Docket No. 50-336

Attachment 2

Millstone Nuclear Power Station, Unit No. 2

Steam Generator Tube Rupture Event

April, 1983

Steam Generator Tube Rupture Event

A Steam Generator Tube Rupture (SGTR) analysis has been performed for the Millstone Nuclear Power Station, Unit No. 2. This analysis was performed by using the RETRAN 02 MOD 2 computer code. The plant simulation included modeling of the Reactor Coolant System (RCS), the Steam Generators (SGs), the main steam and feedwater systems, the charging and letdown systems, the High Pressure Safety Injection (HPSI) System, and reactor core kinetics including fuel and moderator temperature feedback. The pressurizer was modeled as a non-equilibrium volume.

The RETRAN model for Millstone Unit No. 2 has successfully simulated plant steady state operation and plant transient response following a reactor trip. In particular, the RETRAN predictions of primary and secondary system parameters such as steam generator pressure rise, RCS pressure and average temperature following a reactor trip trend well with the actual plant response. The Millstone Unit No. 2 RETRAN model has been successfully used in performing several inhouse analyses prior to this effort.

The analyses presented in this report have been performed in order to assess the plant response to a SGTR incident under the conditions in which additional SG U-tubes are plugged. The current licensing analysis, performed for Cycle 3 operation, assumes a total of 1000 plugged U-tubes while this analysis, presented for Cycle 6, assumes 2500 plugged U-tubes. The results of the Cycle 3 analysis, designated as the reference analysis in this report, are shown graphically in Figures 1 through 5 and by the sequence of events listed in Table 1.

In order to demonstrate the applicability of the RETRAN code and to address the comments presented in Generic Letter 83-11, three cases are presented. The initial case (Case 1) uses input parameters and assumptions for plant operation which, as seen in Table 2, are similar to the reference analysis. Case 2 evaluates the effects due to increased U-tube plugging and an increased SG pressure. Case 3 uses initial conditions which are identical to Case 2 except that assumptions made concerning system operation are consistent with the present mode of operation of the plant. Case 3 is presented to support Cycle 6 operation.

Initial conditions and assumptions for all cases are listed in Table 2. The assumptions and results of each case are discussed below.

Thermal Hydraulic Results

The results of Case 1 are presented graphically in Figures 6 through 10 and in the sequence of events listed in Table 1. As seen in Table 1, the times to the low pressure trip, predicted by the reference analysis and Case 1, agree to within thirty (30) seconds. The difference in reactor trip time is due to the fact that the U-tube break flow is predicted to be slightly higher in Case 1 than in the reference analysis, resulting in a more rapid RCS depressurization and an earlier trip.

Differences also exist in the predicted peak SG pressure (978 psia in Case 1 versus 901 psia in the reference analysis) and in the predicted operation of the steam bypass and atmospheric steam dump systems. These differences are attributable to modeling variances between Case 1 and the reference analysis. The methodology utilized in the Case 1 analysis is supported by the fact that the predicted SG pressure rise following reactor trip is consistent with the pressure rise observed in actual post-trip plant operation.

Two sensitivity studies, not shown in this report, were performed in support of Case 1 to evaluate the post-trip plant performance predicted by RETRAN. The effects of the High Pressure Safety Injection (HPSI) system and the Main Feedwater (MFW) system were investigated in these studies. The first study showed that the operation of the HPSI system (HPSI was not assumed in Case 1) caused the RCS to pressurize to approximately 1170 psia, unlike the results of Case 1 or the reference analysis. The second study, which assumed the post trip MFW flow to be at 132°F and 5% of full power flow, had little effect on the hydraulic transient and did not cause secondary system depressurization and RCS cooldown predicted by the reference analysis. These results are reasonable and further support the analysis performed by NNECO.

Assumptions and initial conditions used in Case 1 are summarized in Table 2. The results of this case serve as a basis of comparison for the subseqent cases of this analysis and demonstrate the applicability of the NNECO model of Millstone Unit No. 2 to SGTR licensing analyses.

Case 2 is performed with assumptions changed in accordance with the parameters listed in Table 3. In Case 2, a total of 2500 SG U-Tubes are assumed to be plugged and the RCS flow rate is correspondingly reduced. Additionally, a higher initial SG pressure is assumed which reflects the upper bound of actual plant operating conditions. Major input assumptions for Case 2 are summarized in Table 2. All other assumptions in Case 2 are identical to those of Case 1.

As shown in Table 3, the maximum and minimum SG pressures (including uncertainties) are 933 psia and 843 psia, respectively. A sensitivity study was performed in support of Case 2 using these bounding conditions for SG pressure. As expected, the U-tube break flow rate at 933 psia was less than that for an initial SG pressure of 843 psia. However, the time to reactor trip was increased by approximately 45 seconds for the higher initial pressure conditions. As a result, the total loss of primary inventory prior to reactor trip was essentially equal for these two conditions. The most important result obtained from this sensitivity study was that a much larger amount of steam dump flow to the atmosphere was predicted when the higher steam generator pressure was assumed. Also, the SG safety valves were predicted to lift following reactor trip under the higher pressure conditions. As a result, the higher initial SG pressure produces more limiting results with respect to offsite dose.

The results of Case 2 are shown graphically in Figures 11 through 15 and in the sequence of events presented in Table 1. The effects due to increased U-tube

plugging, reduced RCS flow rate and increased SG pressures are found by comparing Cases 1 and 2. As shown in Table 1, reactor trip occurs 36-seconds later in Case 2 than in Case 1. This effect is due to the higher core exit coolant temperature, caused by the decreased core flow, and the higher SG pressure. Both effects decrease the break flow rate. Additionally, the SG safety valves are predicted to lift for a short period in Case 2 due to the higher initial SG pressure.

Case 3, presented to support Cycle 6 operation, uses input parameters which are identical to Case 2, as seen in Table 2. However, this case considers the effects of the Auxiliary Feedwater (AFW) system and the HPSI system operation which are consistent with the present mode of operation of the plant. The results of Case 3 are shown in Figures 16 through 20 and the sequence of events is listed in Table 1.

The present Millstone Unit No. 2 emergency procedures require the operator to trip all reactor coolant pumps (RCPs) following the safety injection actuation signal (SIAS) at 1600 psia. At the outset of the SGTR analysis it was not obvious whether the radiological releases would be increased or decreased due to this operator action. A sensitivity study was performed to evaluate the effects of RCPs running and RCPs tripped following SIAS. These results showed that a SGTR incident with RCPs running produces the more limiting radiological consequences. In this sensitivity study, the steam generator safety valves were predicated to lift for equal time periods in both cases. However, the atmospheric dump valves were predicted to close earlier when the RCPs were tripped than when the RCPs were running. The reason for this is that the coolant flow rate through the SGs decreases when the RCPs are tripped. As a result, less heat is transferred temporarily from the primary system to the secondary side of the SGs than when the RCPs are running. Since the steam bypass valves are open during this pump coastdown period and the heat added to the secondary side is less, the SG pressure decreases to a lower value than when the RCPs remain in operation. This causes earlier closure of the atmospheric steam dump valves and a smaller radiological release.

In Case 3, operation of the HPSI system causes the RCS to repressurize to approximately 1185 psia, as shown in Figure 17. As a result of this repressurization, the U-tube break flow is increased, as shown in Figure 20. The delivery of AFW flow (which includes a conservatively long delay time of 240 seconds) removes additional heat from the RCS and results in a decrease in the steam bypass flow to the condenser. The dilution of SG iodine concentration by AFW has a small effect on the offsite dose because the atmospheric steam dump and SG safety/relief valves (the major release path) are predicted to reseat prior to AFW initiation.

