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SUBJECT: Forwards revised FSAR pages describing new main steam line break analysis & resultant containment temp. "Methodology for Predicting Containment Temp Following Main Steam Line Break" also encl.

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TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
400 Chestnut Street Tower II

March 25, 1985

Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ms. Adensam:

In the Matter of the Application of) Docket Nos. 50-438
Tennessee Valley Authority 50-439

The present FSAR analyses on the effects of postulated main steam line breaks (MSLBs) inside containment use assumptions that were developed to ensure that the MSLBs (regardless of break size) would be treated in a conservative manner. While these assumptions are acceptable for small MSLB, the same assumptions, when applied to large breaks, result in peak containment temperatures that are excessively conservative. In the case of Bellefonte, the calculated temperatures are sufficiently high so as to preclude (with few exceptions) obtaining qualified electrical equipment (per 10 CFR 50.49) using presently available designs.

TVA believes that the use of different assumptions (which also provide adequate conservatism) for analyzing postulated large MSLBs inside containment is acceptable. Accordingly, we are providing the following information for your review:

Enclosure 1: Proposed revised FSAR pages in Sections 3.11 and 6.2, which describe the new MSLB analysis and resultant containment temperature.

Enclosure 2: Report "Methodology for Predicting Containment Temperature Following a Main Steam Line Break," Tennessee Valley Authority, describing the technical basis for the new analyses.

TVA believes there is a sound technical basis for the proposed FSAR change related to large MSLBs inside the primary containment. The results presented are conservative and provide adequate design margins. It must be emphasized that there is a real need to reduce the peak temperatures in order to obtain qualified equipment. We would like to meet with you and the cognizant NRC Staff reviewers in Bethesda, Maryland, approximately two to four weeks after receipt of this submittal to discuss any comments or questions you may have.

The revised FSAR pages will be included in the next amendment to the FSAR.

If you have any questions concerning this matter, please get in touch with Amin Kamal at FTS 858-2680.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. H. Shell

R. H. Shell
Nuclear Engineer

Sworn to and subscribed before me
this 25th day of Mar. 1985.
Paulette H. White
Notary Public

My Commission Expires 8-24-88

Enclosures (2)
cc: U.S. Nuclear Regulatory Commission (Enclosures)
Region II
Attn: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

6.2.1.4.1 Steam Line Break

In the event of a steam line break inside containment, the release of the high energy steam will result in high pressures and temperatures. The resulting temperatures may be included in the design basis for safety-related equipment and other structures inside the containment.

The analysis of the pressure and temperature response in the containment for a series of steam line breaks was conducted with the use of TVA's version of CONTEMPT4, entitled MONSTER. Table 6.2.1-26 lists the various steam line break cases which were analyzed, and the resulting pressure and temperature peaks. Analyses were performed for the double-ended break of a 32-inch line with the reactor at three different initial power levels, and for a spectrum of break sizes with the reactor initially at 102-percent EOL. The methods and assumptions used in generating the mass-energy releases for these cases are found in section 15.1.5.

The initial conditions of the containment for a steam line break were modeled to be the same as for a LOCA. Table 6.2.1-3 gives the containment initial conditions for a LOCA.

Of the active containment safeguards, only one set of sprays were modeled. The fans were assumed not to operate during the steam line break since they are not qualified to do so. Thus, minimum safeguards assumptions would reduce to having one set of sprays available for the steam line breaks. The start times for the reactor building sprays were chosen so that the RBS ESFAS set point pressure (Section 7.3.1.1.1) of 10 lb/in^2g and a delay time of 105 seconds would be modeled. Figure 6.2.1-162 shows a breakdown of the RBS delay time.

The walls were modeled as shown in Tables 6.2.1-39 through 41. For all surfaces in the containment except the floor, the CVTR correlation was used to calculate the condensing heat transfer coefficient as recommended in reference 14. To calculate the heat transferred to the walls during a large (area greater than or equal to 0.6 ft^2) break when the wall surface temperatures were below the saturation temperature, the following equation was used:

$$q = H_t A(T_v - T_w)$$

Here q is the heat transfer rate, H_t is the CVTR heat transfer coefficient, A is the heat transfer area, T_w is the wall surface temperature, and T_v is the vapor temperature. The equation is in British units. To calculate the heat transferred to the walls during a small (area less than 0.6 ft2) break, when the wall surface temperatures were below the saturation temperature, the following equation was used:

$$q = H_u A(T_s - T_w)$$

where H_u is the Uchida heat transfer coefficient and T_s is the saturation temperature at the vapor partial pressure. For both large and small breaks, the condensation removal rate (the rate at which condensate is removed from the atmosphere) is calculated using the following:

$$m_c = fq/(h_v - h_l)$$

Here f is the fraction of total heat transferred which is released by the condensate and h_v and h_l are specific enthalpies of the vapor and saturated liquid, respectively. The condensate thus calculated is immediately transferred to the sump with the remaining fraction (1-f) available for revaporization into the containment atmosphere. Ninety-percent of the condensate was transferred directly to the sump during large breaks and 92-percent was transferred during small breaks. When the wall temperature is above the saturation temperature, a natural convection heat transfer coefficient is used with no condensation allowed. For heat transfer to the sump floor, a constant value of 0.4 Btu/(ft^2-hr-^oF) is used for the heat transfer coefficient, as recommended in the CONTEMPT user manual, reference (10).

