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SUBJECT: Forwards reanlayzes of increased main steam flow & inside/ outside containment steamline break, incorporating CE-1 95/95 DNBR limit of 1.31 as basis for fuel rod failure.

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December 13, 1985

Director, Office of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Director
PWR Project Directorate No. 7
Division of PWR Licensing - B
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

The Southern California Edison Company transmitted on December 5, 1985 a formal response to your initial round of questions pertaining to the Cycle 3 Reload Analysis Report for San Onofre Nuclear Generating Station Unit 2. We committed to submit a reanalysis of (1) Increased Main Steam Flow and (2) Inside/Outside Containment Steam Line Break at a later date. Enclosed for your review and approval are the results of these new analyses incorporating the CE-1 95/95 DNBR limit of 1.31 as the basis for fuel rod failure in lieu of the statistical convolution technique employed earlier.

The Increased Main Steam Flow event was reanalyzed with a set of modified initial conditions based on the actual DNBR margin applicable to Cycle 3 and beyond. This has the effect of initiating the subject transient at a larger DNBR value so that the minimum DNBR attained during the transient is larger than the original analysis. The Inside Containment Steam Line Break event was reanalyzed with aid of a Variable Overpower Trip from the CPC's. As a result, the calculated fuel failure is reduced to such a point that the Outside Containment Steam Line Break event becomes limiting for radiological consequences. For this reason, a new analysis of this transient was performed and the results are presented here accordingly.

If you have any questions regarding the enclosed information, please call me.

Very truly yours,

M. O. Medford

Enclosures

cc: Harry Rood, NRC Project Manager
F. R. Huey, USNRC Senior Resident Inspector, Units 1, 2 and 3

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1/1

### 7.1.3 Increased Main Steam Flow

The Increased Main Steam Flow Event is analyzed to ensure that the Departure from Nucleate Boiling Ratio (DNBR) and Fuel Centerline Melt (CTM) Specified Acceptable Fuel Design Limits (SAFDLs) are not violated. This event was reanalyzed due to a more adverse pin census, an increased Doppler multiplier, and the availability of the CPC Variable Overpower Trip (VOPT).

#### 7.1.3.1 Identification of Causes

An Increased Main Steam Flow Event is defined as any rapid increase in steam generator steam flow other than a steam line rupture (discussed in Section 7.1.5) or an inadvertent opening of a secondary safety valve (discussed in Section 7.1.4). Such rapid increases in steam flow result in a power mismatch between core power and steam generator load demand. Consequently, there is a decrease in reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient of reactivity, the decrease in reactor coolant temperature causes an increase in core power.

The High Power Level and Core Protection Calculators (CPCs) trips provide primary protection during this event. Additional protection is provided by other trip signals including Low Steam Generator Water Level and Low Steam Generator Pressure. The approach to the CTM limit is terminated by either the CPC DNB/Local Power Density (LPD) trip, the CPC Variable Overpower Trip (VOPT) or the High Power Level Trip. In this analysis, credit is taken only for the action of the CPC Low DNBR Trip or the VOPT in the determination of the minimum transient DNBR and maximum local linear heat generation rate. The Variable Overpower Trip is described in Reference 7-17.

The following Increased Main Steam Flow Events were examined:

A. An inadvertent increased opening of the turbine admission valves caused by operator error or turbine load limit malfunction. This can result in an additional 10% flow.

B. Failure in the turbine bypass control system which would result in an opening of one or more of the turbine bypass valves. The flowrate of each valve is approximately 11% of the full power turbine flowrate. There are four turbine bypass valves for a total of 45% at full power steam flow.

C. An inadvertent opening of an atmospheric dump valve or steam generator safety valve (see Section 7.1.4) caused by operator error or failure within the valve itself. Each atmospheric dump and safety valve can release approximately 5% of the full power turbine flowrate.

#### 7.1.3.2 Analysis of Effects and Consequences

As in the Reference Cycle analysis (Reference 7-1), the opening of the four steam bypass valves at HFP produces the most adverse results. The opening of the four bypass valves at full power was initiated at the conditions given in Table 7.1.3-1. A moderator temperature coefficient (MTC) of $-3.3 \times 10^{-4}\ \Delta\rho/^{o}F$ was used in the analysis. This MTC, in conjunction with the decreasing coolant inlet temperature, results in an increase in the core heat flux. The most negative fuel temperature coefficient (FTC) with a bias of 25%, was used in the analysis. The minimum CEA worth for shutdown at the time of reactor trip for full power operation is $-6.0\%\Delta\rho$. The pressurizer pressure control system was assumed to be inoperable during the event. This minimizes the RCS pressure for the event and reduces the calculated DNBR. All other control systems were assumed to be in manual mode of operation and have no significant impact on the results for this event. The Reference Cycle cited a coincident loss of AC power as the limiting single failure for this event. The loss of AC power and subsequent reactor coolant pump coastdown occurs such that a coincident CPC Low Flow/VOPT occurs. This timing maximizes both the degradation in DNBR and the quantity of predicted fuel failure.

#### 7.1.3.3 Results

The Increased Main Steam Flow Event plus a single failure (loss of AC power) resulted in a CPC VOPT Trip/Low Flow Trip at 9.75 seconds. The minimum DNBR calculated for the event initiated from the conditions specified in Table 7.1.3-1 was 1.21 compared to the design limit of 1.31. Based on the criteria that all fuel pins with a DNBR value below the design limit fail, less than 1.5% of the fuel pins in the core are predicted to fail. For a conservative estimate of the radiological consequences, a failed fuel percentage of 5% was used. A maximum allowable initial linear heat generation rate of 16.0 kW/ft could exist as an initial condition without exceeding the Acceptable Fuel to Centerline Melt Limit of 21.0 kW/ft during this transient. This amount of margin is assured by setting the linear heat rate LCO based on the more limiting allowable linear heat rate for LOCA (13.9 kW/ft, see Table 7.0-6).