Operator actions are assumed to occur 30 minutes following the initiation of the SGTR event. These actions result in isolation of the affected SG to reestablish containment integrity and depressurization of the RCS to terminate primary to secondary leakage.

Radiological Results

The radiological consequences were analysed conservatively in accordance with Standard Review Plan (SRP) 15.6.3. Additional guidance was obtained from NUREG-0409, "Iodine Behavior in PWR Cooling System Following a Postulated Steam Generator Rupture Accident" and NUREG/CR-2683, "Iodine Behavior in Steam Generator Tube Rupture Accidents."

The primary and secondary coolant systems were modeled to describe the coolant mass balance and iodine activity balance. Radioactive decay was conservatively excluded. The primary coolant system modeling included letdown flow, charging flow, HPSI flow (Case 3) and primary to secondary leakage (both normal and U-tube break). Regarding the effects of iodine spiking, the spike effects were modeled as described in SRP 15.6.3. Secondary coolant system modeling included main and auxiliary feedwater flow, steam flow, liquid carry-over flow, blowdown flow and primary to secondary leakage. Iodine activity transport in the steam generators was modeled in accordance with NUREG-0409 and NUREG/CR-2683.

The main steam system included case specific modeling of safety/relief valve, atmospheric steam dump and main steam dump and bypass to the condenser flows. The safety/relief valve and atmospheric steam dump flows were taken to be ground level releases. The main steam and bypass system flow was taken to be an elevated release via the condenser/steam jet air ejector and then the Unit 1 stack.

The dose due to the plant cooldown following the initiation of operator actions at 30 minutes into the transient has been included. Continued Technical Specification maximum primary to secondary leakage was accounted for in the intact steam generator as well as that activity already contained within the steam generator secondary coolant.

The doses were computed based upon the 5 percent least favorable meteorological conditions over a five year period with Regulatory Guide 1.109 Rev. 1 dose conversion factors. Other important assumptions are listed in Table 4.

The two (2) hour Exclusion Area Boundary (EAB) thyroid dose calculated for Case 1 is 0.118 Rem. This dose is provided for information only and as a basis of comparison for Cases 2 and 3. Case 2 results in a two hour EAB thyroid dose of 0.238 Rem. This value indicates that the radiological consequences are not significantly increased by the additional plugged U-tubes and the higher SG pressure. The predicted increase in dose is due primarily to the SG safety valve flow and the increase in atmospheric steam dump flow. Case 3 results in a predicted two hour EAB dose of 0.239 Rem. Although more U-tube break flow occurred for this case, as compared to Case 2, the predicted doses were essentially equal due to the nearly equal SG safety valve and atmospheric steam dump flows.

Summary and Conclusions

A reanalysis of the SGTR incident was performed to support Cycle 6 operation of Millstone Unit No. 2. The reanalysis was performed assuming additional plugged steam generator U-tubes and the resulting effects of the plugging on plant system parameters.

Case 1 and the associated sensitivity studies demonstrate the applicability of the NNECO model for Millstone Unit No. 2 for SGTR analyses. The difference in radiological doses predicted by Case 1 and the reference analysis are understood. The methodology utilized in the Case 1 analysis is consistent with post-trip operational experience.

Case 2 evaluated the plant response due to a SGTR with additional plugged steam generator U-tubes and an increased steam generator secondary side pressure. The increased steam generator pressure was the dominating parameter causing a larger atmospheric release via the steam generator safety valves and atmospheric dump valves.

Case 3 assessed the SGTR incident assuming limiting conditions for system operation and operator action (i.e., HPSI and AFW operation) as well as the input conditions described for Case 2. The results of this case are presented to support Cycle 6 operation. The offsite dose predicted for Case 3 was 0.239 Rem, well within the limits of 10CFR100.

TABLE 1

SEQUENCE OF EVENTS FOR SGTR CASES
(time in seconds)

EVENT	CYCLE 3 REFERENCE ANALYSIS	CASE 1	CASE 2	CASE 3
Tube Rupture Occurs	0	0	0	0
Low Pressure Trip Condition (1728 psia)	825.2	796.7	833.2	833.2
CEAs Begin Dropping Into Core	826.6	797.6	834.1	834.1
Pressurizer Empties	825.5	804.0	840.0	840.0
Bypass Valves Begin Opening	825.6	800.0	836.0	836.0
Atmospheric Dump Valves Begin Opening	NA	802.0	836.0	836.0
SG Safety Valves Lift	NA	NA	836.0	836.0
Maximum SG Pressure -Time -(psia)	839.0 (901)	806.0 (978)	840.0 (1006)	840.0 (1006)
SG Safety Valves Reseat	NA	NA	844.0	844.0
Bypass Valves Close	913.2	Cycling	Cycling	Cycling
Atmospheric Dump Valves Close	NA	828.0	872.0	872.0
AFW Flow Begins	NA	NA	NA	1148.0
Pressurizer Begins Refilling	NA	NA	NA	1126.0
Emergency Operator Actions Begin	1800	1800	1800	1800

TABLE 2

KEY PARAMETERS AND
ASSUMPTIONS IN SGTR CASES

	CYCLE 3 REFERENCE ANALYSIS	CASE 1	CASE 2	CASE 3
Core Power Level (MW_{th})	2754	2754	2754	2754
Primary Pressure (PSIA)	2300	2300	2300	2300
Core Inlet Temperature (°F)	551	551	551	551
Primary Flow Rate (GPM)	370000	370000	350000	350000
Steam Generator Pressure (PSIA)	860	860	933	933
# Plugged U-Tubes-SG1/SG2	500/500	500/500	1300/1200	1300/1200
Kinetics Conditions	Cycle 3	Cycle 6	Cycle 6	Cycle 6

Note:
- AC available in all cases
- Reactor trip on low pressure (1728 PSIA) in all cases
- Double ended rupture of one U-tube
- RCPs on throughout transient for all cases

TABLE 3

Steam Generator Physical Parameters

	SG #1	SG #2
Number of Tubes:	8519	8519
Number of Plugs:	1300	1200
Heat Transfer Area (ft^2)	75736	76713
Primary Flow Area (ft^2)	16.84	17.07
Total Primary Volume (ft^3) (including tube sheet area)	960.8	973.3

Operating Conditions

	Nominal	Uncertainty/Range
Inlet Temperature (Primary, °F)	549	± 2
Power Level (MWth)	2700	$\pm 2\%$
Primary Flow Rate (GPM)	375,000	---
Minimum Guaranteed Primary Flow Rate (GPM)	---	350,000
Primary Pressure (PSIA)	2250	2200-2300
Steam Generator Pressure* (PSIA)	888	± 35
Feedwater Temperature (°F)	432	± 20

*This value represents the average value of the two generators. Generator #1 runs about 10 psi above the average and generator #2 runs about 10 psi below the average. Thus, the minimum pressure with uncertainties for generator #2 is 843 psia.

