KG&E THE ELECTRIC COMPANY

KANSAS GAS AND ELECTRIC COMPANY

October 1, 1986

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

KMLNRC 86-176
Re: Docket No. STN 50-482
Subj: Large Break LOCA Analysis

Dear Mr. Denton:

The purpose of this letter is to transmit the Large Break Loss-of-Coolant Accident (LOCA) Analysis for Wolf Creek Generating Station (WCGS), Unit No. 1. This analysis is required by the WCGS Facility Operating License NPF-42, License Condition 2.C.12. License Condition 2.C.12 requires submittal of the worst large break LOCA analysis for NRC review and approval using an approved ECCS evaluation model, prior to restart following the first refueling outage.

The LOCA analysis was performed using the revised BART Evaluation Model, which considers the effect of core thimbles, and other recent modifications as described in WCAP-9561-P, Addendum 3. Break sizes with the discharge coefficient, C_D, = 0.4, 0.6, and 0.8 were analysed with minimum safeguards safety injection in order to determine the limiting break size. The worst break size was determined to be the C_D = 0.4 double-ended cold leg guillotine break and resulted in a Peak Clad Temperature (PCT) of 2099.9 deg-F. The C_D = 0.6 break was also analyzed assuming maximum safeguards safety injection to determine the sensitivity of minimum versus maximum safeguards safety injection flow. The minimum safeguards cases were determined to be limiting.

Present Technical Specification values including F_Q (z) and the K (z) curve are bounded by this analysis. This analysis demonstrates conformance for WCGS with the 10 CFR 50.46 requirements for Large Break LOCA analyses. If you have any questions concerning this matter please contact me or Mr. O. L. Maynard of my staff.

Very truly yours,

for John C Bailey

Glenn L. Koester
Vice President - Nuclear

8610090030 861001
PDR ADOCK 05000482
P PDR

GLK:see

Attachment

cc: PO'Connor
JCummins
EJohnson

A001
1/3

Rec'd w/check $150.00

STATE OF KANSAS)
) SS
CITY OF WICHITA)

I, John A. Bailey, of lawful age, being first duly sworn upon oath, do depose, state and affirm that I am Director Engineering and Technical Services of Kansas Gas and Electric Company, Wichita, Kansas, that I have signed the foregoing letter of transmittal for Glenn L. Koester, Vice President - Nuclear of Kansas Gas and Electric Company, know the content thereof; and that all statements contained therein are true.

By John A Bailey
John A. Bailey
Director Engineering and Technical Services

SUBSCRIBED and sworn to before me this 1st day of Oct., 1986.

Teresa G Handley
Notary Public

Expiration Date 9/10/89

TERESA G. HANDLEY NOTARY PUBLIC STATE OF KANSAS

LARGE BREAK LOCA ANALYSIS

USING THE BART COMPUTER CODE

6.2.1.4.5 Additional Information Required for Confirmatory Analysis

No additional information is deemed necessary for the performance of confirmatory analyses.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

The containment backpressure used for the limiting case (C_D = 0.~~6~~4) double-ended cold leg guillotine break for the ECCS analysis presented in Section 15.6.5 is presented in Figure 6.2.1-86. The containment backpressure is calculated, using the methods and assumptions described in Appendix A of Reference 9. Input parameters, including the containment initial conditions, net free containment volume, passive heat sink materials, thicknesses, and surface areas, and starting time and number of containment cooling systems used in the analysis, are described in the following paragraphs.

6.2.1.5.1 Mass and Energy Release Data

The mass and energy releases to the containment during the blowdown and reflood portions of the limiting break transient are presented in Tables 6.2.1-63 and 6.2.1-64.

The mathematical models which calculate the mass and energy releases to the containment are described in Section 15.6.5 and conform to 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." A break spectrum analysis is performed (see references in Section 15.6.5) that considers various break sizes, break locations, and Moody discharge coefficients for the double-ended cold leg quillotines which do not affect the mass and energy released to the containment. This effect is considered for each case analyzed. During refill, the mass and energy released to the containment is assumed to be zero, which minimizes the containment pressure. During reflood, the effect of steam water mixing between the safety injection water and the steam flowing through the reactor coolant system intact loops reduces the available energy released to the containment vapor spaces and therefore tends to minimize containment pressure.

6.2.1.5.2 Initial Containment Internal Conditions

The following initial values were used in the analysis:

a. A containment pressure of 14.7 psia.

b. A containment temperature of 90 F.

c. A refueling water storage tank temperature of 37 F.

d. An outside temperature of -60 F.

e. A relative humidity of 99 percent.

These containment initial conditions are representatively low values anticipated during normal full power operation.

6.2.1.5.3 Containment Volume

The volume used in the analysis was 2.7×10^6 ft^3.

6.2.1.5.4 Active Heat Sinks

The containment spray system and containment air coolers operate to remove heat from the containment.

Pertinent data for these systems which were used in the analysis are presented in Table 6.2.1-65.

The sump temperature was not used in the analysis because the maximum peak cladding temperature occurs prior to initiation of the recirculation phase for the containment spray system. In addition, heat transfer between the sump water and the containment vapor space was not considered in the analysis.

6.2.1.5.5 Steam-Water Mixing

Water spillage rates from the broken loop accumulator are determined as part of the core reflooding calculation and are included in the containment code (COCO) calculational model.

6.2.1.5.6 Passive Heat Sinks

The passive heat sinks used in the analysis, with their thermophysical properties, are given in Table 6.2.1-66. The passive heat sinks and thermophysical properties were derived in compliance with Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation."

6.2.1.5.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients used for heat transfer to the steel containment structures are given in Figure 6.2.1-87 for the limiting break. The containment pressure transient for the limiting break is shown in Figure 6.2.1-86.

6.2.1.5.8 Other Parameters

No other parameters, including the operation of the containment minipurge system, have a substantial effect on the minimum containment pressure analysis.

TABLE 6.2.1-63

MASS AND ENERGY RELEASE DURING BLOWDOWN FOR MINIMUM POST-LOCA CONTAINMENT PRESSURE

TIME (SEC)	MASS FLOW (LB/SEC)	ENERGY FLOW (BTU/SEC)
0.00000	0.0000000E+00	0.0000000E+00
0.05198	5.5471681E+04	3.0897817E+07
0.10133	5.3182319E+04	2.9591798E+07
0.15037	5.4812818E+04	3.0508679E+07
0.20038	5.4747017E+04	3.0471567E+07
0.25177	5.4753027E+04	3.0478936E+07
0.30087	5.4567568E+04	3.0377483E+07
0.35206	5.4480113E+04	3.0332608E+07
0.40117	5.4483882E+04	3.0340161E+07
0.45049	5.4294221E+04	3.0241612E+07
0.50071	5.4092992E+04	3.0136827E+07
0.55112	5.3426158E+04	2.9774612E+07
0.60040	5.3675149E+04	2.9923747E+07
0.65172	5.3212098E+04	2.9676127E+07
0.70092	5.3132940E+04	2.9644813E+07
0.75019	5.2744251E+04	2.9440432E+07
0.80158	5.2780453E+04	2.9477634E+07
0.85118	5.2530991E+04	2.9354424E+07
0.90096	5.2288181E+04	2.9236754E+07
0.95069	5.2245975E+04	2.9232266E+07
1.00061	5.1700297E+04	2.8944708E+07
1.10176	5.0824792E+04	2.8491074E+07
1.20169	5.0075073E+04	2.8107821E+07
1.30057	4.9119764E+04	2.7606434E+07
1.40130	4.7975369E+04	2.6995433E+07
1.50173	4.7807406E+04	2.6941199E+07
1.60103	4.7231284E+04	2.6650928E+07
1.70081	4.6584725E+04	2.6322652E+07
1.80095	4.5742696E+04	2.5880545E+07
1.90081	4.4998504E+04	2.5490844E+07
2.00059	4.3848510E+04	2.4862811E+07
2.10121	4.2419787E+04	2.4071096E+07
2.20089	4.2111267E+04	2.3930334E+07
2.30223	4.0899440E+04	2.3261149E+07
2.40179	4.0380546E+04	2.2997010E+07
2.50084	3.9910826E+04	2.2762017E+07
2.60070	3.9416120E+04	2.2514466E+07
2.70174	3.9052817E+04	2.2339092E+07
2.80150	3.8502569E+04	2.2057703E+07
2.90101	3.8015348E+04	2.1811596E+07
3.00112	3.6712053E+04	2.1100801E+07
3.10018	3.5656440E+04	2.0519039E+07
3.20134	3.4767157E+04	2.0020339E+07
3.30170	3.3803254E+04	1.9477510E+07
3.40101	3.3024994E+04	1.9042628E+07
3.50181	3.2901339E+04	1.8983528E+07
3.60120	3.1770630E+04	1.8363673E+07
3.70140	3.1460784E+04	1.8206857E+07
3.80227	3.1217553E+04	1.8088929E+07
3.90137	3.0382657E+04	1.7638027E+07
4.00196	3.0042356E+04	1.7467426E+07
4.10231	2.9770775E+04	1.7336461E+07
4.20235	2.9367197E+04	1.7133207E+07
4.30165	2.9127611E+04	1.7025421E+07
4.40170	2.8537560E+04	1.6723076E+07
4.50088	2.8212997E+04	1.6572116E+07
4.60172	2.7839474E+04	1.6392899E+07

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TABLE 6.2.1-63 (Sheet 2)

TIME (SEC)	MASS FLOW (LB/SEC)	ENERGY FLOW (BTU/SEC)
4.70122	2.7417373E+04	1.6187338E+07
4.80111	2.7127097E+04	1.6059038E+07
4.90224	2.6730711E+04	1.5870287E+07
5.00027	2.6189058E+04	1.5599065E+07
5.25126	2.4897842E+04	1.4953615E+07
5.50168	2.4500986E+04	1.4798809E+07
5.75091	2.3117509E+04	1.4063434E+07
6.00077	2.2287967E+04	1.3624766E+07
6.25098	2.2180918E+04	1.3562947E+07
6.50148	2.2572270E+04	1.3774071E+07
6.75061	2.2494979E+04	1.3733525E+07
7.00189	2.2263395E+04	1.3621478E+07
7.25173	2.1358049E+04	1.3138851E+07
7.50354	2.1068916E+04	1.3009015E+07
7.75748	2.0736184E+04	1.2862887E+07
8.00503	2.0487904E+04	1.2759719E+07
8.25640	2.0191603E+04	1.2640470E+07
8.50922	1.9846557E+04	1.2507904E+07
8.75236	1.9420245E+04	1.2342759E+07
9.00661	1.8911905E+04	1.2134220E+07
9.25271	1.8560017E+04	1.1969198E+07
9.50864	1.8311603E+04	1.1824228E+07
9.75110	1.8280752E+04	1.1788212E+07
10.00335	1.8127259E+04	1.1690946E+07
10.25557	1.7841968E+04	1.1571381E+07
10.50143	1.7498146E+04	1.1416162E+07
10.75477	1.7154096E+04	1.1257032E+07
11.00174	1.6835157E+04	1.1101395E+07
11.25456	1.6487347E+04	1.0934945E+07
11.50799	1.6073658E+04	1.0744983E+07
11.75185	1.5625563E+04	1.0542684E+07
12.00606	1.5154172E+04	1.0325946E+07
12.25346	1.4719920E+04	1.0114903E+07
12.50804	1.4320361E+04	9.9099033E+06
12.75320	1.3965716E+04	9.7234253E+06
13.00699	1.3611567E+04	9.5314936E+06
13.25527	1.3295572E+04	9.3511070E+06
13.50352	1.3021775E+04	9.1854583E+06
13.75701	1.2758802E+04	9.0187561E+06
14.00277	1.2511215E+04	8.8610237E+06
14.25290	1.2267639E+04	8.7017153E+06
14.50633	1.2029078E+04	8.5461835E+06
14.75292	1.1772510E+04	8.3815388E+06
15.00414	1.1519351E+04	8.2185176E+06
15.25181	1.1313308E+04	8.0720024E+06
15.50243	1.1162995E+04	7.9509680E+06
15.75443	1.1041060E+04	7.8360011E+06
16.00430	1.0948689E+04	7.7367441E+06
16.25391	1.0897746E+04	7.6565393E+06
16.50135	1.0896614E+04	7.6024577E+06
16.75075	1.0847835E+04	7.5325571E+06
17.00030	1.0732635E+04	7.4279312E+06
17.25118	1.0592407E+04	7.3209811E+06
17.50111	1.0399776E+04	7.1944207E+06
17.75027	1.0143065E+04	7.0433374E+06
18.00298	9.8723522E+03	6.8842983E+06
18.25066	9.6570097E+03	6.7434150E+06
18.50178	9.4259227E+03	6.5890321E+06
18.75217	9.1202712E+03	6.4127294E+06
19.00162	8.8032475E+03	6.2338981E+06
19.25272	8.5975171E+03	6.0915650E+06
19.50093	8.5280632E+03	5.9991130E+06
19.75294	8.3641867E+03	5.8814290E+06
20.00088	8.0047030E+03	5.7056319E+06

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TABLE 6.2.1-63 (Sheet 3)

TIME (SEC)	MASS FLOW (LB/SEC)	ENERGY FLOW (BTU/SEC)
20.25164	7.6370674E+03	5.5216255E+06
20.50219	7.5378839E+03	5.4304132E+06
20.75082	7.2300689E+03	5.2734818E+06
21.00178	6.8671034E+03	5.0968258E+06
21.25082	6.5589053E+03	4.9416451E+06
21.50250	6.2767557E+03	4.7912035E+06
21.75129	6.0346490E+03	4.6537124E+06
22.00107	5.8246557E+03	4.5255457E+06
22.25070	5.6323622E+03	4.4126039E+06
22.50093	5.4772775E+03	4.3147879E+06
22.75098	5.3080005E+03	4.2130852E+06
23.00025	5.1407904E+03	4.1116144E+06
23.25064	5.0101564E+03	4.0221117E+06
23.50081	4.8465388E+03	3.9234639E+06
23.75150	4.6672789E+03	3.8213058E+06
24.00064	4.4545966E+03	3.6879070E+06
24.25086	4.2498845E+03	3.5393754E+06
24.50160	4.0694507E+03	3.3898178E+06
24.75125	3.9445860E+03	3.2329349E+06
25.00063	4.0039609E+03	3.1293860E+06
25.25029	3.9730381E+03	3.0504008E+06
25.50024	3.7212447E+03	2.8970860E+06
25.75072	3.4577684E+03	2.7384361E+06
26.00110	3.1852908E+03	2.5688543E+06
26.25091	2.9431291E+03	2.4150158E+06
26.50169	3.3665568E+03	2.4925442E+06
26.75078	4.3836646E+03	2.7737051E+06
27.00052	4.9123266E+03	2.9050786E+06
27.25034	5.0817554E+03	2.9201573E+06
27.50099	5.2777059E+03	2.9454353E+06
27.75164	5.4301659E+03	2.9523375E+06
28.00099	5.5320037E+03	2.9381545E+06
28.25109	5.5975522E+03	2.9090508E+06
28.50137	5.6353159E+03	2.8688634E+06
28.75215	5.6512027E+03	2.8200543E+06
29.00210	5.6478407E+03	2.7639809E+06
29.25023	5.6293904E+03	2.7022407E+06
29.50101	5.5966430E+03	2.6343909E+06
29.75169	5.5474158E+03	2.5607102E+06
30.00164	5.4853462E+03	2.4829958E+06
30.25019	5.4077013E+03	2.4001150E+06
30.50113	5.3146917E+03	2.3120893E+06
30.75067	5.2111214E+03	2.2216267E+06
31.00133	5.0973920E+03	2.1275869E+06
31.25113	4.9703102E+03	2.0319775E+06
31.50097	4.8550772E+03	1.9440761E+06
31.75033	4.6732127E+03	1.8345207E+06
32.00069	4.8094215E+03	1.8427693E+06
32.25042	4.7569756E+03	1.8094467E+06
32.50027	4.5402999E+03	1.7206751E+06
32.75027	4.4993140E+03	1.6766193E+06
33.00022	4.4501772E+03	1.6342281E+06
33.25096	4.3001117E+03	1.5635108E+06
33.50114	4.1406660E+03	1.4862247E+06
33.75062	4.0418902E+03	1.4246201E+06
34.00066	3.9376645E+03	1.3643279E+06
34.25090	3.7840416E+03	1.2923804E+06
34.50099	3.7454213E+03	1.4209963E+06
34.75021	3.4783349E+03	1.1923785E+06
35.00079	3.2944506E+03	1.1035883E+06
35.25033	3.0861869E+03	1.0102918E+06
35.50019	2.9130371E+03	9.2434722E+05
35.75037	2.8079573E+03	8.5953626E+05
36.00028	2.8466842E+03	8.4270724E+05