NSSS cooldown is two hours in duration resulting in offsite doses of less than 7 REM thyroid and a whole body dose of less than 2 REM. These results are no more limiting than those presented in the Reference Cycle for Increased Main Steam Flow Events with a single failure.

Table 7.1.3-2 presents the sequence of events for the event initiated at HFP conditions. Figures 7.1.3-1 to 7.1.3-5 present the NSSS response of core power, core heat flux, RCS pressure, RCS temperatures and steam generator pressure. The DNBR response for Cycle 3 as a function of time is presented in Figure 7.1.3-6.

The results of the Increased Main Steam Flow without a single failure would be no more adverse than those presented in the Reference Cycle.

#### 7.1.3.4 Conclusions

For the Increased Main Steam Flow Events with a single failure, the radiological doses are well within (<25%) the 10CFR100 limits of 300 REM for thyroid and 25 REM for whole body. For the Increased Main Steam Flow Event without a single failure, the DNBR and CTM limits are not exceeded.

### 7.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

The results are bounded by the Reference Cycle.

Table 7.1.3-1

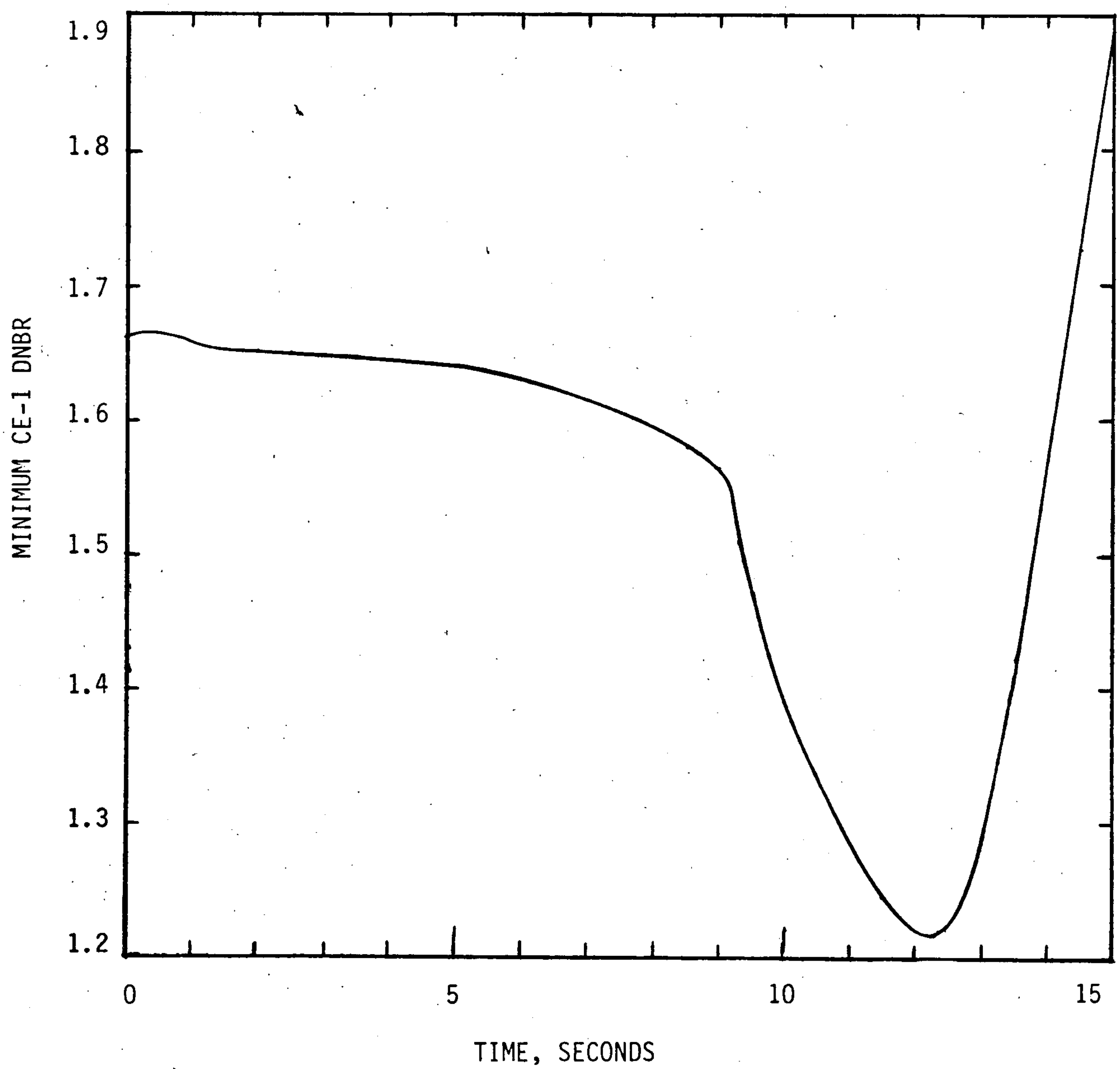
Key Parameters Assumed for the Increased Main Steam Flow Event

| Parameter | Units | Reference Cycle Value | Cycle 3 Value |
|---|---|---|---|
| Total RCS Power (Core Thermal Power + Pump Heat) | MWt | 3478 | 3478 |
| Initial Core Coolant Inlet Temperature | $^oF$ | 560 | 560 |
| Initial Reactor Coolant System Pressure | psia | 2200 | 2200 |
| Initial RCS Vessel Flow Rate | gpm | 396,000 | 396,000 |
| Moderator Temperature Coefficient | $x10^{-4} \Delta\rho/^oF$ | -3.3 | -3.3 |
| CEA Worth at Trip | % $\Delta\rho$ | -4.5 | -6.0 |
| Doppler Coefficient Multiplier | | 1.15 | 1.25 |

