	• manual igniter action in ERG
Accident Monitoring		• SG radiation • containment pressure • containment temperature • containment hydrogen concentration • containment water level • containment radiation	• accident management • emergency response • emergency action levels	• required by 10 CFR 50.34(f)

Table D-4 (Sheet 1 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION DURING TIME FRAME 2 - IN-VESSEL CORE MELTING AND RELOCATION

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS	• CVS • RNS	• RCS pressure • containment pressure • CVS flow • RNS flow	• cool core debris	• RCS injection needed to cool in-vessel debris for reasonable assurance of controlled, stable state
Inject Into Containment				• active operation completed in Time Frame 1
Decay heat removal	• PRHR HX • High Pressure - MFW - SFW • Low Pressure - Condensate - Fire Water - Service Water	• IRWST water level • SG WR water level • steamline pressure	• cool core debris • isolate containment in SGTR • scrub fission products	• also requires RCS depressurization for success of SG injection
Depressurize RCS				• active operation completed in Time Frame 1
Depressurize SGs				• active operation completed in Time Frame 1
Containment Heat Removal		• PCS flowrate • PCS tank level		• active operation completed in Time Frame 1
Containment Isolation	• containment shell • penetrations	• containment pressure	• containment integrity	• containment isolation system active operation completed in Time Frame 1
Control Hydrogen	• igniters	• containment hydrogen monitors • containment atmosphere sampling function	• containment integrity	• active operation continues in Time Frame 2 • monitors only required initially to verify hydrogen igniter operation

Table D-4 (Sheet 2 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION DURING TIME FRAME 2 - IN-VESSEL CORE MELTING AND RELOCATION

Action	Equipment	Instrumentation	Purpose	Comment
Control Fission Product Releases	• fire (spray) pump • spray valve	• spray flowrate • containment pressure	• scrub aerosols	• manual action within SAMG
Accident Monitoring		• containment pressure • containment temperature • containment atmosphere sampling function • aux bldg. radiation monitors • SG radiation monitors	• accident management • emergency response • emergency action levels	• active operation continues in Time Frame 2

Table D-5 (Sheet 1 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION DURING TIME FRAME 3 - EX-VESSEL CORE RELOCATION

Action	Equipment	Instrumentation	Purpose	Comment
Inject into RCS				• not needed in Time Frame 3
Decay heat removal				• injection of CMTs and accumulators in Time Frame 1 provides reasonable assurance of water coverage to ex-vessel core debris
Inject into SGs				• not needed in Time Frame 3
Depressurize RCS				• not needed in Time Frame 3
Depressurize SGs				• not needed in Time Frame 3
Containment Heat Removal		• PCS flowrate • PCS tank level		• active operation completed in Time Frame 1
Containment Isolation	• containment shell • penetrations	• containment pressure	• containment integrity	• active operation of containment isolation system completed in Time Frame 1
	• RNS hot leg suction MOVs		• containment vent	• manual action within SAMG
Control Hydrogen	• igniters	• containment atmosphere sampling function	• containment integrity	• active operation continues in Time Frame 3

Table D-5 (Sheet 2 of 2)

EQUIPMENT AND INSTRUMENTATION OPERATION DURING TIME FRAME 3 - EX-VESSEL CORE RELOCATION

Action	Equipment	Instrumentation	Purpose	Comment
Control Fission Product Release	• spray pump	• spray flowrate	• scrub fission products	• active operation continues
Accident Monitoring		• containment pressure • containment temperature • containment atmosphere sampling function • aux bldg. radiation monitors • SG radiation monitors	• accident management • emergency response • emergency action levels	• active operation continues in Time Frame 3

Table D-6

SUMMARY OF MAAP4 ANALYSES: EQUIPMENT SURVIVABILITY TIME FRAMES

Key Quantity or Timing	Sequences				
	IGN	**IVR**	**NOIGN**	**CCI**	**GLOB**
Cladding Oxidation In-Vessel (%)	78	48	86		100
Time of Core Uncovery (second)	2481	2483	2481	2285	14
Time (second) Core Exit Gas Temp. >1367°K	3318	3320	3318	3672	160
Time of Initial Core Material Relocation to Lower Plenum (second)	–	11100	–	5940	–
Time Core Material Relocation to Lower Plenum Ends (second)	–	11600	–	9000	–
Time of Vessel Failure (second)	–	–	–	9288	–

Table D-7 (Sheet 1 of 3)

SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Equipment	
PXS equipment (injection)	The PXS equipment utilized for introduction of cooling water includes component redundancy and is separated into two delivery flow paths. The two flow paths are physically separated into two trains such that if one train is disabled due to a sustained burn from DVI or other line break within that subsystem, the other subsystem will function.
CVS equipment (injection)	The equipment providing for CVS injection is located within the CVS compartment with the exception of the CVS makeup isolation valve. In accordance with the above, a sustained burn will not occur within the CVS compartment and, therefore, the equipment within this compartment utilized for CVS makeup will be operable. The CVS makeup isolation valve is normally in the correct position for severe accident scenario and is considered operable.
RNS equipment (injection)	Injection via the RNS is dependent only upon check valves within containment and, therefore, is not susceptible to sustained burning effects.
Main Feedwater (high pressure injection into the SG)	The operability of main feedwater system to inject high pressure feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Startup Feedwater (high pressure injection into the SG)	The operability of startup feedwater system to inject high pressure feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Condensate (low pressure injection into the SG)	The operability of the condensate system to provide makeup for low pressure feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Fire Water (low pressure injection into the SG), containment spray, and external containment vessel cooling	The operability of the fire water system to provide makeup for low pressure feedwater to steam generators, for containment spray and for external containment vessel cooling is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.
Service Water (low pressure injection into the SG)	The operability of the service water system to provide makeup for low pressure feedwater to steam generators is not dependent upon equipment located within containment and, therefore, is not susceptible to sustained burning effects.

Table D-7 (Sheet 2 of 3)

SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Equipment	
Containment Shell	The operability of the containment shell during sustained burning is addressed by Reference D-5.
Igniters	Igniters are specified and designed to withstand the effects of sustained burning and, therefore, are considered operable for these events.
Instrumentation	
RCS Pressure	There are four RCS pressurizer pressure transmitters. Two transmitters are located at a distance greater than 75 feet from the vent from the PXS valve/accumulator room and are, therefore, beyond the distance that potentially causes operability concerns from a sustained flame. The other two transmitters are located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns.
Containment Pressure	There are three extended range containment pressure transmitters. The three transmitters are located such that they cannot all be exposed to a sustained flame from either of the vents from the PXS valve/accumulator room into the maintenance floor at the base of the CMTs. Therefore, continued operability of the containment pressure function is provided.
SG 1 Wide Range Level	There are four steam generator wide range levels for SG 1. Two of the transmitters are located at a distance of greater than 20 feet from a CMT and are, therefore, beyond the distance that could potentially cause operability concerns from a sustained flame from the vent from the PXS valve/accumulator room into the maintenance floor at the base of the CMT. The other two transmitters are located over 20 feet below the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns.
SG 2 Wide Range Level	Based on the layout of the four steam generator wide range levels for SG 2, at least two of the transmitters will not be exposed to a sustained flame from either of the vents from the PXS valve/accumulator room into the maintenance floor at the base of the CMTs. Therefore, continued operability of the SG 2 wide range level indication function is provided.

Table D-7 (Sheet 3 of 3)

SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT

EQUIPMENT AND INSTRUMENTATION	SUSTAINED HYDROGEN COMBUSTION SURVIVABILITY ASSESSMENT
Instrumentation	
Containment Hydrogen Monitors	There are 3 distributed containment hydrogen monitors. There are no sustained burns that could potentially affect the two sensors that are located at an elevation of 164 feet or the sensor located within the dome.
Containment Atmosphere Sampling Function	The capabilities to perform containment atmosphere sampling are discussed in Section 9.3.3.1.2.2 – Post-Accident Sampling. Successful containment atmosphere sampling is dependent on the availability of either of the hot leg sample source isolation valves and the containment isolation valves in series with the isolation valve. The sample isolation valve from reactor coolant hot leg number 1 is located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns. The sample isolation valve from reactor coolant hot leg number 2 is located in a different room from the fourth stage ADS valves. This precludes radiative heating, which could potentially cause operability concerns. The containment isolation valves are located less than 20 feet from a CMT. However, a steel shroud around base of the CMT prevents a sustained flame existing on the containment side of that CMT and, therefore, affecting the operability of either of the containment isolation valves.