As can be seen in Table 6.2.1-27, the 32-inch double-ended break at 102-percent EOL with main steam isolation valve (MSIV) failure and loss of offsite power resulted in the highest peak pressure for these cases, which was 42.1 lb/in^2a. The 32-inch double-ended break at 80-percent EOL with MSIV failure and loss of offsite power resulted in a peak temperature of 371.4^F, which was the highest peak temperature of the cases analyzed. Cases were also analyzed to find the effects of various single failures. These failures included an MSIV failure, main feedwater isolation valve (MFIV) failure, and minimum safeguards. The MSIV and MFIV failures were analyzed with and without a loss of offsite power (LOOP). For all these cases, one set of containment sprays and the walls were modeled as discussed above and no credit was taken for the fan coolers. The results of this analysis are listed in Table 6.2.1-27. The mass and energy release for the two peak cases are given in Tables 6.2.1-28a and 6.2.1-28b. The plot of pressure versus time for the double-ended break at 102-percent EOL with MSIV failure and loss of offsite power is shown in Figure 6.2.1-163. Figures 6.2.1-164a, -164b, -164c, and -164d are plots of the RB vapor temperature, heat transfer coefficient, containment liner temperature, and interior concrete temperature versus time for the 80-percent EOL with an MSIV failure and LOOP case.

REFERENCES

9. W. M. Kays and A. L. London, Compact Heat Exchangers, Second Edition, McGraw Hill, Inc., 1964.

10. L. L. Wheat, CONTEMPT LT - A computer program for predicting containment pressure - temperature response to a Loss-of-Coolant Accident, ANCR-1219, Aerojet Nuclear Company, Idaho Falls, Idaho, June 1975.

11. H. E. Zittel, "Post-Accident Hydrogen Generation from Protective Coatings in Power Reactors," Nuclear Technology, Volume 17, February 1973, pp. 143-146.

12. J. H. Cudlin, P. W. Daggett, "TRAP 2-FORTRAN Program for Digital Simulation of the Transient Behavior of the Once-Through Steam Generator and Associated Reactor Coolant System" BAW-10128, Babcock and Wilcox, August 1976.

13. C. C. Francis, "Emperial Solution of Radiation Streaming Problems in the Air Gap Around Pressure Vessel of a PWR," Presented at the Fifth International Conference on Reactor Shielding, Knoxville, Tennessee, April 1977.

14. William D. Crouch, R. H. Bryan, and R. G. Irby, "Methodology for Predicting Containment Temperature Following a Main Steam Line Break," Tennessee Valley Authority.

TABLE 6.2.1-26
Steam Line Break - Peak Pressure and Temperature

Break	Peak RB Pressure lb/in^2g	Time of Peak Pressure Seconds	Peak RB Vapor Temp ^oF	Time of Peak Temp
32" O.D. double-ended at 102% EOL	38.8	27	353	24
32" O.D. double-ended at 80% EOL	35.8	24	360	15
32" O.D. double-ended at 40% EOL	31.9	60	350	60
4.76 ft^2 split-ended at 102% EOL	36.5	26	334	20
2.38 ft^2 split-ended at 102% EOL	34.7	38	335	25
1.19 ft^2 split-ended at 102% EOL	33.8	70	332	70
0.6 ft^2 split-ended at 102% EOL	30.9	140	283	136
0.3 ft^2 split-ended at 102% EOL	31.9	220	344	211
0.15 ft^2 split-ended at 102% EOL*	29.9	600	304	255

*Mass energy for this break is assumed to end at 600 seconds after rupture.

TABLE 6.2.1-27
TVA 32-Inch Double-Ended Steam Line Break
Single Failure Analysis

Power Level % of Rated Power	Single Failure	Time of RB Peak Pressure Seconds	Peak Pressure lb/in^2g	Peak Vapor Temperature °F	Time of Peak RB Temperature Seconds
102	Minimum Safeguards	27	38.8	353	24
102	MSIV Failure	27	40.8	359	24
102	MFIV Failure	30	40.6	355	28
102	MSIV Failure with LOOP	29	42.1	358	27
102	MFIV Failure with LOOP	28	40.2	352	28
80	Minimum Safeguards	24	35.8	360	15
80	MSIV Failure	23	37.1	369	15
80	MFIV Failure	25	36.9	368	17
80	MSIV Failure with LOOP	30	39.2	371	21
80	MFIV Failure with LOOP	30	38.0	369	22

