Consumers Power Company

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Harold R Denton, Director
Office of Nuclear Reactor Regulation
Division of Licensing
US Nuclear Regulatory Commission
Washington, DC 20555

MIDLAND PROJECT
MIDLAND DOCKET NOS 50-329, 50-330
FOLLOW-UP RESPONSE TO OPEN ITEMS OF DRAFT SER SECTION 5.2.2
FILE: 0505.805 SERIAL: 13785

Reference (a): J W Cook letter to H R Denton, Serial No 13364, dated July 23, 1981

Enclosures (1) Follow-up Response to Open Item 2(a) of Draft SER Section 5.2.2

(2) Follow-up Response to Open Item 2(c) of Draft SER Section 5.2.2

Reference (a) committed CP Co to provide to the NRC follow-up responses to Draft SER Section 5.2.2 Open Items 2(a) and 2(c) by September 10, 1981. Enclosed are the follow-up responses to these items.

With the responses provided in this letter, we believe Open Items 2(a) and 2(c) can be closed out by the Staff.

James W. Cook

JWC/PEP/fms

CC RJCook, Midland Resident Inspector
DSHood, US NRC
TPSpeis, US NRC
DBMiller, Midland Construction (3)
RWHuston, Washington

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Enclosure 1

Conservatism and input assumptions used in the Midland DYSID analysis of erroneous actuation of one HPI train are as follows:

1. The initial pressurizer pressure used was 550 psig which is the open setpoint for the PORV. In actual plant operation, the pressure would be lower. At 280°F RCS temperature, the minimum pressure required for RC pump operation is 295 psig, so the plant operator would control the pressure between 295 and 550 psig. The middle of this allowable pressure "window" would be 425 psig which is considerably below the 550 psig initial pressure used in the analysis. A lower initial pressure would reduce the pressure reached at 10 minutes.

2. The initial pressurizer level used was 180 inches. The range of the level measurement is 0 to 400 inches and the normal level for power operation is 220 inches. The 180 inch level will be a Technical Specification limit in the 280°F to 330°F RCS temperature range as noted by Item No 7 in the response to FSAR Question No 211.105. During actual plant operation, the operator would maintain the level below the Technical Specification limit. A lower initial level would reduce the pressure reached at 10 minutes.

3. The HPI flow rate into the RCS was based on a presetting of the HPI valves which allows the HPI pump to reach the pump run out flow limit. In plant preoperational testing, the HPI valves will be set to achieve a flow rate between the minimum required flow rate and the pump run out flow rate limit. A lower HPI flow rate would reduce the pressure reached at 10 minutes.

4. An insurge temperature into the pressurizer of 300°F was used. The Technical Specification pressure limit is lowest at 280°F (see FSAR Figure 5.2-1). Use of the lower insurge temperature would tend to reduce the pressure reached at 10 minutes.

5. No credit was taken for: (1) any letdown flow from the RCS which would decrease the insurge, (2) any pressurizer spray which would slow the pressure increase rate, or (3) any relief by the PORV.

The Technical Specification pressure limit at 280°F RCS temperature is 1540 psig which includes a pressure decrease adjustment for possible instrument error on pressure and temperature. The DYSID analysis, with the above listed conservatisms, resulted in a pressure of 1455 psig at 10 minutes which is 85 psi below the lowest pressure limit in the 280°F to 330°F RCS temperature range where operator action is one of the two overpressure protection methods.

CADDS BENCHMARKS

A comparison has been made of the CADDS computer program with actual operating data. The results indicate excellent agreement with that of reactimeter data taken during actual transients. The two cases investigated were a turbine trip at 72% FP at Oconee Unit 1, and a unit generator trip at 96% FP at Three Mile Island Unit 1. Both of these transients resulted in a high RC pressure trip.

Primary side parameters which best represented the reactor steady state operation (core life, power level, boron concentration, etc.) prior to the transient, were used in each of the CADDS calculations. Actual secondary side pressure and feedwater flow responses derived from the reactimeter data were input to CADDS to describe the secondary side behavior prior to and during the transients. The calculated primary side responses thus represent the effect on the primary side of transient changes in the secondary side.

Oconee Unit I Trip:

Due to a malfunction in the speed controller in the "B" main feedwater pump, a sequence of events was initiated which resulted in the "A" Main Feedwater Pump (MFDWP) tripping due to the loss of the auxiliary oil pump, and the "B" MFDWP tripping on high discharge pressure. This caused a turbine trip (immediately) which resulted in a secondlary side pressure surge due to the closing of the turbine stop valves. This is shown in Figure 1. The RC pressure and temperature increased (see Figures 2 and 3) due to the reduction in steam flow and the loss of feedwater. The reactor tripped on high RC pressure approximately 20 seconds after the start of the transient. The reactor started a power runback due to the loss of feedwater, and peak prior to the high RC pressure trip via the positive moderator coefficient as seen in

Figure 4. All measured data was recorded at 3 second intervals, so the response times can only be approximated. A comparison of the measured and CADDS calculated responses are shown in Figures 2, 3, and 4. With the exception of the power peak just prior to the reactor trip, the CADDS calculated results agree very well with the recorded responses. Differences in peak power may be attibutable to the moderator coefficient used and NI calibration.