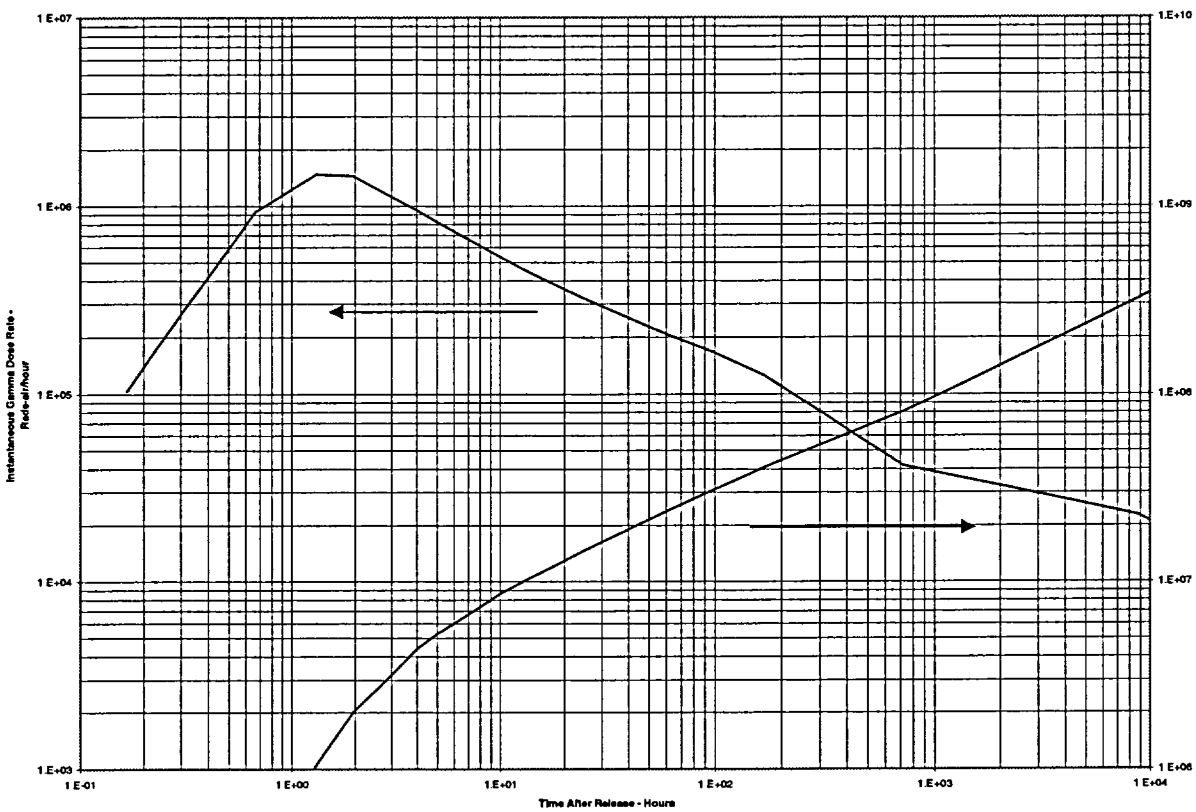


Figure D-1

Post-LOCA Gamma Dose and Dose Rate Inside Containment

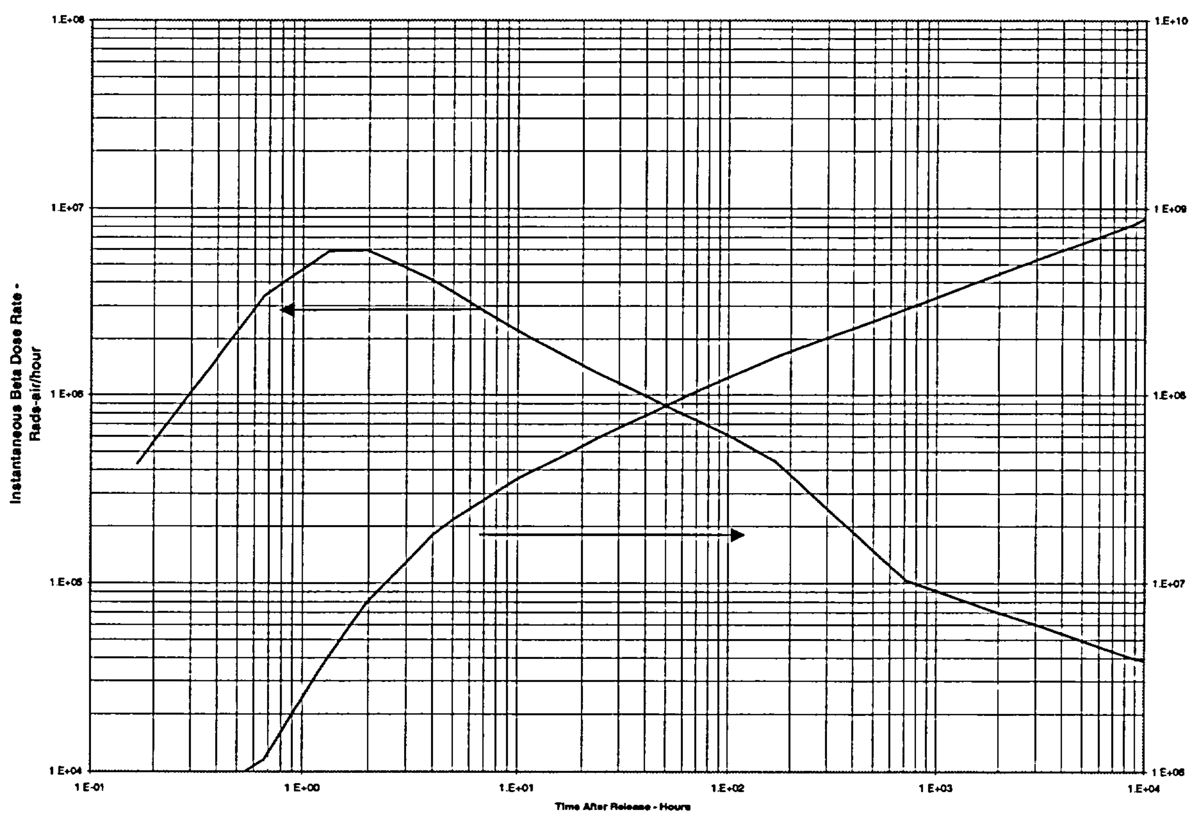


Figure D-2

Post-LOCA Beta Dose and Dose Rate Inside Containment

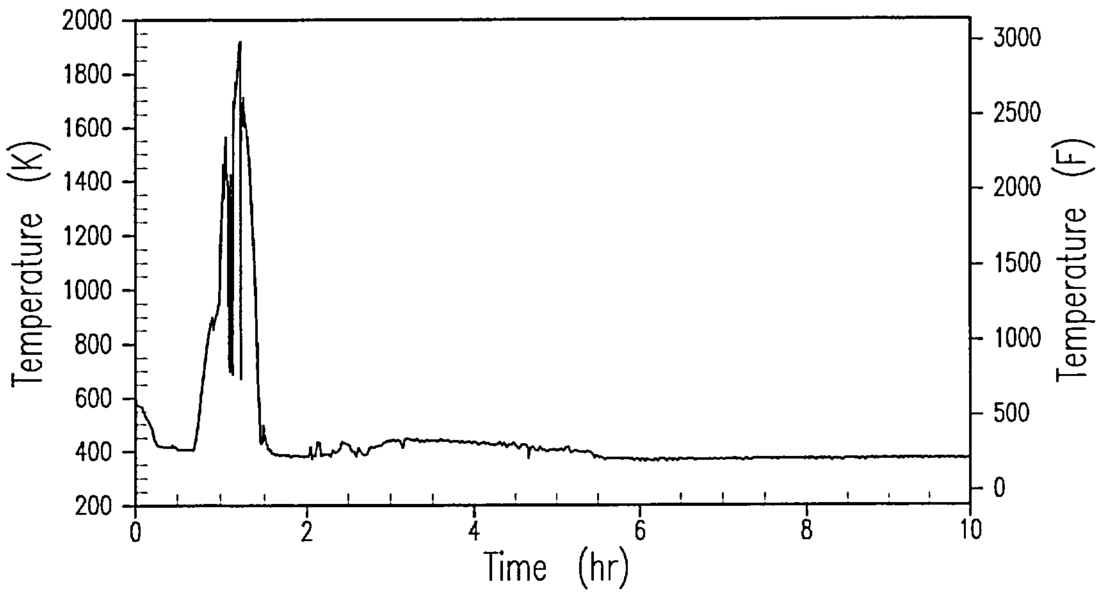


Figure D-3

Equipment Survivability Case IGN – Hydrogen Burning at Igniters
Upper Plenum Gas Temperature

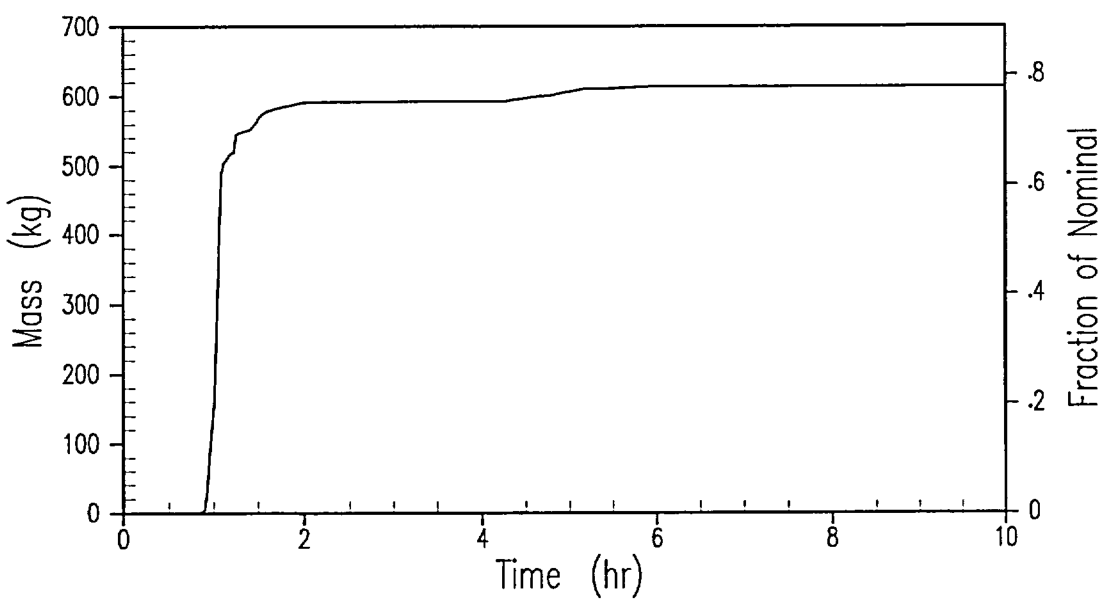


Figure D-4

Equipment Survivability Case IGN – Hydrogen Burning at Igniters
In-Vessel Hydrogen Generation

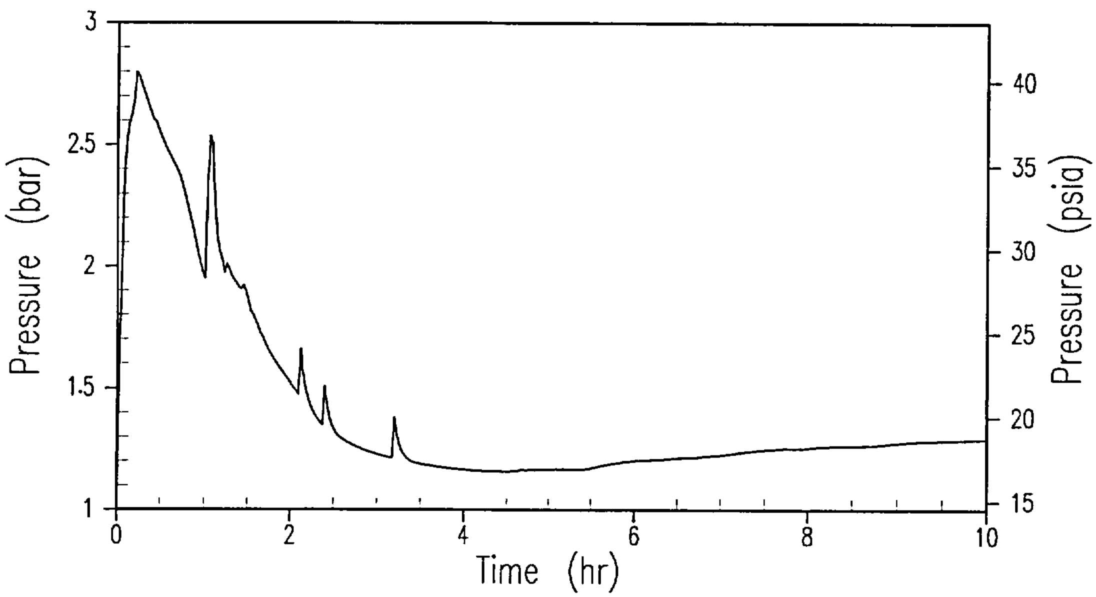


Figure D-5

Equipment Survivability Case IGN – Hydrogen Burning at Igniters Containment Upper Compartment Pressure

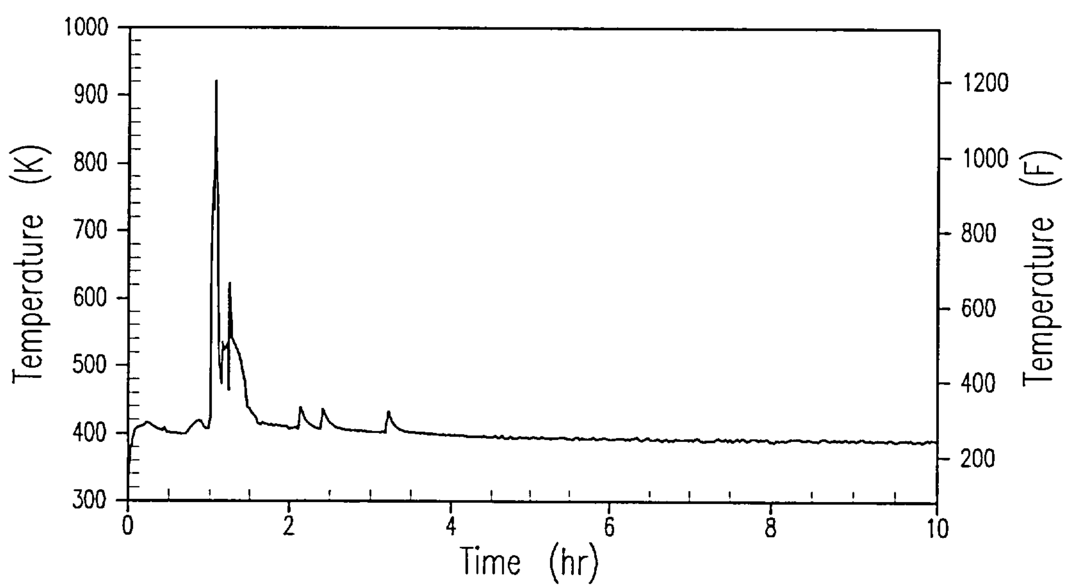


Figure D-6

Equipment Survivability Case IGN – Hydrogen Burning at Igniters SG Compartment 1 Gas Temperature

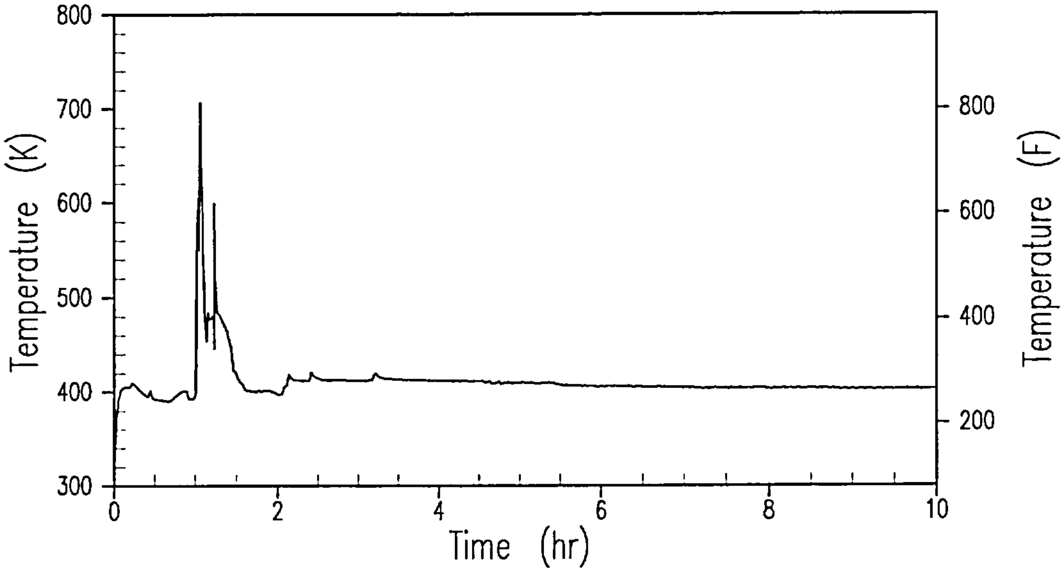


Figure D-7

Equipment Survivability Case IGN – Hydrogen Burning at Igniters
SG Compartment 2 Gas Temperature

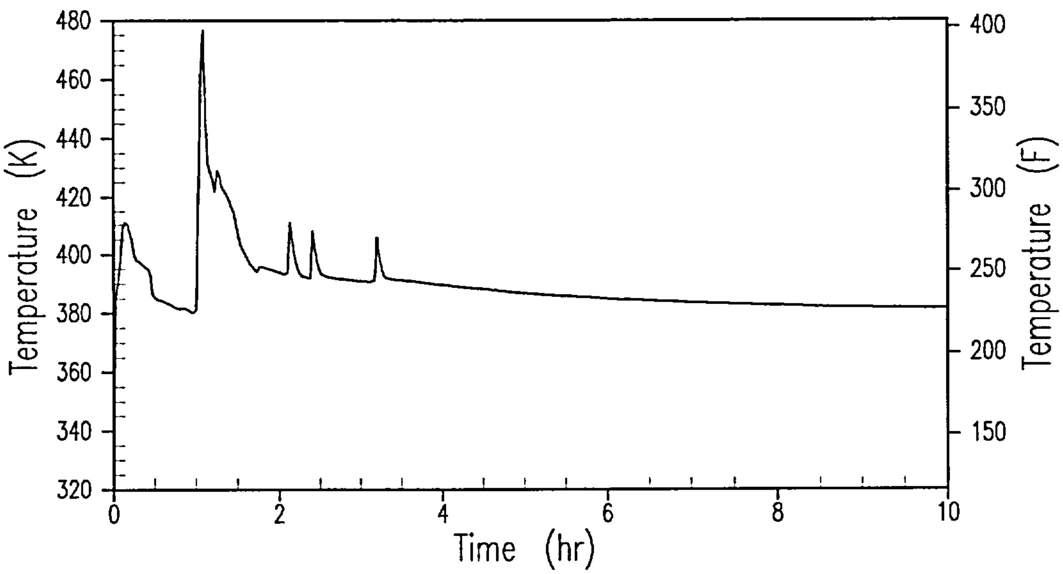


Figure D-8

Equipment Survivability Case IGN – Hydrogen Burning at Igniters
CMT Room Gas Temperature