TABLE 6.2.1-63 (Sheet 4)

TIME (SEC)	MASS FLOW (LB/SEC)	ENERGY FLOW (BTU/SEC)
36.25075	2.8600360E+03	8.3022828E+05
36.50098	2.8322902E+03	8.0865418E+05
36.75098	2.7951990E+03	7.8520833E+05
37.00061	2.7668287E+03	7.6533479E+05
37.25039	4.4474145E+03	1.1524620E+06
37.50081	6.2567636E+03	1.5694592E+06
37.75062	3.7984953E+03	9.3433220E+05
38.00098	4.1584691E+03	1.0005845E+06
38.25095	1.4055922E+03	3.2699096E+05
38.50073	2.0597712E+02	2.7118549E+04
38.75073	4.6468470E+02	6.4076078E+04
39.00018	-1.2212861E+02	-1.4247354E+05
39.25106	-2.0157203E+02	-2.3515111E+05
39.48105	-2.3064814E+02	-2.6907091E+05

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TABLE 6.2.1-64

MASS AND ENERGY RELEASE DURING REFLOOD
FOR MINIMUM POST-LOCA CONTAINMENT PRESSURE

TIME (sec)	$\dot{m}$(Total) (lbm/sec)	$\dot{m}$h(Total) (BTU/sec)
~~40.66~~ 55.408	0	0
~~44.966~~ 56.783	~~34.47~~ 0.033	~~4.45+4~~ 42.97
~~52.903~~ 66.205	~~295.76~~ 38.44	~~1.91+5~~ 4.92+4
~~66.103~~ 86.305	~~353.77~~ 66.31	~~1.99+5~~ 8.02+4
~~82.603~~ 108.005	~~374.29~~ 91.26	~~1.96+5~~ 1.12+5
~~101.203~~ 129.605	~~386.33~~ 242.77	~~1.90+5~~ 1.71+5
~~121.603~~ 150.605	~~395.66~~ 350.08	~~1.84+5~~ 1.95+5
~~167.603~~ 173.105	~~412.02~~ 361.50	~~1.68+5~~ 1.89+5
~~222.103~~	~~420.15~~	~~1.51+5~~
~~289.303~~	~~446.41~~	~~1.33+5~~
~~386.703~~	~~472.87~~	~~6.22+4~~

TABLE 6.2.1-65

ACTIVE HEAT SINK DATA
FOR MINIMUM POST-LOCA CONTAINMENT PRESSURE

<u>Containment Spray System Parameters</u>	
Number of pumps operating	2
Runout flow rate (total), gpm	7754
Temperature of spray, F	37
Actuation time (full flow), sec	25
<u>Containment Air Cooler Parameters</u>	
Number of fan coolers operating	4
Actuation time, sec	35

TABLE 6.2.1-66

STRUCTURAL HEAT SINKS

Thickness (ft)	Area (ft^2)
.021 Carbon steel 4.0 Concrete	64,919
.021 Carbon steel 3.0 Concrete	34,129
1.5 Concrete .021 Carbon steel 10.0 Concrete	13,538
1.0 Concrete	8,564
2.0 Concrete	43,497
2.5 Concrete	17,061
.021 Carbon steel 2.0 Concrete	7,821
.021 Stainless steel 2.0 Concrete	8,708
.0001083 Zinc coating .005 Carbon steel 2.0 Concrete	8,081
.0001083 Zinc Coating .0104 Carbon steel	186,183
.0104 Carbon steel	17,746
.0208 Carbon steel	114,205
.0417 Carbon steel	49,101
.0833 Carbon steel	31,372
.1667 Carbon steel	5,631
.3333 Carbon steel	8,355
.6667 Carbon steel	503
.0833 Carbon steel	9,726
.0104 Stainless steel	19,779
.0417 Stainless steel	10,885

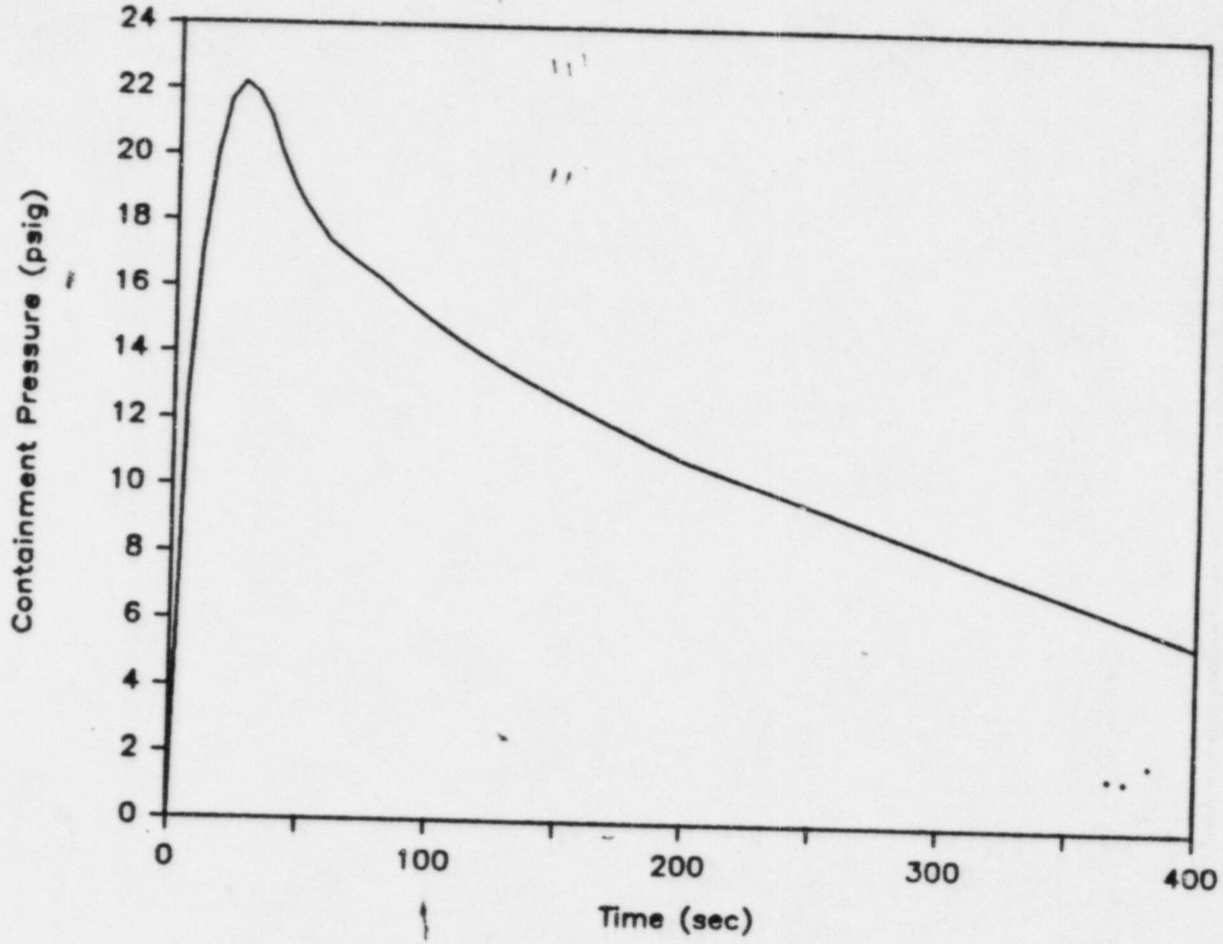

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FIGURE 6.2.1-86

CONTAINMENT BACKPRESSURE – DECLG (C_D = 0.4)

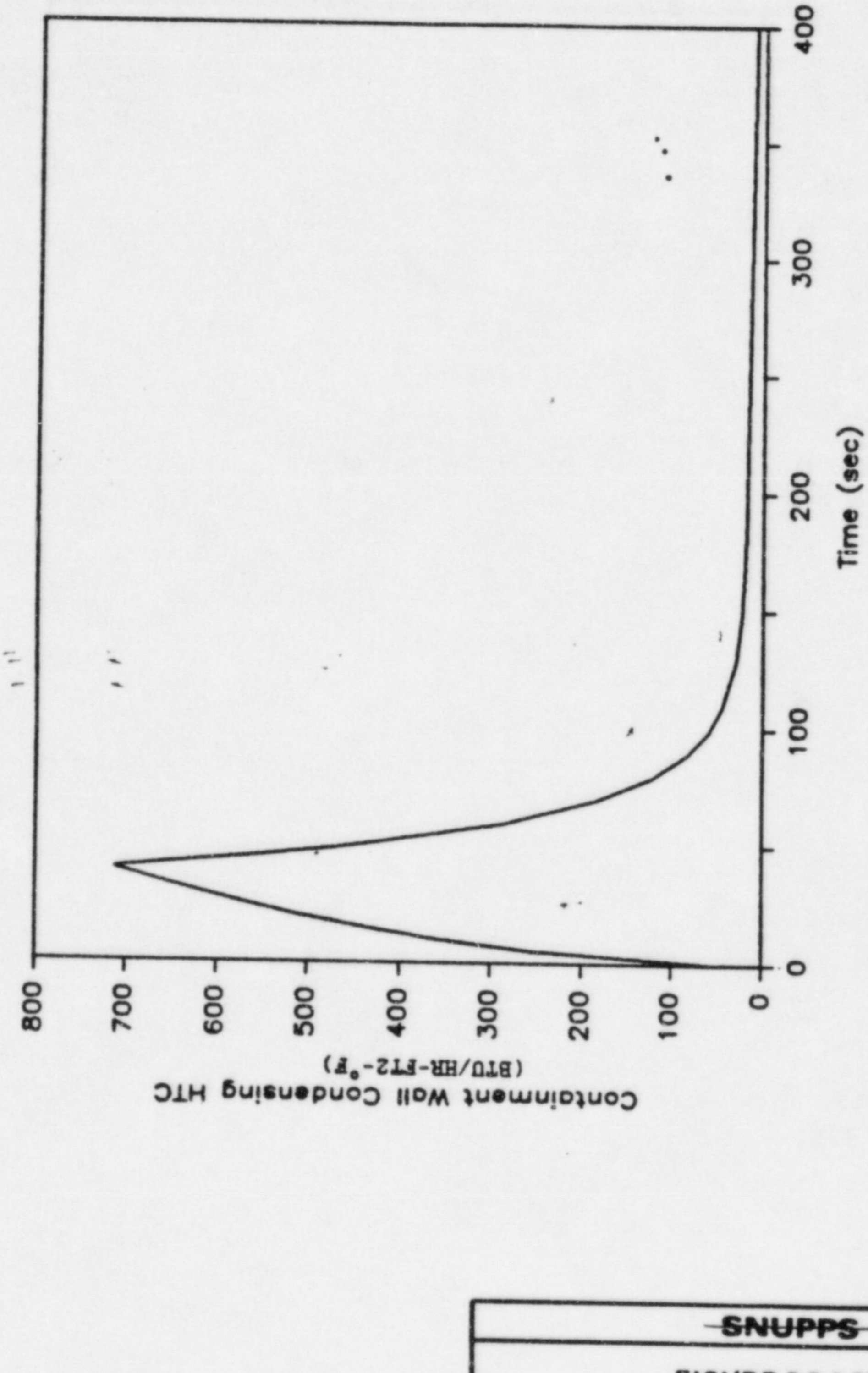

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FIGURE 6.2.1-87

CONDENSING HEAT TRANSFER COEFFICIENTS – DECLG (C_D = 0.4)

15.6.3.3.3.2 Doses to Receptor at the Exclusion Area Boundary and Low-Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of a postulated SGTR have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour period at the exclusion area boundary and for the duration of the accident (0 to 8 hours) at the low-population zone outer boundary. The results are listed in Table 15.6-5. The resultant doses are well within the guideline values of 10 CFR 100.

15.6.3.4 Conclusions

A steam generator tube rupture will cause no subsequent damage to the RCS or the reactor core. An orderly recovery from the accident can be completed, even assuming simultaneous loss of offsite power.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE OF CONTAINMENT

This section is not applicable to SNUPPS.

15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the RCS pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 square foot (ft^2). This event is considered an ANS Condition IV event, a limiting fault, in that it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see Section 15.0.1).

A minor pipe break (small break), as considered in this section, is defined as a rupture of the reactor coolant pressure boundary (see Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event, in that it is an infrequent fault which may occur during the life of the plant.

The Acceptance Criteria for the LOCA is described in 10 CFR 50.46 as follows:

a. The calculated peak fuel element clad temperature is below the requirement of 2,200 F.

b. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

c. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limits of 17 percent are not exceeded during or after quenching.

d. The core remains amenable to cooling during and after the break.

e. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in the emergency core cooling system (ECCS) performance following a LOCA. Reference 2 includes a recent study of the probability of the occurrence of RCS pipe ruptures.

In all cases, small breaks (less than 1.0 ft^2) yield results with more margin to the Acceptance Criteria limits than large breaks.

15.6.5.2 <u>Sequence of Events and Systems Operations</u>

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement the void formation in causing the rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken in the LOCA analysis for the boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large-break analysis.

b. Injection of borated water provides for heat transfer from the core and prevents excessive clad temperatures.

Description of Large-Break LOCA Transient

The sequence of events following a large-break LOCA is presented in Figure 15.6-4.

Before the break occurs, the unit is assumed to be in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of heat removal from the primary, secondary system pressure increases, and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2,250 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the bypassing of emergency core cooling water injected into the RCS are calculated not to be effective. At this time (called end-of-bypass), refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel, which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the later stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and, subsequently, supply water to maintain a full downcomer and complete the reflooding process.

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with the dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the engineering safety features sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure. Approximately 24 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small-Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for a small-break LOCA there are only three characteristic stages, i.e., a gradual blowdown in which the decrease in water level is checked, core recovery, and long-term recirculation.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50.

Large-Break LOCA Evaluation Model

The analysis of a large-break LOCA transient is divided into three phases: 1) blowdown, 2) refill, and 3) reflood. There are three distinct transients analyzed in each phase, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in References 3, 18, 28, 29, and 30. Th~~is~~ese documents describe~~s~~ the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes which are used in the LOCA analysis are described in detail in References 4 through 7; code modifications are specified in References 8 through 10. insert A These codes are used to assess core heat transfer and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown, and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. insert B The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown phase, these data are transferred to the WREFLOOD code. Also at the end-of-blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during this first phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core inlet flow rate and enthalpy, the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters the bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the containment through the break. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. ~~WREFLOOD is also linked to the LOCTA-IV code in that thermal-hydraulic parameters from WREFLOOD are used by LOCTA-IV in its calculation of the fuel temperature.~~ Replace with insert D LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core. insert E

~~The large-break analysis was performed with the February 1978 version of the evaluation model, which includes modifications delineated in References 15 through 18.~~ Replace with insert C

Insert A

The BART code is described in References 29 and 30.