TABLE 6.2.1-28a
Mass-Energy Release
TVA 32-Inch Double-Ended Steam Line Break at 102-Percent EOL, MSIV Failure Loop

Time Period, Seconds	Mass Release Rate lbm/sec	Specific Enthalpy Btu/lbm
0.000-- 1.0	13030.0	1221.8
1.001-- 2.0	11850.0	1210.1
2.001-- 4.0	11130.0	1208.2
4.001-- 5.0	10120.0	1204.0
5.001-- 7.0	9761.0	1204.4
7.001-- 9.0	8231.0	1204.7
9.001-- 10.0	7468.0	1204.7
10.001-- 11.0	6886.0	1204.6
11.001-- 12.0	6201.0	1204.4
12.001-- 14.0	5432.0	1204.2
14.001-- 16.0	4803.0	1203.8
16.001-- 18.0	4404.0	1203.3
18.001-- 19.0	3928.0	1202.9
19.001-- 21.0	3471.0	1202.3
21.001-- 22.0	2885.0	1201.6
22.001-- 24.0	2610.0	1200.7
24.001-- 25.0	2112.0	1237.1
25.001-- 26.0	1750.0	1250.7
26.001-- 27.0	1440.0	1258.5
27.001-- 28.0	1164.0	1258.7
28.001-- 29.0	893.0	1258.8
29.001-- 30.0	643.0	1253.8
30.001-- 31.0	455.0	1252.4
31.001-- 32.0	350.0	1237.5
32.001-- 36.0	105.0	1246.4
36.001--3600.0	0.0	1300.0

TABLE 6.2.1-28b
Mass-Energy Release
TVA 32-Inch Double-Ended Steam Line Break at 80-Percent EOL, MSIV Failure Loop

Time Period, Seconds	Mass Release Rate lbm/sec	Specific Enthalpy Btu/lbm
0.000-- 1.0	12860.0	1239.9
1.001-- 2.0	11460.0	1240.3
2.001-- 4.0	10040.0	1232.3
4.001-- 6.0	8864.0	1232.2
6.001-- 8.0	7835.0	1234.6
8.001-- 9.0	7159.0	1240.7
9.001-- 10.0	6317.0	1240.7
10.001-- 11.0	5137.0	1240.5
11.001-- 12.0	4452.0	1238.7
12.001-- 13.0	3946.0	1240.2
13.001-- 15.0	3348.0	1245.0
15.001-- 17.0	2852.0	1238.0
17.001-- 19.0	2494.0	1210.0
19.001-- 21.0	2024.0	1197.6
21.001-- 22.0	3414.0	1196.2
22.001-- 24.0	2418.0	1194.5
24.001-- 26.0	2130.0	1191.5
26.001-- 27.0	1751.0	1189.6
27.001-- 28.0	1447.0	1188.8
28.001-- 29.0	1231.0	1189.0
29.001-- 30.0	1079.0	1186.2
30.001-- 31.0	797.0	1202.9
31.001-- 32.0	584.0	1206.7
32.001-- 33.0	438.0	1209.5
33.001-- 39.0	165.0	1212.1
39.001--3600.0	0.0	1300.0

TABLE 6.2.1-39
Passive Heat Sinks - Mesh Spacing
Used in Main Steam Line Break Analysis

	Location	Region	Material	Mesh Points	Mesh Point Thickness
1.	Containment wall	1	Paint	17	1.625 mils
		2	Steel	21	0.0125 in
		3	Concrete	121	0.05 in
2.	Containment dome	1	Paint	17	0.625 mils
		2	Steel	21	0.0125 in
		3	Concrete	121	0.05 in
3.	Containment floor	1	Paint	17	0.625 mils
		2	Concrete	121	0.05 in
4.	Internal concrete	1	Paint	17	0.625 mils
		2	Concrete	121	0.05 in
5.	Hatches	1	Paint	17	0.625 mils
		2	Steel	121	0.05 in
6.	Crane girder	1	Paint	17	0.625 mils
		2	Steel	41	0.009 in
7.	Stairs	1	Paint	17	0.625 mils
		2	Steel	21	0.00625 in
8.	1/8" steel ductwork, piping	1	Steel	21	0.00625 in
9.	1/4" steel ductwork, piping	1	Steel	41	0.00625 in
10.	Miscellaneous steel	1	Paint	17	0.625 mils
		2	Steel	81	0.00625 in