TMI Unit 1 Trip:

A planned unit generator trip transient was initiated at Three Mile Island Unit 1 by opening the breakers while the reactor was operating at 96% FP. The loss of load caused the turbine to overspeed and initiate closing of the turbine stop valves. The removal of steam flow to the turbine initiated a pressure rise on the secondary side to the code safety value of 1065 psia in approximately 4.5 seconds. A feedwater pump speed "kicker" initiated by the ICS, upon a turbine trip, helped to keep up the feedwater flow. (See Figure 5 for secondary pressure and feedwater flow response).

The reduction in steam flow caused a pressure and temperature rise in the reactor coolant system. (See Figure 6 and 7). The ICS started a rod runback (20%/min.) as can be seen in the power response in Figure 8. The reactor tripped on high RC pressure in approximately 3.5 seconds. The change in pressurizer level during the transient is shown in Figure 9. A comparison of the measured and CADDS calculated responses are indicated in Figures 6, 7, 8, and 9.

The calculated pressure response, as shown in Figure 6, took almost 2 seconds longer than the recorded response to turn fully around. This may be due to the code's inability to respond quickly to the feedwater recovery near the trip time which appears to be responsible for the recorded response.

The CADDS calculated RC outlet temperature agrees well with the recorded results (see Figure 7). Delays in temperatures are difficult to account for due to differences in the code nodes and actual sensor locations.

The agreement is good, between the calculated and measured pressurizer level response during the transient (see Figure 9).

No rod runback was used in the CADDS calculation (Figure 8), since in the 3.5 seconds prior to reactor trip, the power level decreased only ~2% FP, and should have little effect on RC pressure and temperature response. A slightly negative moderator coefficient was estimated for this case based on the initial power level, EFPD, and boron concentration. The calculated effect on the power level was negligible; the measured effect was marked by the rod runback.