Insert B

The BART computer code is used to calculate the fluid and heat transfer conditions in the core during reflood.

Insert C

The large break analysis was performed with the approved December, 1981 version of the Evaluation Model (Reference 28), with the approved 1984 version of BART (Reference 29).

Insert D

With input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the hydraulic and heat transfer conditions in the core during reflood.

Insert E

A schematic representation of the computer code interfaces for large break calculations is shown in Figure 15.6-5.

The analysis in this section was performed with the upper head fluid temperature equal to the reactor coolant system cold leg fluid temperature, achieved by increasing the upper head cooling flow (Ref. 23).

Small-Break LOCA Evaluation Model

The WFLASH program used in the analysis of the small-break LOCA is an extension of the FLASH-4 code (Ref. 11) developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the RCS.

The RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied through the system. A detailed description of WFLASH is given in Reference 12.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the cross-over leg during a loss-of-coolant transient.

Clad thermal analyses are performed with the LOCTA-IV code (Ref. 7), which uses the RCS pressure, fuel rod power history, steam flow past the uncovered part of the core, and mixture height history from the WFLASH hydraulic calculations as input.

Figure 15.6-44 presents the hot rod power shape utilized to perform the small-break analysis. This power shape was chosen because it provides a conservative distribution of power versus core height, and also local power is maximized in the upper regions of the reactor core (10' to 12'). This power shape is skewed to the top of the core with the peak local power occurring at the 10.5-foot core elevation.

This is limiting for the small-break analysis, because of the core uncovery process for small breaks. As the core uncovers, the cladding in the upper elevation of the core heats up and is sensitive to the local power at that elevation. The cladding temperatures in the lower elevation of the core, below the two-phase mixture height, remain low. The peak clad temperature occurs above 10 feet.

Schematic representations of the computer code interfaces are given in Figures 15.6-5 and 15.6-6.

The small-break analysis was performed with the October 1975 version of the Westinghouse ECCS Evaluation Model (Ref. 7, 12, 13, and 14).

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6-9 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature. The effect of using the cold leg temperature in the reactor vessel upper head is described in Reference 23. In addition, the analysis in this section utilized the upflow barrel-baffle methodology described in Reference 19.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from sensitivity studies (Ref. 20, 21, 22). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core-peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

15.6.5.3.3 Results

<u>Large-Break Results</u>

Based on the results of the LOCA sensitivity studies (Ref. 20, 21, and 22), the limiting large break was found to be the double-ended cold leg guillotine (DECLG) break. Therefore, only the DECLG break is considered in the large-break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.6-10 and 15.6-11. The worst break in the spectrum of break sizes analyzed was a discharge coefficient (C_D) of 0.64 This worst break was analyzed with a 12-second diesel generator start time in order to study the effect of a 2-second delay in the start of the diesel generator. Two cases are presented in Table 15.6-11 for the C_D = 0.6 DECLG break: the minimum safety injection case and the maximum safety injection case (Reference 27 methodology). ~~The maximum safety injection case proved to be the most limiting.~~

The mass and energy release data for the break resulting in the highest calculated peak clad temperature are presented in Section 6.2.1.5.

Figures 15.6-7 through 15.6-30 present the parameters of principal interest from the large-break ECCS analyses. For all cases analyzed, transients of the following parameters are presented:

a. Hot spot clad temperature

b. Coolant pressure in the reactor core

c. Water level in the core and downcomer during reflood

d. Core reflooding rate

e. Thermal power during blowdown

The containment pressure transient resulting from a LOCA is presented in Section 6.2.1.5.

For the limiting break analyzed, the following additional transient parameters are presented:

a. Core flow during blowdown (inlet and outlet)

b. Core heat transfer coefficients

c. Hot spot fluid temperature

d. Mass released to containment during blowdown

e. Energy released to containment during blowdown

f. Fluid quality in the hot assembly during blowdown

g. Mass velocity during blowdown

h. Accumulator water flow rate during blowdown

i. Pumped safety injection water flow rate during reflood

The maximum clad temperature calculated for a large break is 2100 ~~2174~~ F, which is less than the Acceptance Criteria limit of 2200 F of 10 CFR 50.46. The maximum local metal-water reaction is 4.54 ~~6.18~~ percent, which is well below the embrittlement limit of 17 percent, as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, compared with the 1-percent criterion of 10 CFR 50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop, and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

Small-Break Results

As noted previously, the calculated peak clad temperature resulting from a small-break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies (Ref. 20) the limiting small break was found to be less than a 10-inch-diameter rupture of the RCS cold leg. Therefore, a range of small-break analyses is presented which establishes the limiting break size. The results of these analyses are summarized in Tables 15.6-10 and 15.6-12.

Figures 15.6-31 through 15.6-44 present the principal parameters of interest for the small-break ECCS analyses. For all cases analyzed, the following transient parameters are presented:

a. RCS pressure

b. Core mixture height

c. Hot spot clad temperature

For the limiting break analyzed, the following additional transient parameters are presented:

a. Core steam flow rate

b. Core heat transfer coefficient

c. Hot spot fluid temperature

The maximum calculated peak clad temperature for all small breaks analyzed is 1790 F. These results are well below all Acceptance Criteria limits of 10 CFR 50.46, and in all cases are not limiting when compared to the results presented for large breaks.

15.6.5.4 Radiological Consequences

15.6.5.4.1 Method of Analysis

15.6.5.4.1.1 Containment Leakage Contribution

PHYSICAL MODEL - Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the ECCS limits the clad temperature to well below the melting point and ensures that the reactor core remains intact and in a coolable geometry, minimizing the release of fission products to the containment. However, to demonstrate that the operation of a nuclear power plant does not represent any undue radiological hazard to the general public, a hypothetical accident involving a significant release of fission products to the containment is evaluated.

15.6.5.4.3.2 Doses to a Receptor at the Exclusion Area Boundary and Low Population Zone Outer Boundary

The potential radiological consequences resulting from the occurrence of the postulated LOCA have been conservatively analyzed, using assumptions and models described in previous sections.

The total-body dose due to immersion and the thyroid dose due to inhalation have been analyzed for the 0-2 hour dose at the exclusion area boundary and for the duration of the accident at the LPZ outer boundary. The results are listed in Table 15.6-5. The resultant doses are within the guideline values of 10 CFR 100.

15.6.5.4.3.3 Doses to Control Room Personnel

Radiation doses to control room personnel following a postulated LOCA are based on the ventilation, cavity dilution, and dose model discussed in Section 15A.3.

Control room personnel are subject to a total-body dose due to immersion and a thyroid dose due to inhalation. These doses have been analyzed, and are provided in Table 15.6-8. The resultant doses are within the limits established by GDC-19.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to SNUPPS.

15.6.7 REFERENCES

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TABLE 15.6-9

INPUT PARAMETERS USED IN THE ECCS ANALYSIS

Licensed core power*, MWt	3411
Peak linear power, includes 102 percent factor, kW/ft	~~12.88~~ 13.439
Total peaking factor, F^T_Q	~~2.32~~ 2.42
Axial peaking factor, F_Z	~~1.453~~ 1.5613
Power shape	
Large break	Chopped cosine
Small break	See Figure 15.6-44
Fuel assembly array	17 x 17
Accumulator water volume, nominal, ft^3/accumulator	850
Accumulator tank volume, nominal, ft^3/accumulator	1390
Accumulator gas pressure, minimum psia	600
Safety injection pumped flow	See Figures 15.6-~~20~~ 30 and 15.6-43
Containment parameters	See Section 6.2
Initial loop flow, lb/sec	~~9913~~ 9606.39
Vessel inlet temperature, F	~~556.0~~ 559.47
Vessel outlet temperature, F	~~616.8~~ 627.19
Reactor coolant pressure, psia	2250
Steam pressure, psia	~~982~~ 984.07
Steam generator tube plugging level, %	~~0~~ 10

*Two percent is added to this power to account for calorimetric error.

TABLE 15.6-10

TIME SEQUENCE OF EVENTS FOR LOSS-OF-COOLANT ACCIDENTS

Accident	Event	Time (sec)
Large break LOCA		
a. DECLG C_D=0.8	Start	0.0
	Reactor trip signal	0.774 ~~0.82~~
	Safety injection signal	1.15 ~~1.0~~
	Accumulator injection begins	13.00 ~~13.9~~
	End-of-bypass	25.531 ~~24.2~~
	Pump injection begins	28.15 ~~26.0~~
	End-of-blowdown	25.531 ~~27.0~~
	Bottom of core recovery	39.291 ~~37.4~~
	Accumulator empty	51.126 ~~47.9~~
b. DECLG C_D=0.6 (Minimum SI ~~with 12-second diesel generator start~~)	Start	0.0
	Reactor trip signal	0.794 ~~0.83~~
	Safety injection signal	1.32 ~~1.15~~
	Accumulator injection begins	15.4 ~~15.9~~
	End-of-bypass	29.482 ~~26.5~~
	Pump injection begins	28.32 ~~28.6~~
	End-of-blowdown	29.482 ~~29.1~~
	Bottom of core recovery	43.495 ~~40.9~~
	Accumulator empty	54.248 ~~50.2~~
c. DECLG C_D=0.6 (Maximum SI ~~with 12-second diesel generator start~~)	Start	0.0
	Reactor trip signal	0.794 ~~0.83~~
	Safety injection signal	1.32 ~~1.15~~
	Accumulator injection begins	15.4 ~~15.9~~
	End-of-bypass	29.482 26.5
	Pump injection begins	28.32 ~~28.6~~
	End-of-blowdown	29.482 ~~29.1~~
	Bottom of core recovery	42.826 ~~40.5~~
	Accumulator empty	55.469 ~~51.3~~
d. DECLG C_D=0.4	Start	0.0
	Reactor trip signal	0.829 ~~0.82~~
	Safety injection signal	1.62 ~~1.42~~
	Accumulator injection begins	20.0 ~~20.7~~
	Pump injection begins	28.62 ~~26.4~~
	End-of-bypass	39.481 ~~34.6~~
	End-of-blowdown	39.481 ~~37.7~~
	Bottom of core recovery	55.408 ~~48.9~~
	Accumulator empty	60.711 ~~56.9~~

~~SNUPPS~~ Wolf Creek

TABLE 15.6-10 (Sheet 2)

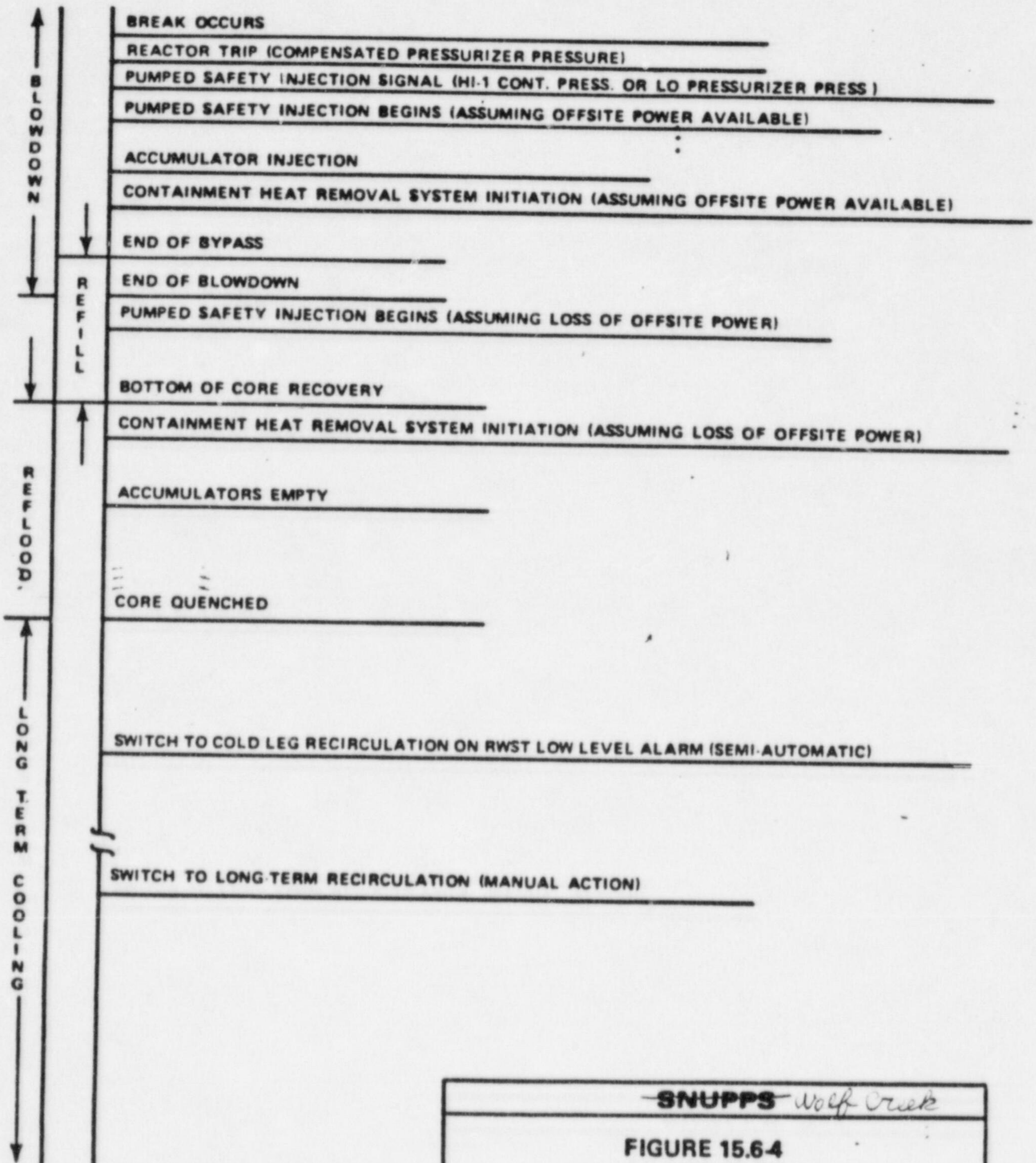
Accident	Event	Time (Sec)
Small break LOCA		
a. 3 inch	Start	0.0
	Reactor trip signal	29.7
	Top of core uncovered	623
	Accumulator injection begins	N/A
	Peak clad temperature occurs	1351
	Top of core covered	2300
b. 4 inch	Start	0.0
	Reactor trip signal	17.3
	Top of core uncovered	324
	Accumulator injection begins	785
	Peak clad temperature occurs	836
	Top of core covered	846
c. 6 inch	Start	0.0
	Reactor trip signal	13.3
	Top of core uncovered	135
	Accumulator injection begins	330
	Peak clad temperature occurs	359
	Top of core covered	364

TABLE 15.6-11

LARGE BREAK LOCAL RESULTS*
FUEL CLADDING DATA

Results	DECLG C_D=0.8	DECLG C_D=0.6		DECLG C_D=0.4
	Case a	Case b	Case c	Case d
Peak clad temperature, F	1867.7 ~~2053.6~~	1985.8 ~~2114.5~~	~~2174.2~~ 1977.0	~~1624~~ 2099.9
Peak clad location, ft	7.25 ~~7.5~~	7.25 ~~7.5~~	~~7.5~~ 7.25	~~7.5~~ 7.25
Local Zr/H_2O reaction, max. (%)	2.20 ~~4.38~~	2.79 ~~5.38~~	~~6.18~~ 2.73	~~0.85~~ 4.54
Local Zr/H_2O location, ft	6.5 ~~7.5~~	6.0 ~~7.5~~	~~7.5~~ 7.25	~~7.5~~ 6.0
Total Zr/H_2O reaction, %	<0.3	<0.3	<0.3	<0.3
Hot rod burst time, sec	52.0 ~~29.2~~	46.8 ~~27.8~~	~~27.8~~ 47.0	~~N/A~~ 61.8
Hot rod burst location, ft	6.0	6.0	6.0	~~N/A~~ 6.0