Table 7.1.3-2

Sequence of Events for the Increased Main Steam Flow Event Plus a Single Failure

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Quick Open Signal Generated, Four Bypass Valves Start to Open | --- |
| 1.0 | Four Bypass Valves Full Open | 145% of full steam flow |
| 8.95 | Loss of All On and Offsite Power, Turbine Admission Valves and Bypass Valves Start to Close, Feedwater Begins to Coast Down, Reactor Coolant Pumps Begin to Coast Down | --- |
| 9.75 | CPC VOPT Trip/Low Flow Signal Generated | 116% of 3410 MWt, 95% of shaft speed |
| 10.0 | Reactor Trip Breakers Open, Turbine Trip | --- |
| 10.3 | CEAs Begin to Drop in the Core | --- |
| 10.7 | Maximum Core Power | 117.6% of 3410 MWt |
| 12.1 | Maximum Core Heat Flux | 110.6% of 3410 MWt |
| 12.3 | Minimum DNBR Occurs (CE-1) | 1.21 |
| 12.75 | Turbine Admission Valves and Bypass Valves Closed | --- |
| 16.7 | Steam Generator Safety Valves Open | 1100 psia |
| 28.05 | Feedwater Flow Reaches 5% of Full Power | |

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INCREASED MAIN STEAM FLOW
PLUS SINGLE FAILURE
DNBR VS TIME

FIGURE 7.1.3-6

## 7.1.5 Steam System Piping

Failures in the main steam system piping were analyzed to ensure that a coolable geometry is maintained and that the site boundary doses do not exceed 10CFR100 guidelines.

### 7.1.5a Steam System Piping Failures: Inside and Outside Containment Pre-Trip Power Excursions

This event was analyzed to evaluate the maximum number of calculated fuel pin failures for the site boundary dose calculation due to changes in the pin census and the availability of the CPC Variable Overpower Trip (VOPT).

#### 7.1.5a.1 Identification of Causes

A rupture in the main steam system piping increases steam flow from the steam generators. This increase in steam flow increases the rate of RCS heat removal by the steam generators and causes a decrease in core coolant inlet temperature. In the presence of a negative moderator temperature coefficient of reactivity (MTC), this decrease in temperature causes core power to increase.

The excursion in core power is terminated by the action of one of the following Reactor Protection System (RPS) trips: Core Protection Calculators (CPCs), Low Steam Generator Pressure (LSGP), High Linear Power Level, or High Containment Pressure.

#### 7.1.5a.2 Analysis of Effects and Consequences

Steam Line Breaks (SLBs) inside containment may be postulated to have break areas up to the cross section of the largest main steam pipe (7.41 $ft^2$). Those SLBs occurring outside the containment building have break areas limited by the areas of the flow restrictors (4.13 $ft^2$) which are located upstream of the containment penetrations.

Inside containment SLBs may cause environmental degradation of sensor input to the CPCs and pressure measurement systems. Additionally, the high linear power level trip undergoes temperature decalibration due to RCS cooldown. The only credit taken for CPC action during this event is the CPC VOPT. The required input into the VOPT includes output from the Resistance Temperature Detectors (RTDs) and ex-core reactor flux power detectors. The qualification of these sensors for degraded environmental conditions is discussed in Reference 7-18. Other trips which may occur for inside containment SLBs are: LSGP, High Linear Power Level or High Containment Pressure. Additionally, the environmentally degraded value of the Delta Pressure Low Flow trip is used to determine the most adverse timing of a Loss of AC Power (LOAC).

Outside containment SLBs are not subject to the same environmental effects as the inside containment breaks. Therefore, the full array of RPS trips including the CPC Low DNBR trip, are credited for these breaks.

In the Reference Cycle, an extensive parametric analysis in both MTC and break area was performed on the inside containment SLB event. This parametric analysis identified the limiting inside containment SLB event in terms of fuel pin failure caused by the pre-trip power excursion. Table 7.1.5a-1 of the Reference Cycle (Reference 7-1) lists the values of key parameters used in the parametric analysis.

The pre-trip SLB event was reanalyzed in Cycle 3 to accommodate a more adverse pin census. Changes in other Key Parameters for Cycle 3 are within the ranges used for the Reference Cycle Parametric Study. The Reference Cycle results (heat flux, RCS temperatures, pressure and flow rate) were combined with the pin census to yield a value for predicted fuel failure. Consideration of the VOPT was also included for Cycle 3.

The Reference Cycle identified the limiting break location to be inside containment. The analysis for Cycle 3 indicates that the limiting break in terms of radiological consequences is located outside of the containment building. This is because crediting the action of the CPC VOPT reduced the amount of calculated fuel failure for the inside containment SLB relative to the Reference Cycle. In addition, the radiological consequences of the inside containment SLB were calculated using a Steam Generator Iodine Partition Factor (PF) of 100, as specified in Reference 7-19. The Reference Cycle used a more conservative PF of 10. For the outside containment break, the affected steam generator dries out and a PF of 1 was used.