FIGURE 1

OCONEE I - TRIP #21

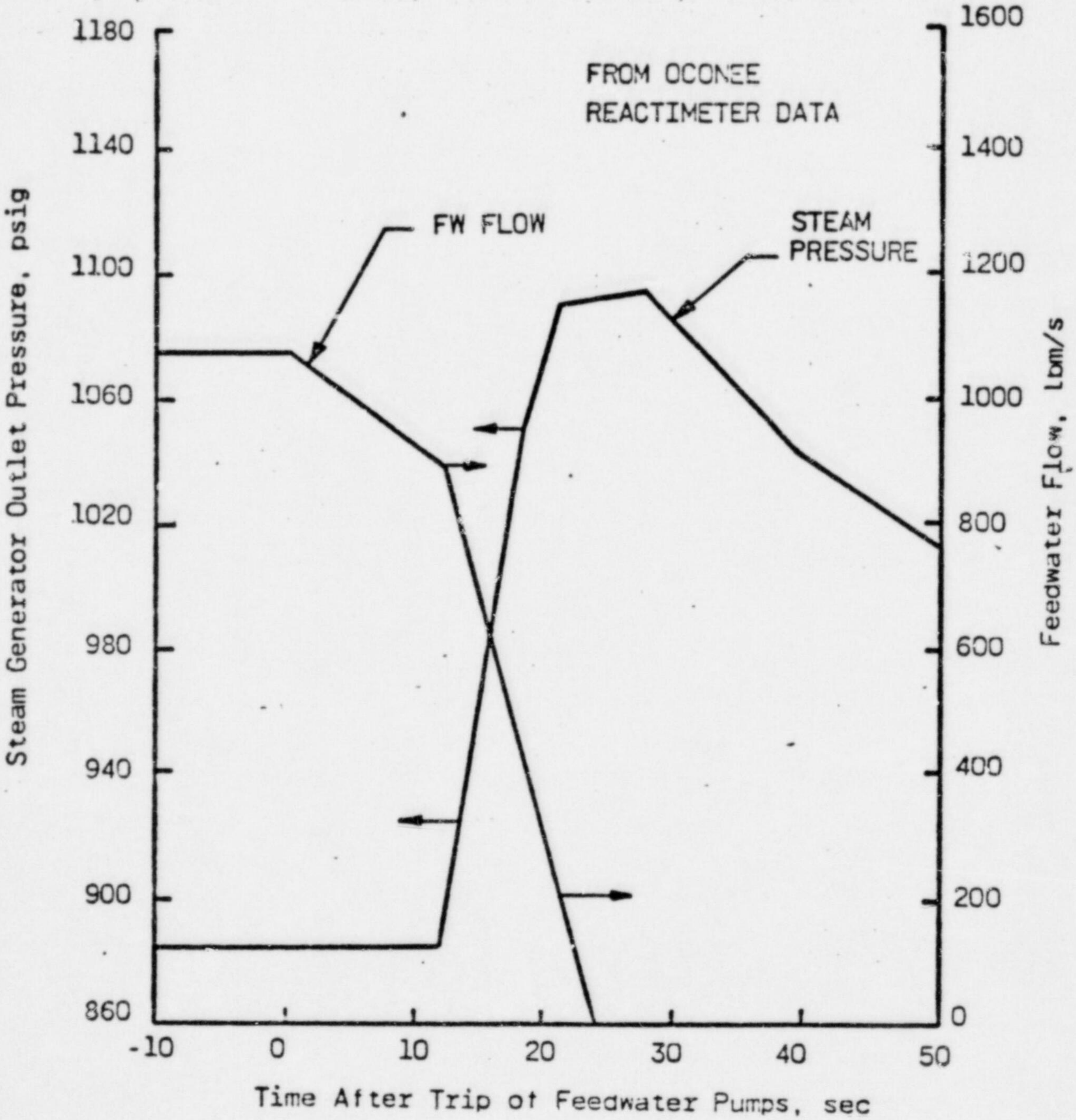

FIGURE 2

OCONEE 1 - TRIP #21