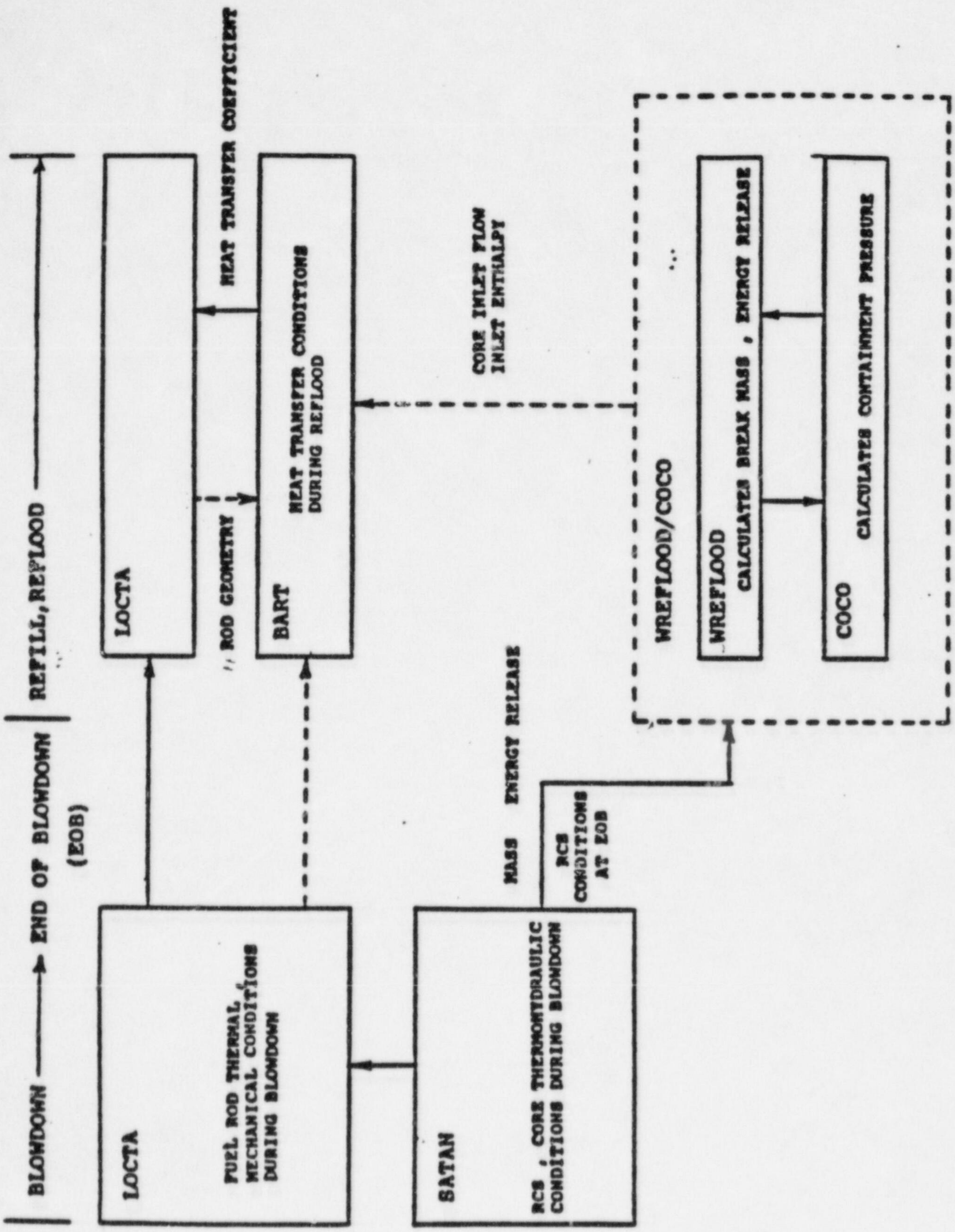
*Refer to Section 15.6.5.3.3 and Table 15.6-10 for a definition of cases a through d.



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-4

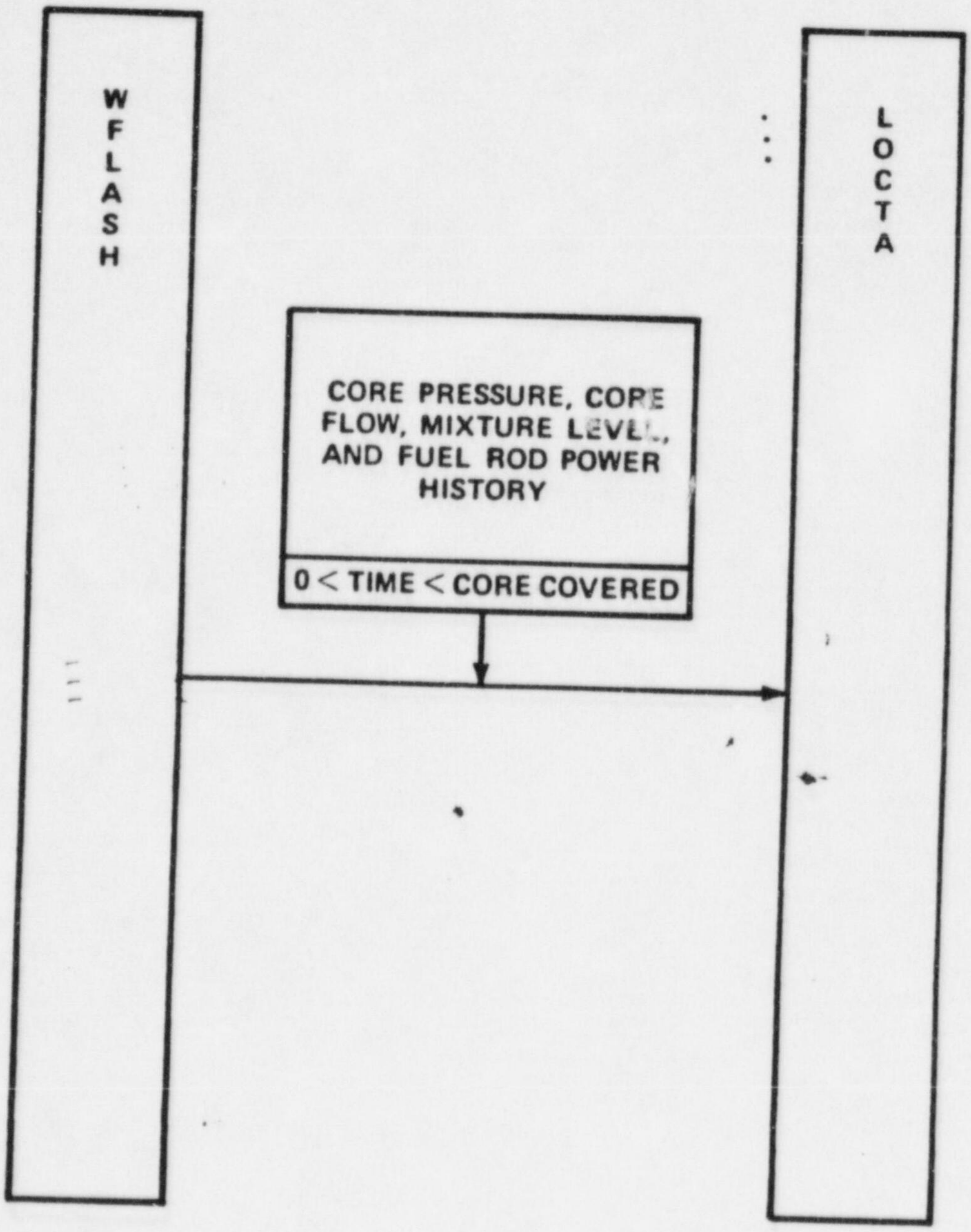
SEQUENCE OF EVENTS FOR LARGE BREAK LOCA ANALYSIS



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-5

CODE INTERFACE DESCRIPTION FOR LARGE BREAK MODEL



~~SNUPPS~~ Wolf Creek

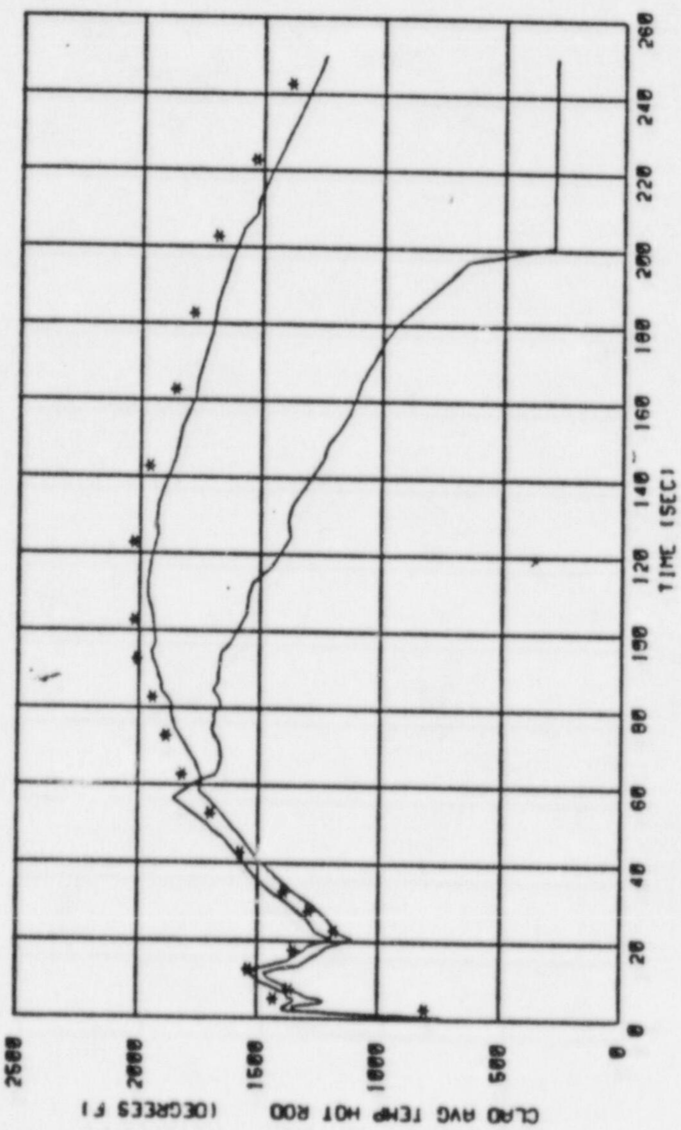
FIGURE 15.6-6

CODE INTERFACE DESCRIPTION FOR SMALL BREAK MODEL

LEGEND:

LINE MARKED BY AN * IS AT THE PEAK NODE ELEVATION (7.25 ft FOR ALL CASES)

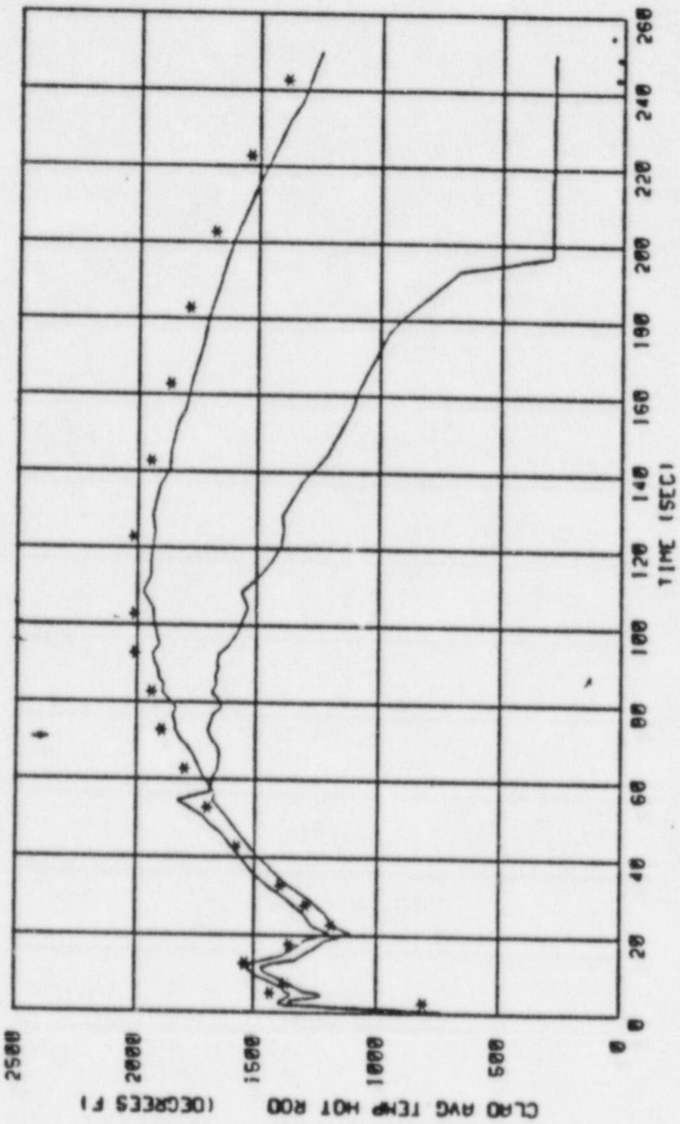
UNMARKED LINE IS AT THE BURST NODE ELEVATION (6.00 ft FOR ALL CASES)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-7

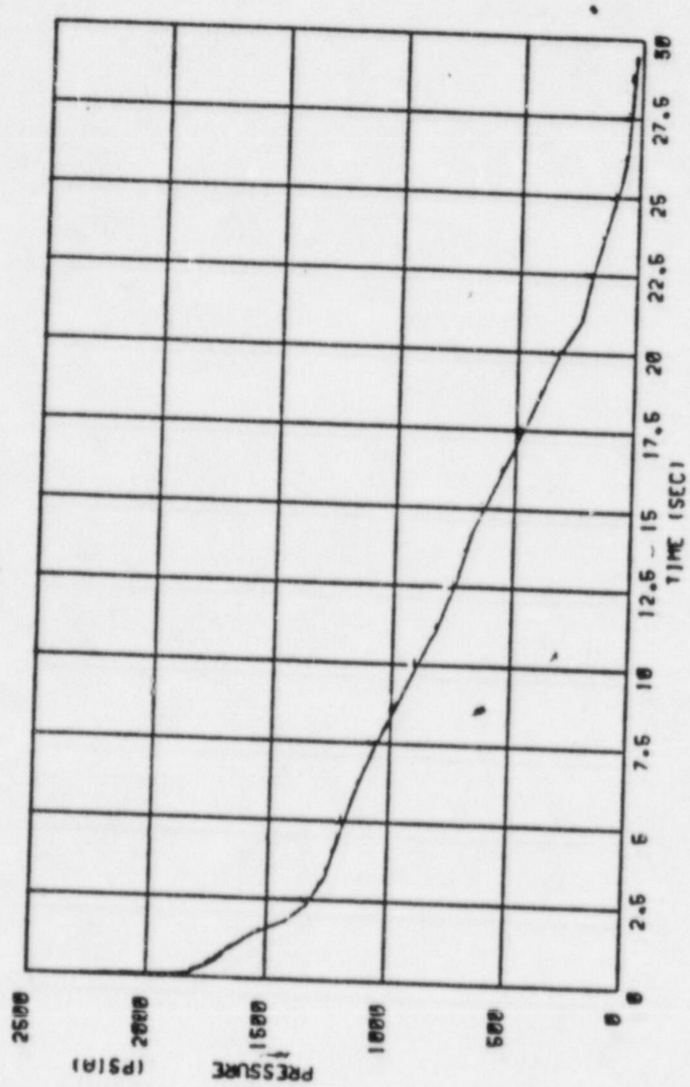
PEAK CLAD TEMPERATURE – DECLG (C_D = 0.6 - MIN SI)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-7a