Table 4

Millstone Unit No. 2 SGTR Event

Radiological Assumptions

1.	Initial Reactor Coolant System Iodine Concentration due to pre-event iodine Spike	=	60 uCi/gm DE I-131
2.	Initial Steam Generator Secondary Side Iodine Concentration	=	0.1 uCi/gm DE I-131
3.	HPSI Cooling Flow Iodine Concentration (Case 3 only)	=	1.00×10^{-4} uCi/gm DE I-131
4.	Main/Auxiliary Feedwater System Iodine Concentration (Case 3 only)	=	1.78×10^{-3} uCi/gm DE I-131
5.	Steam Generator Iodine Partition Factor	=	.01
6.	Condenser/Steam Jet Air Ejector Iodine Partition and Decontamination Factor	=	.0005
7.	Carry Over Fraction in Steam Generators	=	0.0025
8.	Breathing Rate	=	3.47×10^{-4} m^3/sec
9.	Atmospheric Dispersion Coefficients:		
	EAB (0-1hr) Ground Level Release	=	5.41×10^{-4} sec/m^3
	EAB (fumigation) Elevated Release	=	1.03×10^{-4} sec/m^3

Figure 1

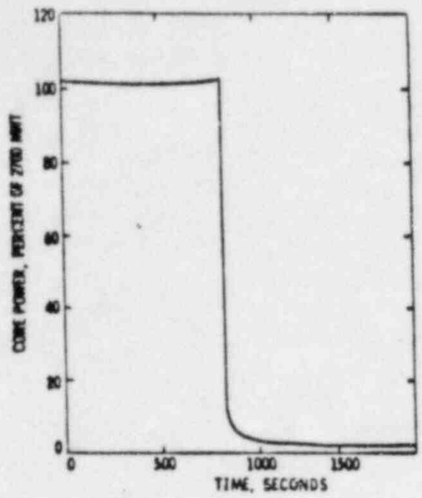


Figure 2

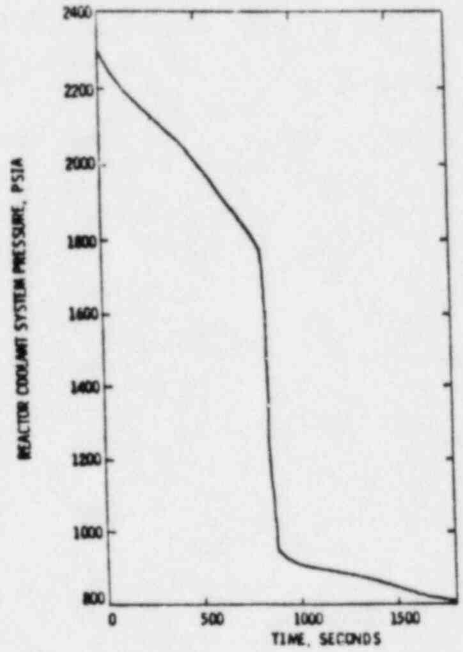


Figure 3

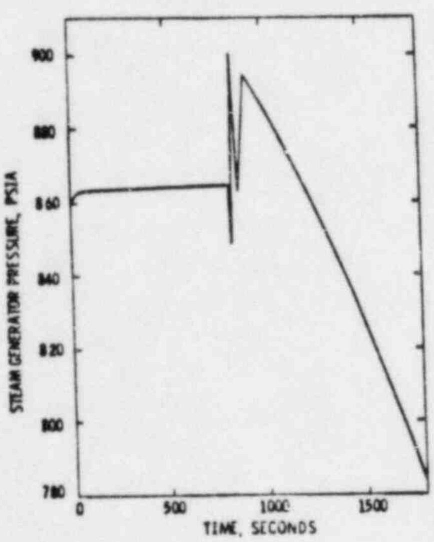
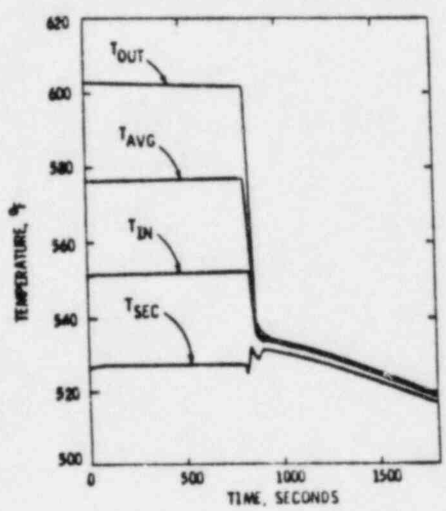


Figure 4



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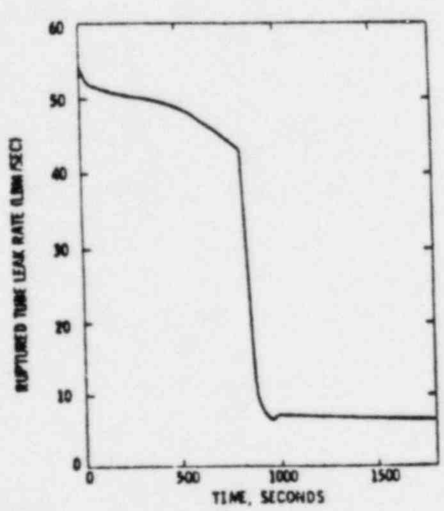

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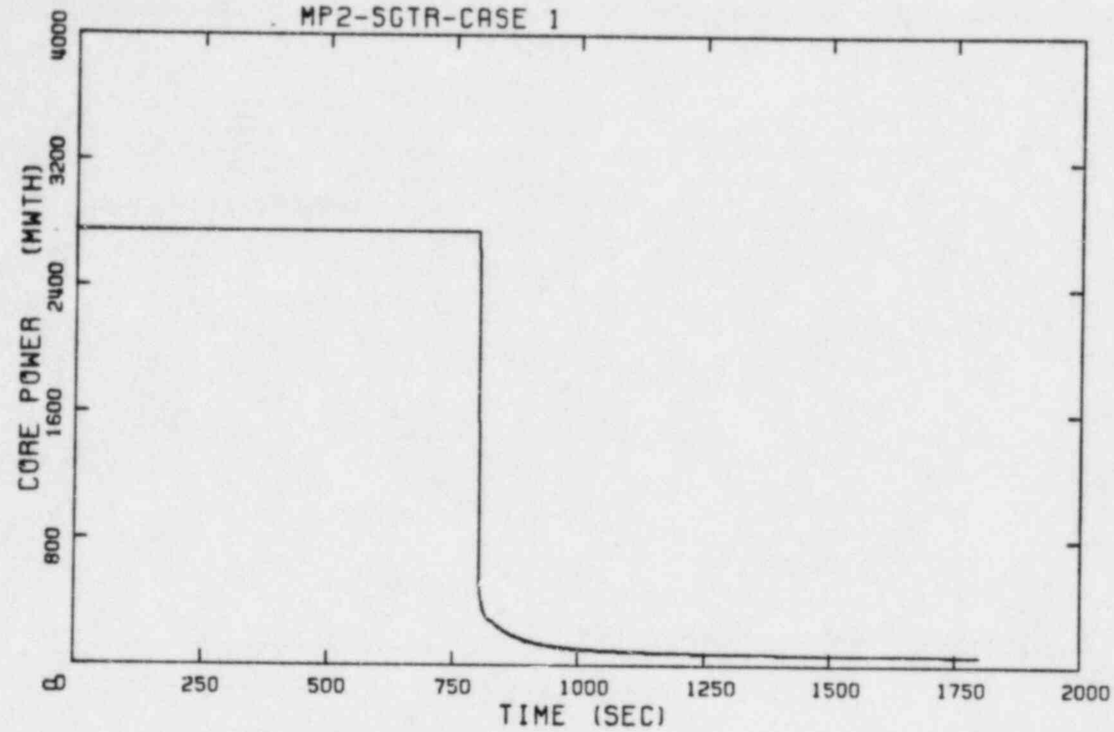

