ANALYSIS SUMMARY IN SUPPORT OF
INADEQUATE CORE COOLING GUIDELINES FOR A
LOSS OF RCS INVENTORY

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CONTENTS

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1.0 INTRODUCTION

The TMI-2 Lessons Learned Task Force Status Report, NUREG-0578, contains two sections addressing inadequate core cooling. First, Section 2.1.9 requires that Licensees provide the analysis, emergency procedures, and training needed to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling. Secondly, Section 2.1.3 requires that:

"Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided."

In response to NUREG-0578, an extensive program for inadequate core cooling has been developed. The objectives of this program are as follows:

1. Develop operating guidelines that will allow the reactor operator to recognize and respond to conditions of inadequate core cooling under the following conditions:

   a. Power Operation with portions of the core in DNB.

   b. Loss of RCS inventory without the reactor coolant pumps operating.

c. Loss of RCS inventory with the reactor coolant pumps operating.

d. Loss of the Decay Heat Removal System and Loss of RCS Inventory During Refueling Operations.

e. Loss of natural circulation due to loss of heat sink.

2. Provide recommendations for any additional instrumentation required to indicate inadequate core cooling under the conditions listed above. Included with the recommendations will be:

   a. A description of the functional design requirements for the additional instrumentation.

   b. A description of the Operating Guidelines to be used with the proposed equipment.

   c. A description of the analyses used in developing these guidelines.

   d. Installation schedules for additional instrumentation (if required).

To date, operating guidelines and supportive analyses are complete for the following conditions within the scope of the Inadequate Core Cooling Program:

1. Loss of RCS inventory without the reactor coolant pumps operating.

2. Loss or RCS inventory with the reactor coolant pumps operating.

3. Loss of natural circulation due to a loss of heat sink.

This report is thus a partial submittal. Additional guide-lines/supportive analysis for power operation - DNB condition and refueling operations will be submitted, along with instrumentation-related recommendations in a subsequent report.

For the inadequate core cooling conditions examined herein, guidelines for operator action and a description of the plant behavior, for use in operator training sessions, have been prepared. This information is provided in the revisions to Parts I and II of the Small Break Operating Guidelines References 3, 4, and 5. Supportive analyses and information relating to the expected behavior of the out of core detectors and loop flow measurements during inadequate core cooling conditions is provided in Section 2.0.

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2.0 ANALYSIS SUMMARY

Guidelines for inadequate core cooling and a description
of the plant behavior to support additional operator
training are presented in Parts I and II of the Small
Break Operating Guidelines. These guidelines are
in part based on the operators ability to assess the
transient. Section 2.1 describes the basis and results
of an analysis performed to correlate fuel rod conditions
based on the pressure and temperature conditions of the
RCS. This information provides a means to detect and to
initiate corrective actions for an inadequate core cooling
event. In addition, Section 2.2 and 2.3 provide a qualitative
assessment of the behavior of the source range out of core
neutron detectors and loop flow measurements during periods
of inadequate core cooling.

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2.1 Correlation of Cladding Temperature to Reactor Coolant Pressure-Temperature Condition

During the small break LOCA, without the reactor coolant pumps operating, core cooling is accomplished by keeping the core covered by a steam-water mixture. However, should the core uncover, the uncovered portion of the fuel rod would be cooled only by the steam produced by boiling in the covered portion of the rod. Under this situation, elevated cladding temperatures, which could result in cladding rupture and/or a significant production of hydrogen due to metal-water reaction, would result. The inadequate core cooling guide-lines have been developed to allow the operator to determine if core uncovery has occurred and to define appropriate actions to prevent significant cladding damage and/or hydrogen generation.

The core exit thermocouples, which measure the core outlet fluid temperature, are the most direct indicators available to the operator for determining the core status during a small break LOCA. If these thermocouples indicate superheated fluid conditions, core uncovery is in progress. This behavior allows an assessment of core cooling. To develop operator guidelines, a series of computer calculations were performed to develop a correlation between the measured core outlet fluid temperatures and the peak cladding temperature. Using the above correlation, various levels of inadequate core cooling were defined and appropriate operator actions were developed (see Appendix).