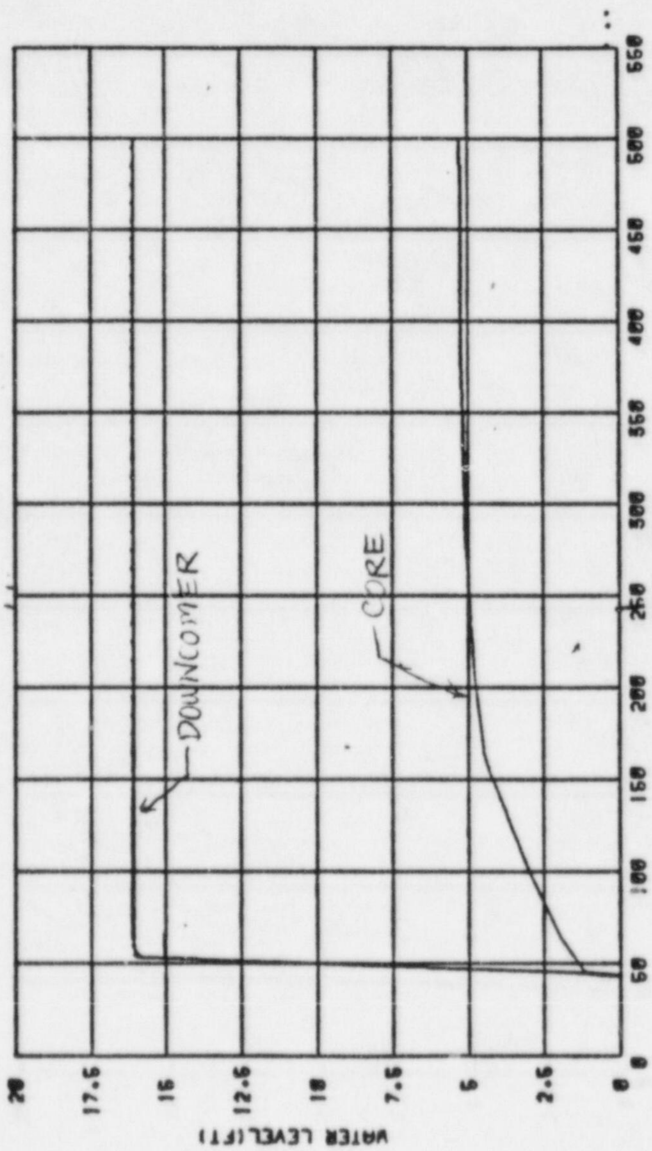
PEAK CLAD TEMPERATURE DECLG (C_D = 0.6 - MAX SI)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-8

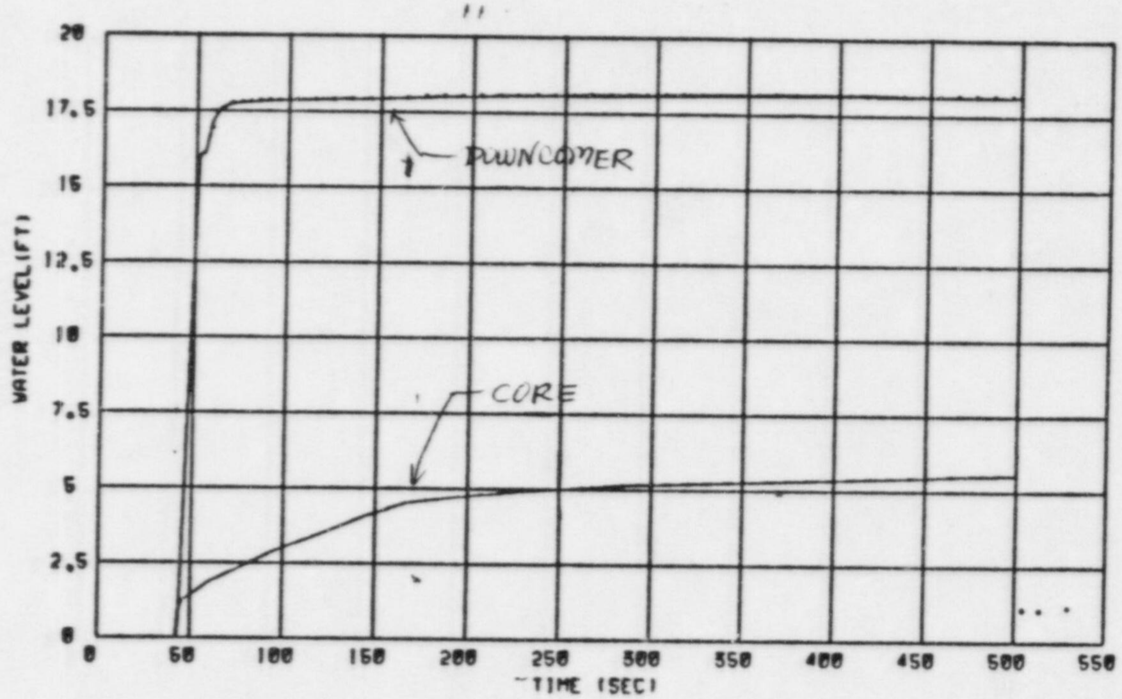
CORE PRESSURE – DECLG (C_D = 0.6)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-9

DECLG (C_D = 0.6 - MIN SI
DOWNCOMER AND
CORE WATER LEVELS DURING REFLOOD

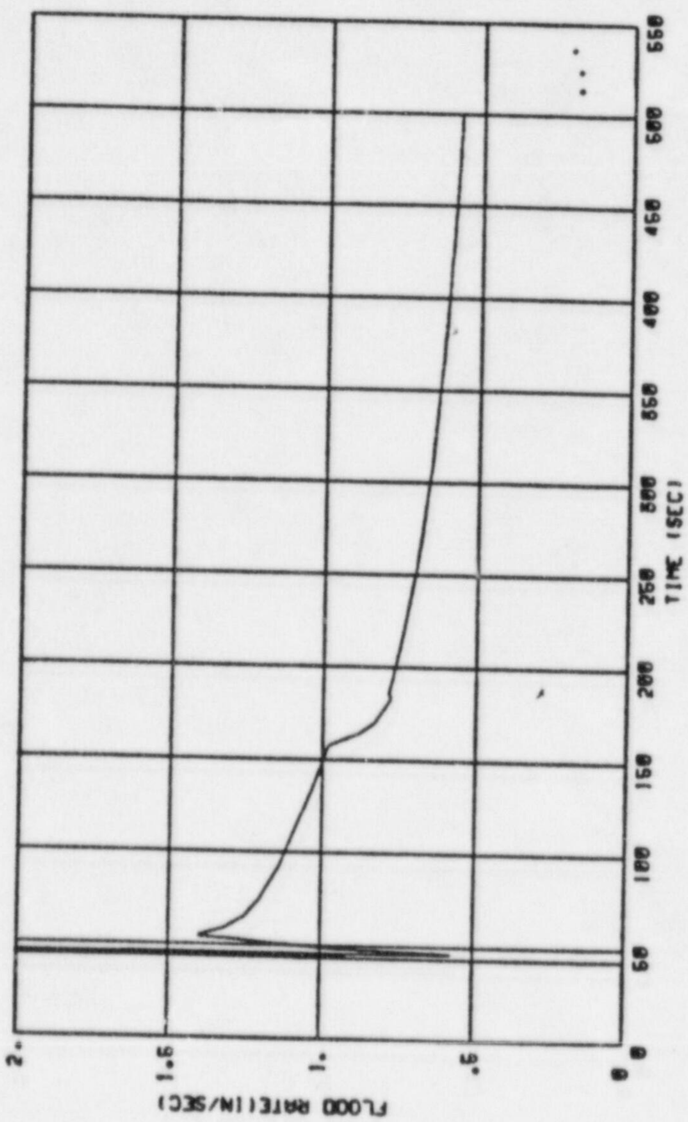


~~SNUPPS~~ Wolf Creek

FIGURE 15.6-9a

DECLG (C_D = 0.6 - MAX SI)

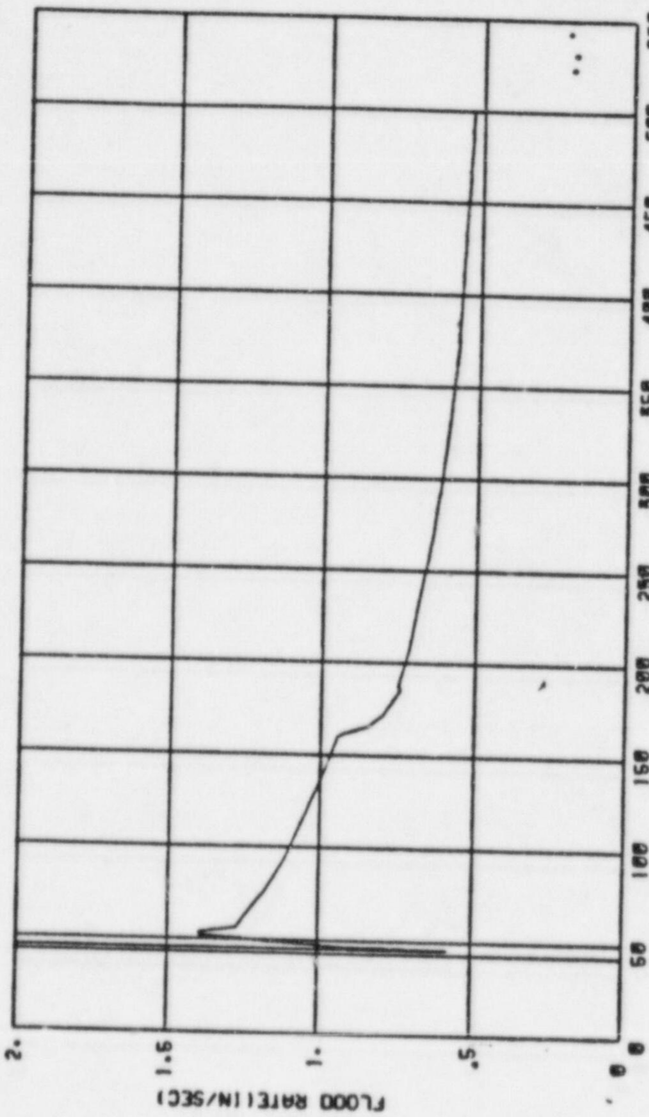
DOWNCOMER AND CORE WATER LEVELS DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-10

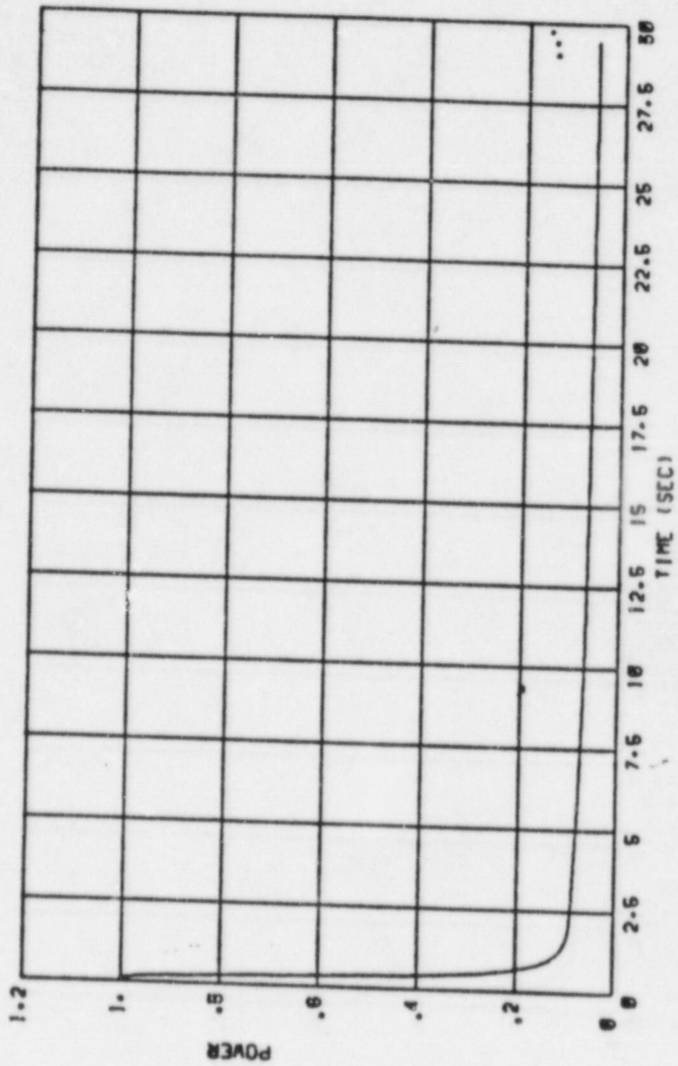
DECLG (C_D = 0.6 - MIN SI)
CORE INLET
VELOCITY DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-10a

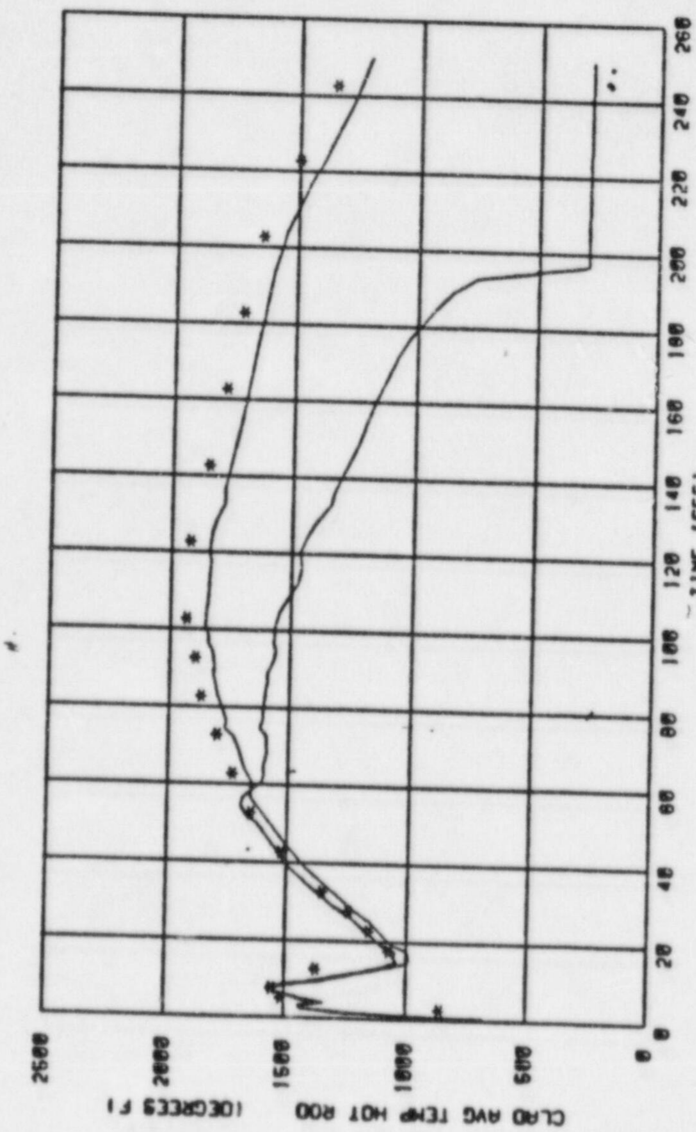
DECLG (C_D = 0.6 - MAX SI)
CORE INLET
VELOCITY DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-11