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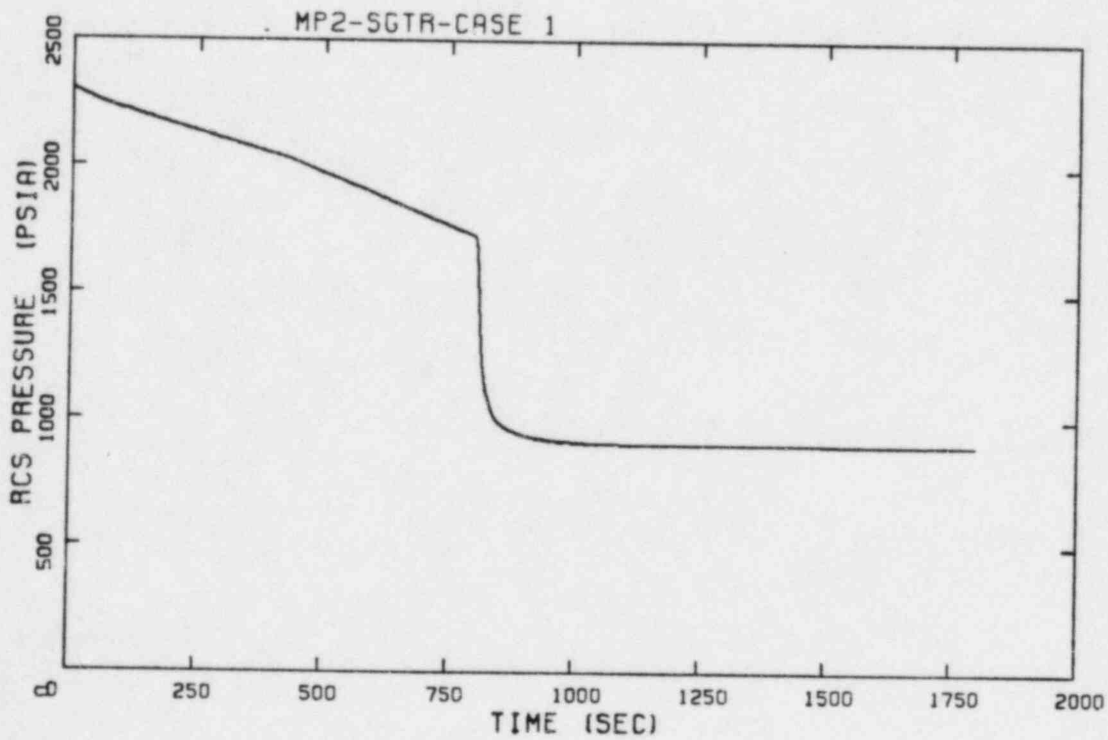

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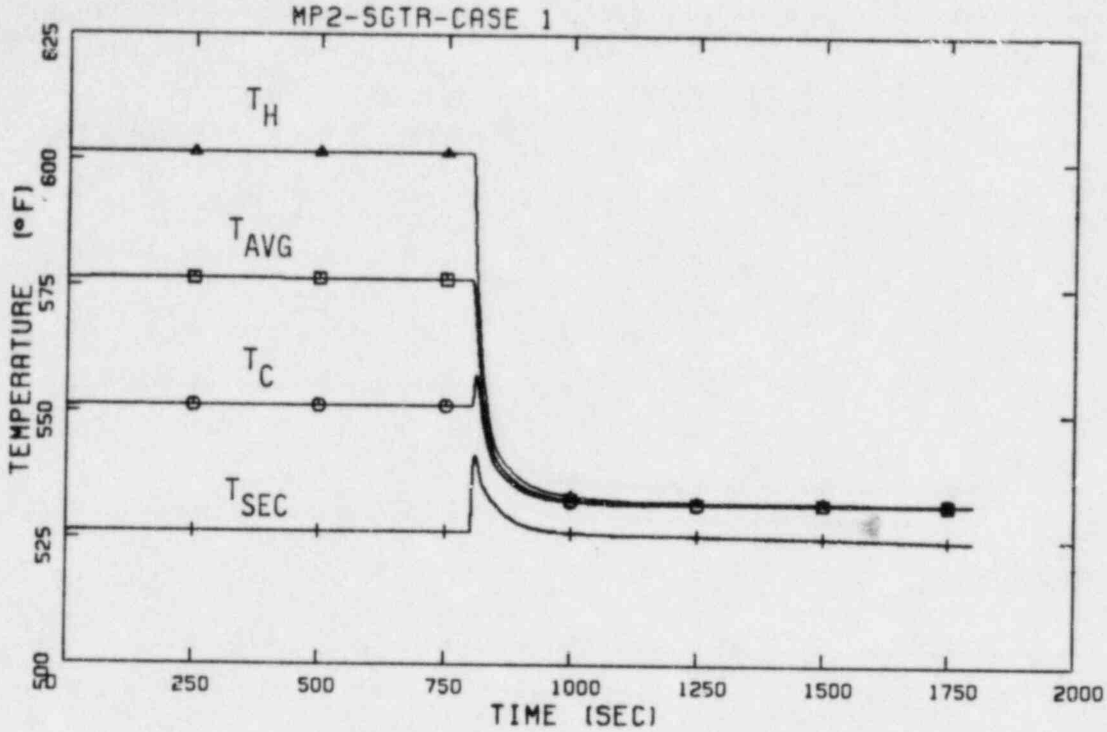