### 7.1.5a.3 Results

Table 7.1.5a-1 contains the sequence of events for the outside containment pre-trip SLB.

Radiological consequences were considered for three cases. Each case assumed that all iodine transported to the secondary side was released to atmosphere during periods of steam generator dryout. Each case used the maximum Technical Specification steam generator leakage of 1 gpm to the affected steam generator.

The first case was for an outside containment SLB with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike. The FSAR analysis of this case remains valid. The resultant doses were a small fraction of 10CFR100 limits.

The second case was for an outside containment SLB with a pre-existing iodine spike. The FSAR analysis of this case remains valid. The resultant doses were within 10CFR100 limits.

The third case was for an outside containment SLB with a small amount of predicted fuel failure. Use of the criteria that all fuel pins with a DNBR value less than the 95/95 limit of 1.31 are assumed to fail results in the prediction of less than 1.5% failed fuel. The resultant calculated doses were less than 200 REM thyroid and less than 17 REM whole body. This is within 10CFR100 limits.

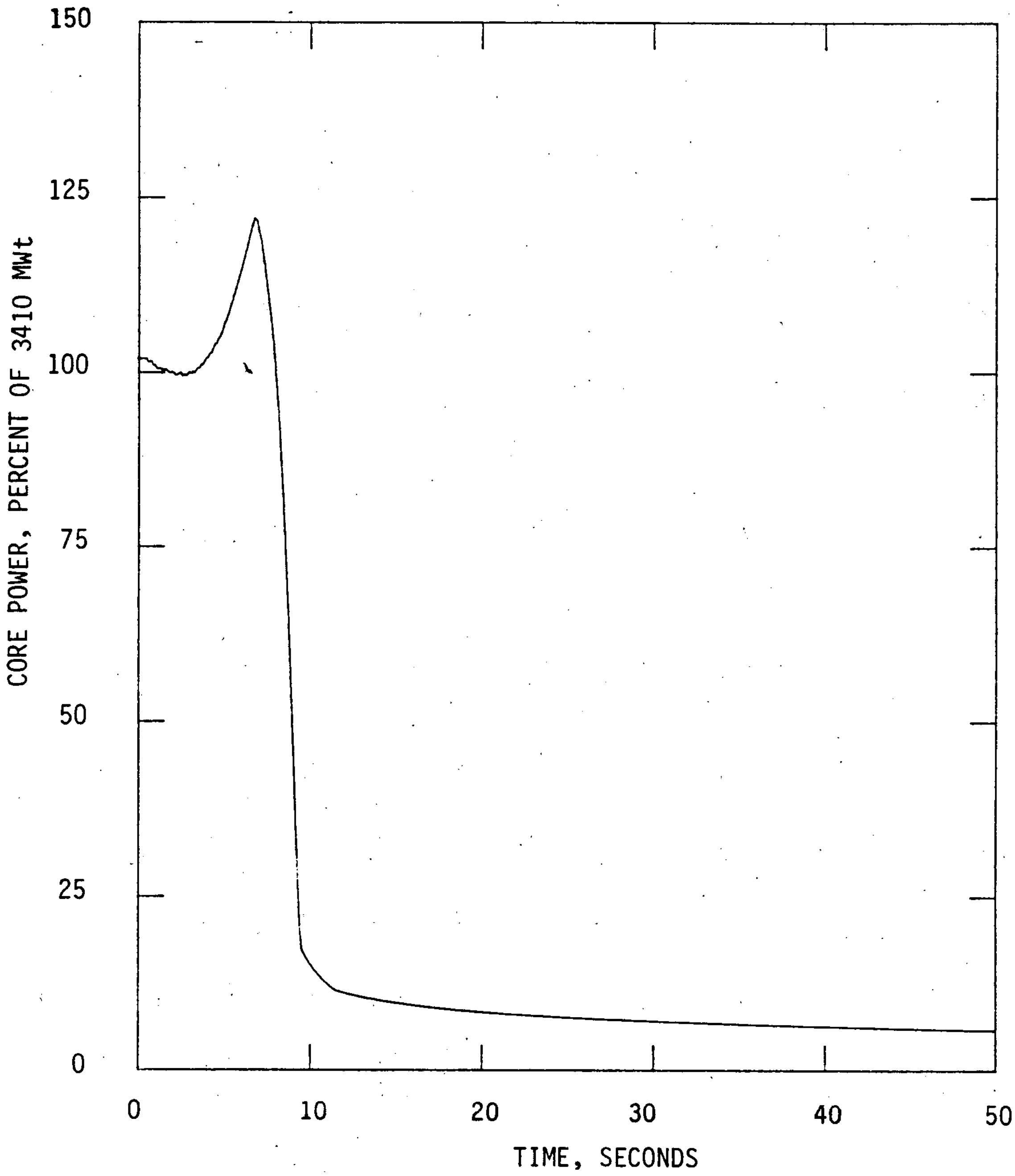
### 7.1.5a.4 Conclusions

The results of this analysis demonstrate that a coolable geometry is maintained during this event as the number of fuel pins calculated to fail is less than 1.5 percent. For a coincident iodine spike, the site boundary dose is a small fraction (<10%) of the 10CFR100 limits. For a pre-existing iodine spike or for the predicted fuel failure, the resultant doses are within the 10CFR100 limits.