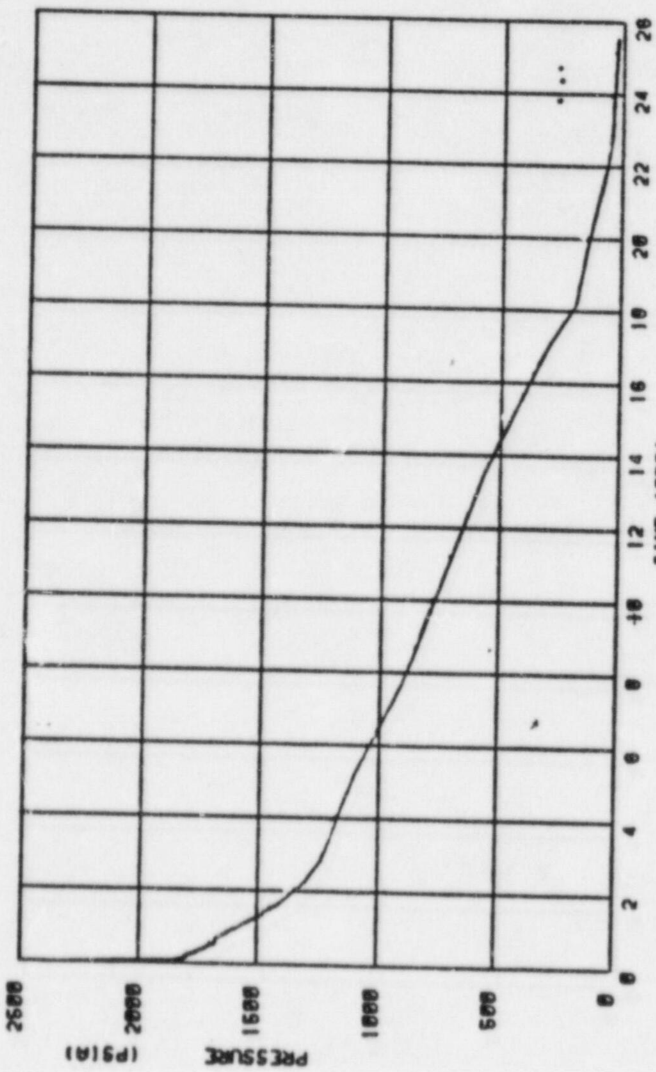
CORE POWER TRANSIENT — DECLG (C_D = 0.6)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-12

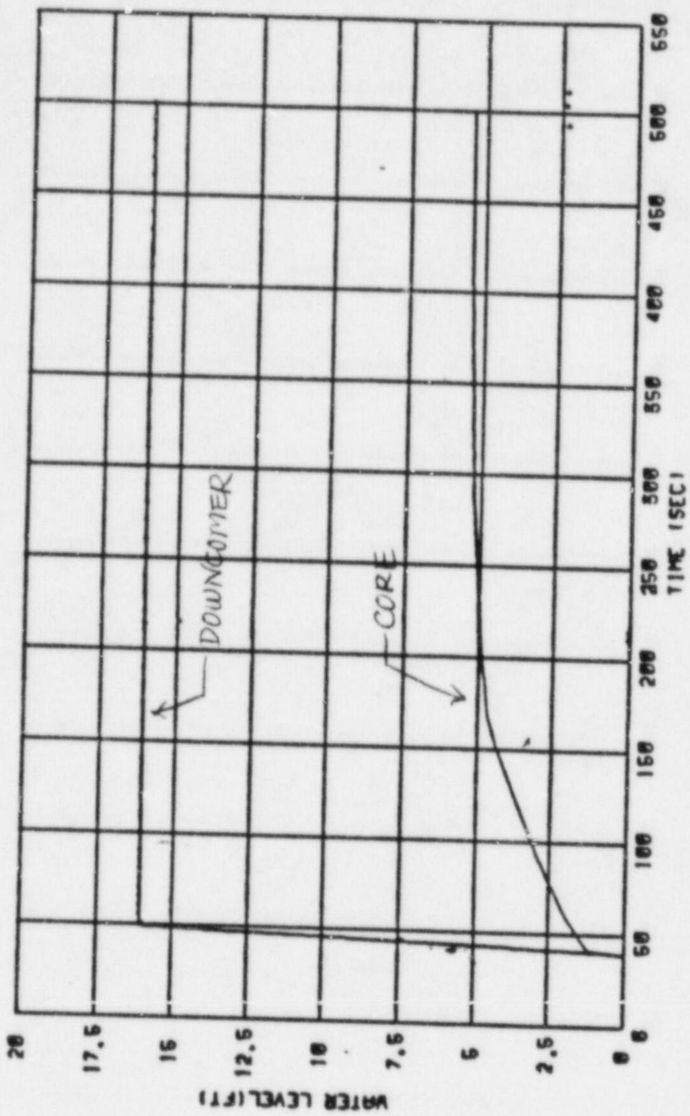
PEAK CLAD TEMPERATURE — DECLG (C_D = 0.8)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-13

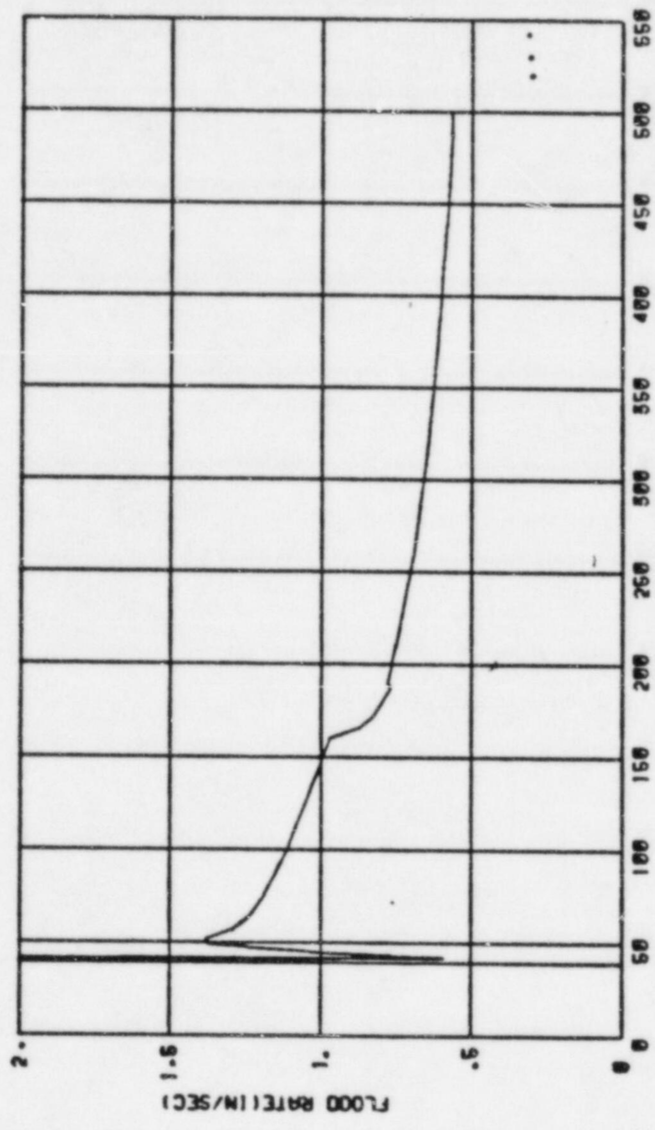
CORE PRESSURE –
DECLG (C_D = 0.8)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-14

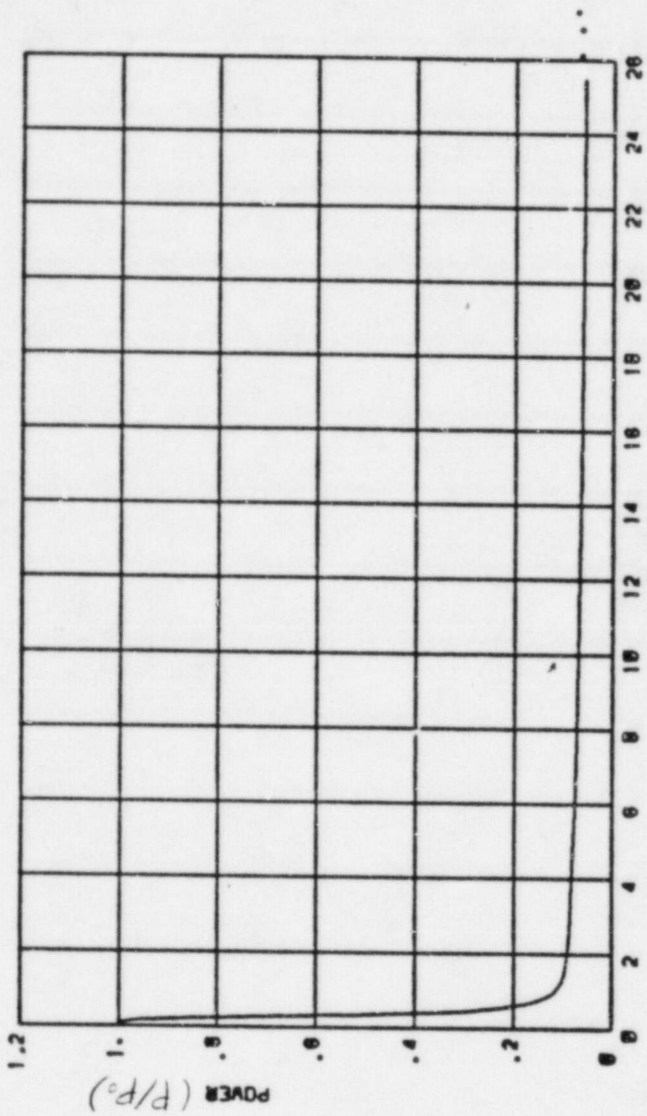
DECLG (C_D = 0.8) DOWNCOMER AND CORE WATER LEVELS DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-15

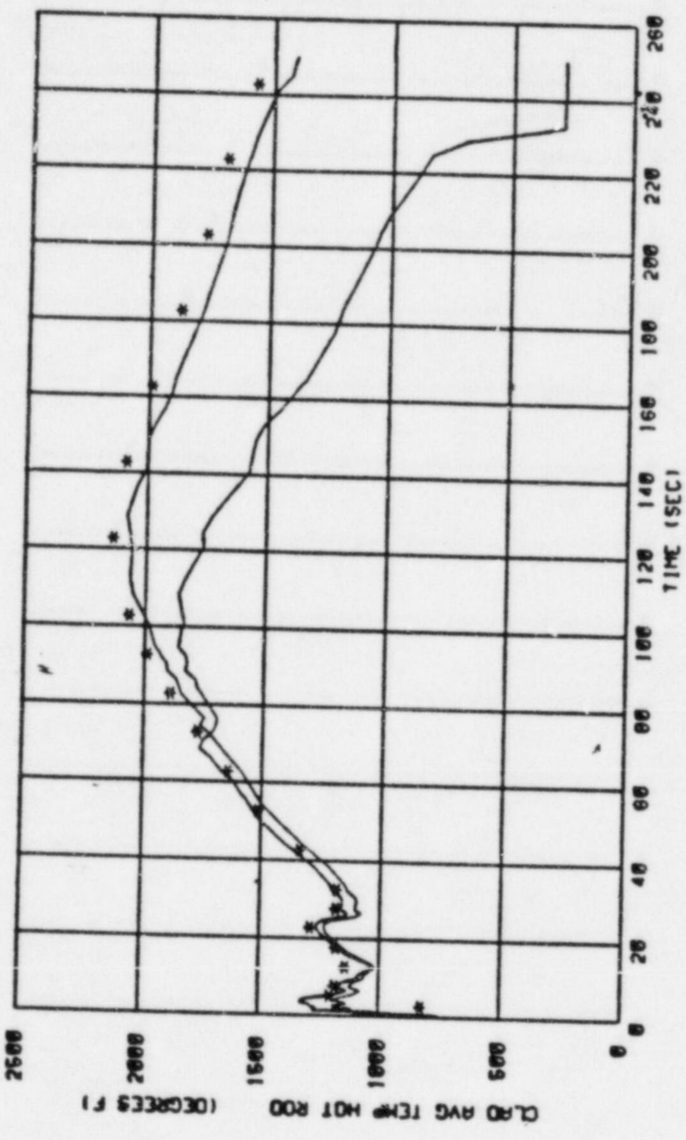
DECLG (C_D = 0.8) CORE INLET VELOCITY DURING REFLOOD



~~**SNUPPS**~~ Wolf Creek

FIGURE 15.6-16

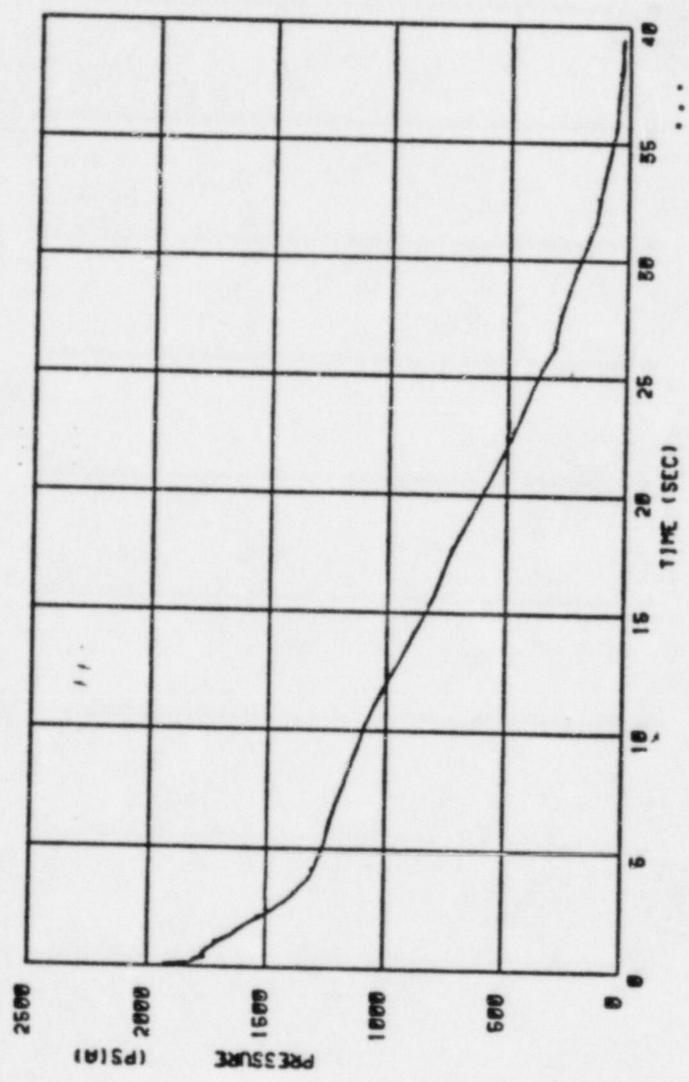
CORE POWER TRANSIENT — DECLG (C_D = 0.8)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-17