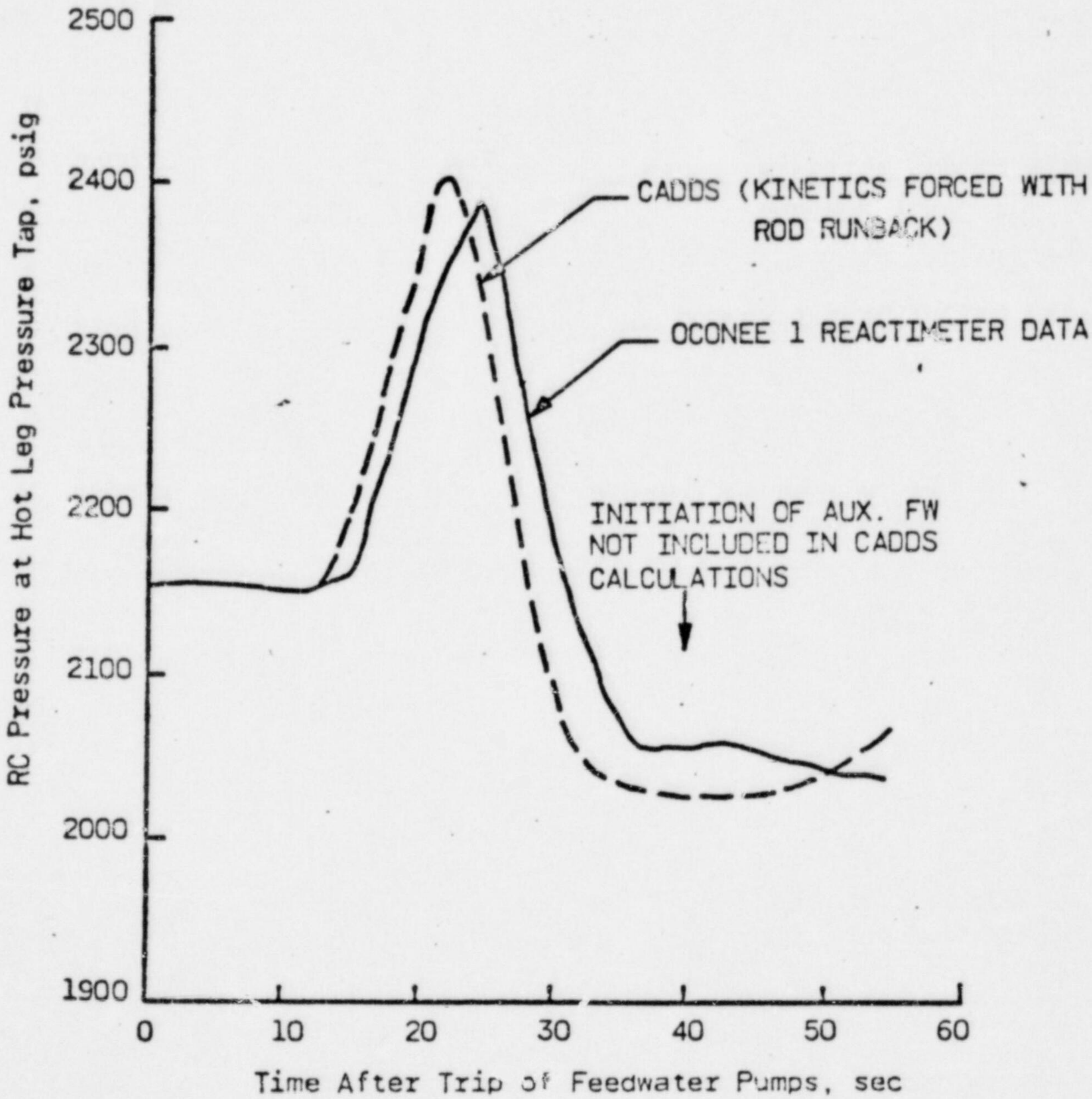

FIGURE 3

OCONEE 1 - TRIP #21

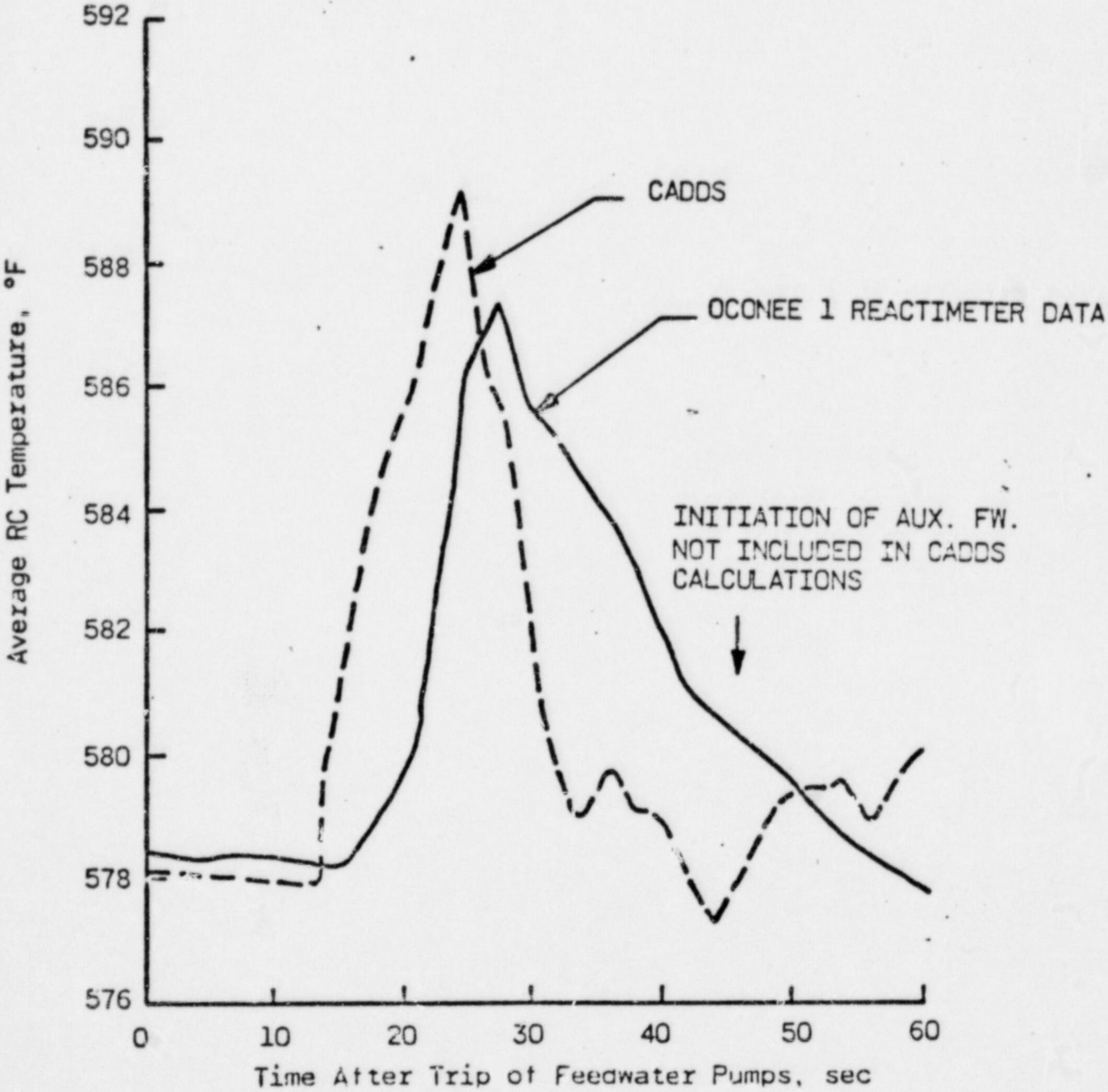

FIGURE 4