The approach taken for this analysis was to non-mechanistically reduce the Reactor Coolant System Inventory in order to develop the correlation between clad temperature and outlet fluid temperature. Core decay heat, based on 1.2 times the 1971 ANS standard for infinite operation, at 200 seconds after scram was utilized for this evaluation. Core uncovery for small breaks should not occur any earlier than 200 seconds; thus this assumption will maximize power and the peak cladding temperature in the uncovered portion of the fuel rod. Five power shapes, given in Figures 1 through 5, were analyzed to cover a reasonable spectrum of core conditions and to ensure that an outlet fluid temperature indication used for operator action would correlate to a peak cladding temperature less than a selected value. Radial peaking factors were chosen such that the maximum local power was equal to the LOCA limit value.

The FOAM[1] code was utilized to predict the peak cladding temperature and core exit fluid temperature. Table 1 provides a summary of the cases analyzed. A brief outline of the procedure utilized in the FOAM code is as follows:

1. Using the input total core power, axial power shape, system pressure, and solid water level, the core mixture height is determined. This mixture height is based on a radial peaking factor of 1.0 and reflects the average core swell level.

2. Assuming that all decay heat is removed in the covered portion of the fuel rod, the core steaming is calculated. As with the core mixture level, the steaming rate is based on a radial peaking factor of 1.0.

3. Using the average core steaming rate, the fluid temperature, in the uncovered portion of the fuel rod, for the hot pin is computed. This calculation uses the input radial peaking factor. In determining the fluid temperature, as a function of elevation in the core, it is assumed that all the core energy is removed by the steam.

4. Using the core steaming rate and the local fluid properties in the uncovered portion of the fuel rod, a surface heat transfer coefficient, based on the Dittus-Boelter correlation[2], is calculated.

5. Steady-State, hot pin cladding temperatures are then determined based on the local fluid properties obtained by Step 3 and the surface heat transfer coefficient obtained by Step 4.

Figures 6 and 7 summarize the results of the calculations performed for the five power shapes analyzed. These curves correlate the calculated core exit fluid temperatures for peak cladding temperatures of 1400F and 1800F, respectively. From these results, a bounding set of curves, shown on Figure 8, was obtained for use in the operating guidelines.

The small break operating guidelines include a provision
for prompt tripping of the RC pumps upon receipt of a
low pressure ESFAS signal. If the RC pumps cannot be
tripped, core cooling will be provided by the continued
forced circulation of fluid throughout the RCS. There
are two ways that inadequate core cooling can occur for
a small break with the RC pumps operating. First, with
the RC pumps operating, the fluid in the RCS can evolve
to a very high void fraction. Should the RC pumps trip
at a time when the system void fraction is greater than
approximately 70%, the amount of water left in the RCS
would not be sufficient to keep the core covered and
an inadequate core cooling situation may exist. For
this situation, the analysis described in the previous
paragraphs apply directly.

Secondly, if little or no ECCS flow is provided to the RCS,
the fluid being circulated by the RC pumps will eventually
become superheated steam due to the continued energy addition
to the fluid provided by the core decay heat. Under these
circumstances, an inadequate core cooling situation will
start to exist. Due to the forced circulation of the
superheated steam through the core under these conditions,
even with only one RC pump operating, the heat removal process
is better than the steam cooling mode described for the pumps
off situation. Thus, the indications and operator responses
determined for no RC pumps operating are appropriate for
controlling an inadequate core cooling situation with the
RC pumps operating.

2.2 Excore Neutron Detector Behavior

The excore source range neutron detectors are available to provide indications of anomalous incore behavior, although they cannot uniquely quantify inadequate core cooling. A departure from expected response is anticipated for conditions that lead to inadequare core cooling. Therefore, the behavior of the source range detectors may, in some instances, be used to confirm other indications of inadequrte core cooling.

The behavior of the neutron flux following a reactor trip is monitored and recorded by the source range count rate instrumentation following reactor trip. An example of this trace is presented in Figure 5. Normally the detector count rate falls at rates characteristic of the various mechanisms of neutron production that exist following the trip. During a trip, the neutron flux undergoes a prompt decrease associated with the negative reactivity of the control rods. Following the prompt decrease the neutron flux decays with an 80 second period, characteristic of the decay of the longest-lived delayed neutron group. The neutron flux continues to decay at this rate until it approaches the level produced by neutron sources and subcritical multiplication. Two types of neutron sources are important in the determination of neutron level following delayed neutron decay, namely:

(1) Fixed startup sources

(2) Natural sources

The most important of the natural sources is the photoneutron production (γ, n) resulting from the interactions of high energy fission product gammas with deuterium (D_2O). The photoneutron level decreases consistent with the decay of fission products (primarily Kr^{88} and La^{140}).