TABLE 6.2.1-40
Passive Heat Sinks
Used in Main Steam Line Break Analysis

	Exposed Material				Secondary Material		
	Material	Thickness (in)	Exp. Surface Area (ft^2)	Mass 10^5 lb	Material	Thickness (in)	Mass 10^5 lb
1. Containment wall	Steel[a]	0.25	97120	9.9142	Concrete	42.0	492.884
2. Containment dome	Steel[a]	0.25	21200	2.1641	Concrete	36.0	92.220
3. Containment floor	Concrete[a]	36.0	20920	91.002	Steel	0.25	2.1355
4. Internal concrete	Concrete[a]	21.0	127800	324.2929	---	---	---
5. Hatches	Steel[a]	2.0	400	0.3266	---	---	---
6. Crane girder	Steel[a]	0.35	74000	10.5758	---	---	---
7. Stairs	Steel[a]	0.125	5000	0.2552	---	---	---
8. 1/8" steel ductwork, piping	Steel[b]	0.125	30000	1.5313	---	---	---
9. 1/4" steel ductwork, piping	Steel[b]	0.25	5000	0.5104	---	---	---
10. Miscellaneous steel	Steel[a]	0.5	70000	14.5775	---	---	---

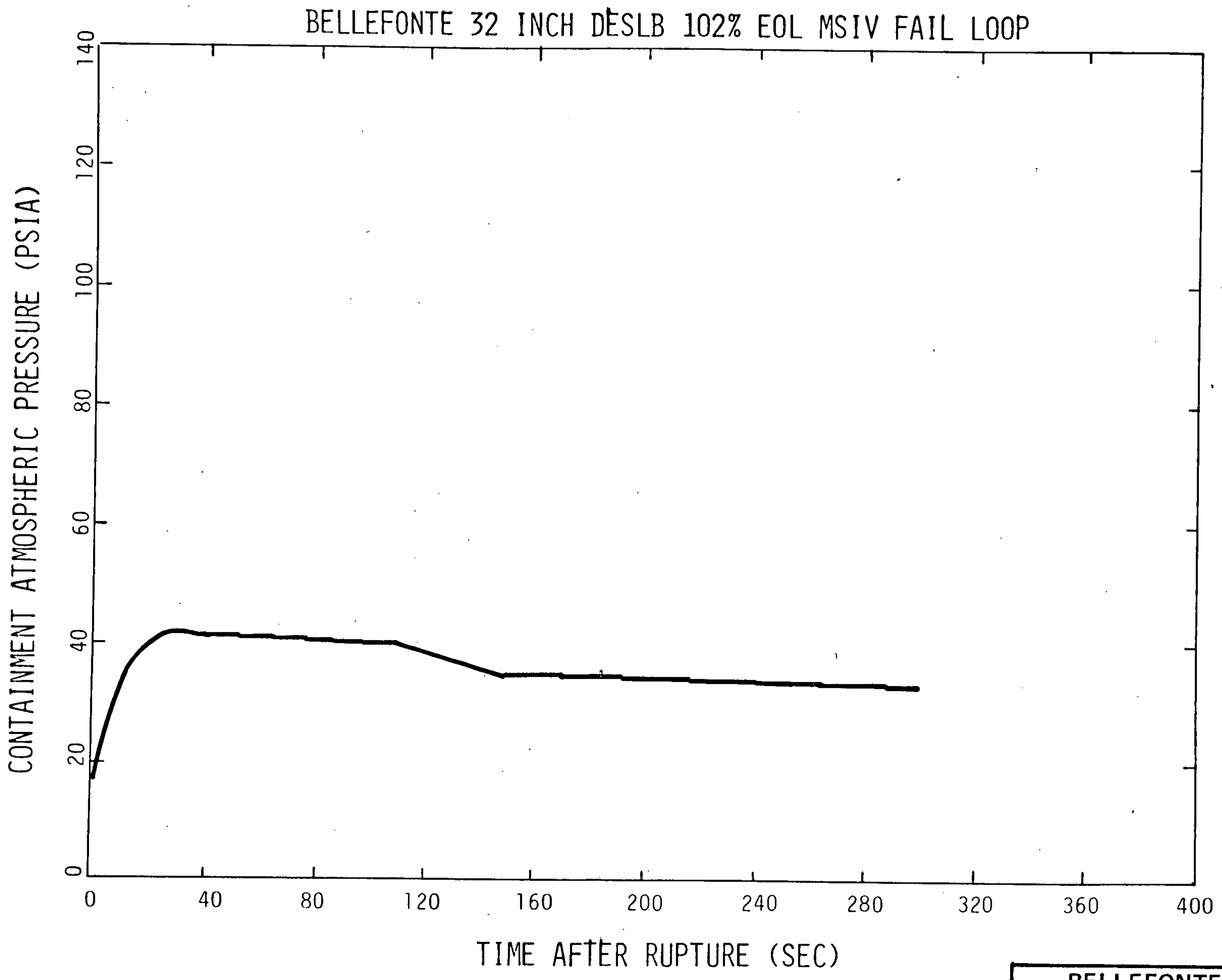
[a] Painted - paint 0.010" thick

[b] Unpainted

TABLE 6.2.1-41

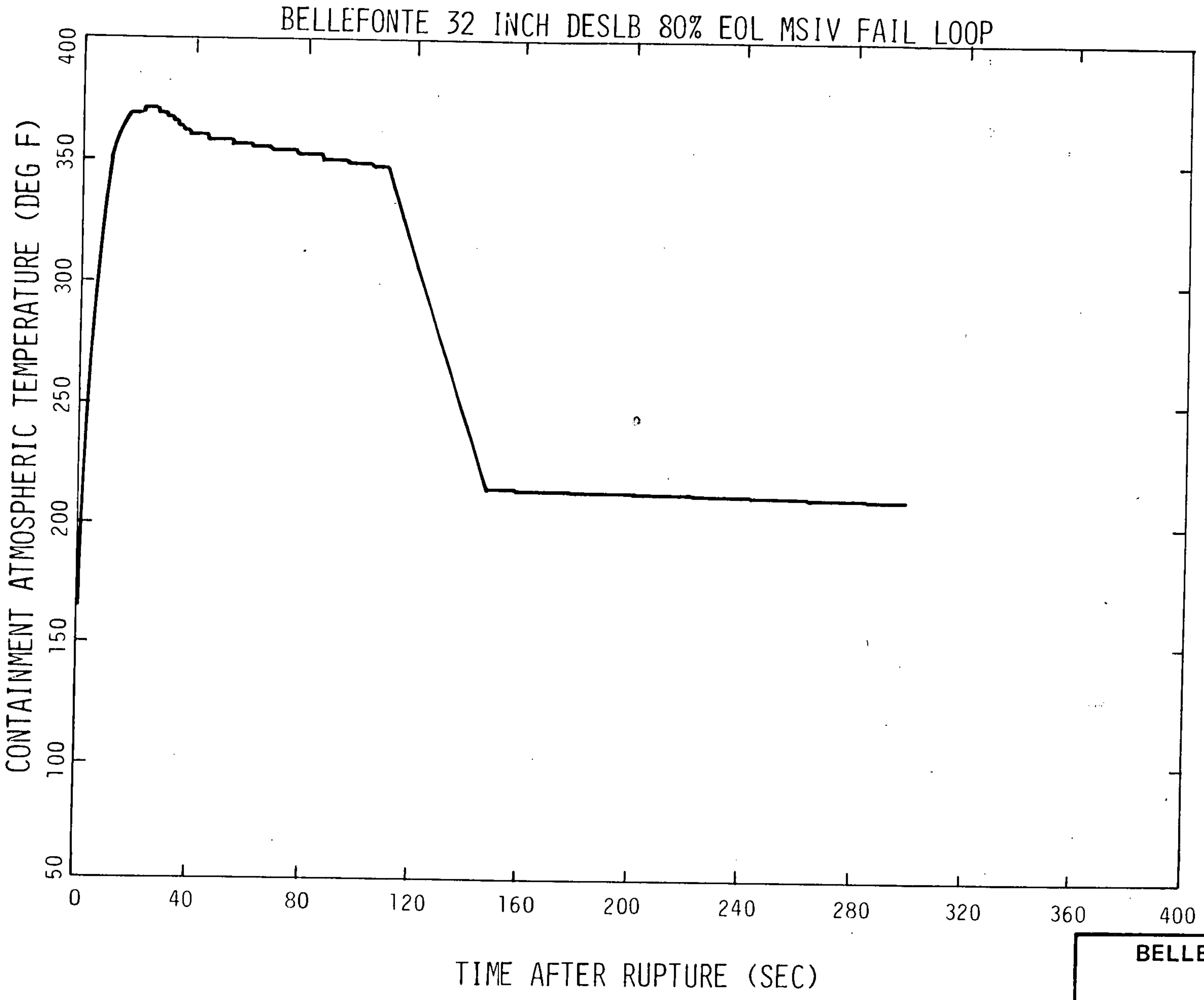
Passive Heat Sinks - Thermophysical Properties of Passive Heat Sink Material for Main Steam Line Break

Material	Density, lb/ft^3	Specific heat, $Btu/lb-^oF$	Thermal conductivity, $Btu/h-ft-^oF$	Volumetric heat capacity, Btu/ft^3-^oF
Concrete	145	0.156	0.9	22.62
Steel	490	0.120	27.0	58.8
Paint	---	---	0.083	35.0