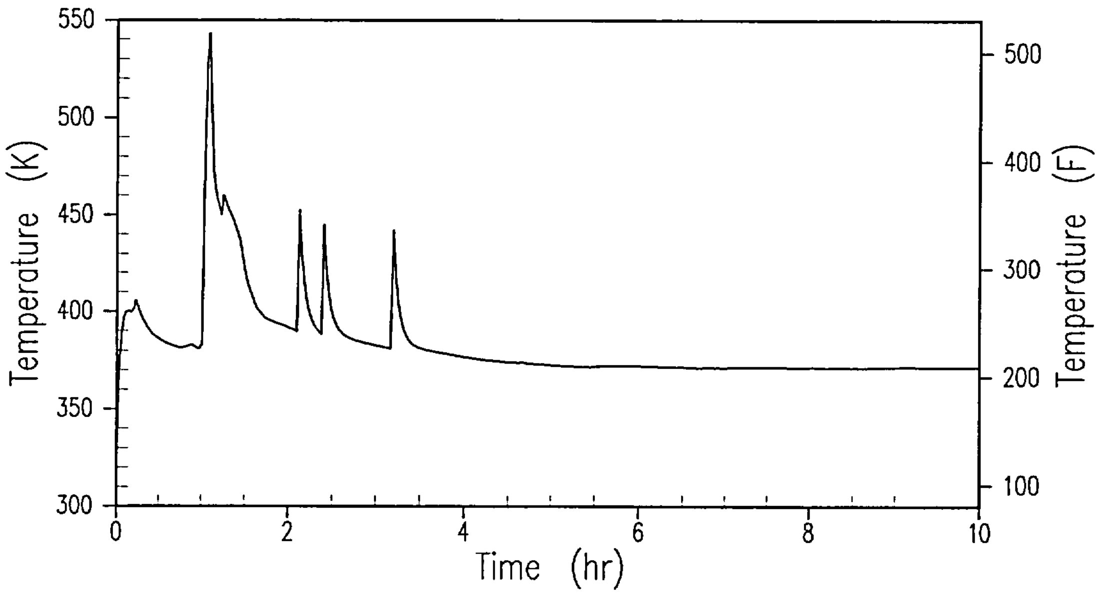


Figure D-9

Equipment Survivability Case IGN – Hydrogen Burning at Igniters Upper Compartment Gas Temperature

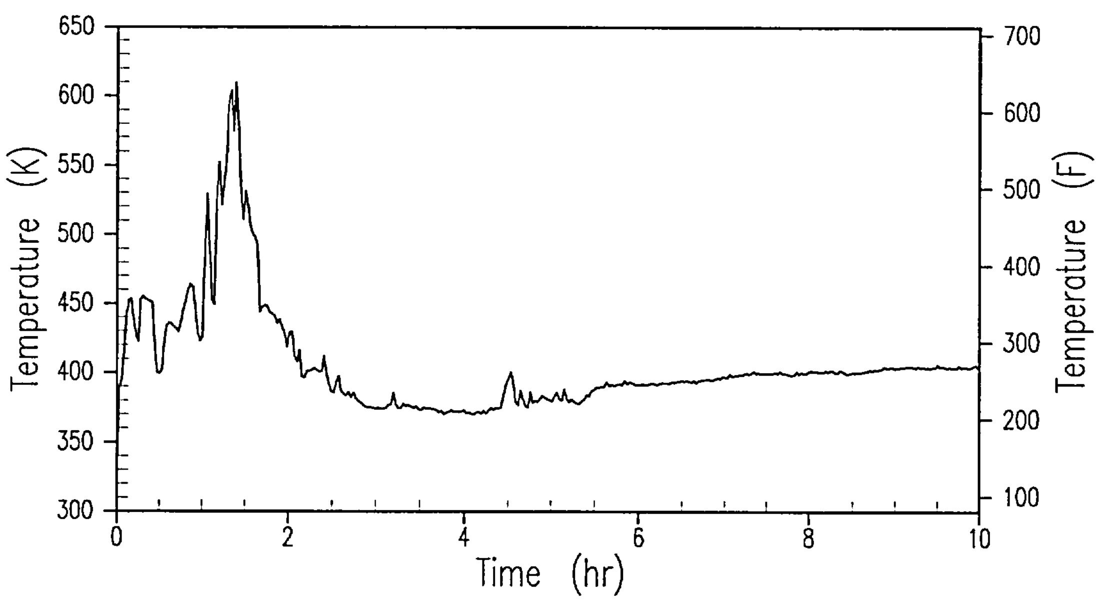


Figure D-10

Equipment Survivability Case IGN – Hydrogen Burning at Igniters Faulted PXS Compartment Gas Temperature

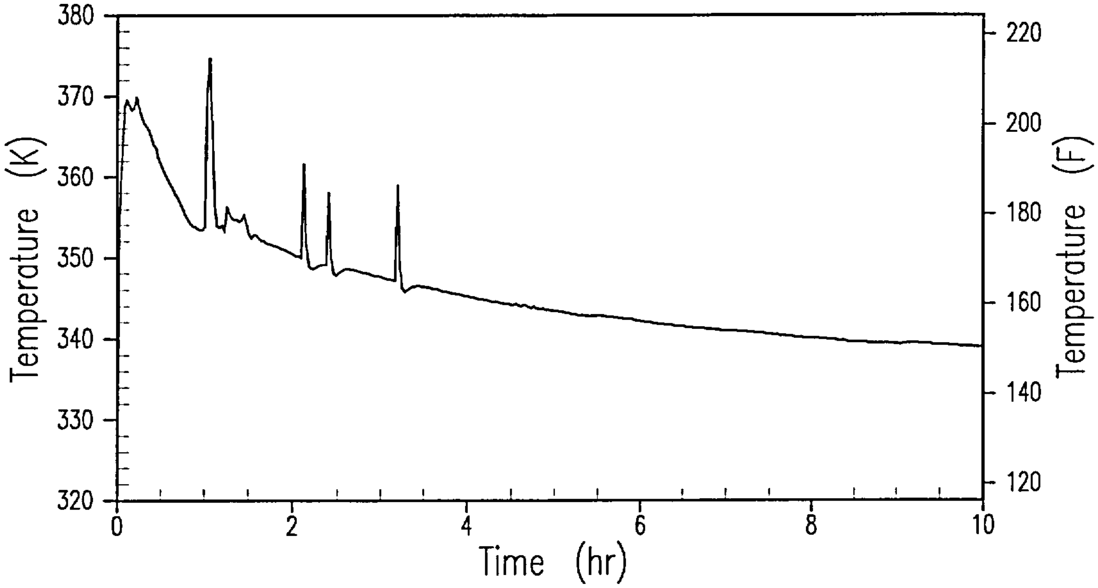


Figure D-11

Equipment Survivability Case IGN – Hydrogen Burning at Igniters
Intact PXS Compartment Gas Temperature

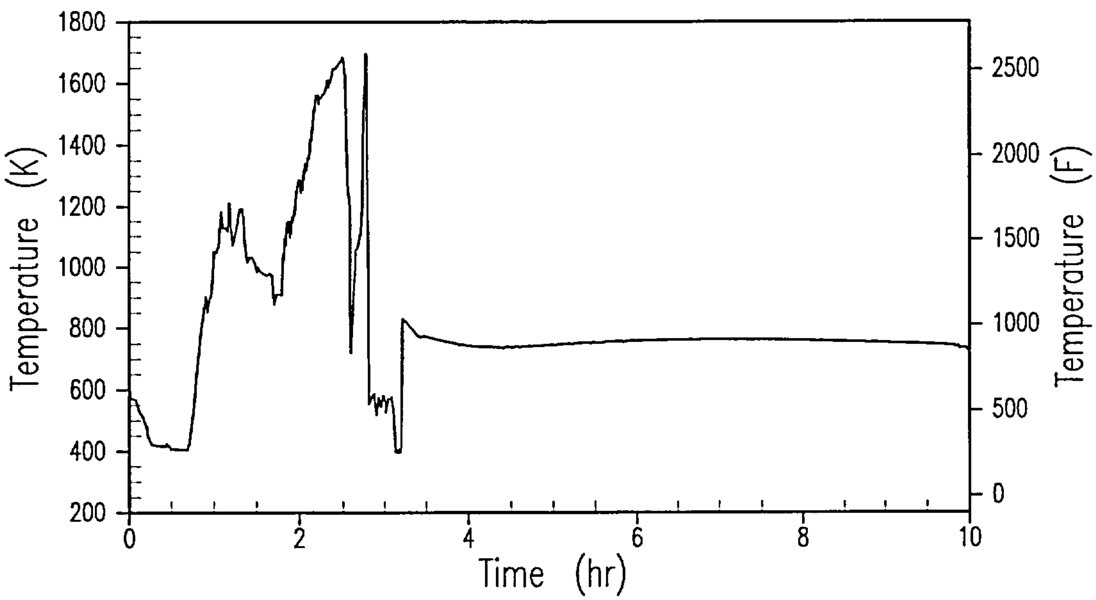


Figure D-12

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris
Upper Plenum Gas Temperature

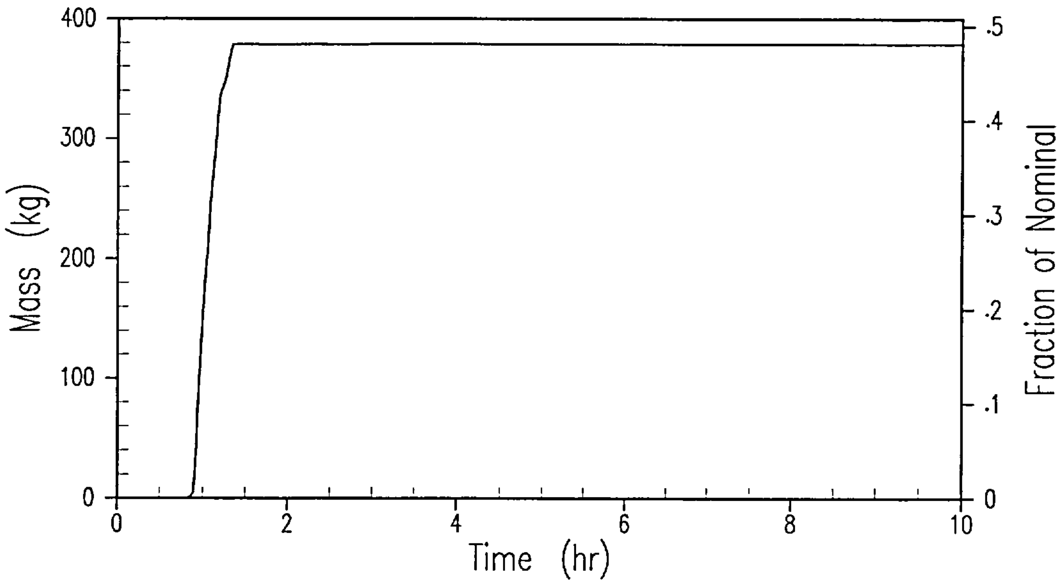


Figure D-13

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris In-Vessel Hydrogen Generation

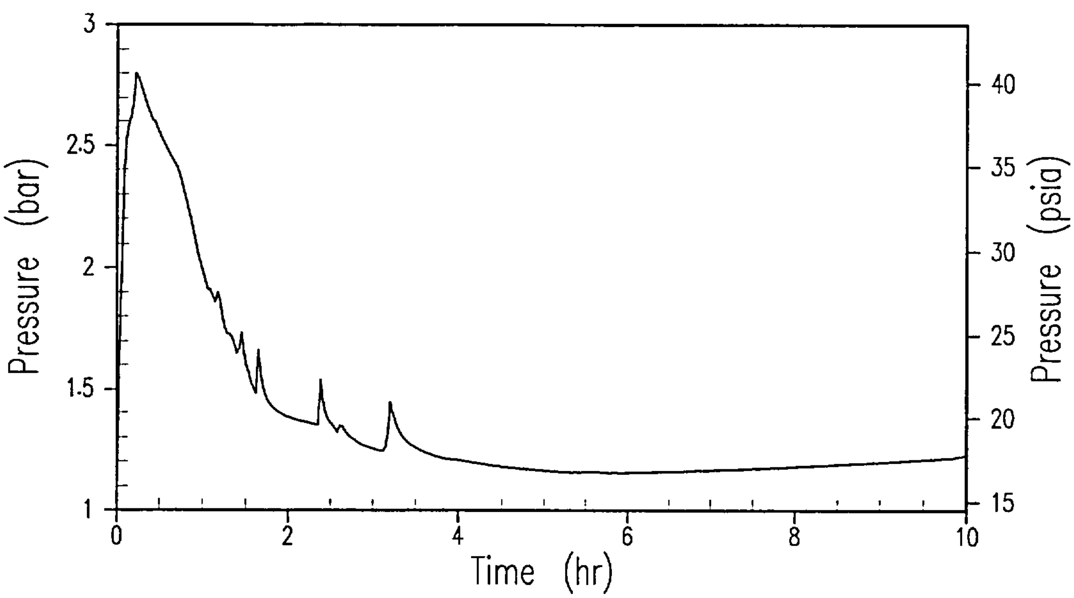


Figure D-14

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris Containment Upper Compartment Pressure