The source range detectors will respond to a decrease in water density through several mechanisms.

(1) Reduced water density will <u>enhance</u> neutron transmission from core to detectors.

(2) Reduced water density will <u>decrease</u> the neutron sources (i.e., photoneutrons from the γ, n reaction in D_2O).

(3) The reduced water density will <u>decrease</u> the core multiplication factor due to the negative moderator coefficient.

Scoping calculations with a 1-dimensional transport code have shown that the dominant effect is the improved neutron transmission from core to detector. Thus, the source range detector count rate will increase or the rate of decrease will be altered, depending on the magnitude of the change in water density, even though the core is becoming more subcritical and the photoneutron source strength is decreasing.

The source range detectors cannot unambiguously detect inadequate core cooling because voiding in different regions of the core will have different effects on the excore flux

levels. If the reactor coolant pumps are operating while the primary system is partially voided, the steam voids are expected to be evenly distributed. Under these conditions, the source range detectors are expected to read a higher than normal count rate. If the reactor coolant pumps are not operating, the steam and water will separate. In order for the core to be inadequately cooled, the water level must drop below the top of the core. When this happens the source range detector count rate should increase. However, as the level continues to drop, the continued decrease in the quantity of available water could reduce the photoneutron production and subcritical multiplication to the point where the source range detector output could begin to decrease. Because of this complex behavior, the source range detector should only be used to confirm other indications of inadequate core cooling.

A correlation has been made between the source range detector response and several key events that followed the TMI-2 accident on March 28, 1979. This correlation is shown in Figure 10. The following is a discussion of the significant events referenced to the source range detector behavior shown in Figure 10. As was discussed above, there is considerable uncertainty in interpreting the behavior of the source range detectors. Any interpretation should therefore be used with caution.

1. Time 0400 - The neutron power in the reactor core de-
creased rapidly to the source range, as is typical of reactor trip.

2. Time 0408 - Emergency feedwater was established to the
steam generators approximately 8 minutes after reactor trip. The
PORV had stuck open, and it continued to relieve reactor coolant.
During the first 3 hours following reactor trip, high-pressure
injection (HPI) was initiated automatically several times as
system pressure decreased. Each time the automatic system activated,
the plant operators took manual control of the HPI system.
Information on HPI flow rates and times of injection was not
available and had to be inferred from makeup tank levels, operator
interviews, and the alarm printer.

3. Time 0420 to 0540 - Reactor coolant pumps were circulating
a saturated, two-phase flow. The void fraction was increasing
due to loss of coolant through the PORV. From the SR plot,
circulating two-phase flow can be inferred from the noisy, gradually
increasing signal prior to point A. The noise in the signal
is due to the turbulent action in the two-phase fluid. A decrease
in moderator density results in an increase in SR level due to
reduced attentuation. At 0514, the RC pumps in loop B were turned
off by the operator, but no fuel damage is believed to have occurred
during this time interval since calculations have shown that the
circulating fluid from loop A provided adequate heat removal.

4. Time 0540 - The RC pumps in loop A were turned off by
the operator. As a result, the flow decreased rapidly, with a
corresponding separation of steam and water. The calculated water
level was at the bottom of the core inlet pipes, which are 3
feet above the top of the active core. The calculation was based

on coolant quality just prior to trip and was inferred to consist of 30 to 50% voids. This inference is consis.ent with gentile tube flow measurements and source range data.

5. Time 0540 to 0615 - The water level in the core gradually decreased between points B and D. The change in slope of the SR detector level at point C was interpreted to indicate the start of detector uncovering. This is supported by the reflux boiler calculation and the coolant loss through the PORV. During this time, the RCS was acting as a reflux boiler; that is, steam was being created in the core region, condensing in the steam generators, and returning to the core by the cold legs. The return of cold water to the reactor vessel was verified by the subcooled temperatures observed in the cold legs during this period. Reactor coolant continued to be lost from the system through the PORV.

6. Time 0615 to 0654 - The block valve upstream of the PORV was closed at 0615, preventing further loss of reactor coolant. The core was approximately 50% uncovered at this point and remained near this level until 0654. During this time interval, system pressure increased rapidly from 620 to 2150 psig. System pressure was then manually regulated using the block valve.