BELLEFONTE NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

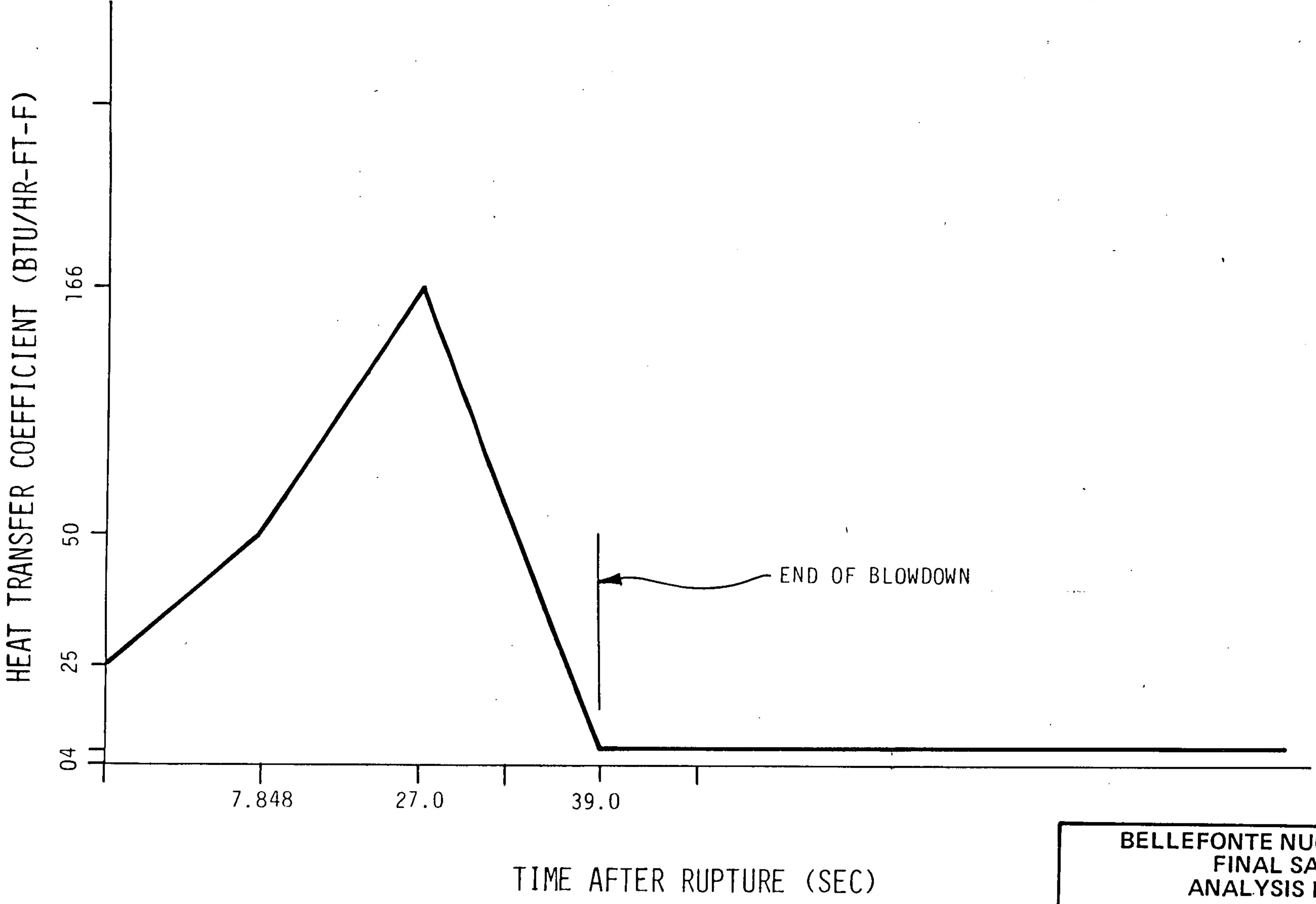
32" DOUBLE ENDED
SLB 102 EOL WITH
MSIV FAILURE & LOOP
FIGURE 6.2.1-163



**BELLEFONTE NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT**

**32" DOUBLE ENDED SLB
80% EDL WITH
MSIV FAILURE & LDOP
FIGURE 6.2.1-164a**