OCONEE 1 - TRIP #21

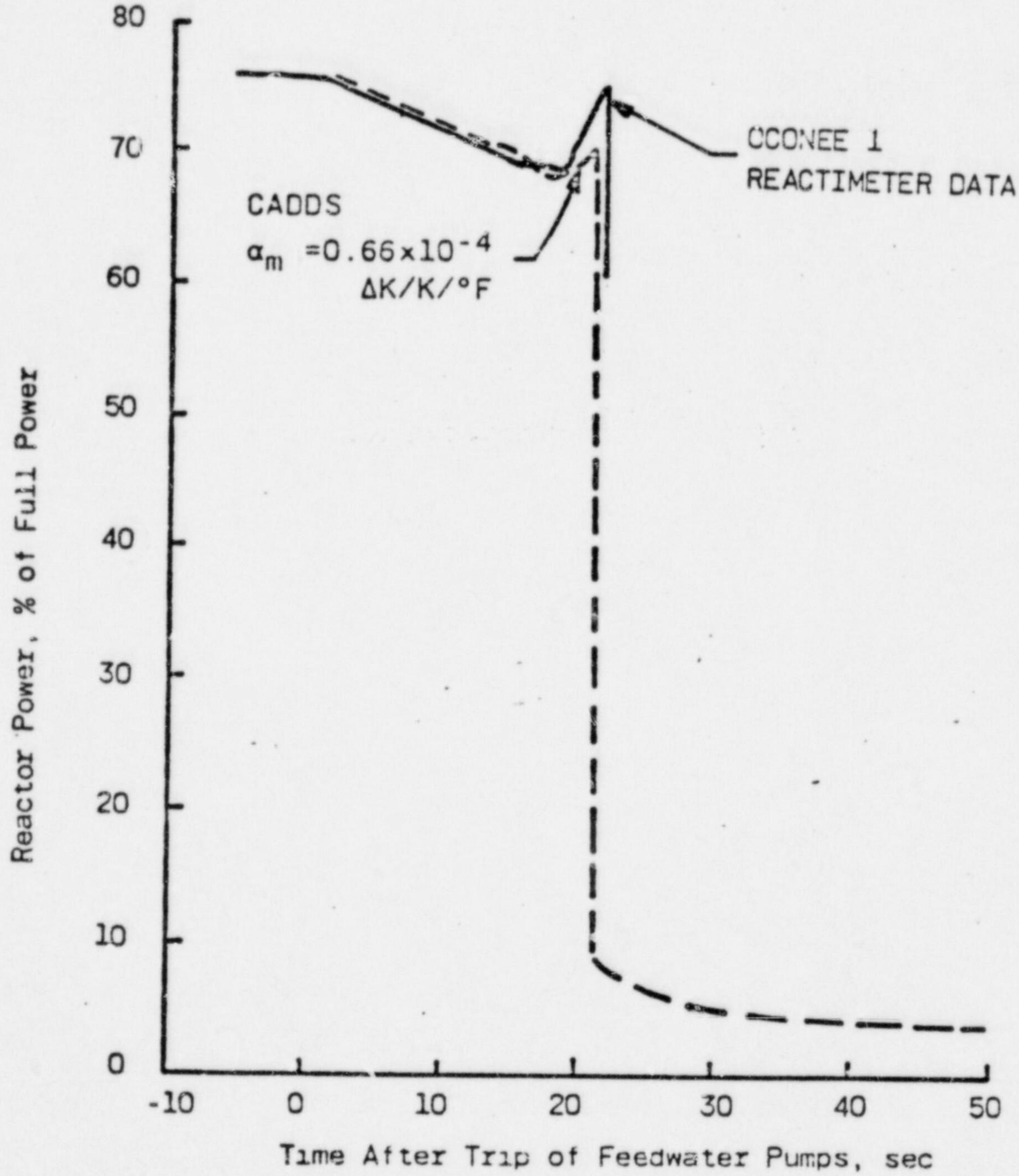

FIGURE 5

TMI-1 GENERATOR TRIP @ 96% POWER