7. Time 0654 - Based on alarm messages, it was concluded that RC pump '-B was started and ran either intermittently or continuously for approximately 19 minutes. The core coolant level increased with at most 2 to 3 feet of the core remaining uncovered. This inferred level is supported by some incore thermocouple readings which came on scale and read below 700F. In addition, the SR levels from E to F indicate a rapid increase in coolant density.

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8. <u>Time 0654 to 0724</u> - The open PORV block valve (0713), core boil off, and the turning off of RC pump 2-B dropped the reactor coolant level so that approximately 4 to 5 feet of the core were uncovered during this time period.

9. <u>Time 0724</u> - The alarm printer indicated that high-pressure injection of about 1000 gpm was started and continued for about 2 minutes before the operator took control. After this time, HPI flow is uncertain but apparently was at least reduced in flow. During this period, the core was partially refilled until only 2 to 3 feet of the core was uncovered. The temperature in the peripheral incore thermocouples decreased rapidly to the 600-700F range.

10. <u>After 0724</u> - The water level in the core gradually increased with minor perturbations. This was determined from some peripheral thermocouples that came back on scale, indicating the temperature was below the saturation level of 600F. Core covering was further substantiated by the return of the SR detector readings to corrected normal shutdown levels. At about 0730 the PORV was closed as determined by the RC pressure increase.