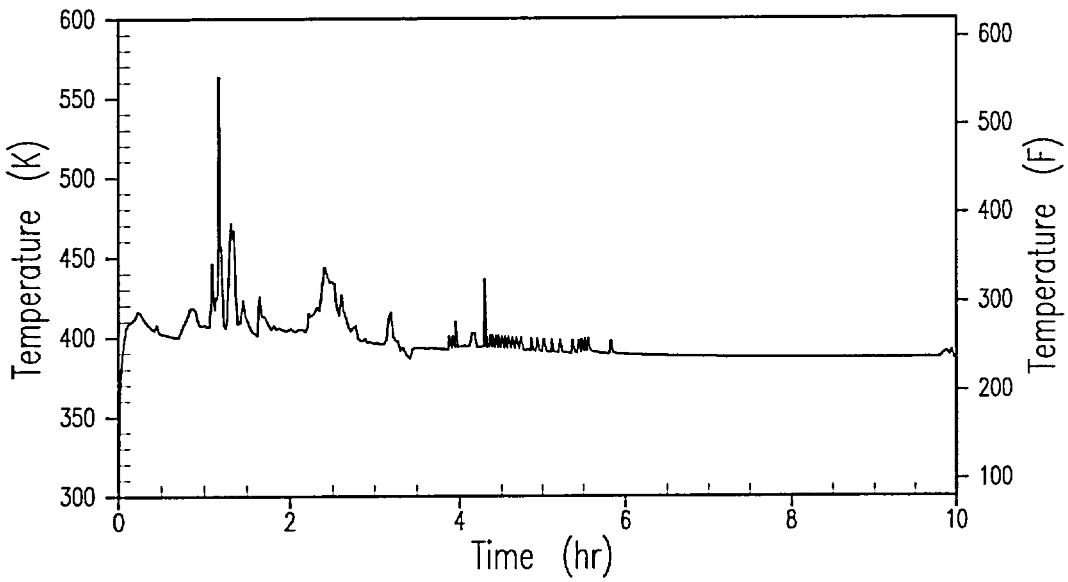


Figure D-15

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris
SG Compartment 1 Gas Temperature

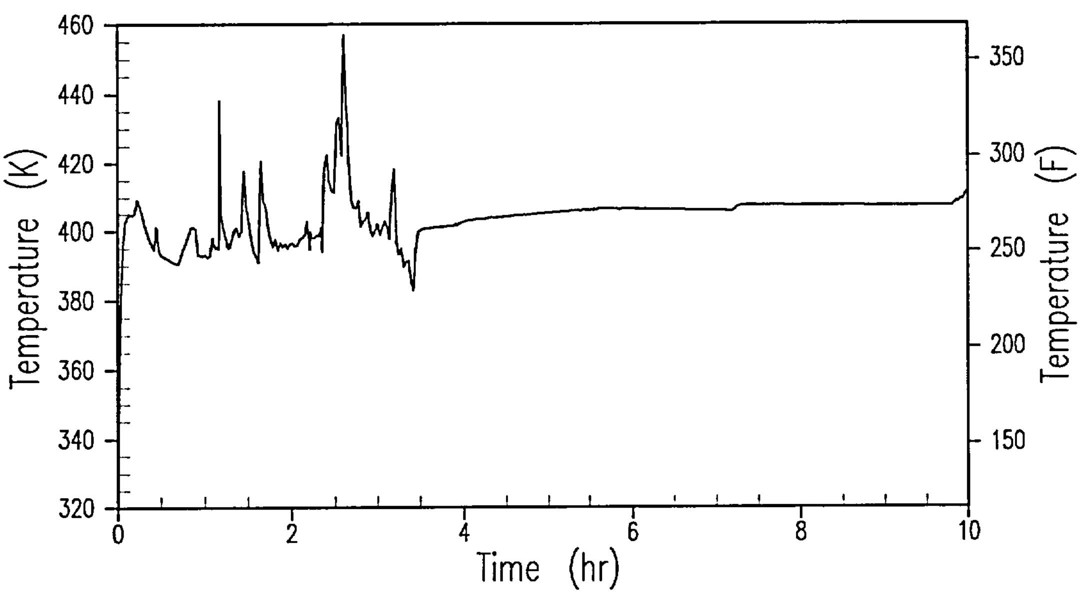


Figure D-16

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris
SG Compartment 2 Gas Temperature

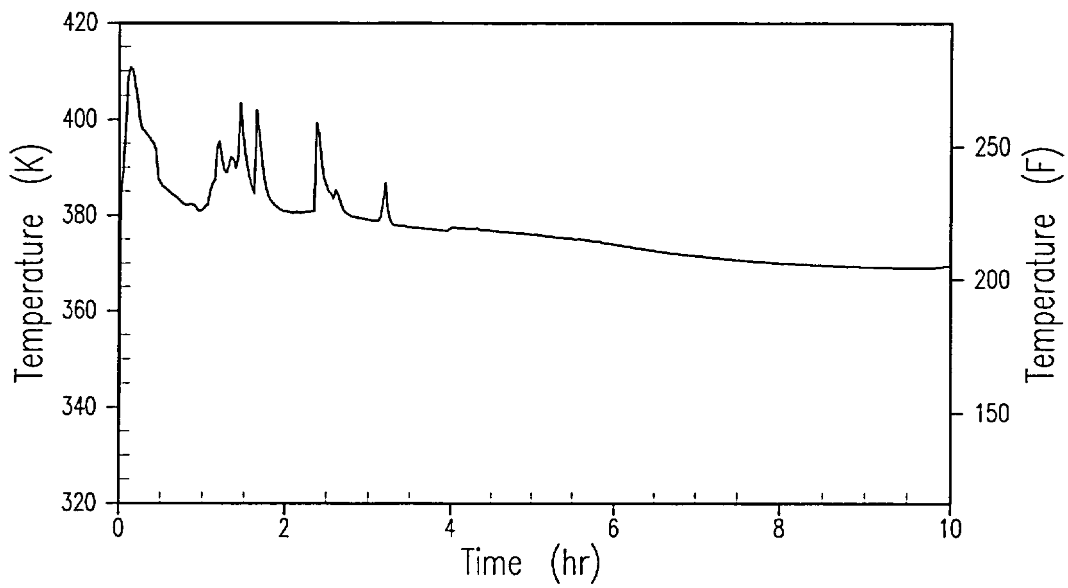


Figure D-17

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris
CMT Room Gas Temperature

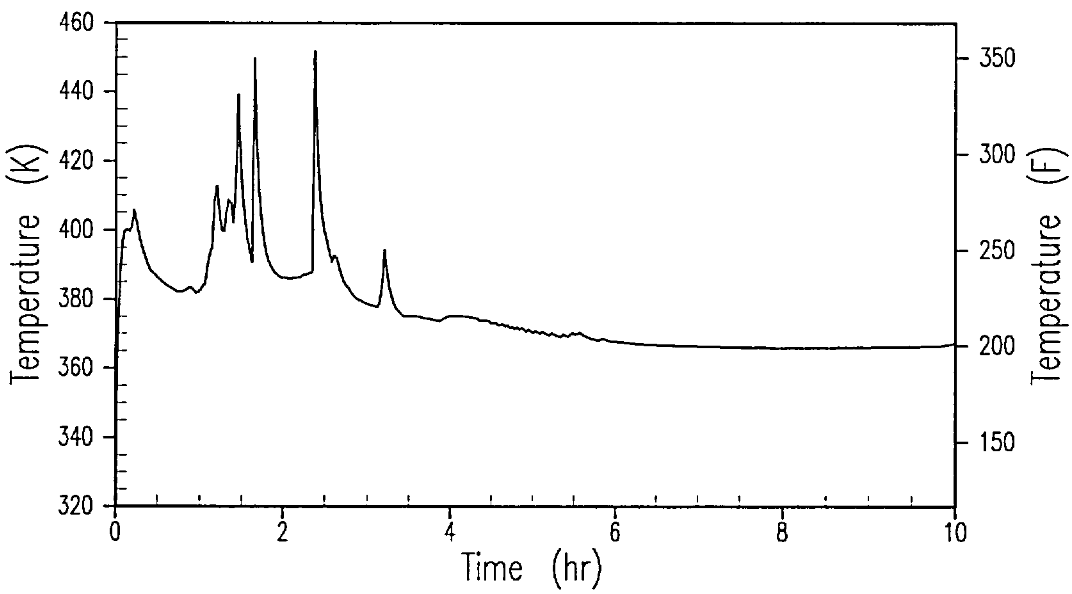


Figure D-18

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris
Upper Compartment Gas Temperature

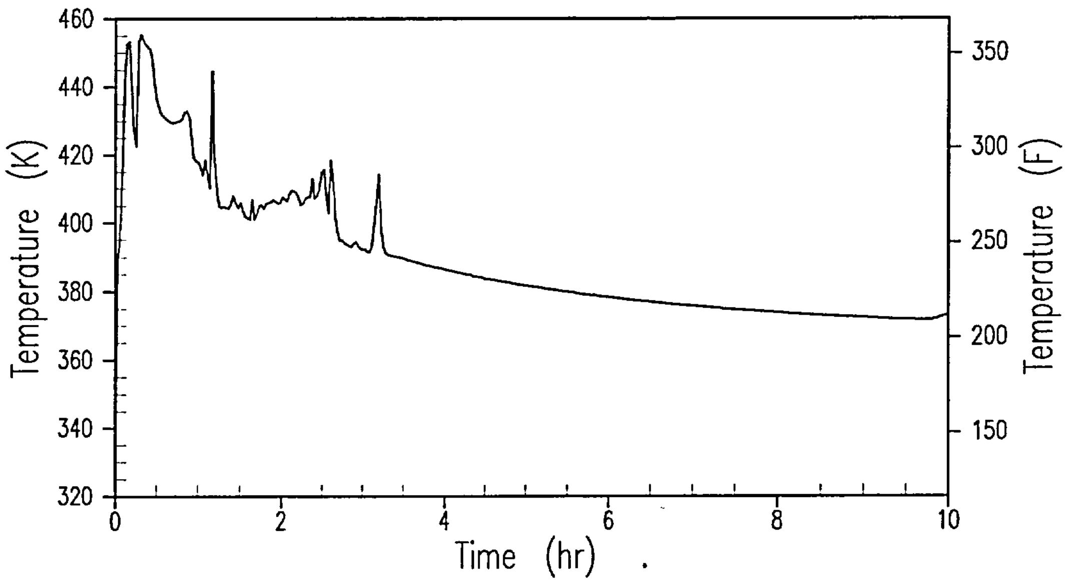


Figure D-19

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris Faulted PXS Compartment Gas Temperature

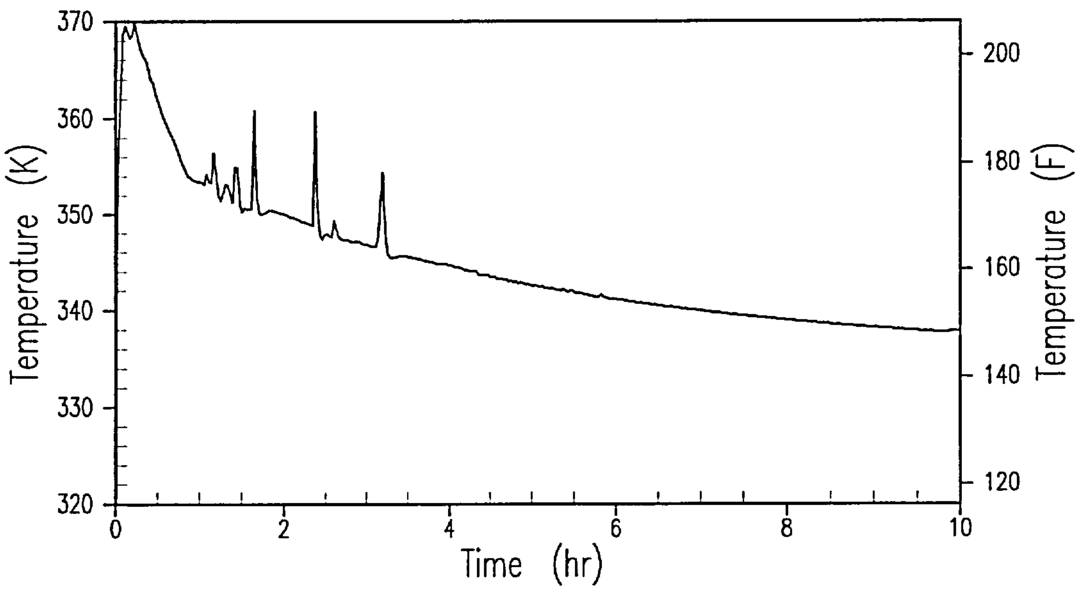


Figure D-20

Equipment Survivability Case IVR – In-Vessel Retention of Core Debris Intact PXS Compartment Gas Temperature