Table 7.1.5a-1

Sequence of Events for the Steam System Piping Failure Event
Pre-Trip Power Excursions

| Time (sec) | Event | Setpoint or Value |
|---|---|---|
| 0.0 | Failure in the Main Steam System Piping | 7.41 $ft^2$ |
| 5.8 | CPC VOPT RPS Trip Generated<br>Main Steam Isolation Signal | 113.0% of 3410 MWt,<br>675 psia |
| 6.1 | Trip Breakers Open | ---- |
| 6.1 | LOAC Occurs, Reactor Coolant Pumps Begin to Coastdown | ---- |
| 6.4 | CEAs Begin to Enter Core | ---- |
| 6.7 | Main Steam Isolation Valves Begin to Close | ---- |
| 6.8 | Maximum Core Power | 122.1 of 3410 MWt |
| 7.3 | Maximum Core Heat Flux | 111.7% of 3410 MWt |
| 16.7 | Main Steam Isolation Valves Closed | ---- |
| 21.55 | Safety Injection Signal | 1560 psia |
| 33.6 | Pressurizer Empties | ---- |
| 52.75 | Safety Injection Pumps Reach Full Speed | ---- |

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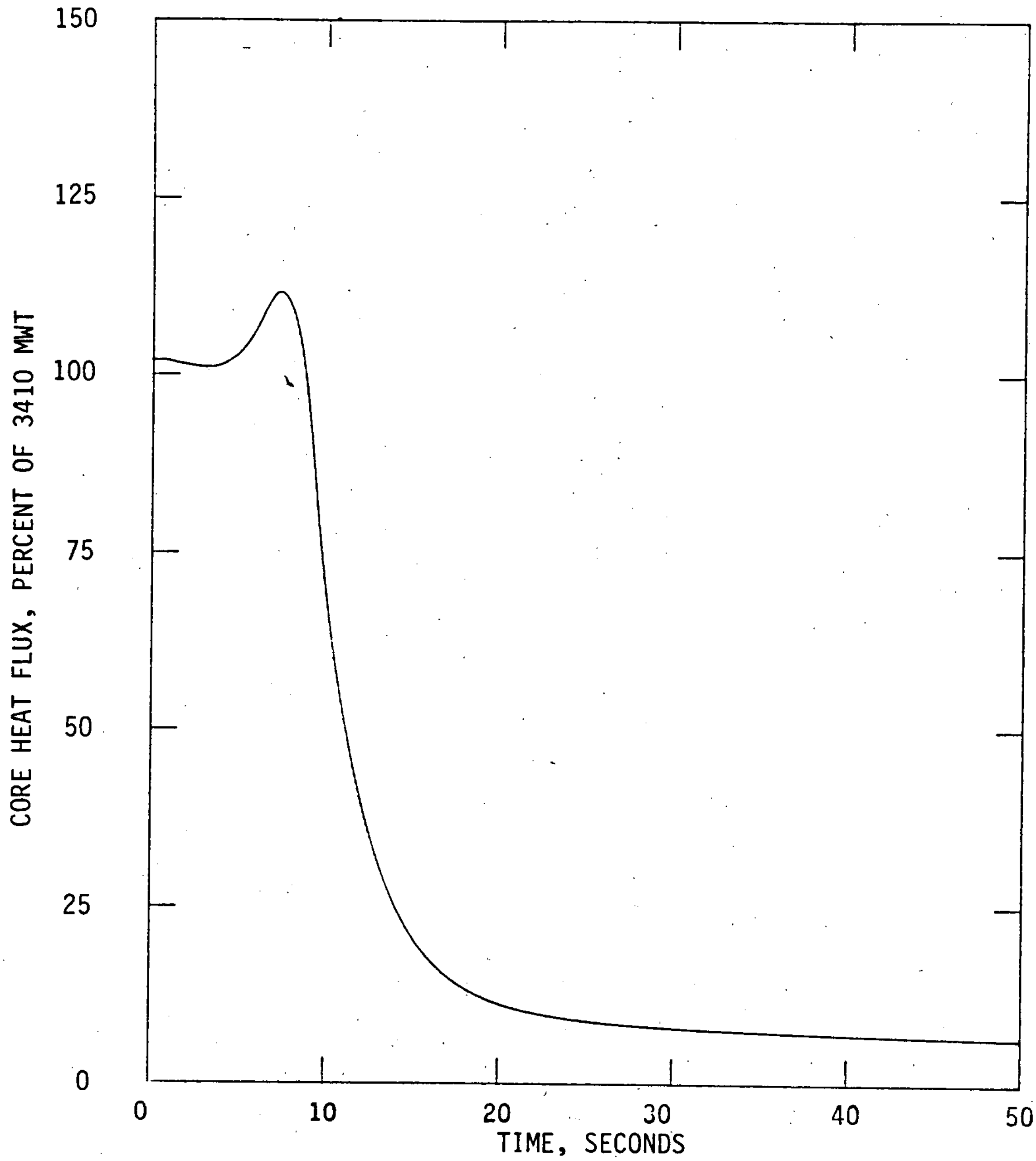
HOT FULL POWER STEAM LINE BREAK
SMALL BREAK WITH LOSS OF AC
CORE POWER VS TIME