2.3   Behavior of Loop Flow Indication

Gentile flow tubes are used to measure mass flow in each
loop. For solid water conditions, the reactor coolant pumps
will act as constant volume pumps with the mass flow changing
as the density of the water varies. If steam voids begin
to form in the loop, the reactor coolant pumps will still
act as constant volume pumps with some degraded performance.
The formation of steam voids in the loop reduces the fluid
density and consequently the mass flow in the loop. For
this two-phase flow condition, the indicated flow will no
longer accurately represent the mass flow   However, the
indicated flow will follow the trend of a decreasing
measured flow with an increasing void fraction. Figure 11
is the measured loop flow during the TMI-2 accident. This
curve illustrates the expected behavior of the measured
loop flow during two phase flow conditions with a gradually
increasing void fraction.
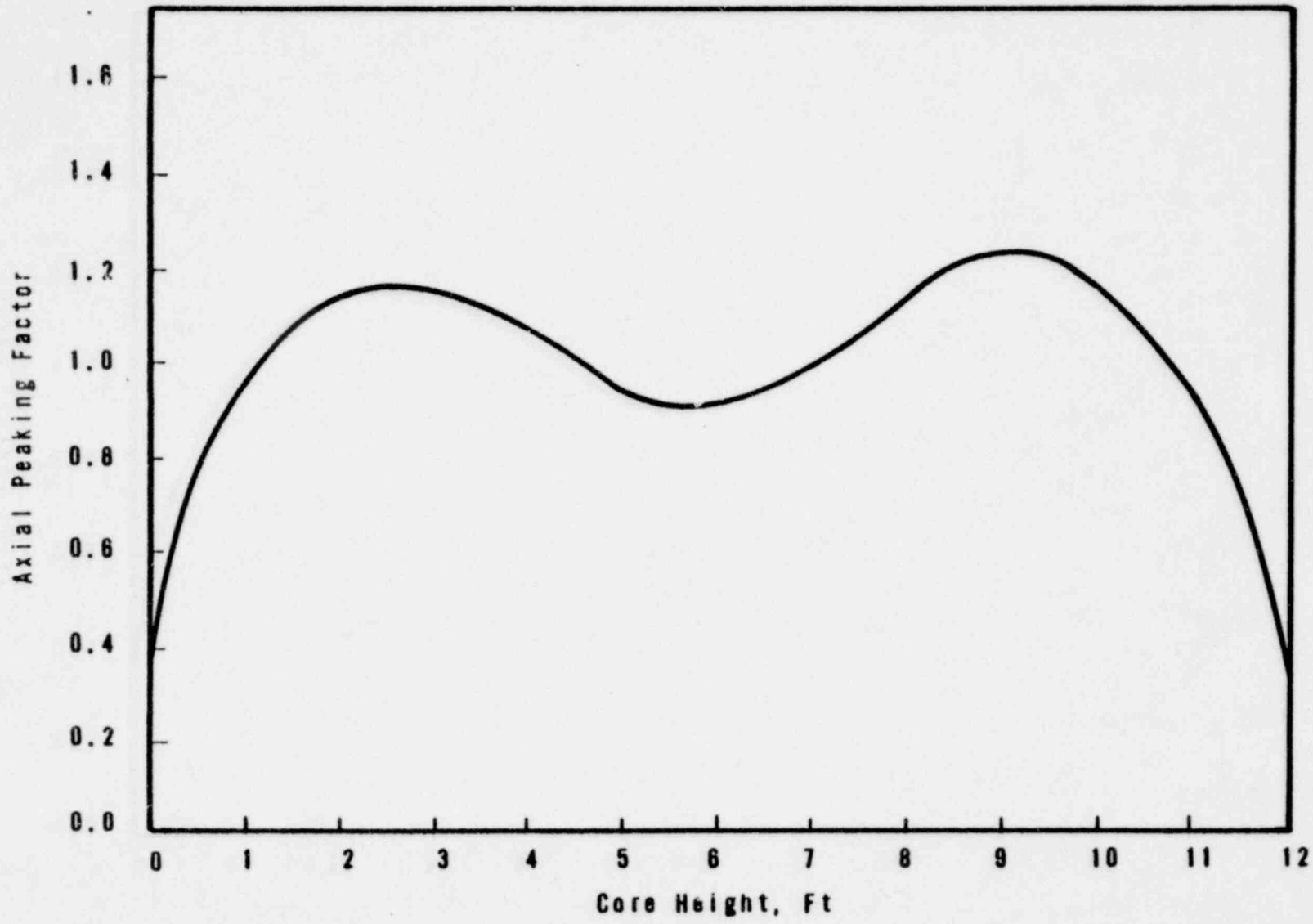
FIGURE 1. LOCA LIMITS POWER SHAPE - 6 FT PEAK
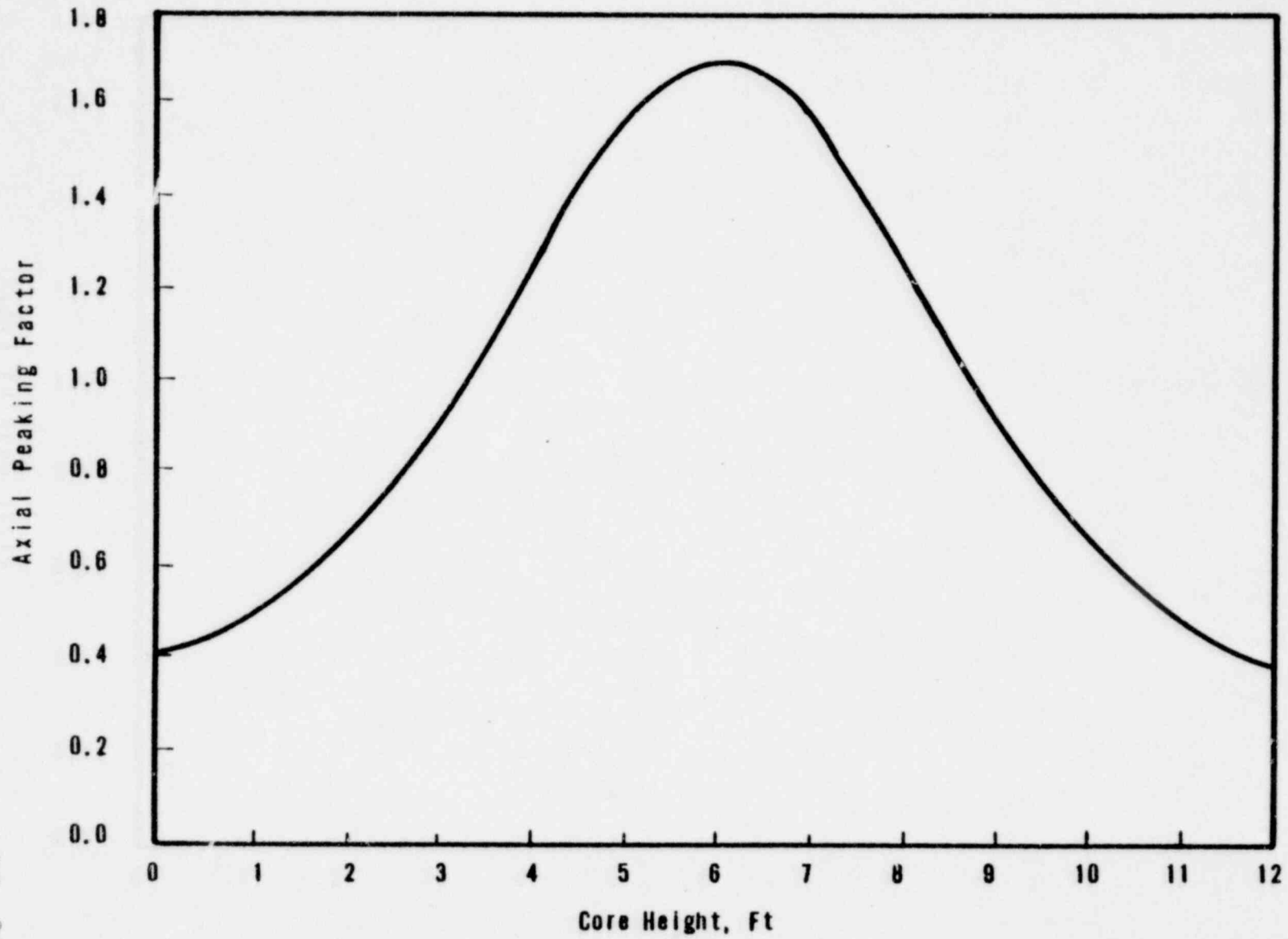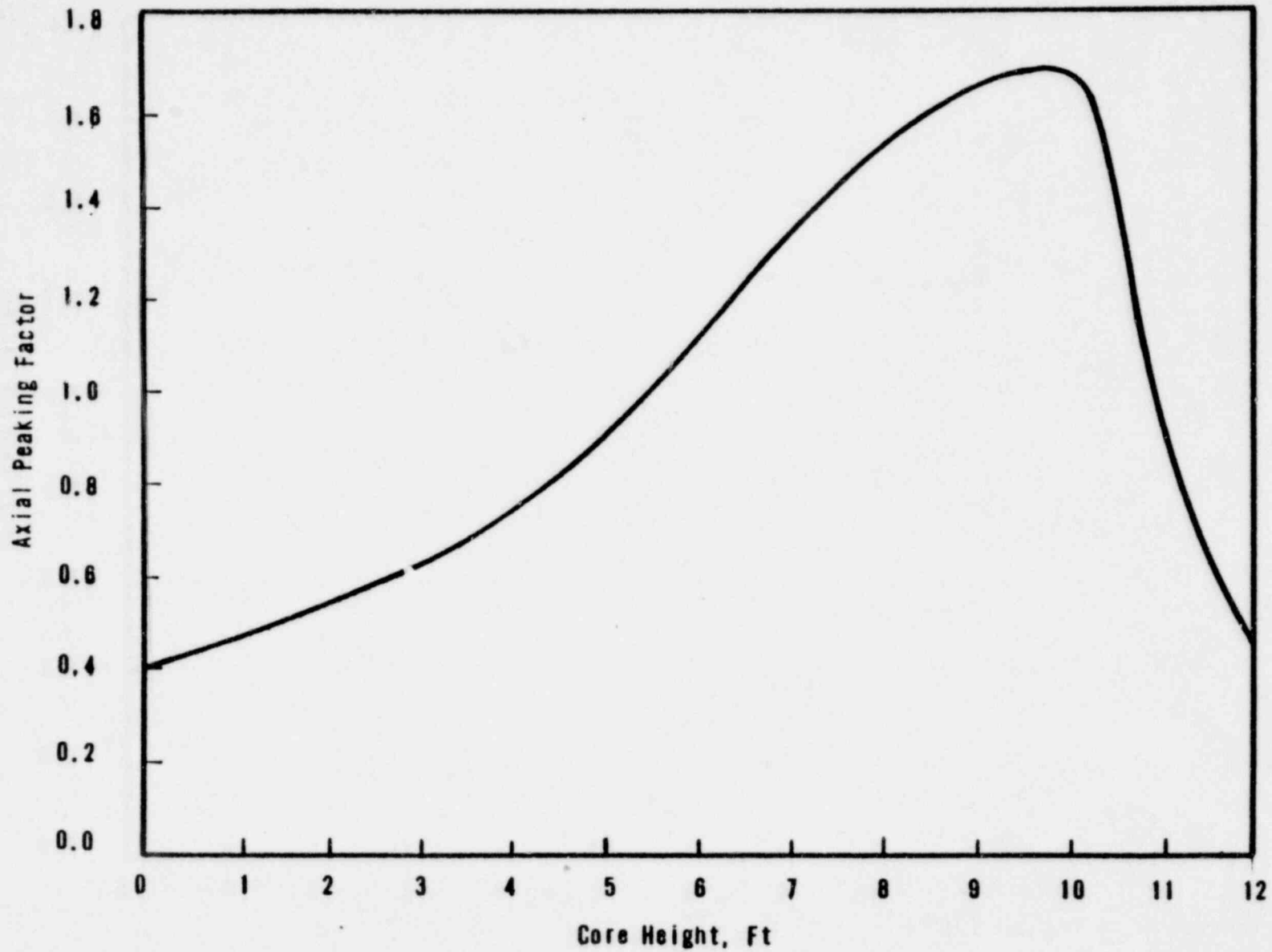
FIGURE 2. SADDLE SHAPE POWER CURVE-UNEQUAL PEAKS
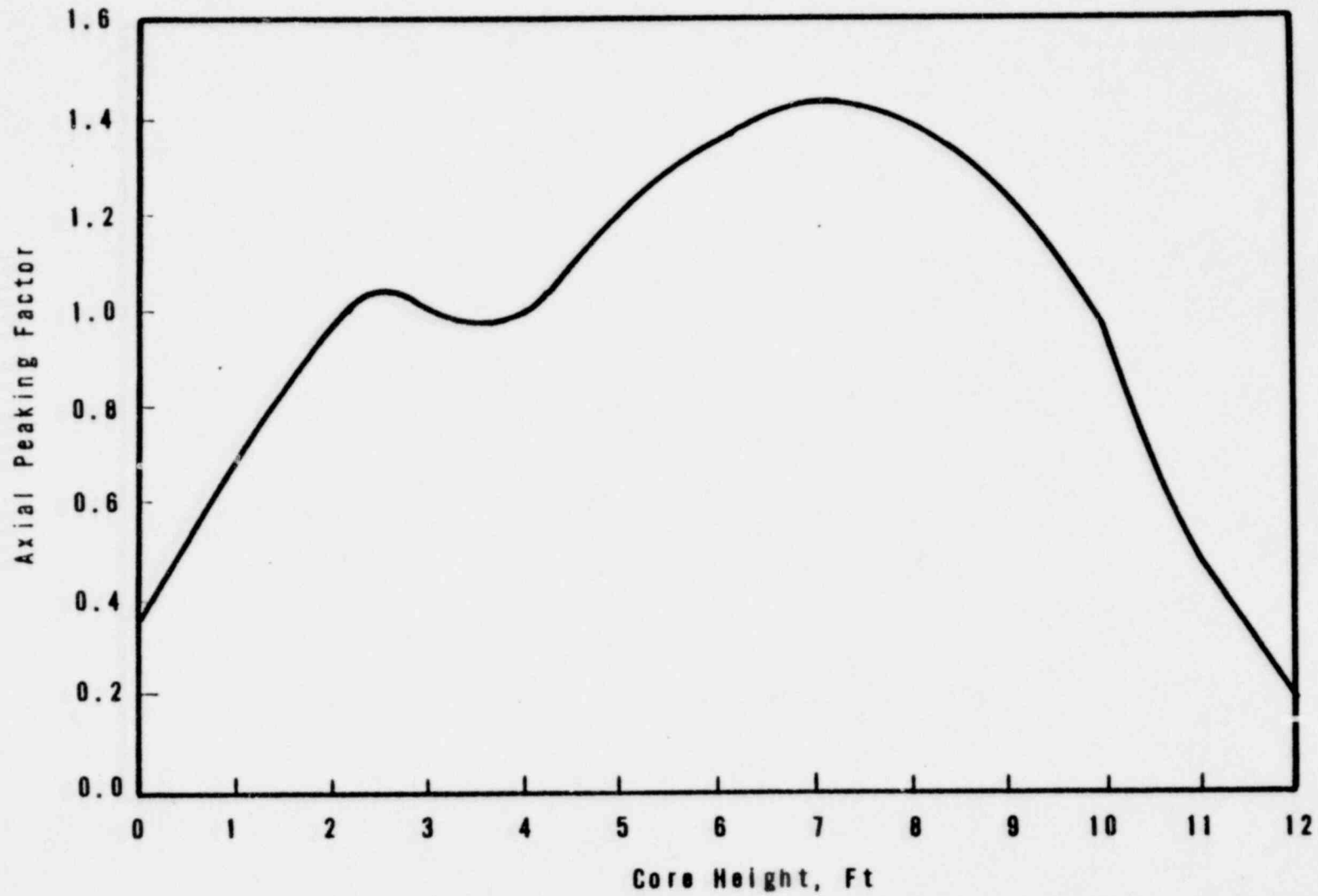
FIGURE 3. SADDLE SHAPE POWER CURVE-EQUAL PEAKS