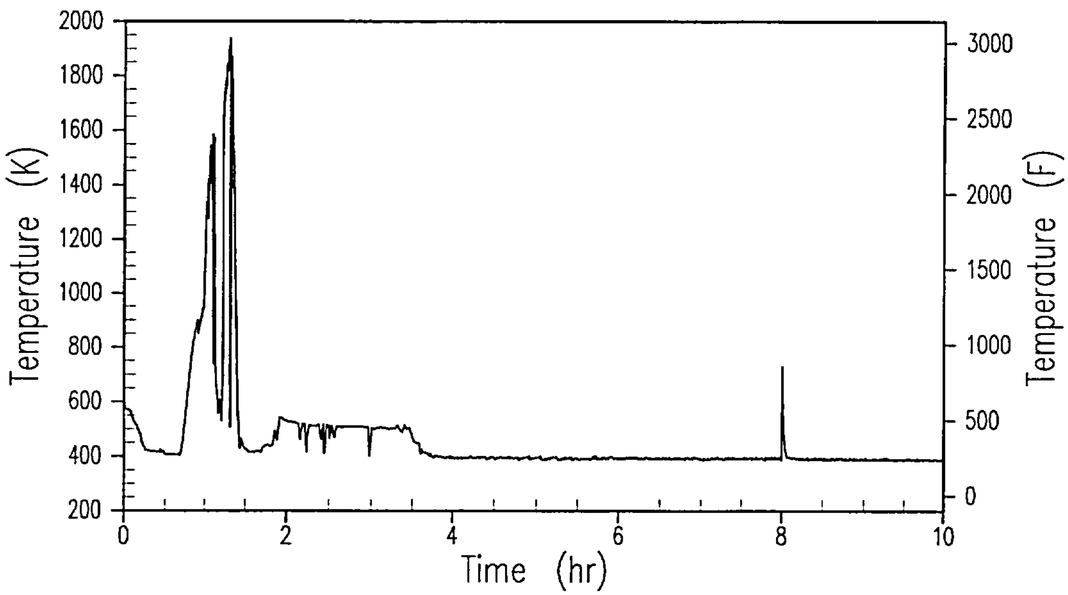


Figure D-21

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation
Upper Plenum Gas Temperature

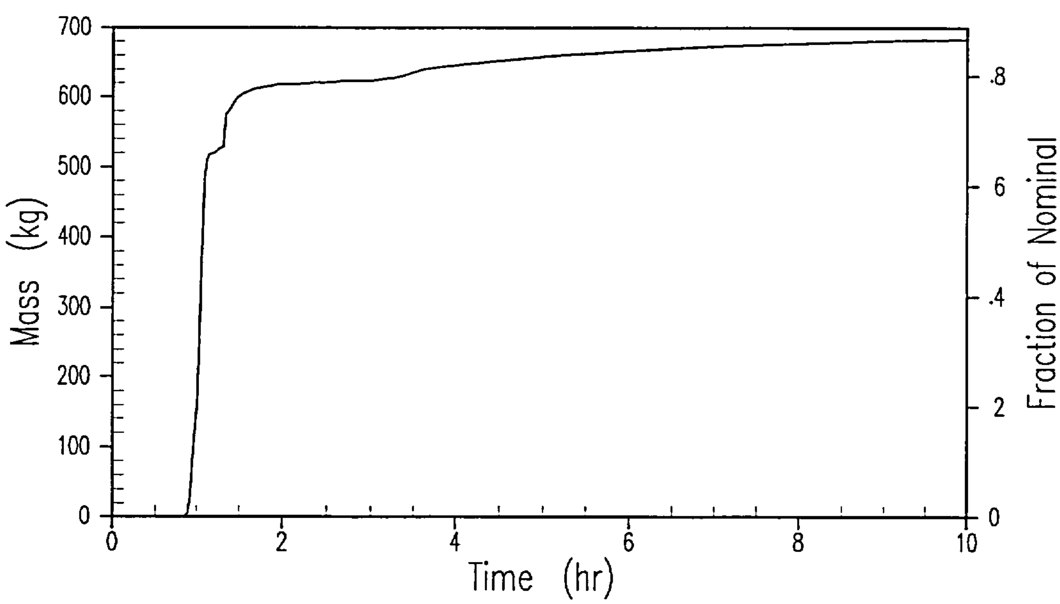


Figure D-22

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation
In-Vessel Hydrogen Generation

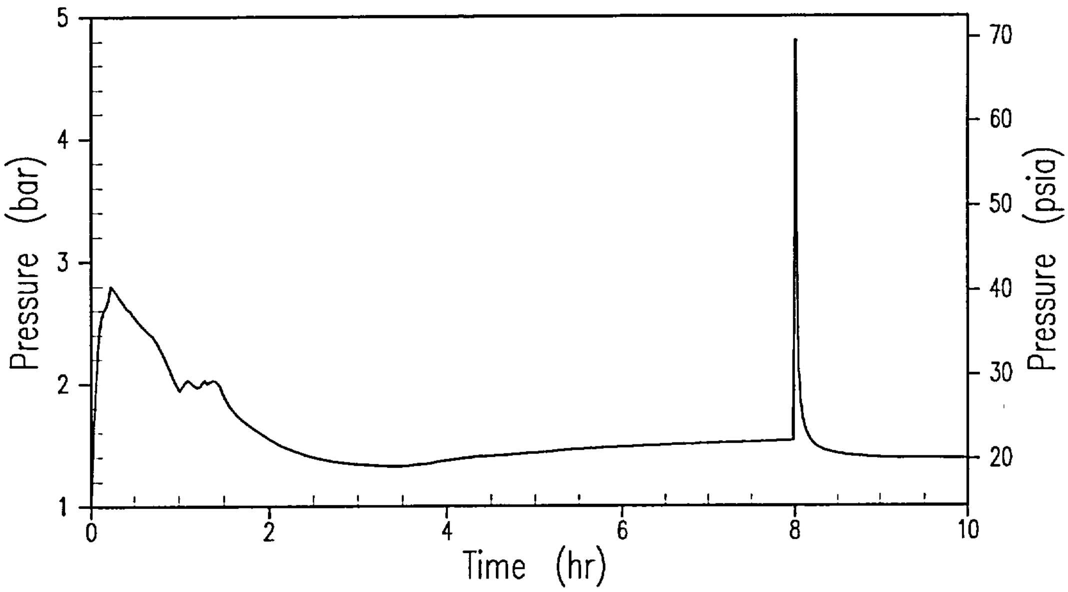


Figure D-23

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation Containment Upper Compartment Pressure

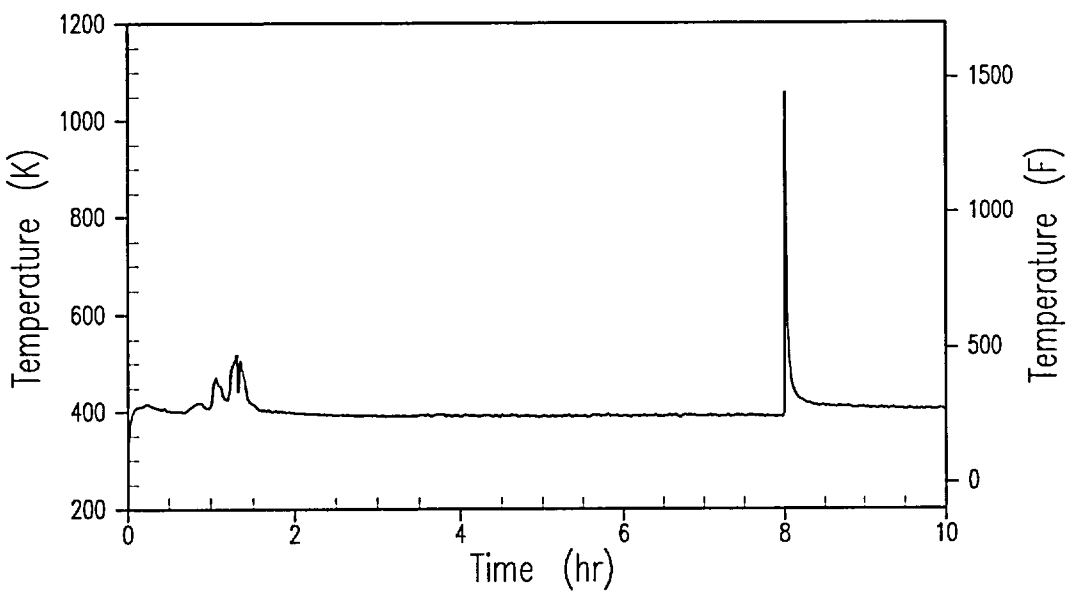


Figure D-24

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation SG Compartment 1 Gas Temperature

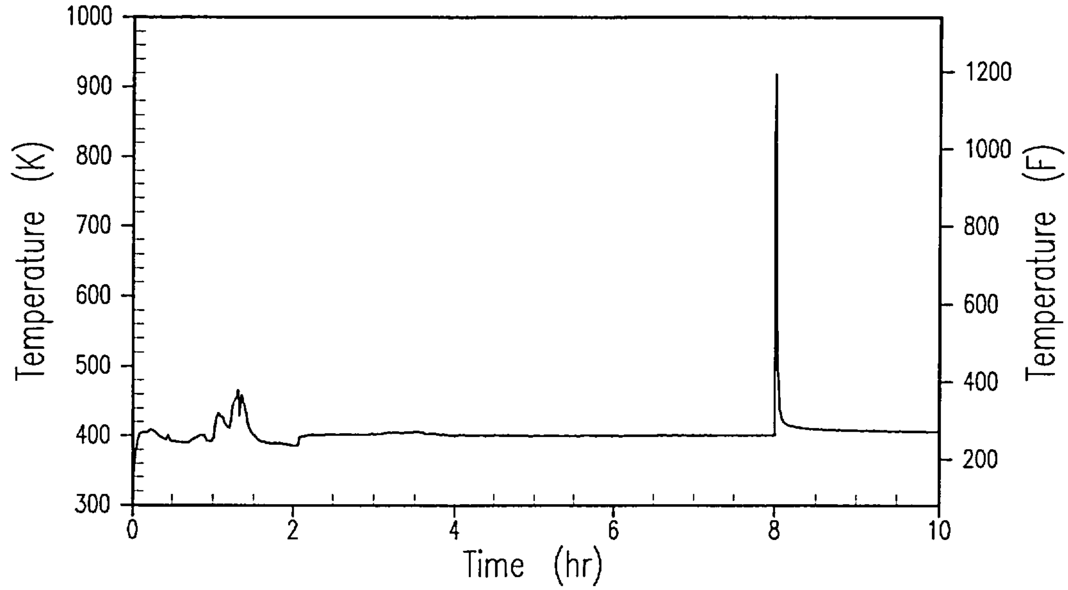


Figure D-25

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation SG Compartment 2 Gas Temperature

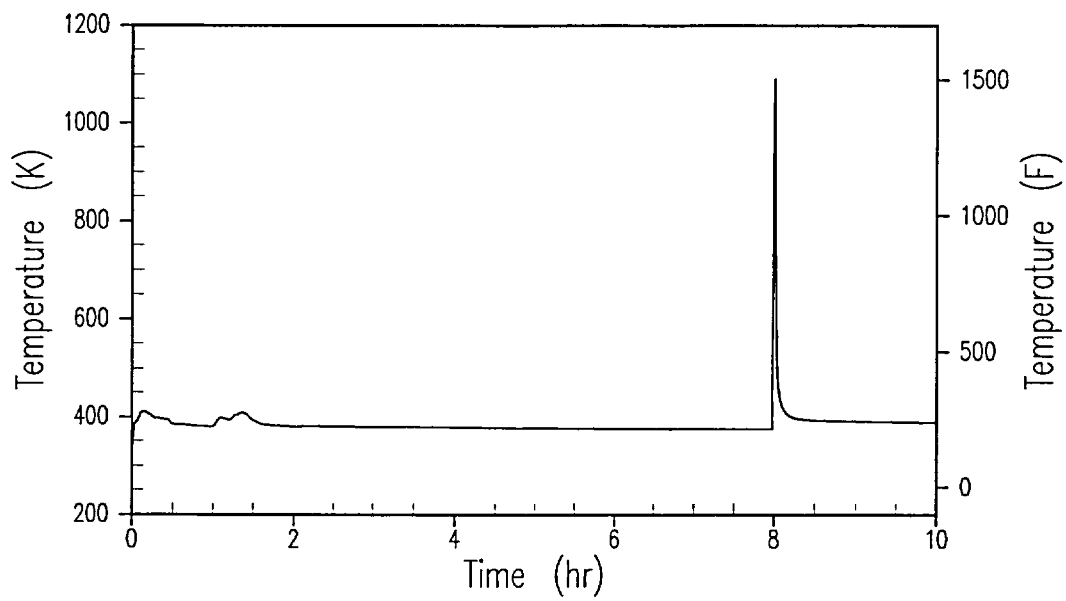


Figure D-26

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation CMT Room Gas Temperature

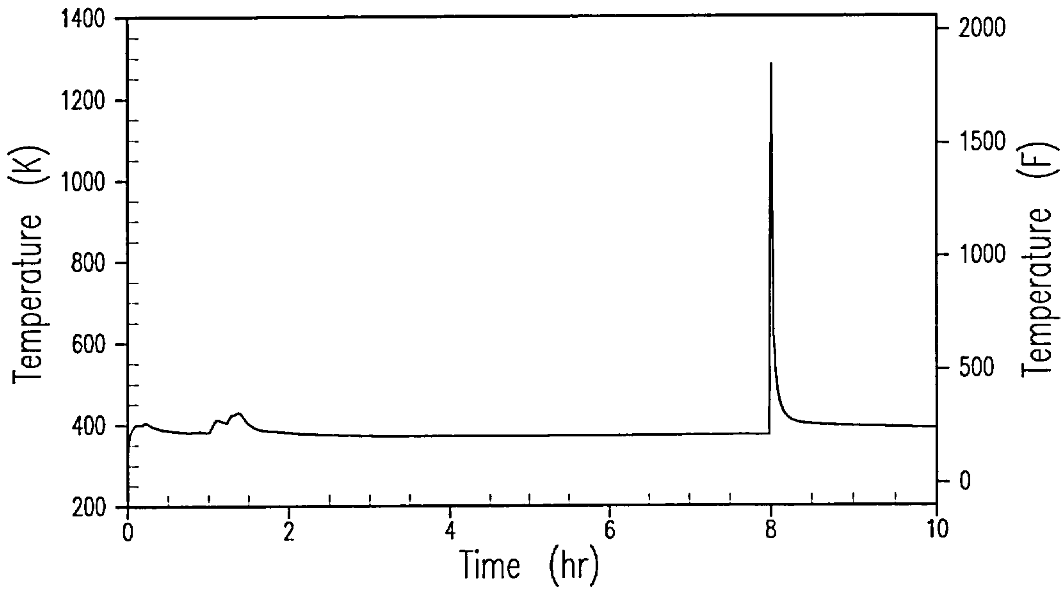


Figure D-27

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation
Upper Compartment Gas Temperature