FIGURE 7.1.5a-1

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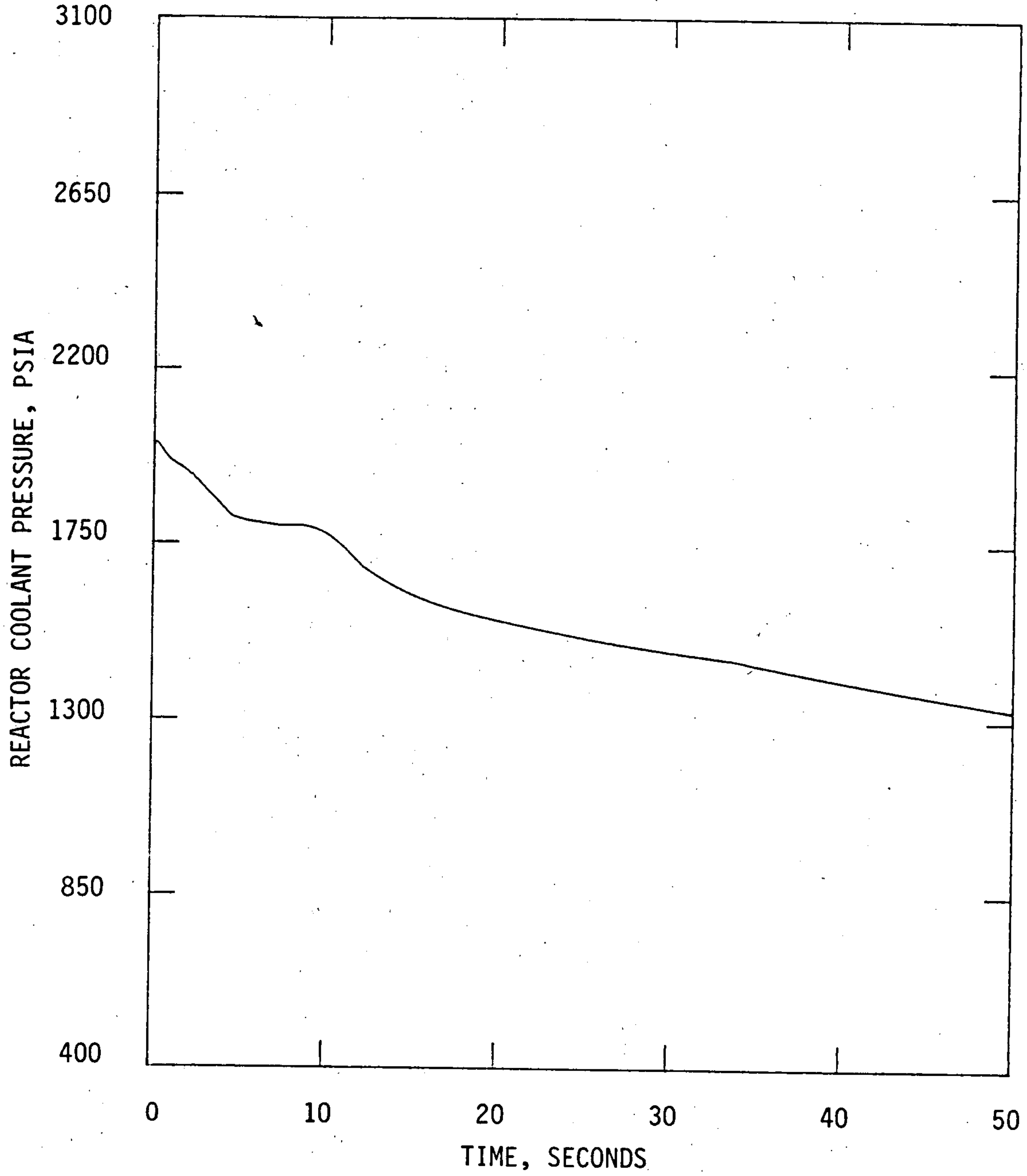
SMALL BREAK WITH LOSS OF AC
CORE HEAT FLUX VS TIME