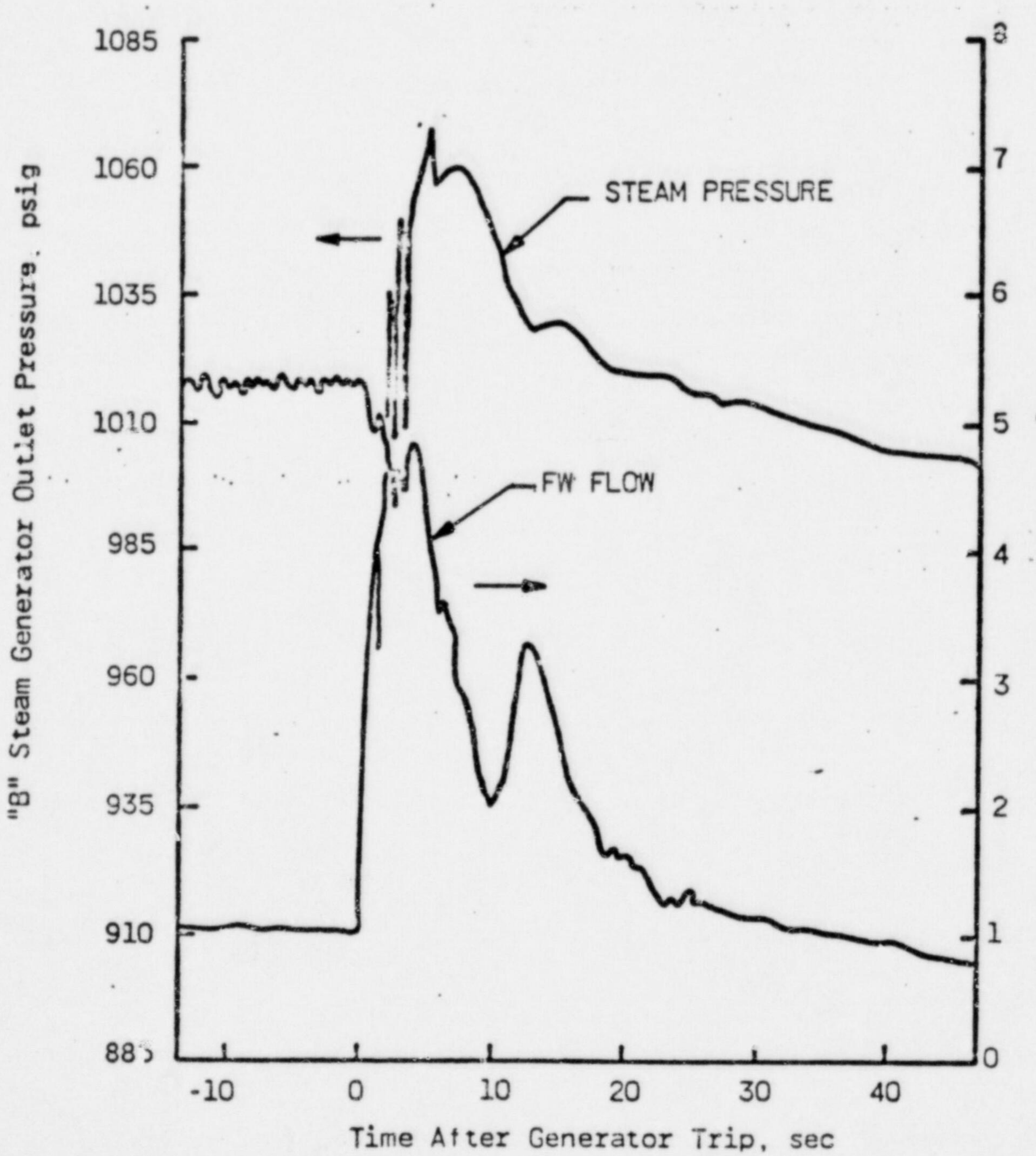

FIGURE 6

TMI-1 GENERATOR TRIP @ 96% POWER

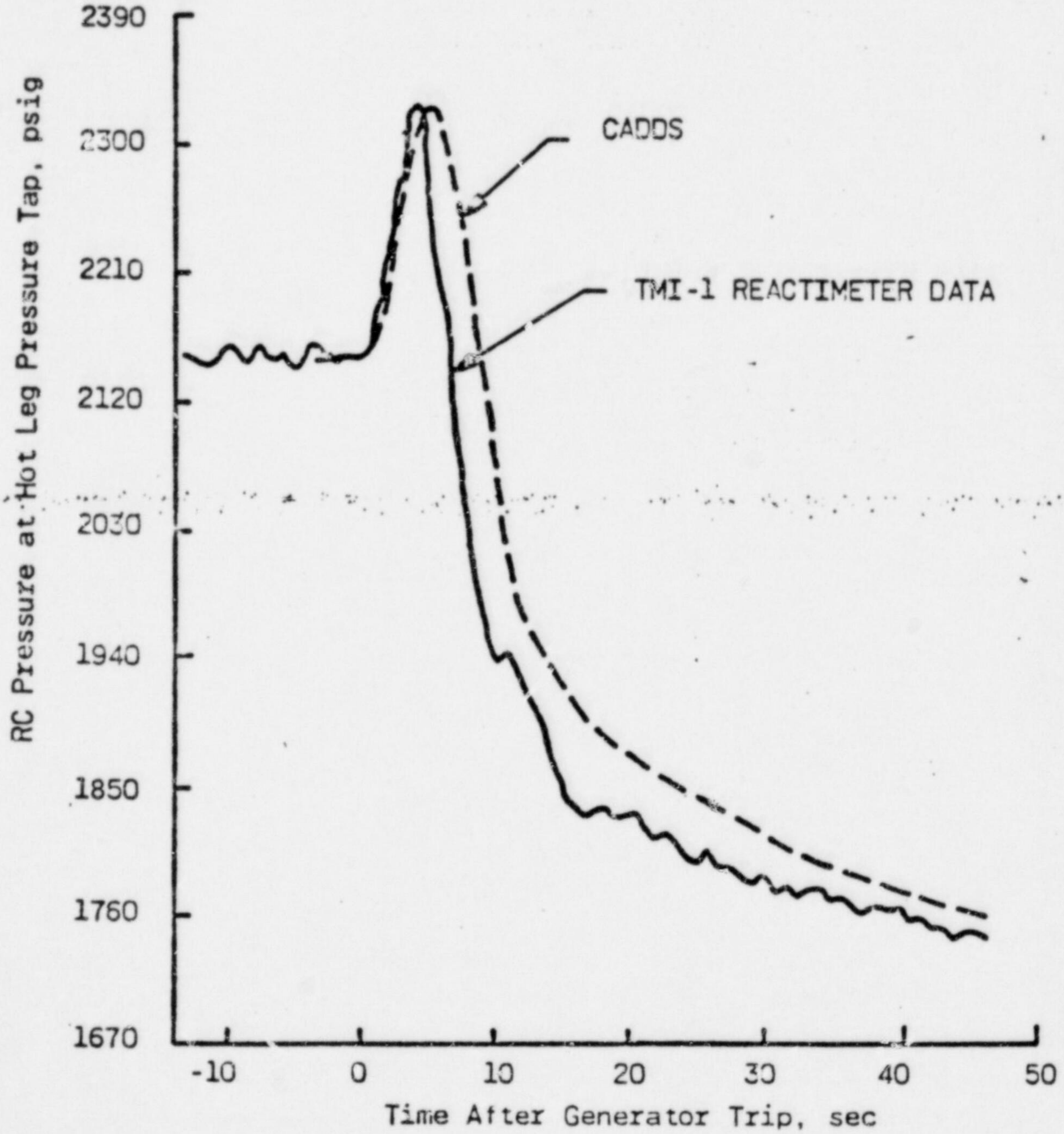