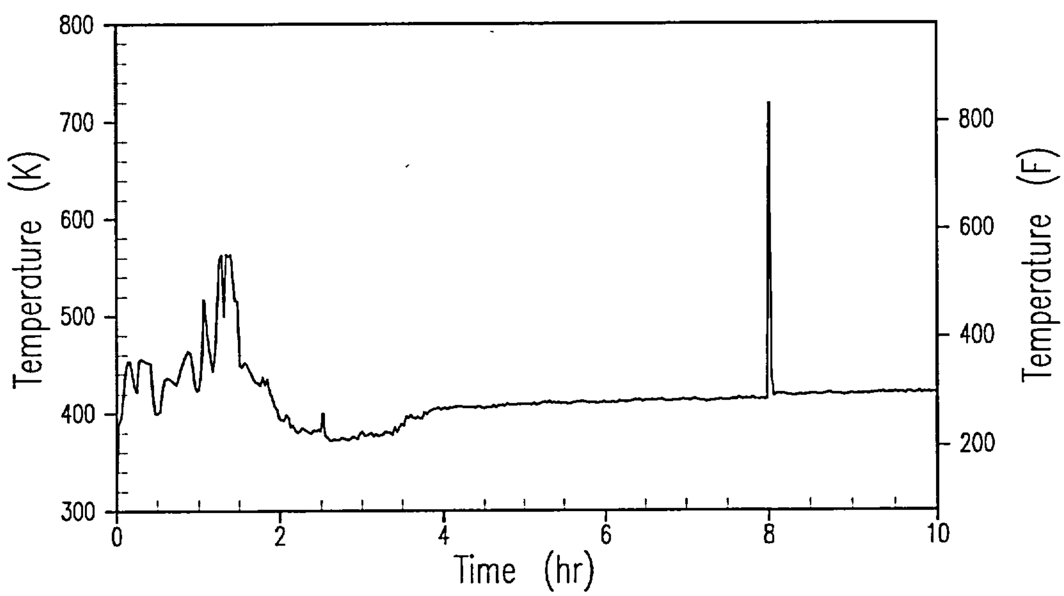


Figure D-28

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation
Faulted PXS Compartment Gas Temperature

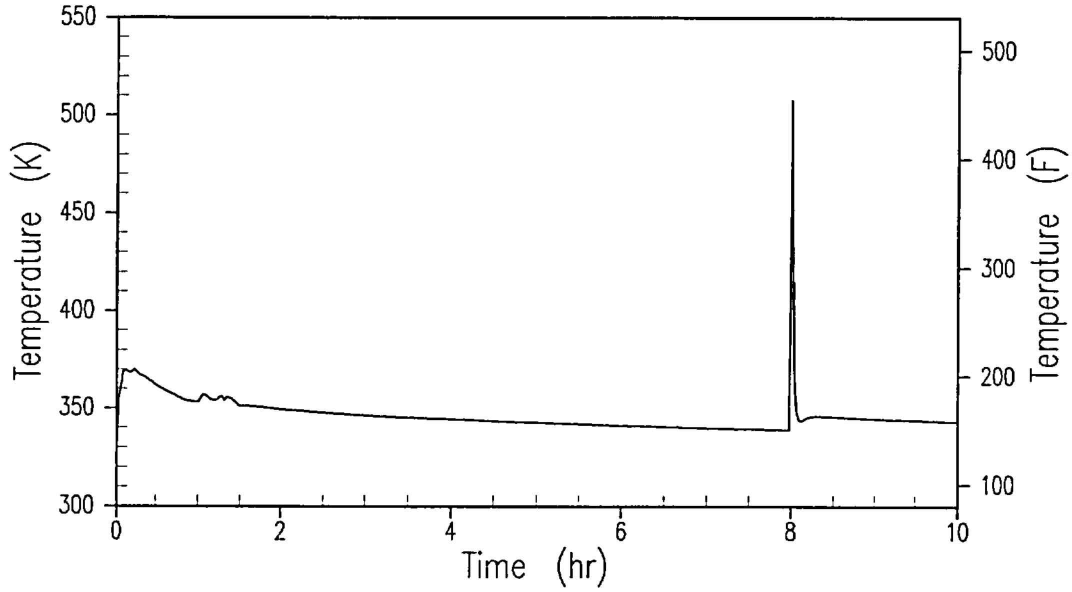


Figure D-29

Equipment Survivability Case NOIGN – DVI Line Break Without Igniter Operation Intact PXS Compartment Gas Temperature

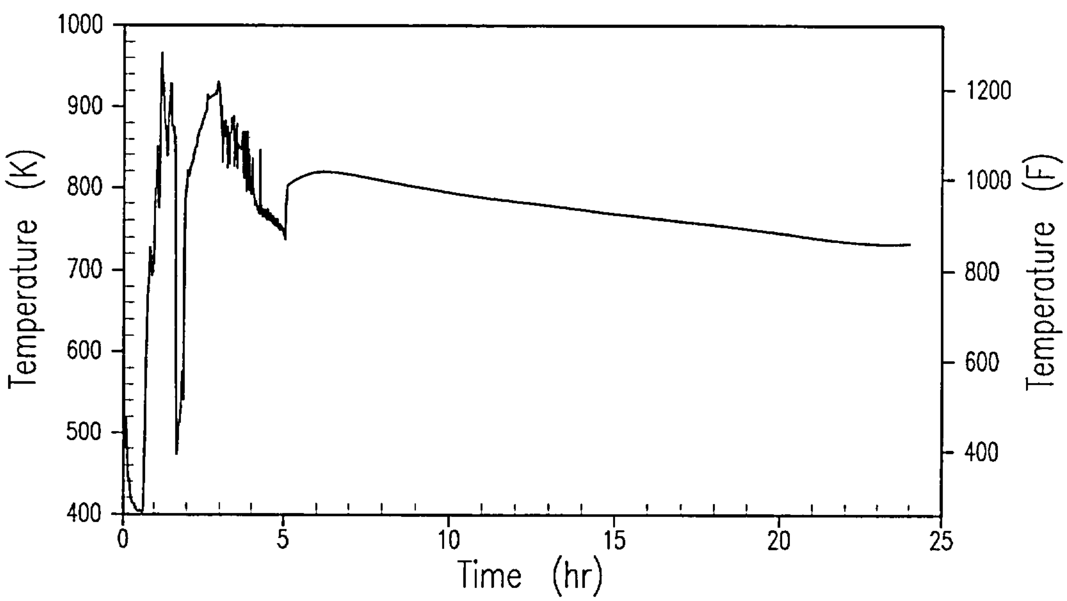


Figure D-30

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity Upper Plenum Gas Temperature

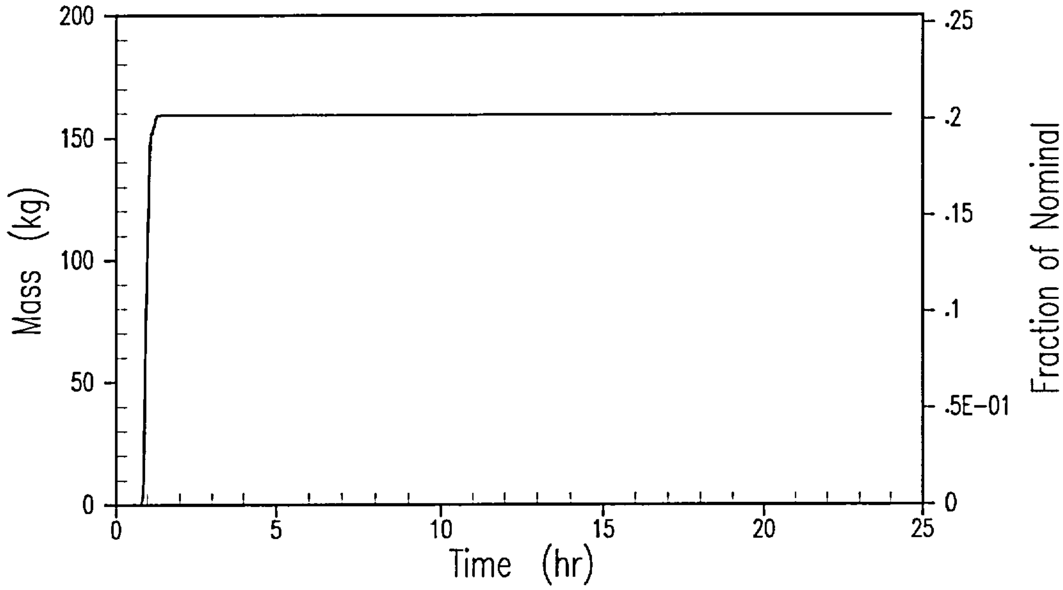


Figure D-31

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity In-Vessel Hydrogen Generation

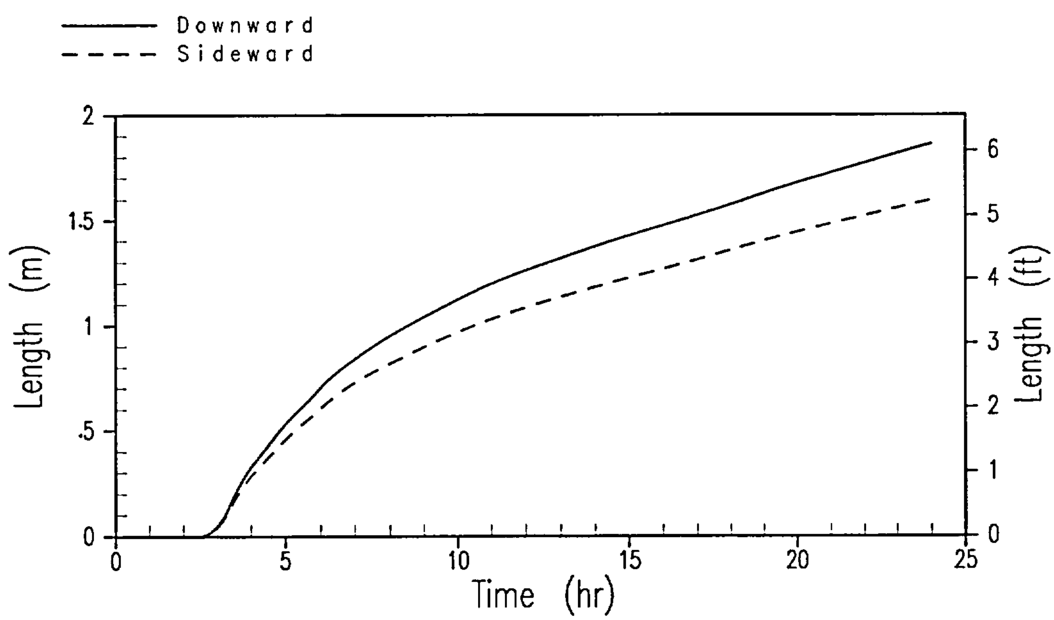


Figure D-32

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity Concrete Penetration Depth

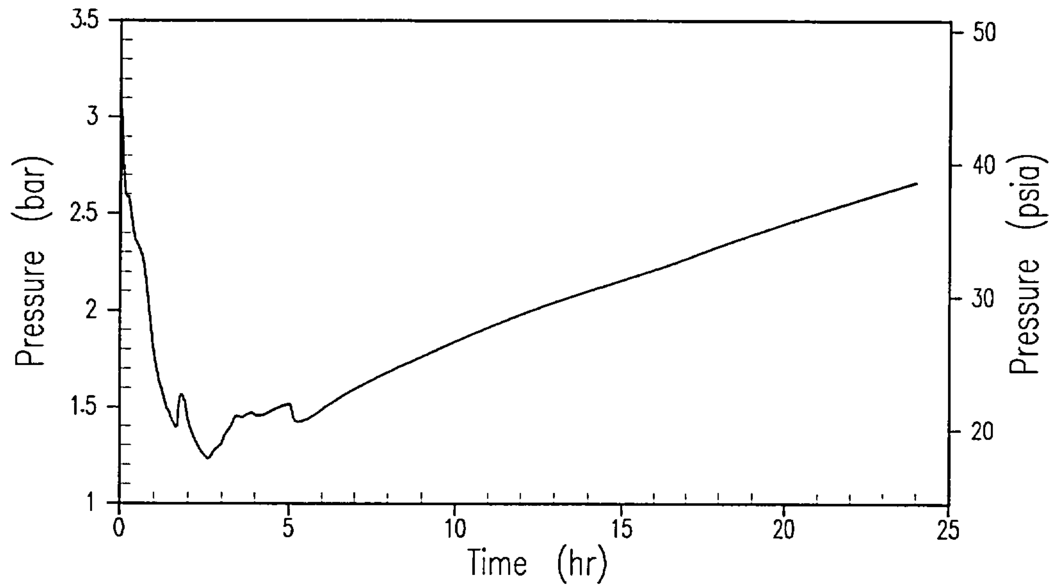


Figure D-33

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity Containment Upper Compartment Pressure