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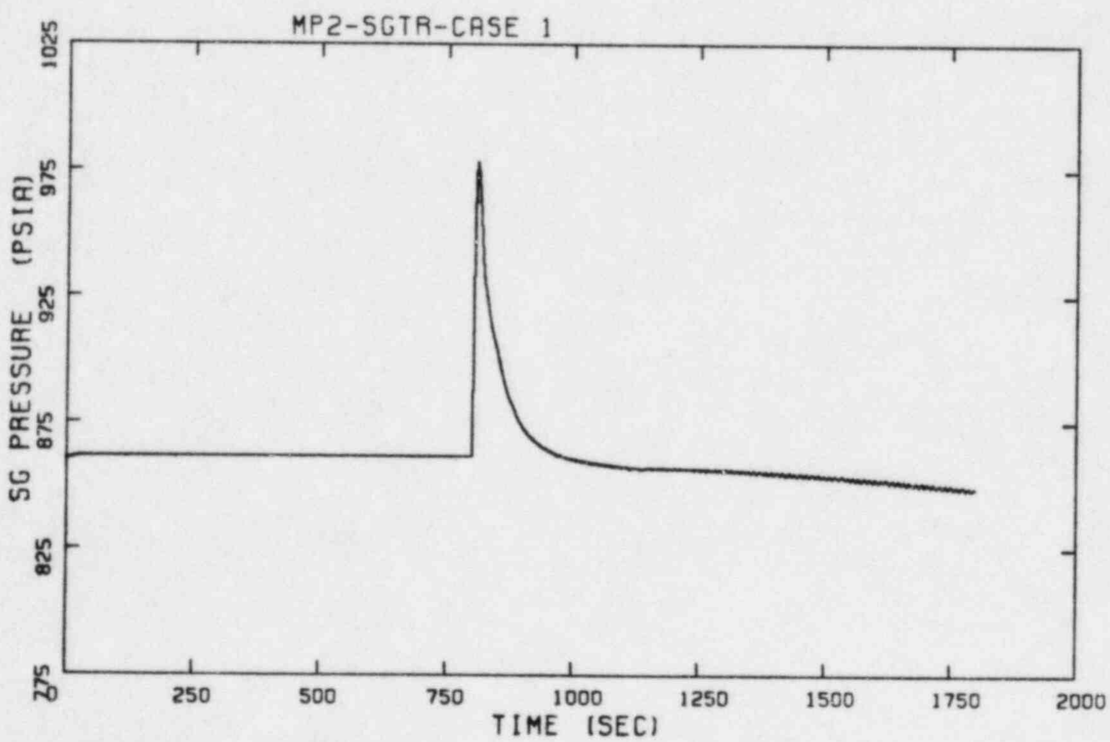
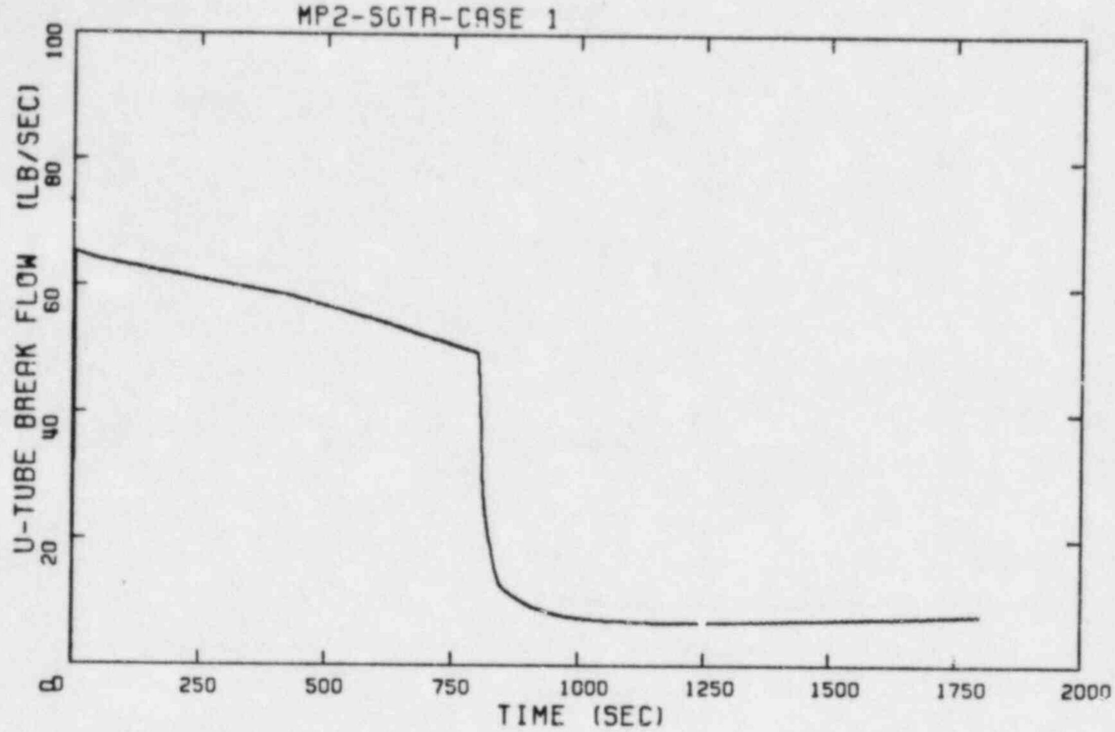


FIGURE 10



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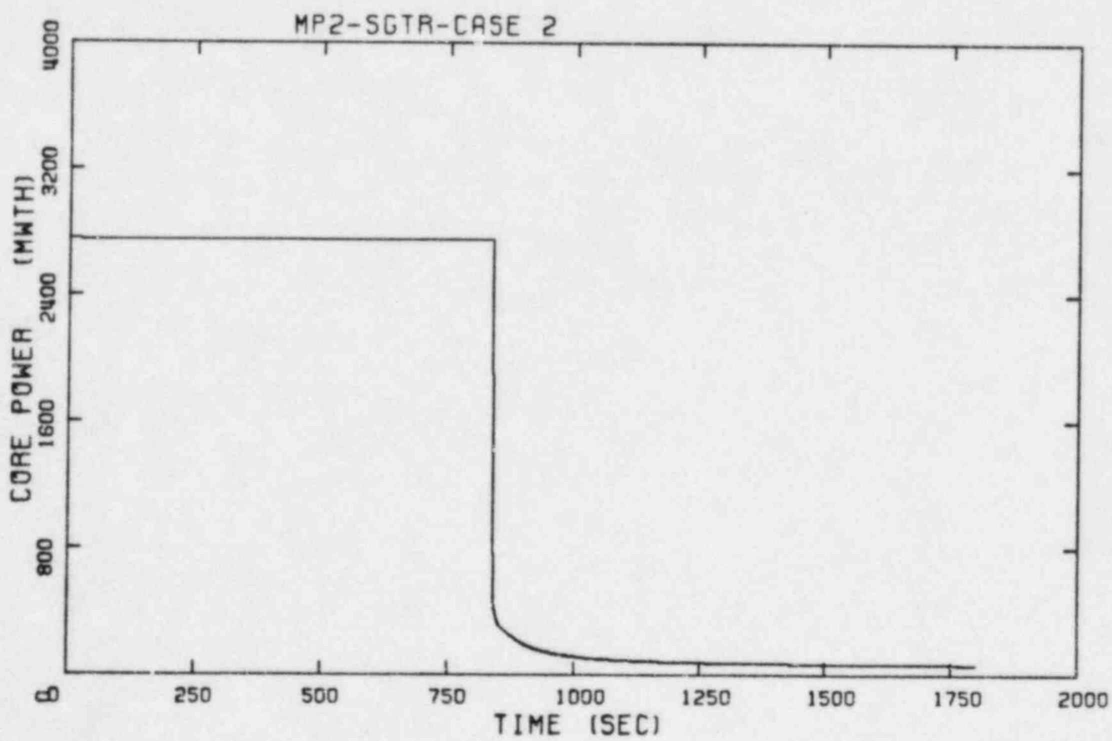

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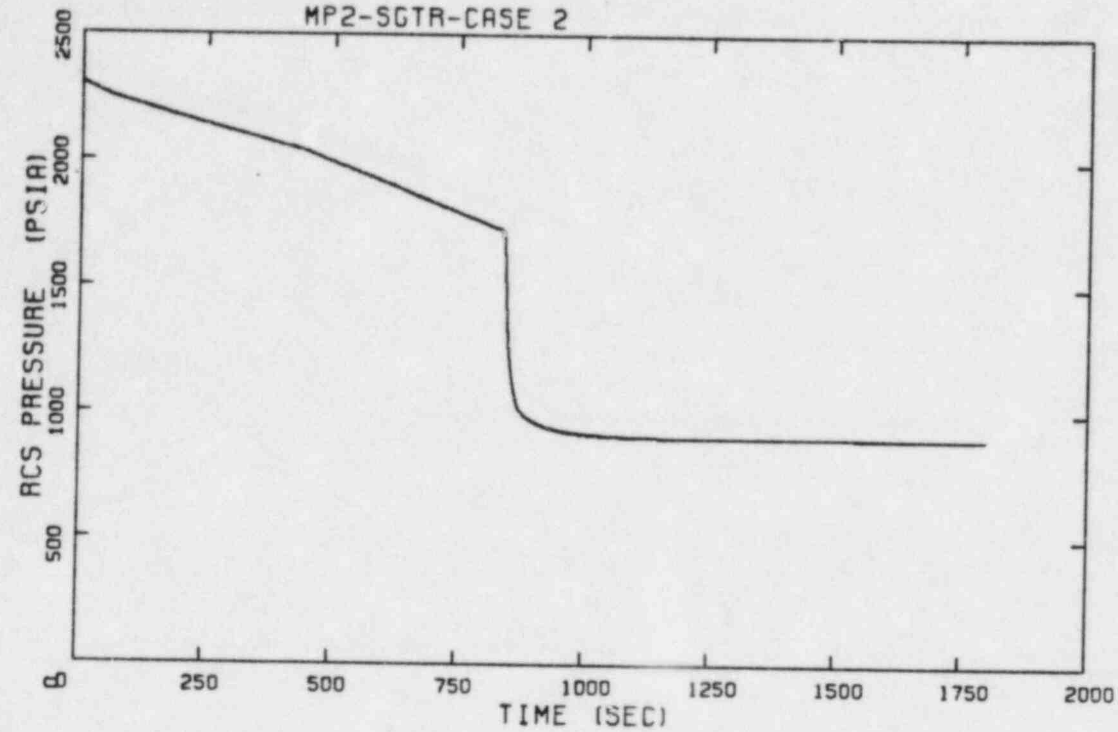