FIGURE 7.1.5a-2

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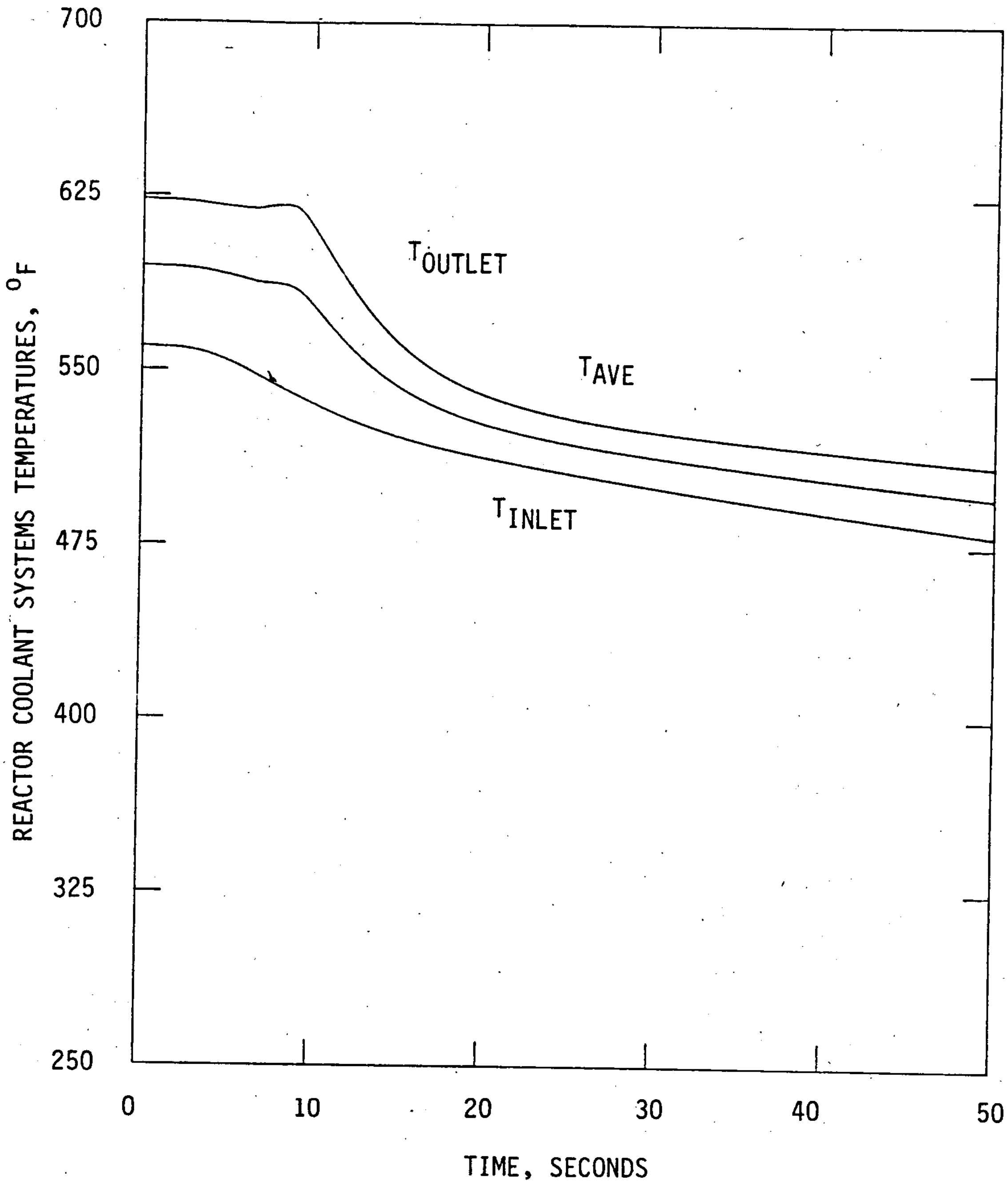
HOT FULL POWER STEAM LINE BREAK
SMALL BREAK WITH LOSS OF AC
RCS PRESSURE VS TIME

FIGURE 7.1.5a-3

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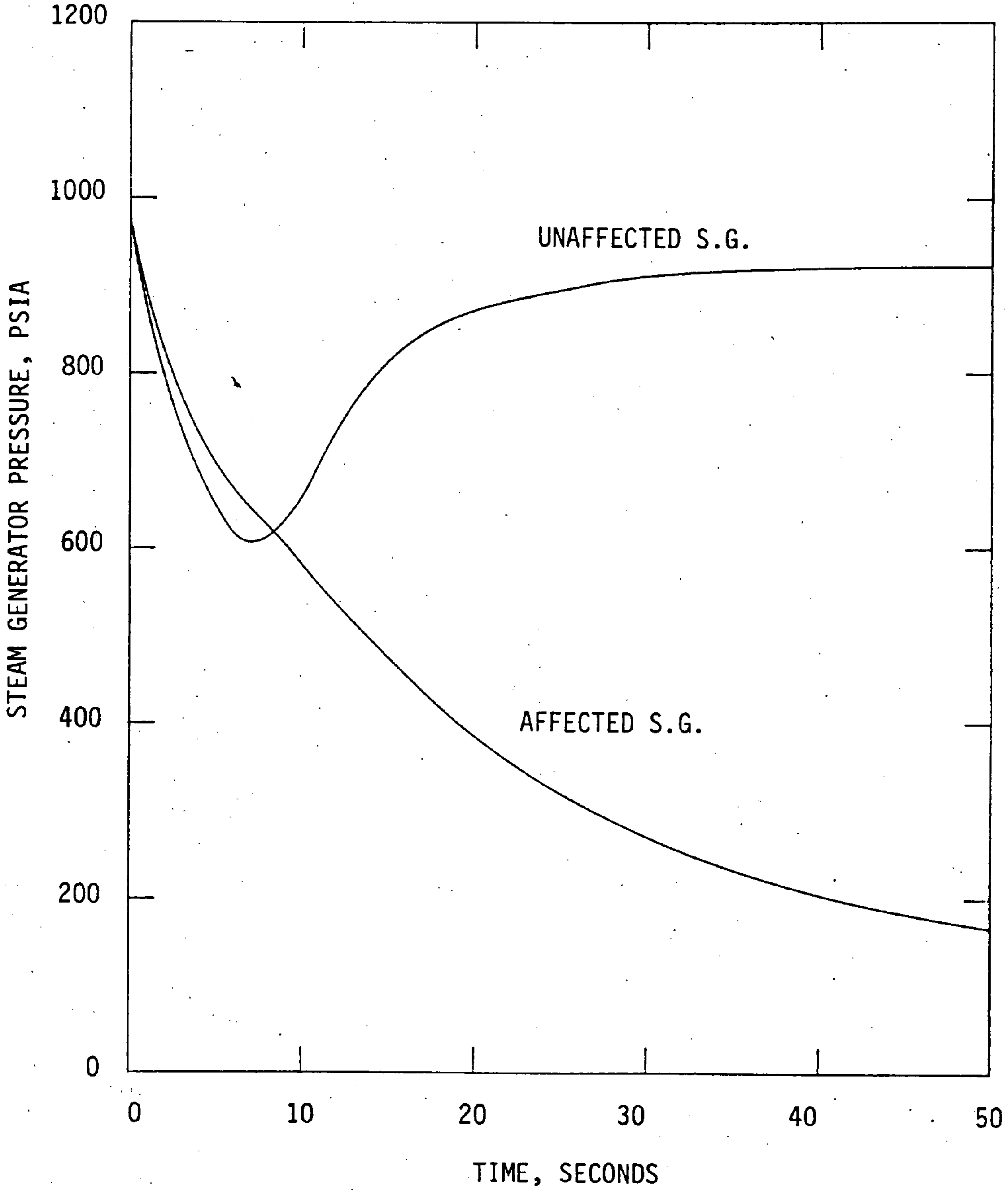
HOT FULL POWER STEAM LINE BREAK
SMALL BREAK WITH LOSS OF AC
RCS TEMPERATURES VS TIME