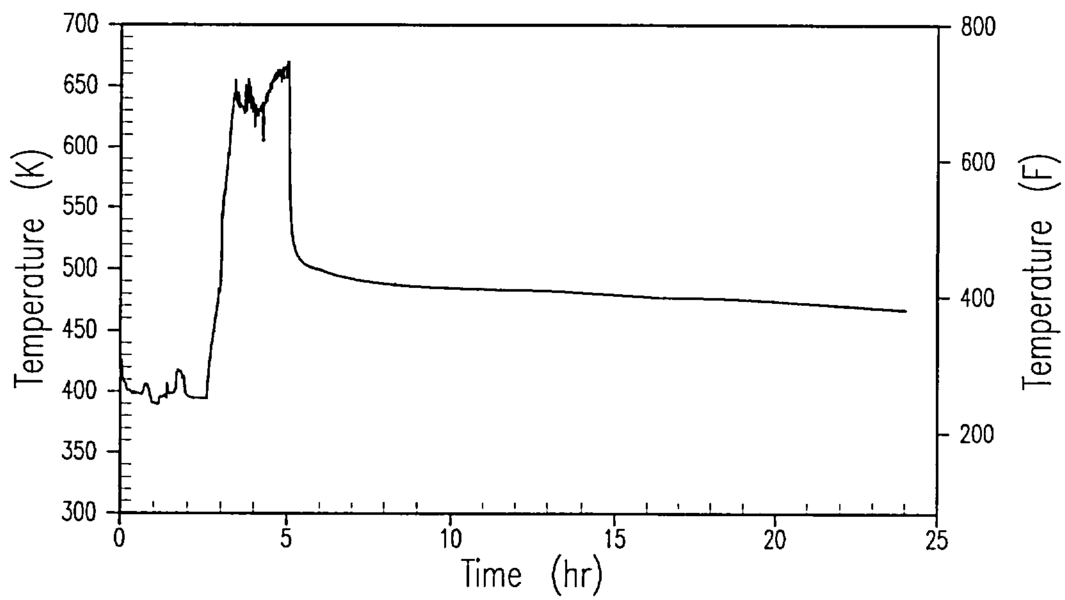


Figure D-34

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity SG Compartment 1 Gas Temperature

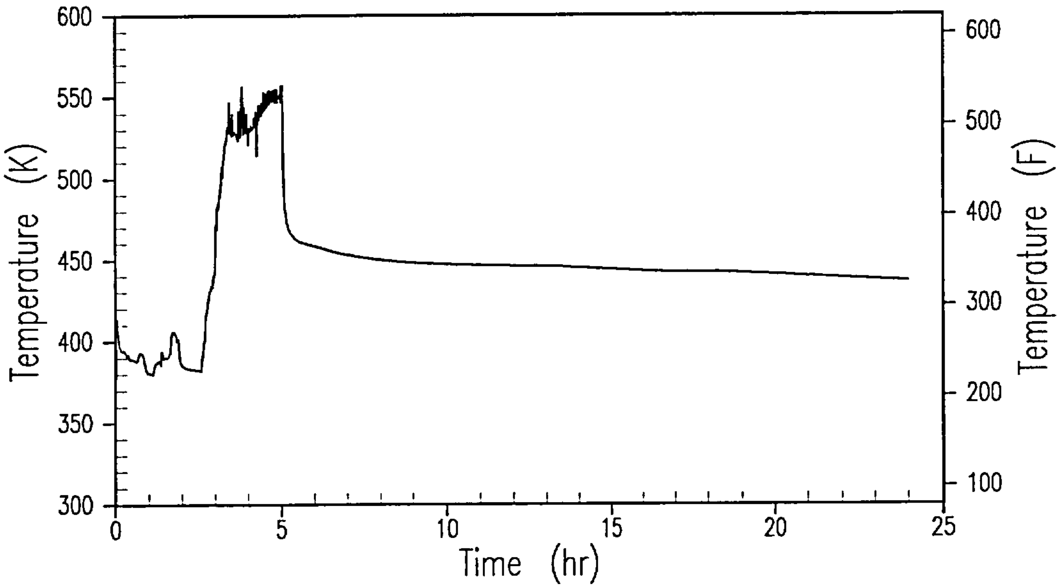


Figure D-35

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity SG Compartment 2 Gas Temperature

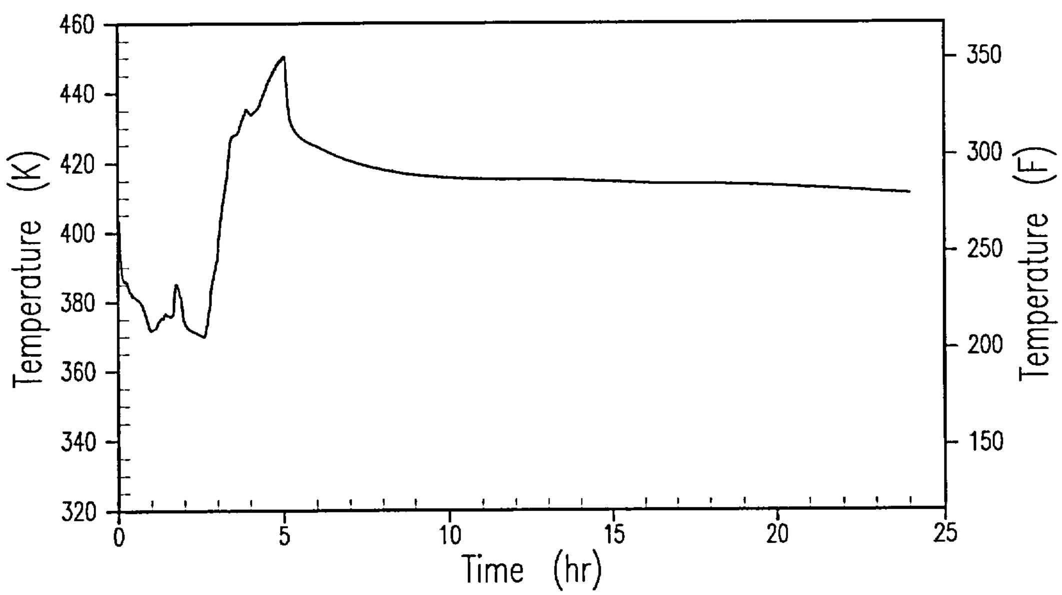


Figure D-36

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity CMT Room Gas Temperature

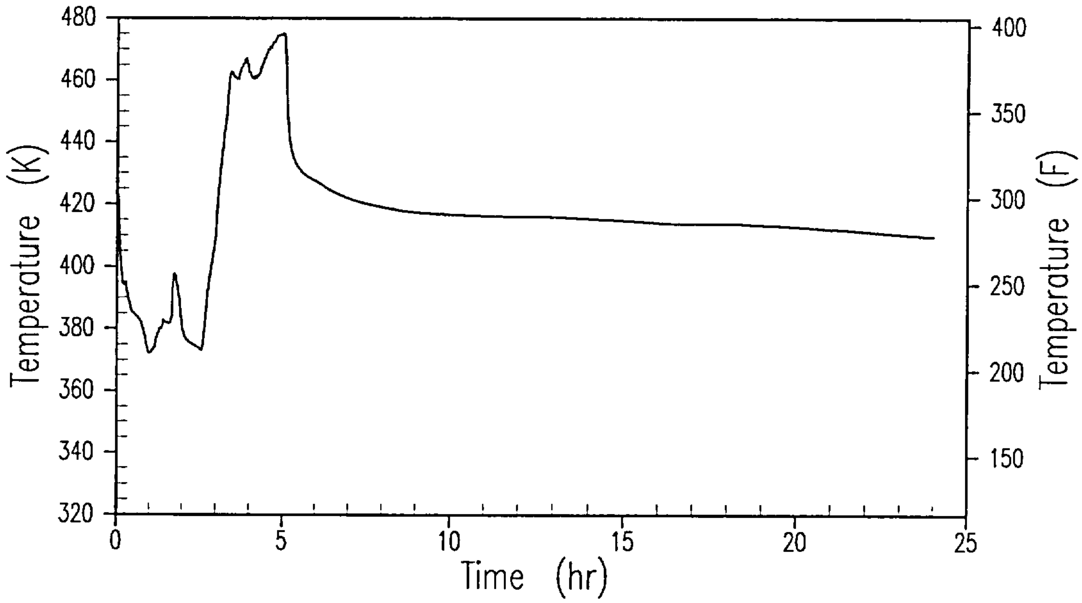


Figure D-37

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity Upper Compartment Gas Temperature

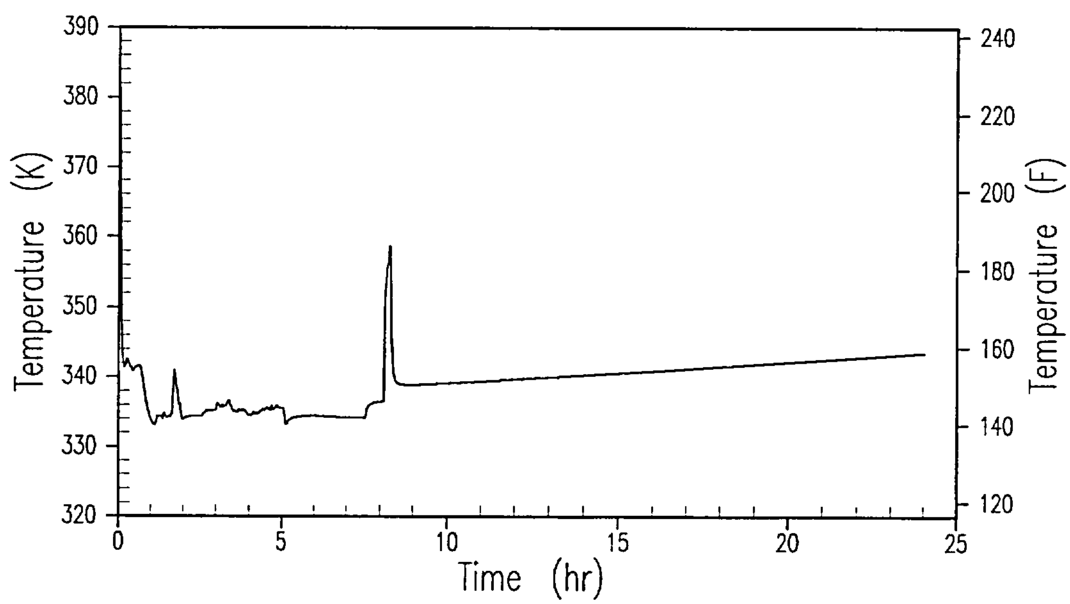


Figure D-38

Equipment Survivability Case CCI – Uncoolable Debris Bed in Reactor Cavity PXS Compartment Gas Temperature

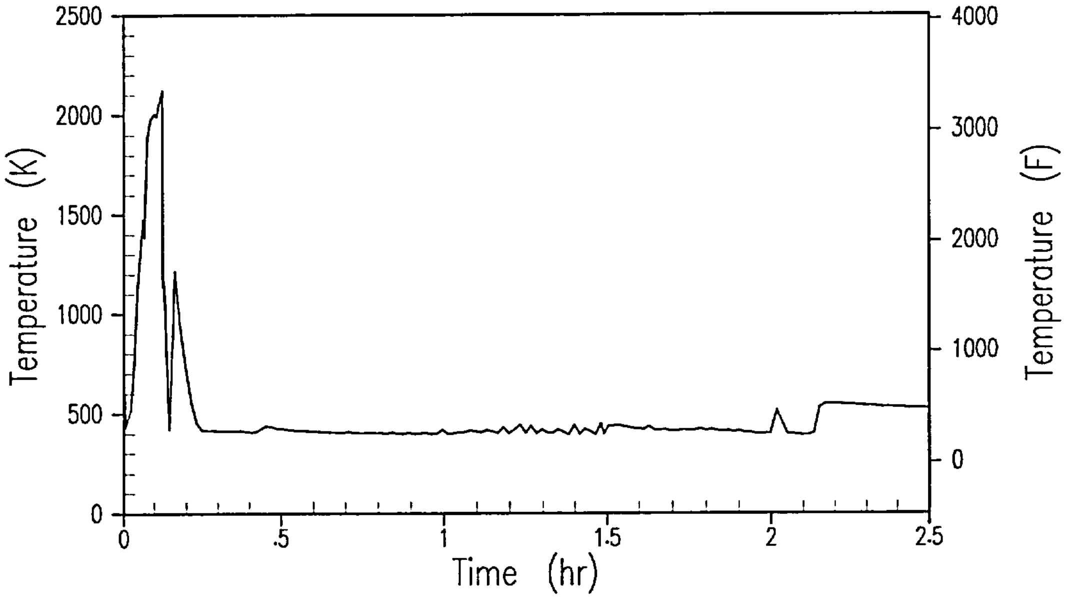


Figure D-39

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – Upper Plenum Gas Temperature

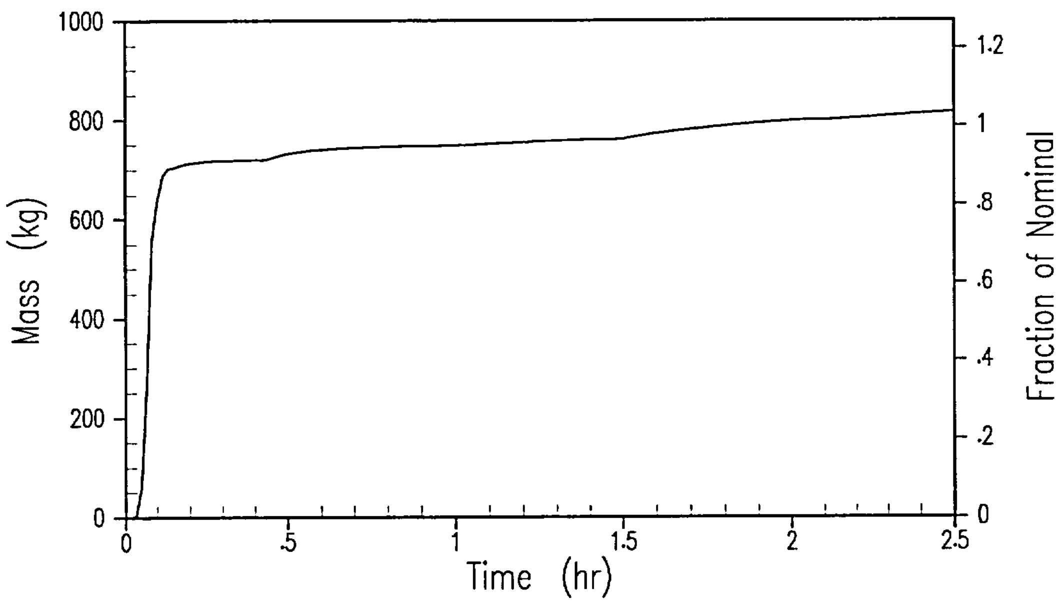


Figure D-40

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – In-Vessel Hydrogen Generation