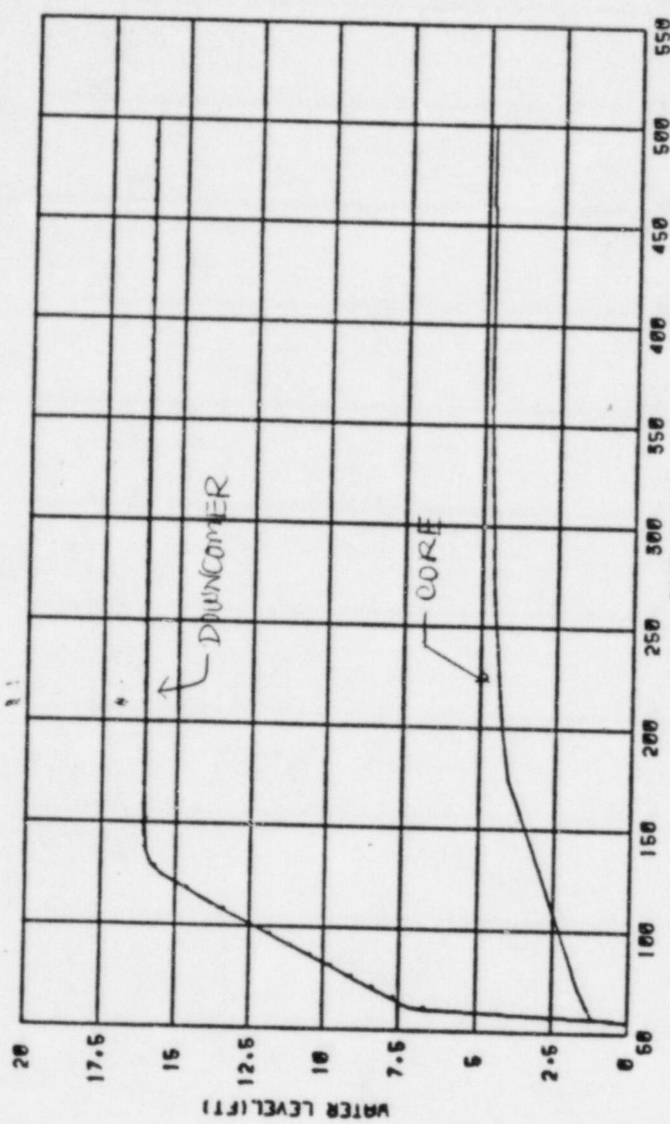
PEAK CLAD TEMPERATURE – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-18

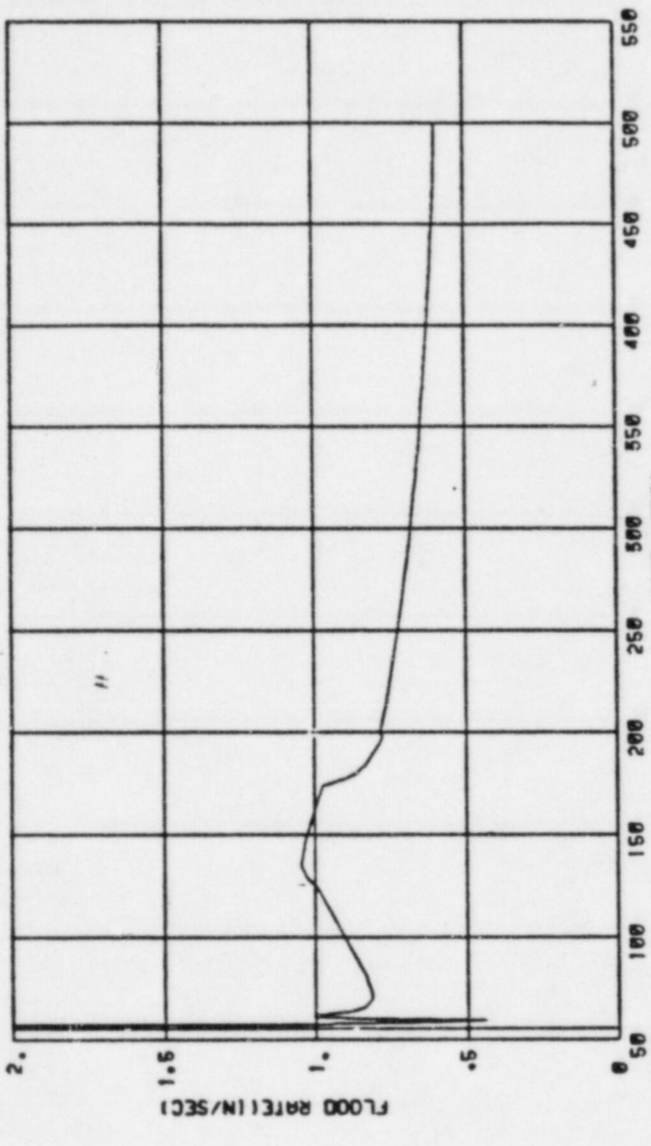
CORE PRESSURE – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-19

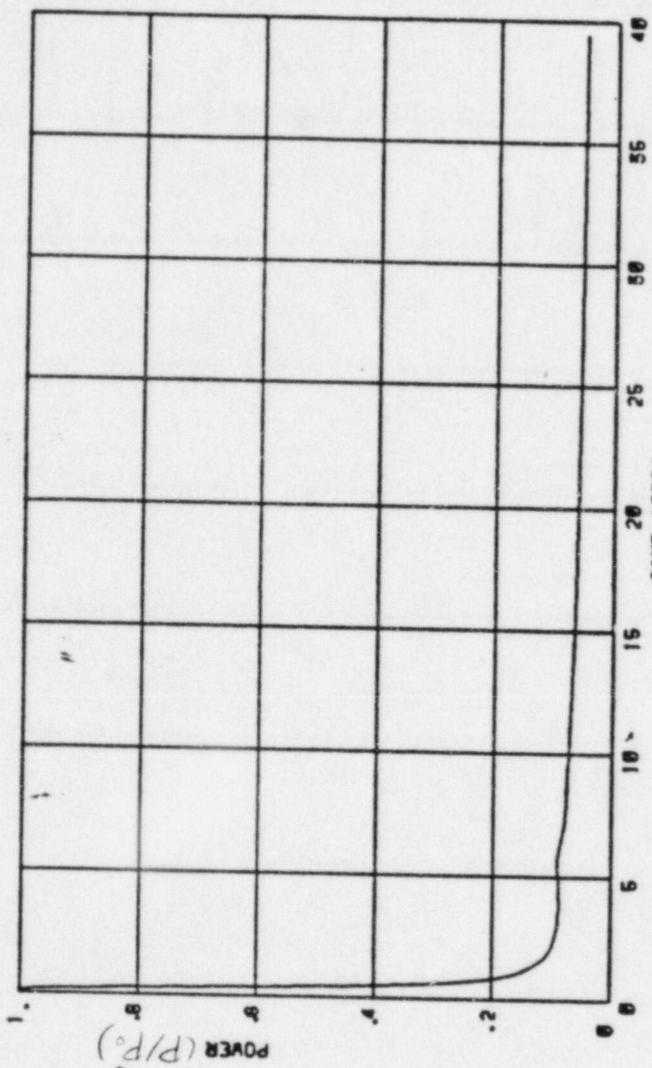
DECLG (C_D = 0.4) DOWNCOMER AND CORE WATER LEVELS DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-20

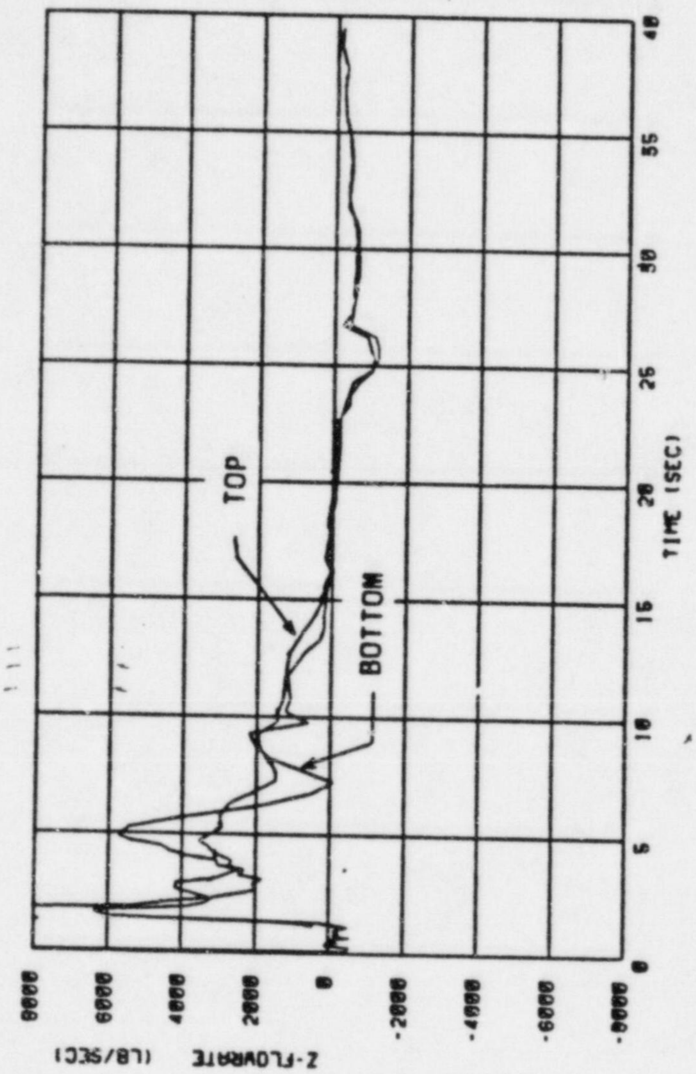
DECLG (C_D = 0.4) CORE INLET VELOCITY DURING REFLOOD



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-21

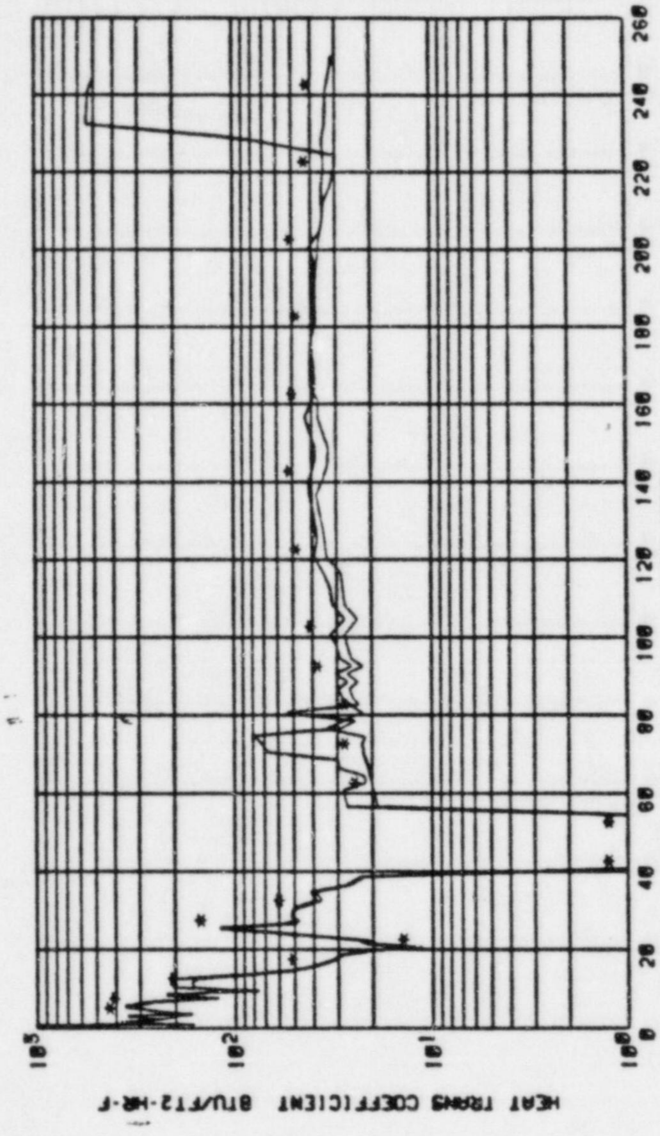
CORE POWER TRANSIENT — DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-22

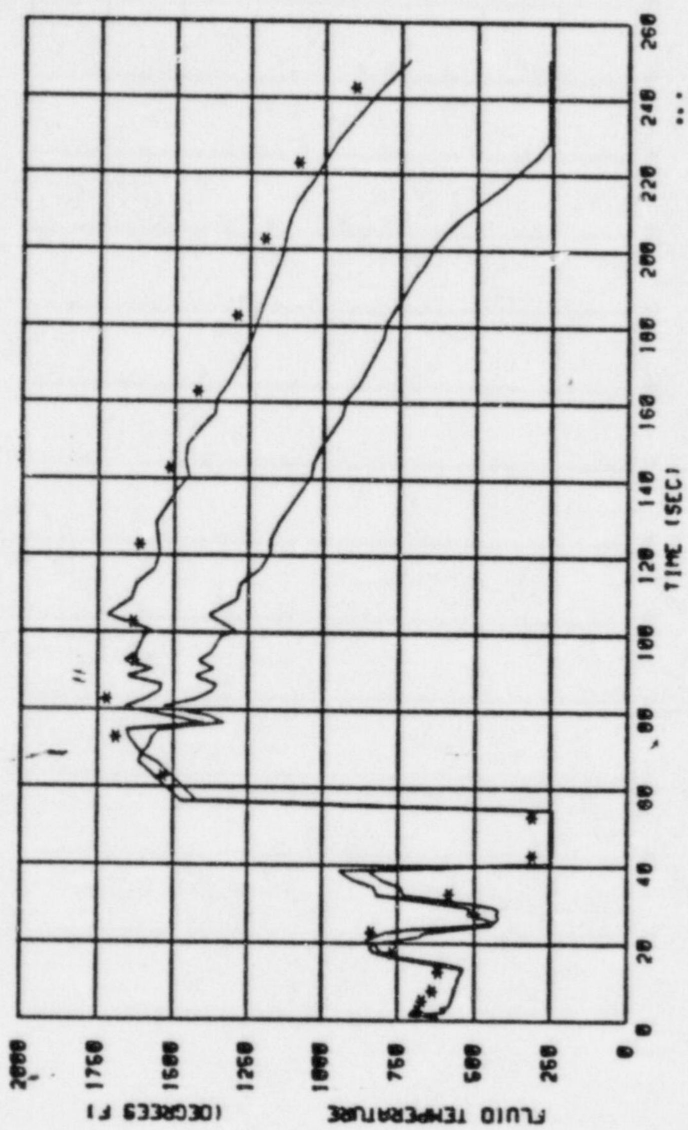
CORE FLOW – TOP AND BOTTOM – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-23