FIGURE 7.1.5a-4

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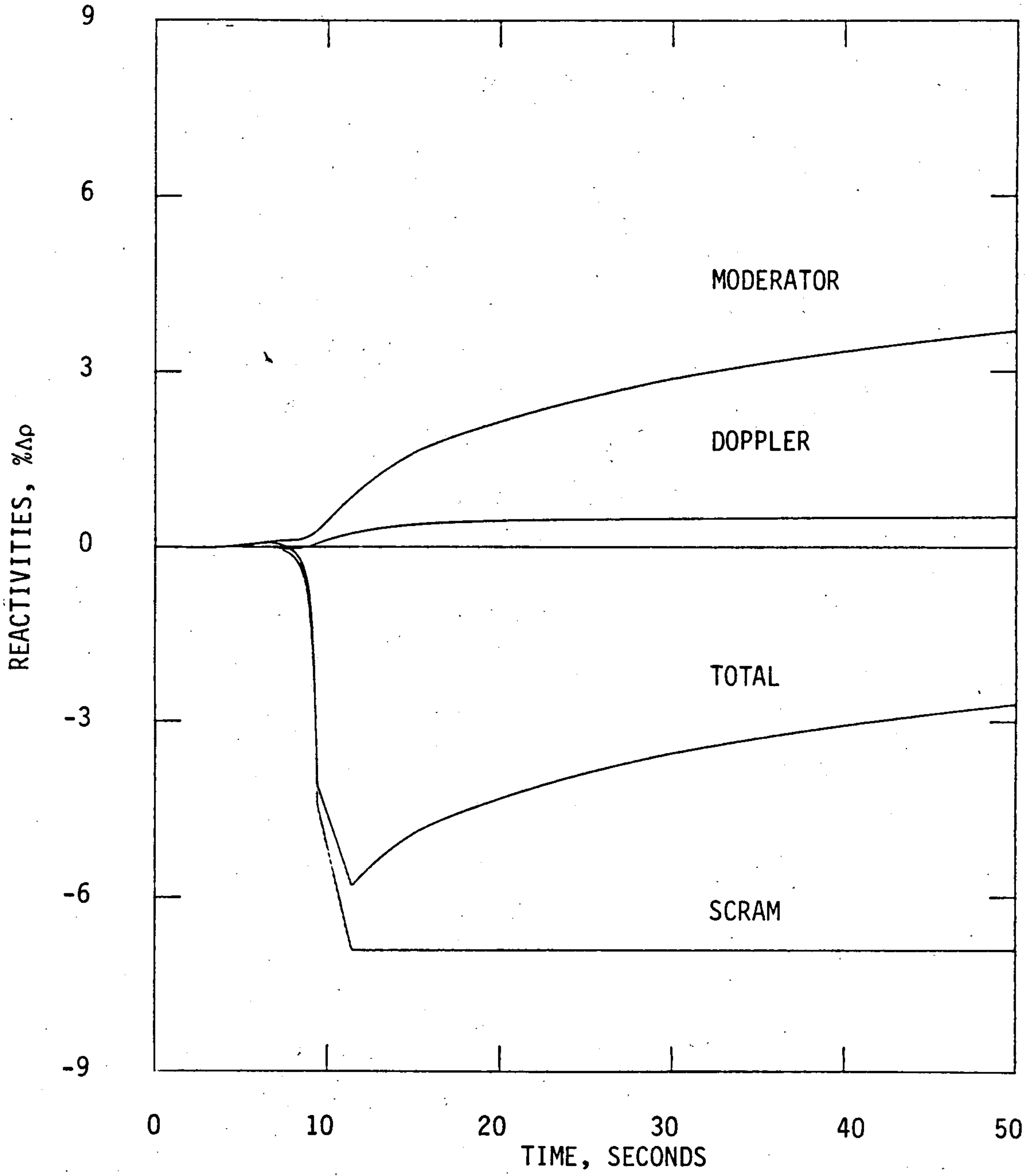
HOT FULL POWER STEAM LINE BREAK
SMALL BREAK WITH LOSS OF AC
STEAM GENERATOR PRESSURE VS TIME

FIGURE 7.1.5a-5

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HOT FULL POWER STEAM LINE BREAK
SMALL BREAK WITH LOSS OF AC
REACTIVITIES VS TIME

FIGURE 7.1.5a-6

Additional Section 7 References:

(7-18) "Environmental Qualification Report Per Requirements of NUREG-0588, Revision 2," San Onofre Nuclear Generating Station Unit 2 and Unit 3, Unit 2 Docket 50-361, Unit 3 Docket 50-362.

(7-19) "Standard Review Plan," NUREG 75/087, Revision 1.