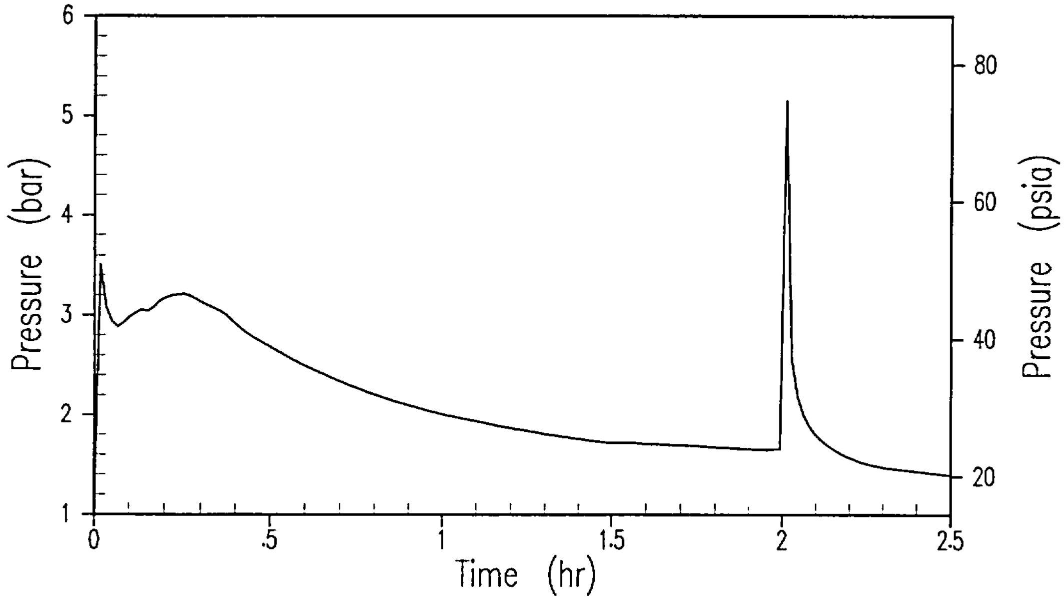


Figure D-41

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – Containment Upper Compartment Pressure

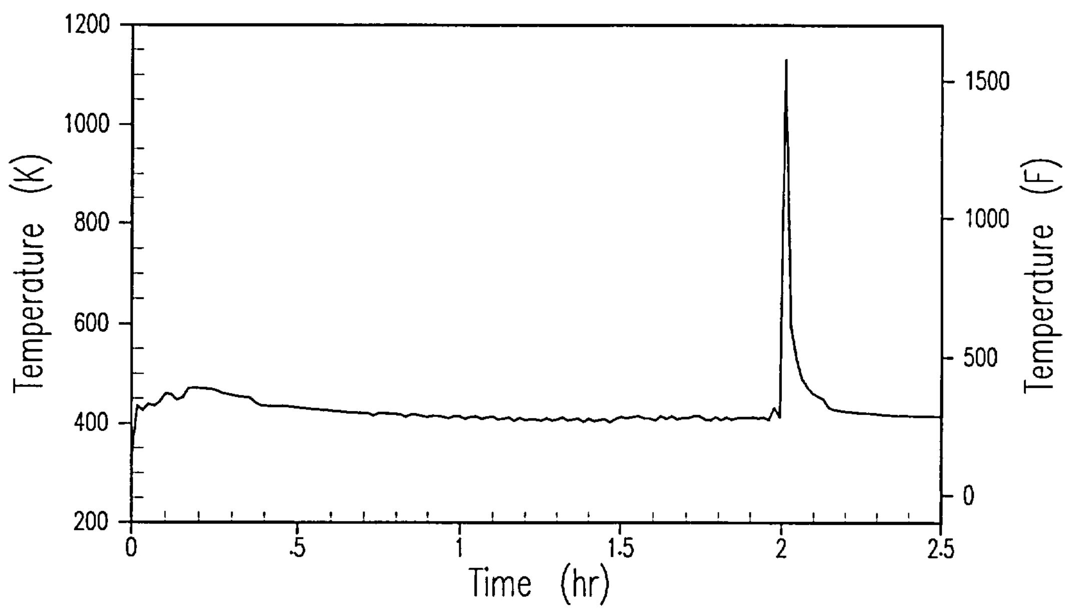


Figure D-42

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation SG – Compartment 1 Gas Temperature

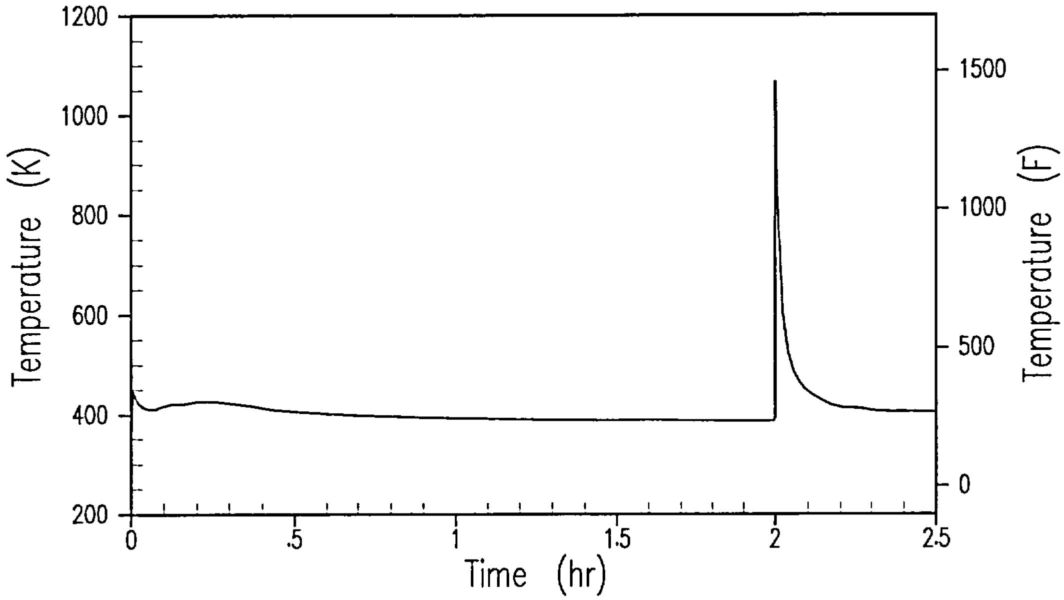


Figure D-43

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – SG Compartment 2 Gas Temperature

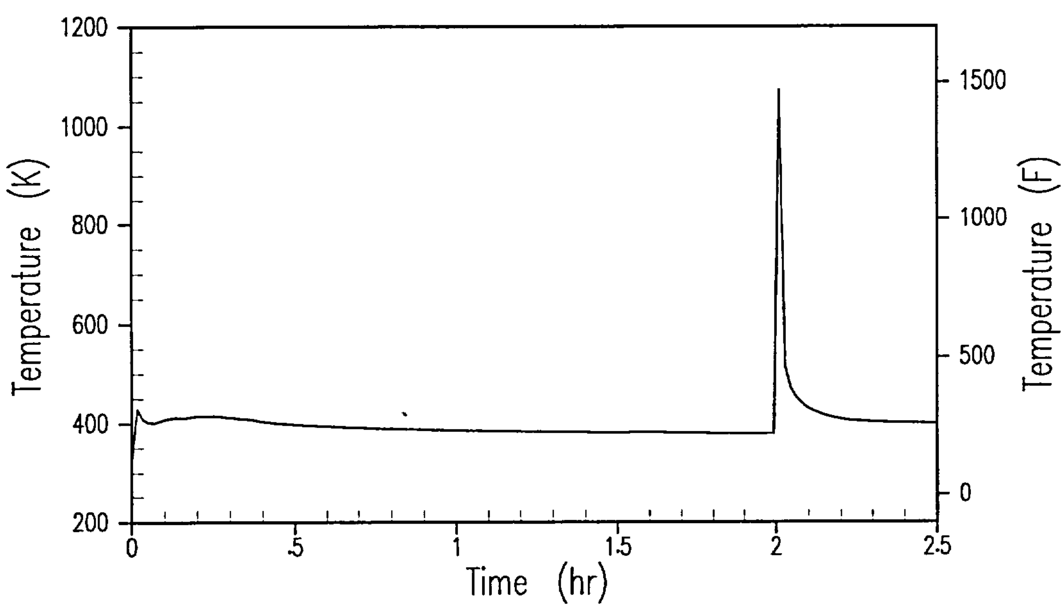


Figure D-44

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – CMT Room Gas Temperature

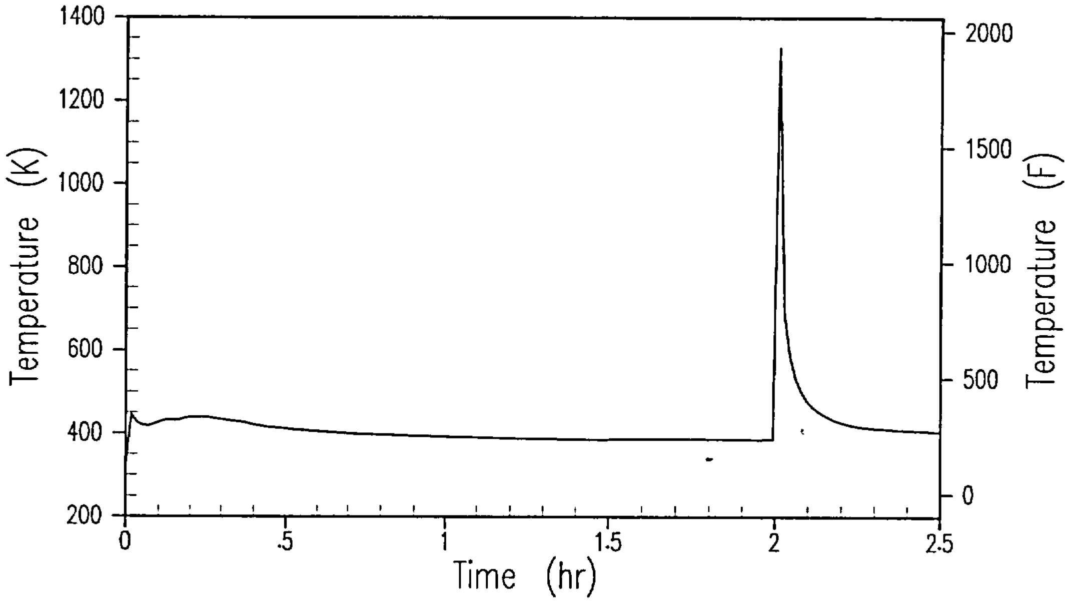


Figure D-45

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – Upper Compartment Gas Temperature

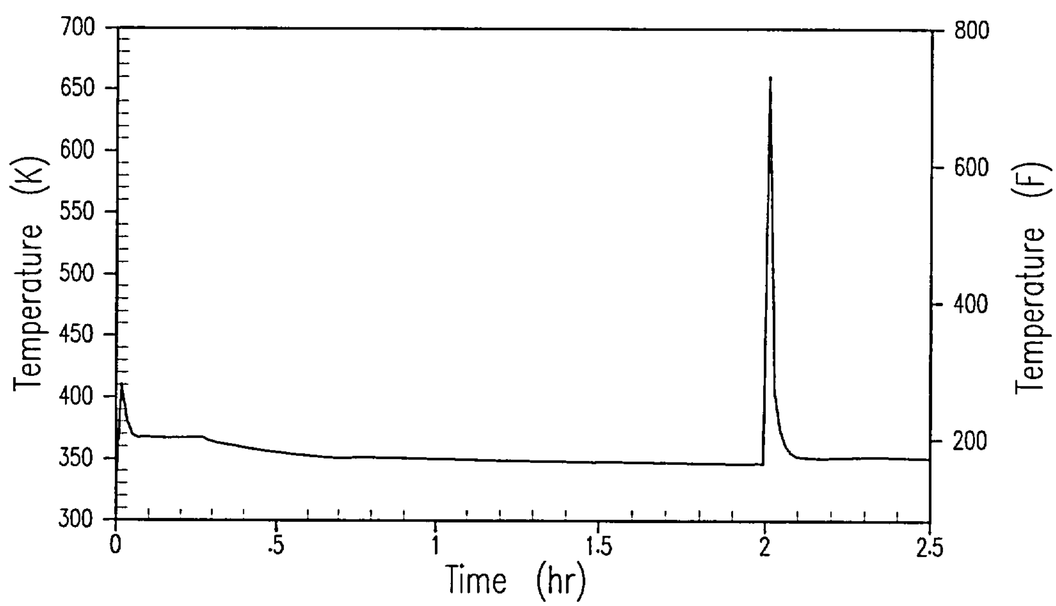


Figure D-46

Equipment Survivability Case GLOB – Global Burning of Hydrogen From 100-Percent Cladding Oxidation – PXS Compartment Gas Temperature