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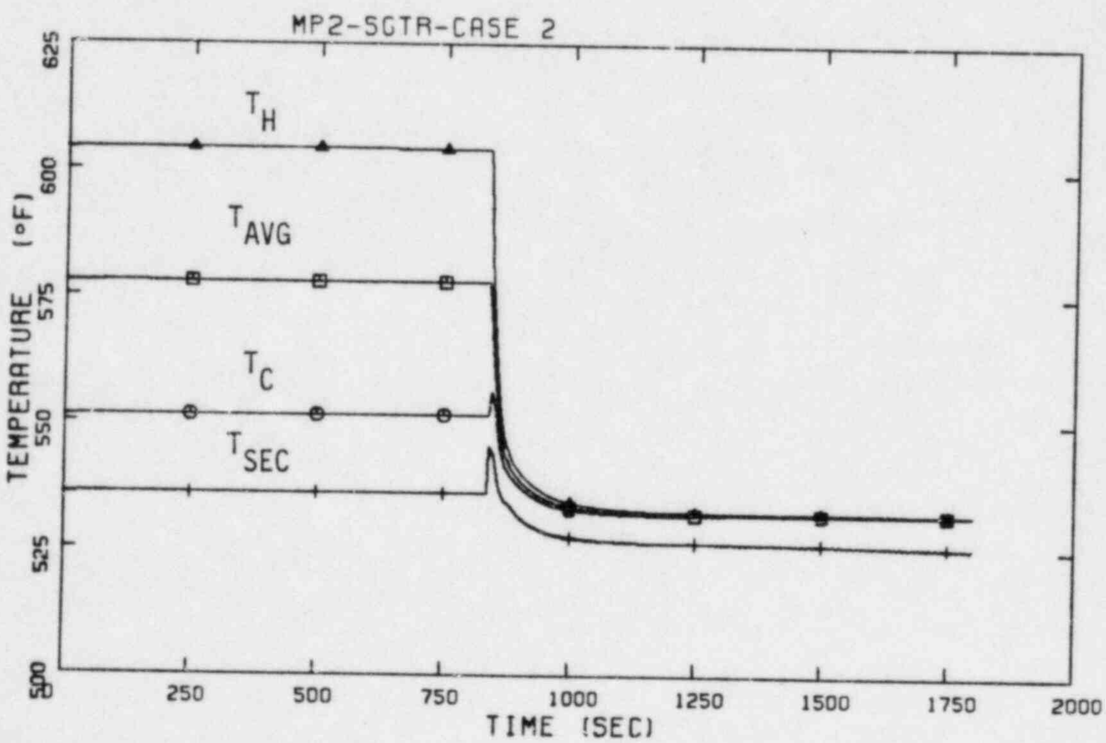

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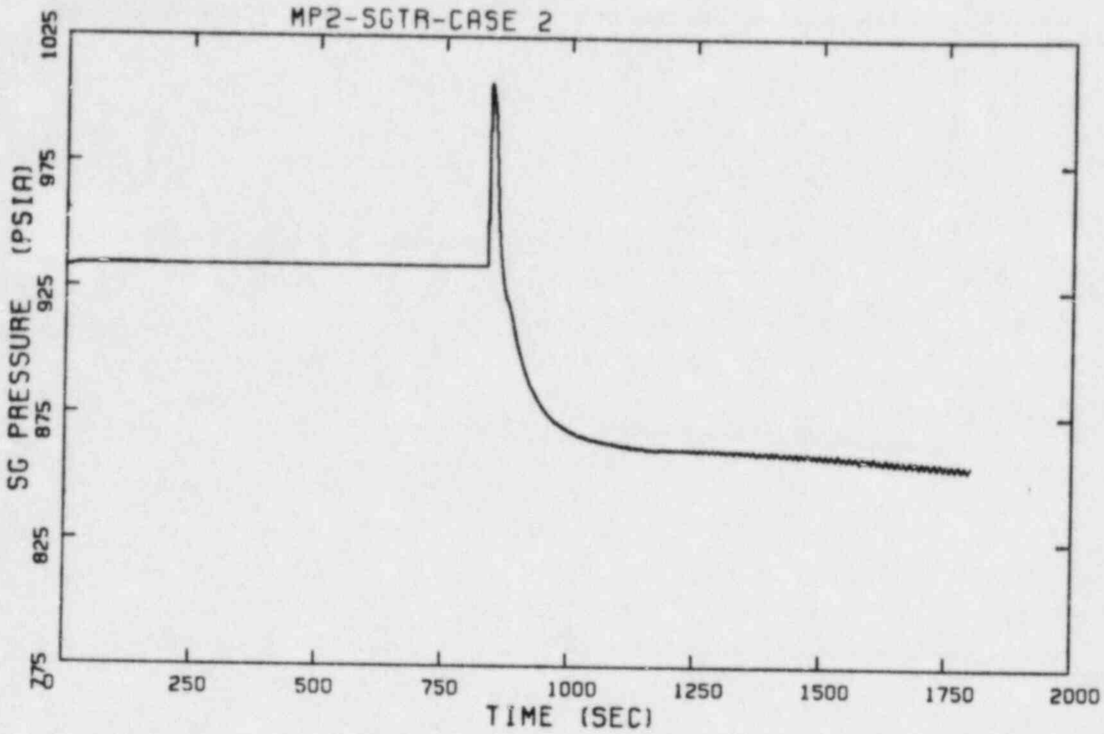