FIGURE 7

TMI-1 GENERATOR TRIP @ 96% POWER

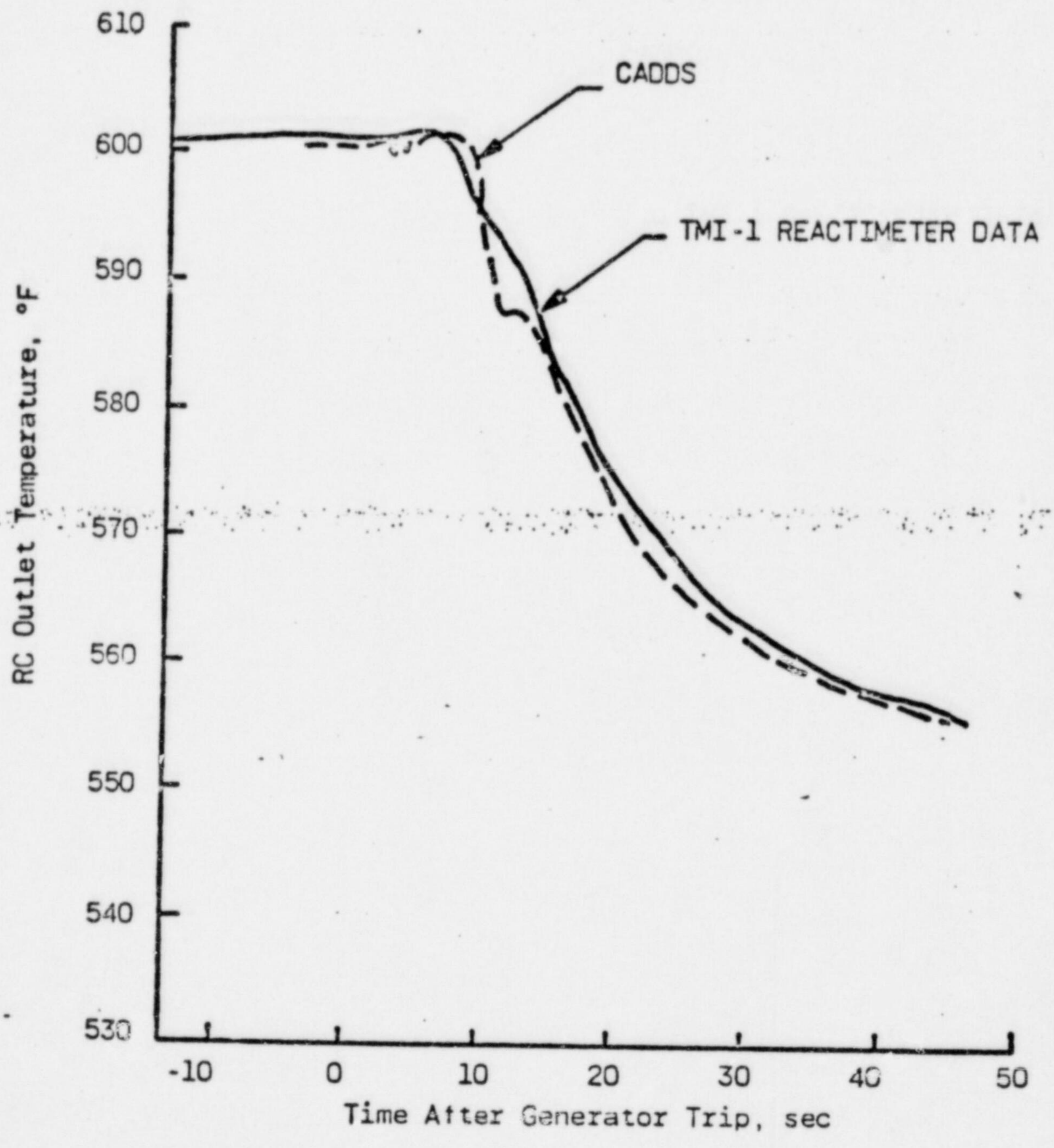

FIGURE 8