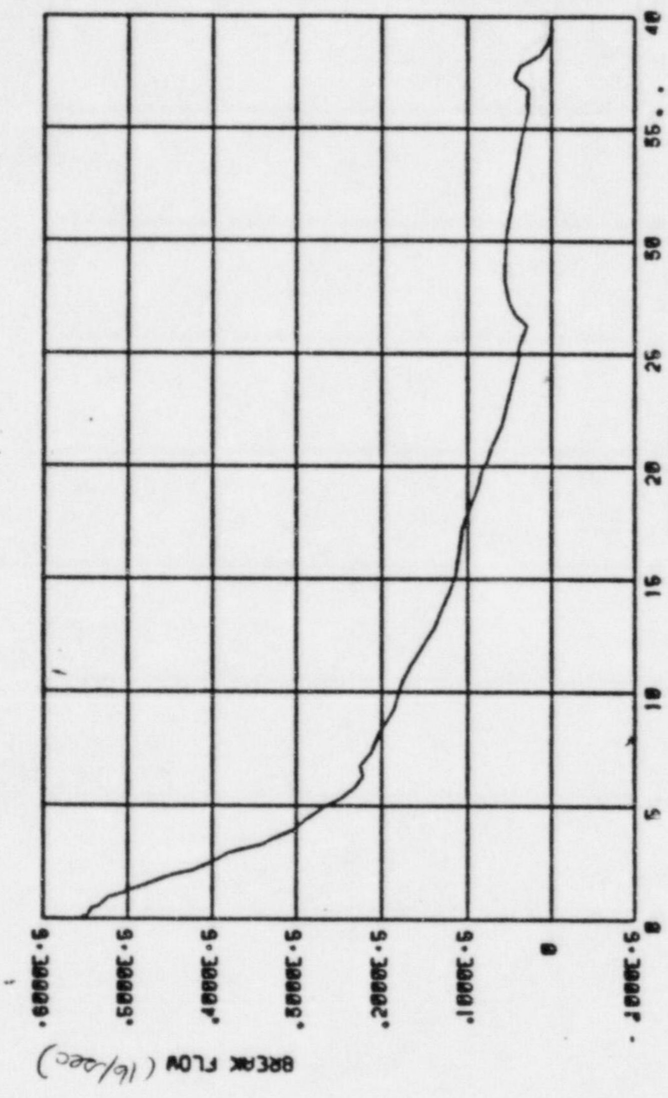
CORE HEAT TRANSFER COEFFICIENT — DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-24

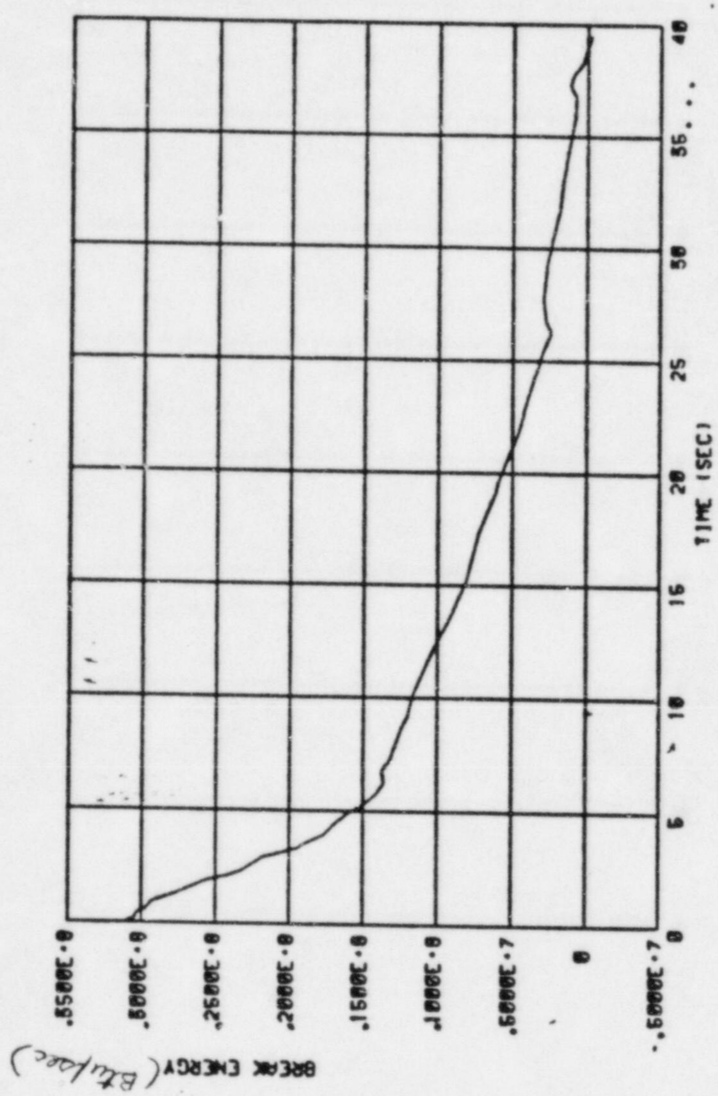
FLUID TEMPERATURE — DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-25

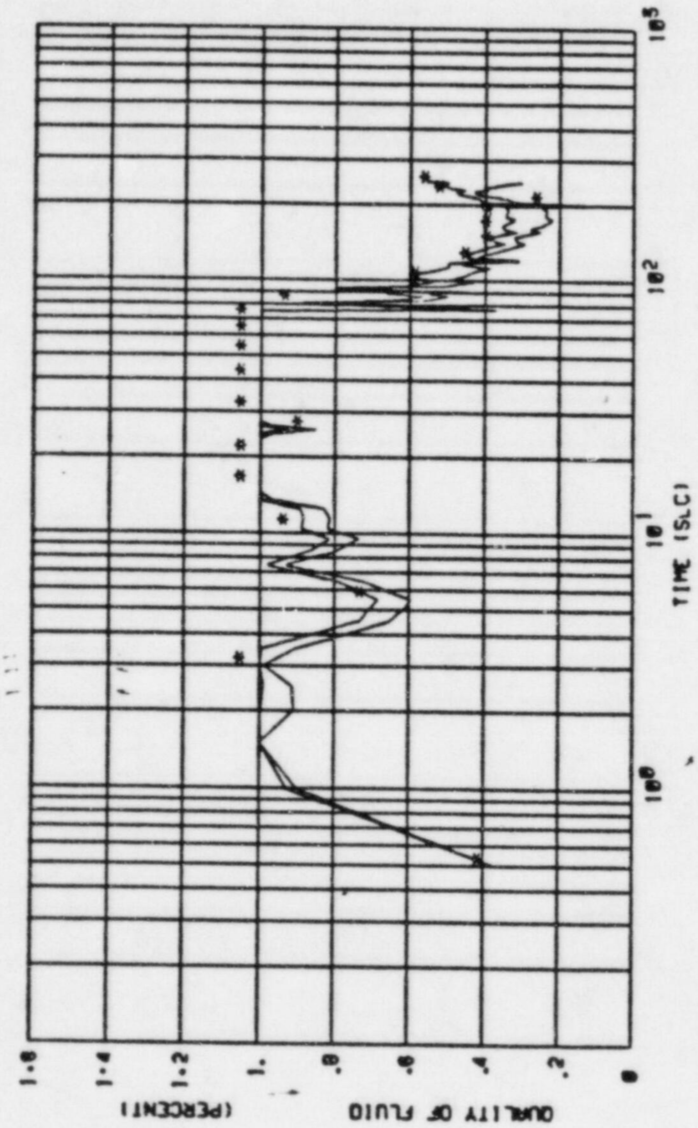
BREAK MASS FLOW RATE – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-26

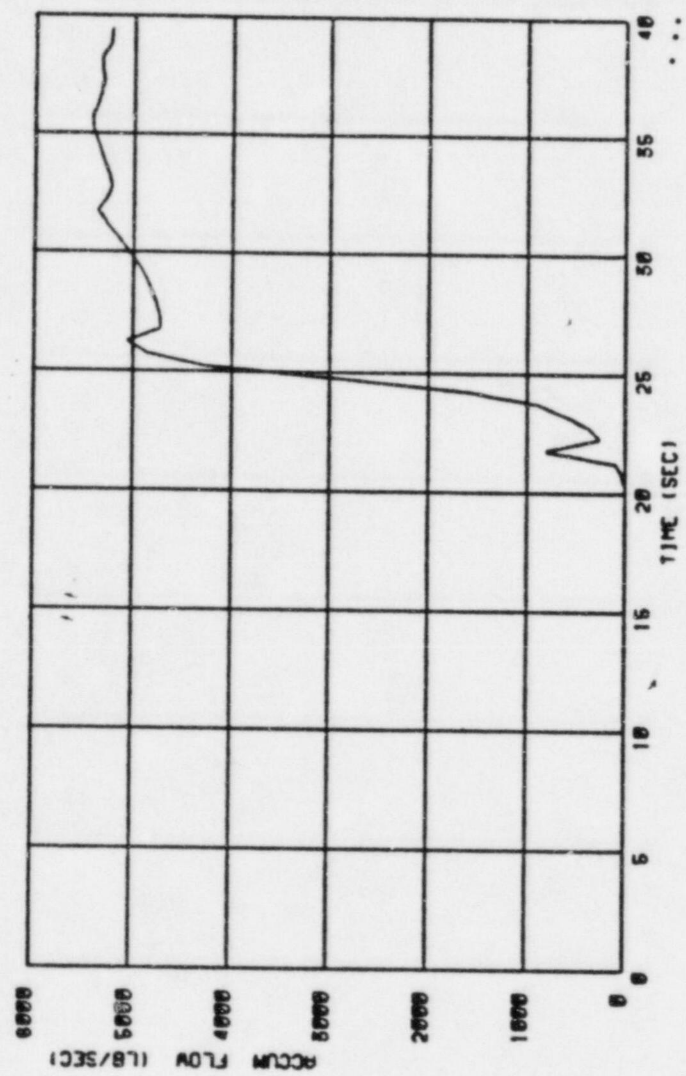
BREAK ENERGY RELEASE RATE – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-27

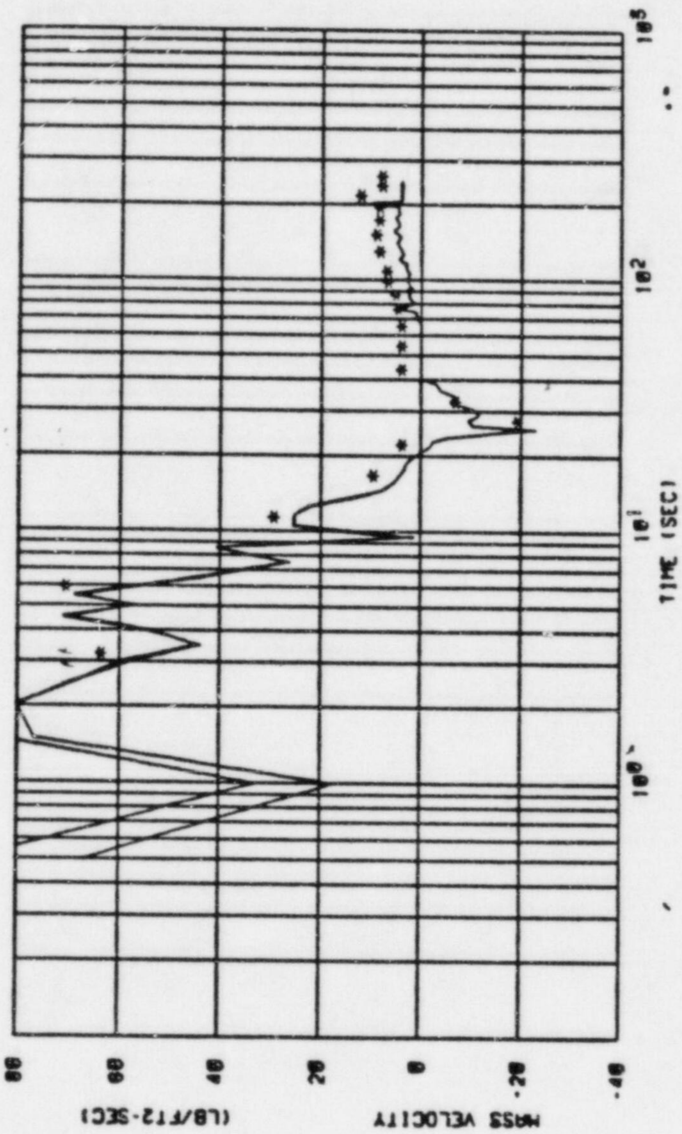
FLUID QUALITY – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-28

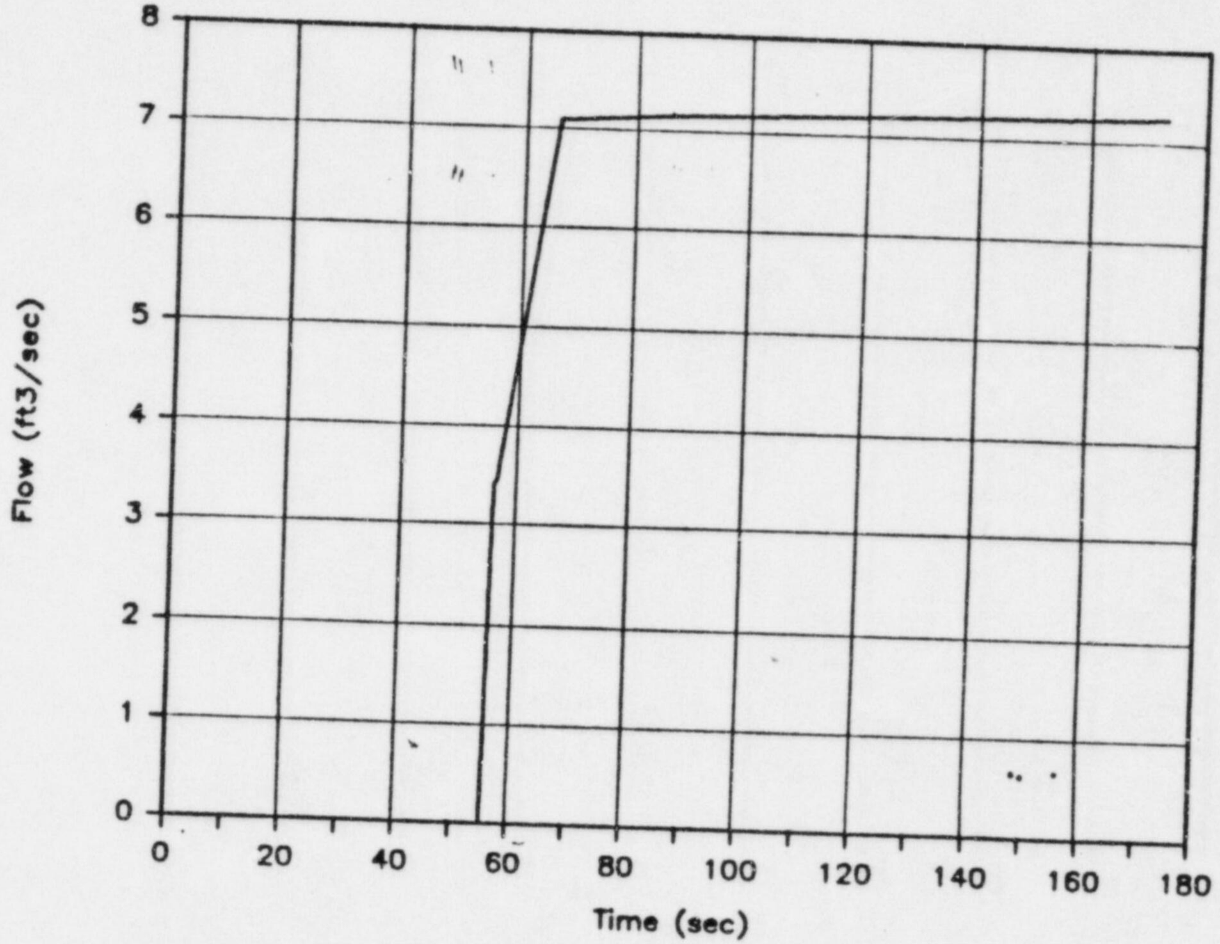
ACCUMULATOR FLOW DURING BLOWDOWN – DECLG (C_D = 0.4)



~~SNUPPS~~ Wolf Creek

FIGURE 15.6-29

MASS VELOCITY – DECLG (C_D = 0.4)



SNUPPS Wolf Creek

FIGURE 15.6-30

PUMPED ECCS FLOW DURING REFLOOD – DECLG (C_D = 0.4)