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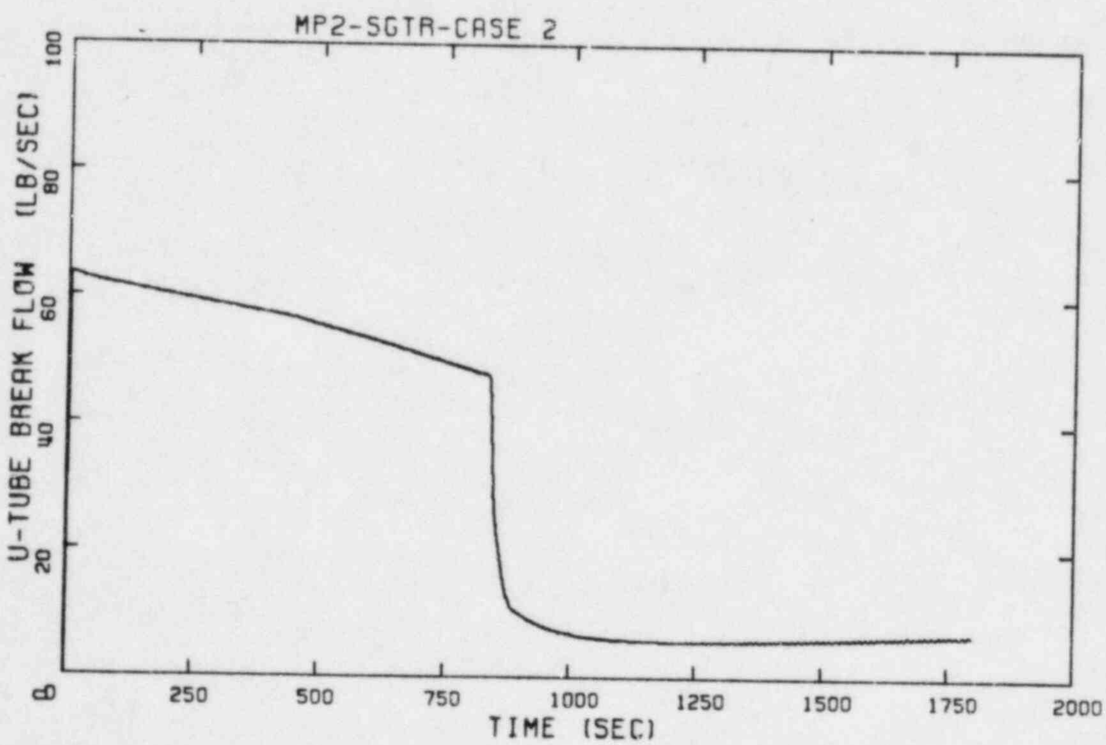
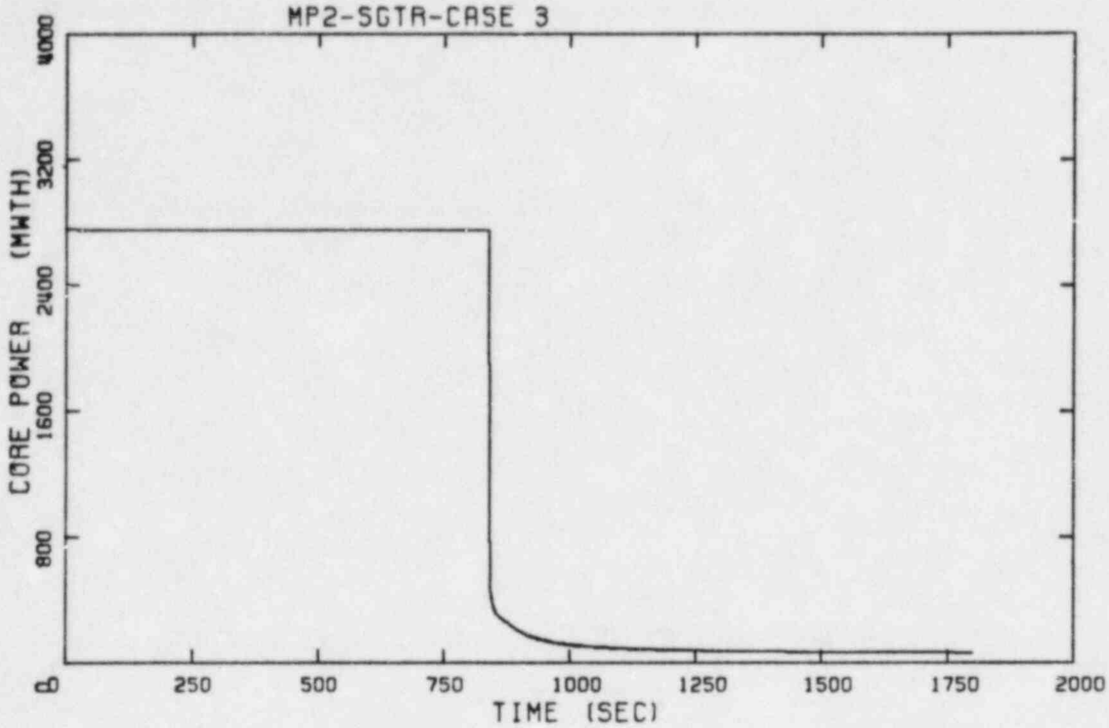


FIGURE 16



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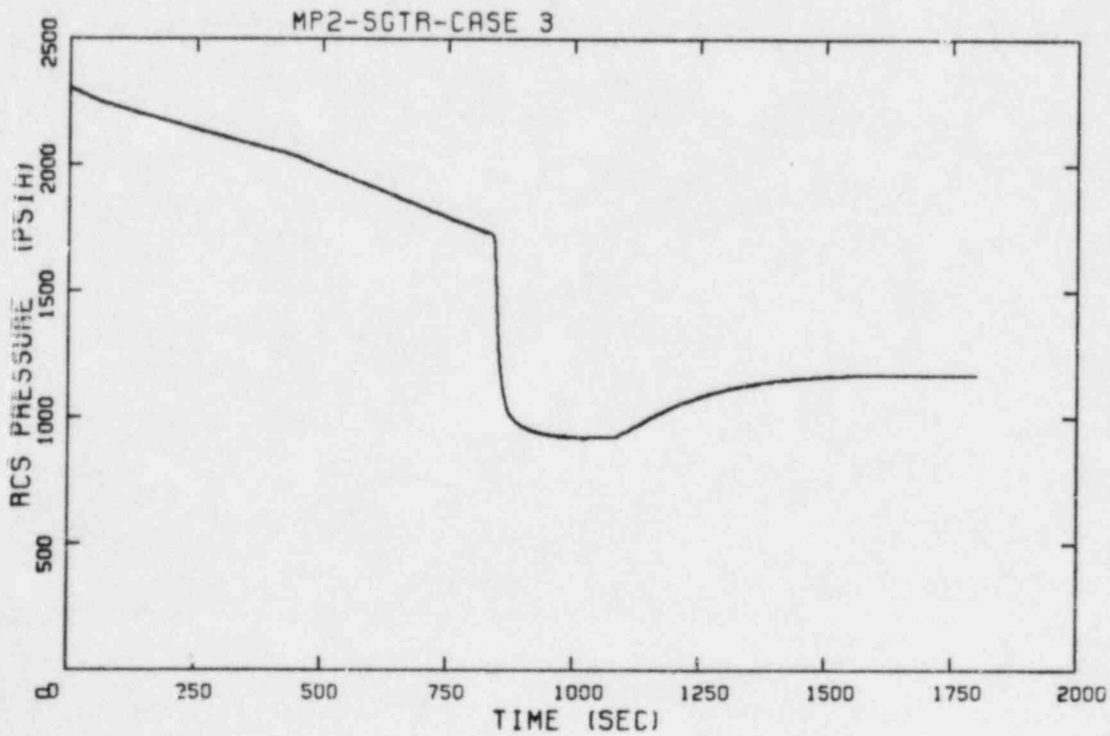