TMI-I GENERATOR TRIP @ 96% POWER

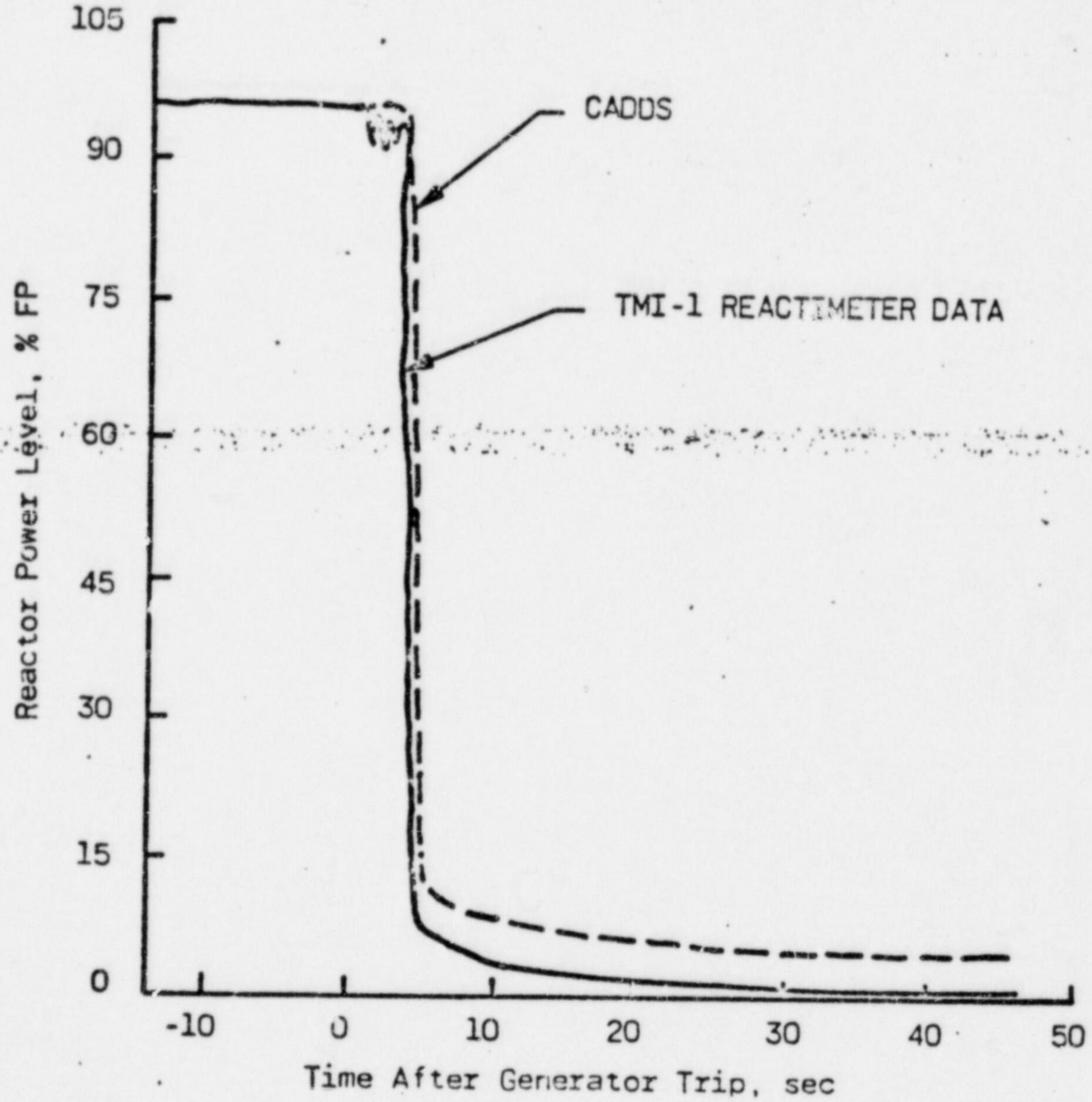

FIGURE 9

TMI-1 GENERATOR TRIP @ 96% POWER