STEEL HEAT TRANSFER COEFFICIENT
32 INCH DESLB 80% EOL MSIV FAILURE LOOP



**BELLEFONTE NUCLEAR PLANT
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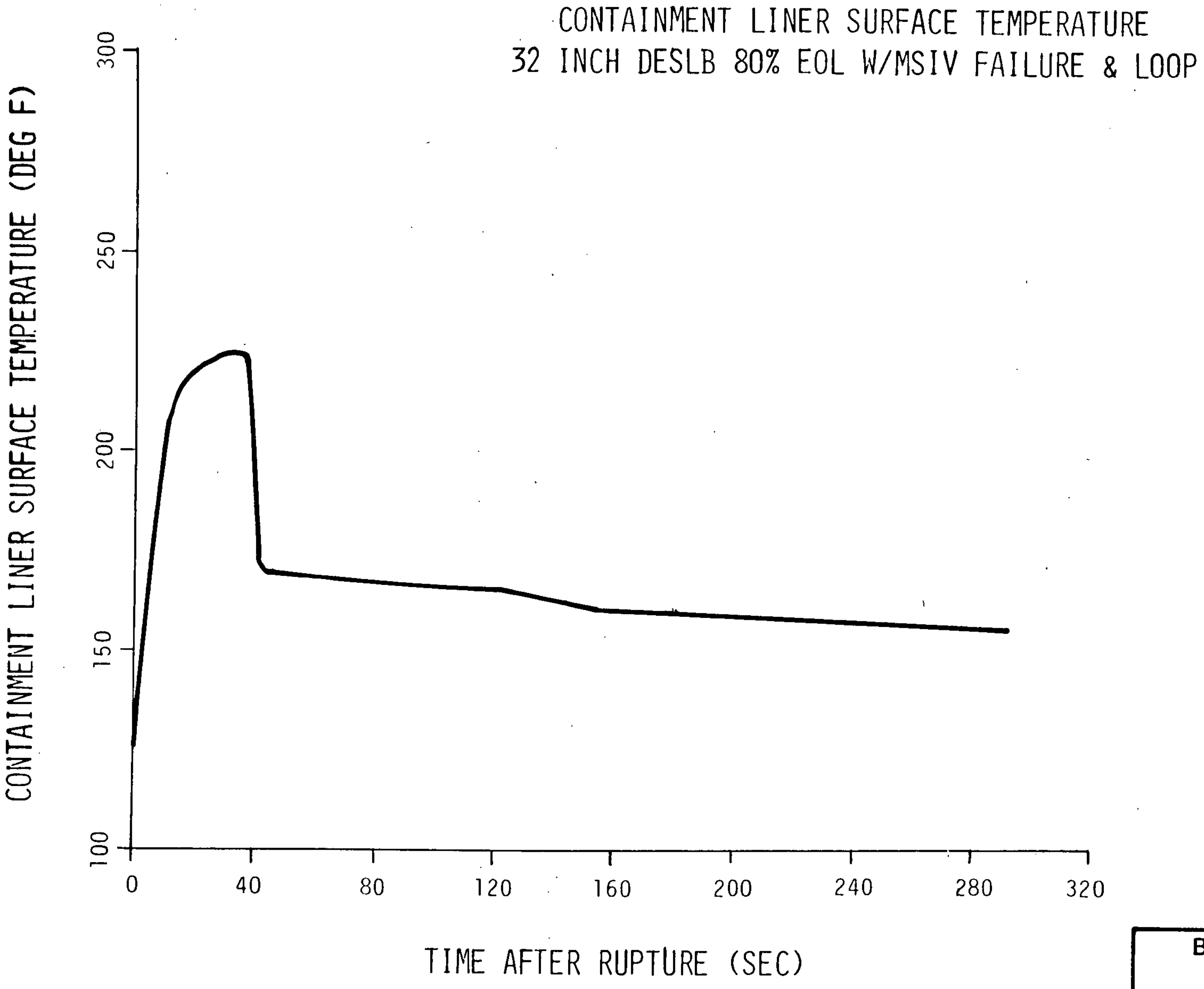
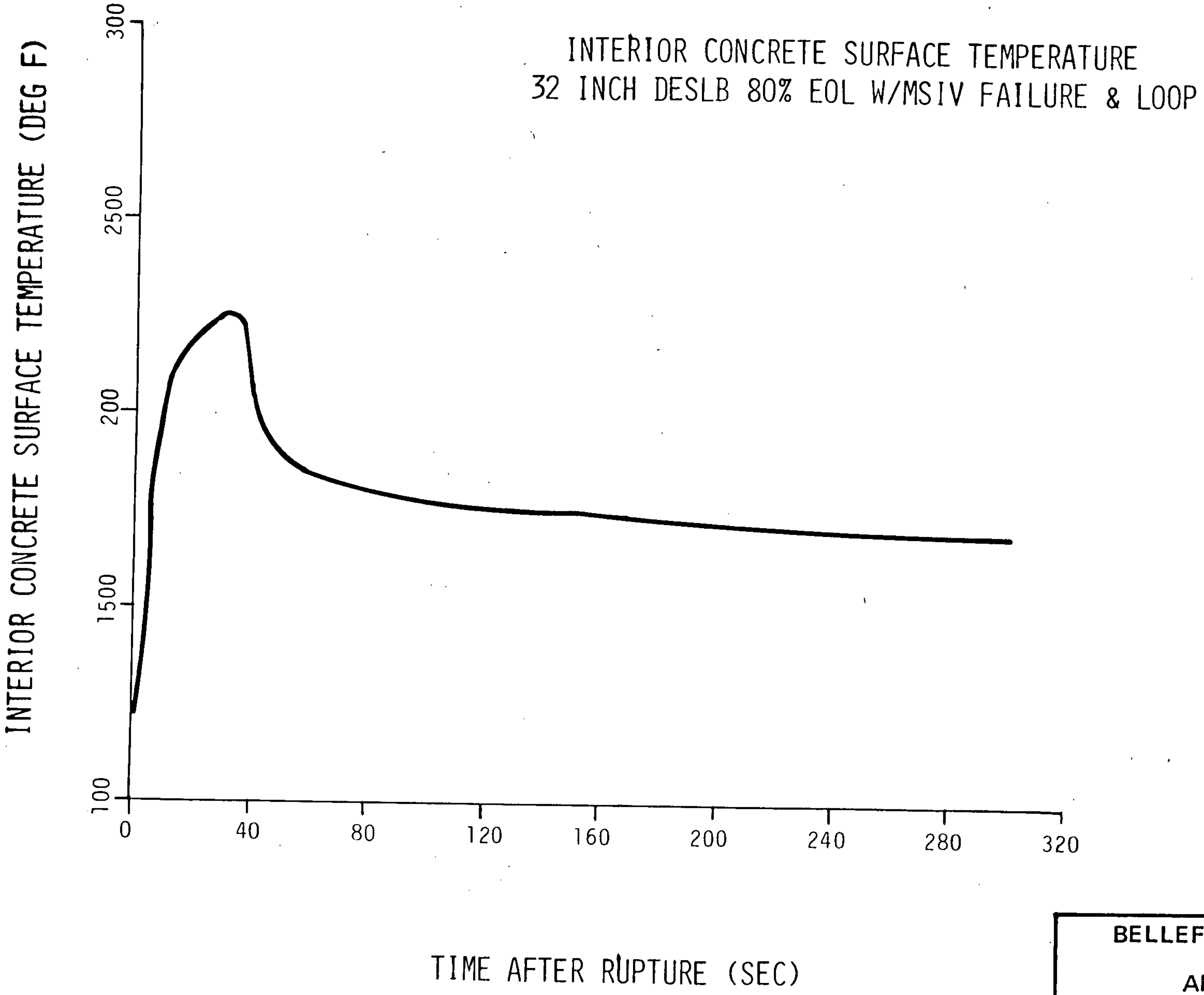
**STEEL HEAT TRANSFER COEFFICIENT
32" DOUBLE ENDED SLB
EOL WITH
MSIV FAILURE & LOOP
FIGURE 6.2.1-164b**