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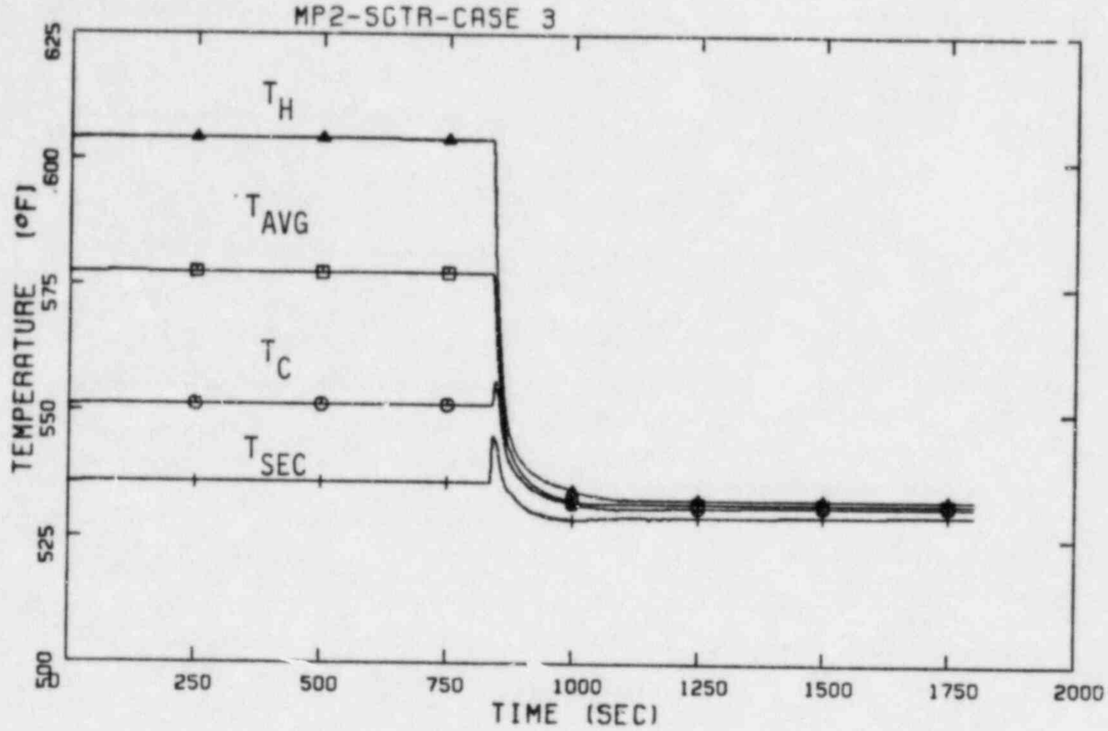

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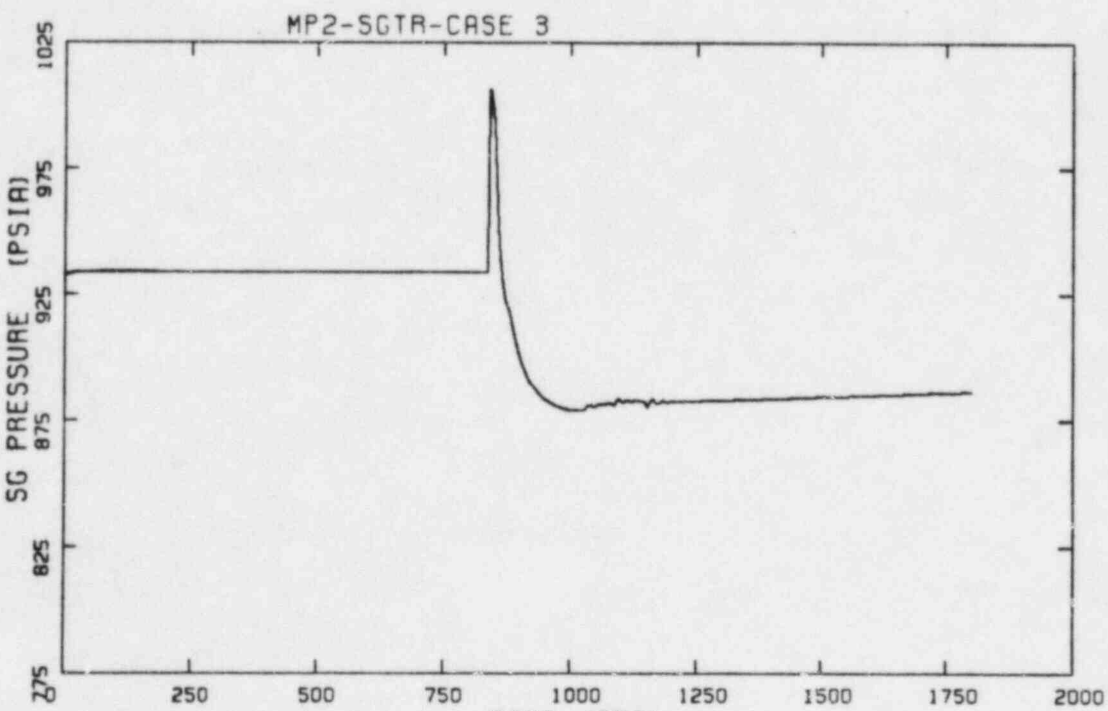

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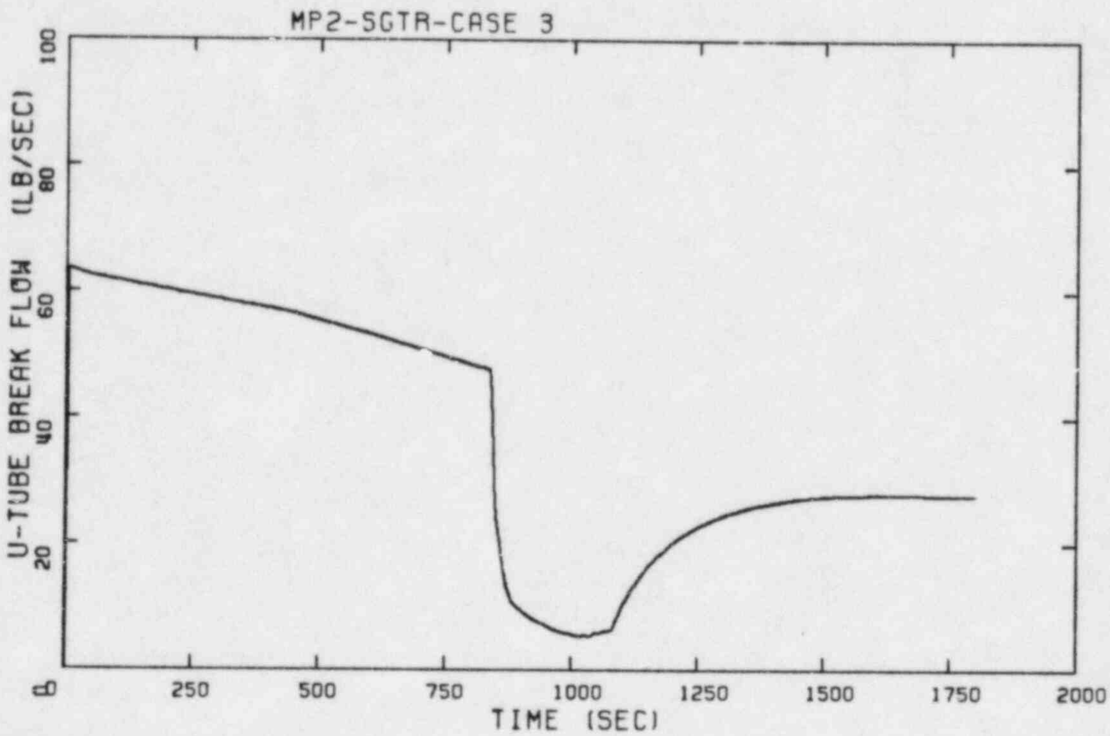

MP2-SGTR-CASE 3
U-TUBE BREAK FLOW (LB/SEC)
100
80
60
40
20
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)