**BELLEFONTE NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT**

**CONTAINMENT LINER
SURFACE TEMPERATURE
32" DOUBLE ENDED SLB
80% EOL WITH
MSIV FAILURE & LOOP
FIGURE 6.2.1-164c**



**BELLEFONTE NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT**

**INTERIOR CONCRETE SURFACE TEMPERATURE
32" DOUBLE ENDED SLB 80% EOL WITH
MSIV FAILURE & LOOP
FIGURE 6.2.1-164d**

TABLE 3.11.1-3
SUMMARY OF ACCIDENT CONDITIONS

Building(1)	Location	Temperature (^{o}F)	Relative Hum. (%)	Pressure lb/in^{2}a	Pressure Dif. (lb/in^2)	Peak Accident Dose Rate(mr/hr)	Integrated Acc. Dose (rads)
Containment	Primary Containment	60-371.4(2)	10-100	64.3(2)	-4.0 to +50	$1.62x10^{10}$	$1.0x10^8$(30-day)(5)
	Secondary Containment	42-120	10-98	14.3	-0.4 to +4(4)	$2.1x10^7$ (3)	$5.0x10^7$ (30-da[illegible]
	Instr. Rms.	60-371.4(2)	10-100	64.3(2)	+4.0 to -4.0	---	$1.0x10^8$ (30-day)
Control	Main Control Room	50-104	10-80	14.3	+0.25 in/water	---	---
Aux. Bldg.	ESF Mech. Equip. Rms.	50-120	10-100	14.3(6)	-0.25 in/water	---	$4.6x10^7$ (30-day)
	RB Interface Area	50-120	10-98	14.3	slight positive pressure	---	$1.5x10^4$ (30-day)

Notes:

1. The design accident conditions for all other rooms shall be equivalent to the "upset" temperature, humidity, peak normal dose rate, and integrated dose shown in Table 3.11.1-2.
2. See Figures 6.2.1-10 and 6.2.1-22 for containment temperature and pressure versus time for a loss of coolant accident (LOCA). See Tables 6.2.1-26 and -27, and Figure 6.2.1-164a for the containment temperature results for main steam line breaks.
3. These levels will be experienced in the vicinity of the secondary containment air cleanup filters.
4. Pressure differential of +4.0 lb/in^2 with respect to primary containment.
5. RPS is qualified for 15 seconds; ESFAS is qualified for 1.0 minute; and PAMS is qualified for 12 hours.
6. Pipe break may result in locally higher pressures.