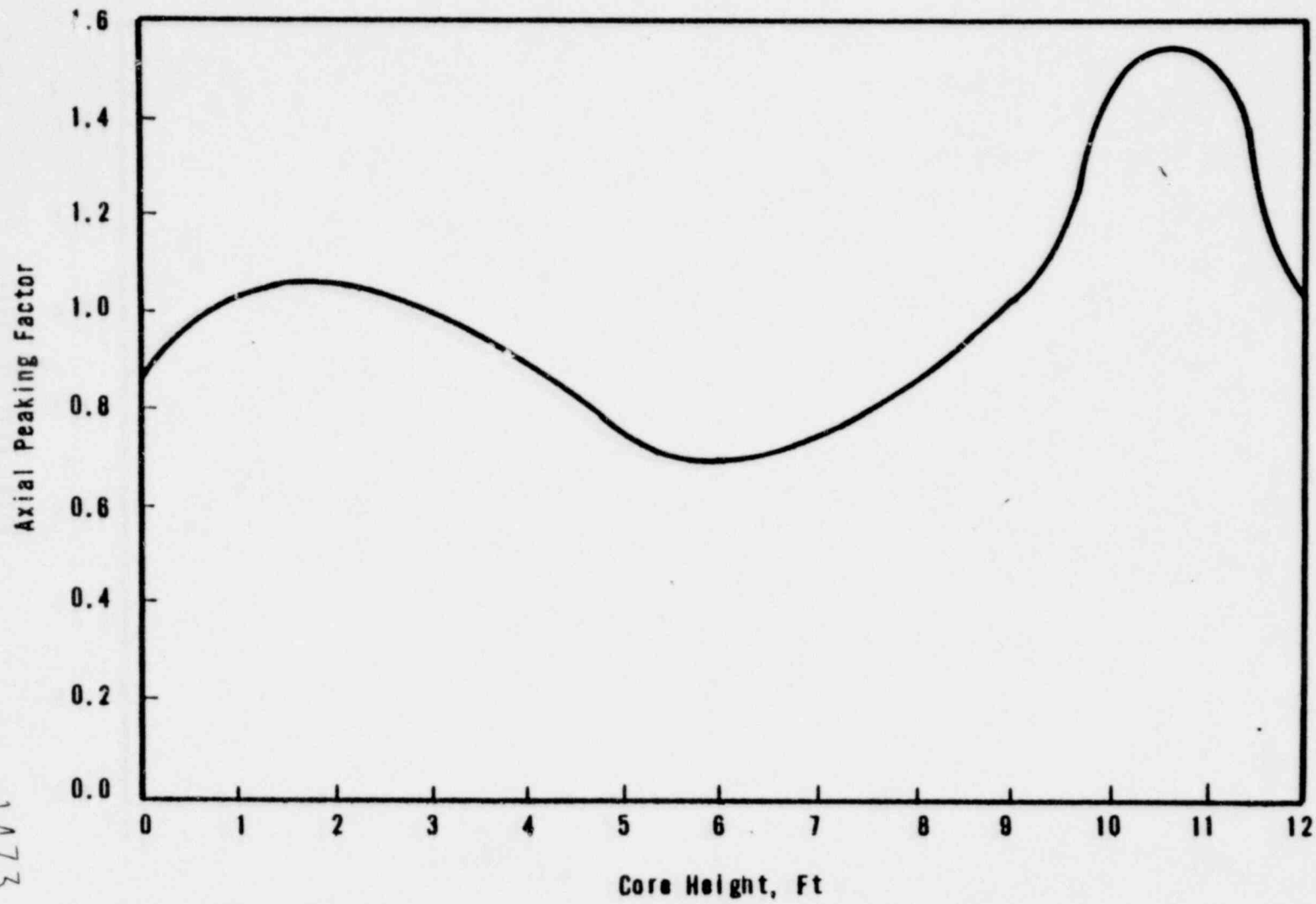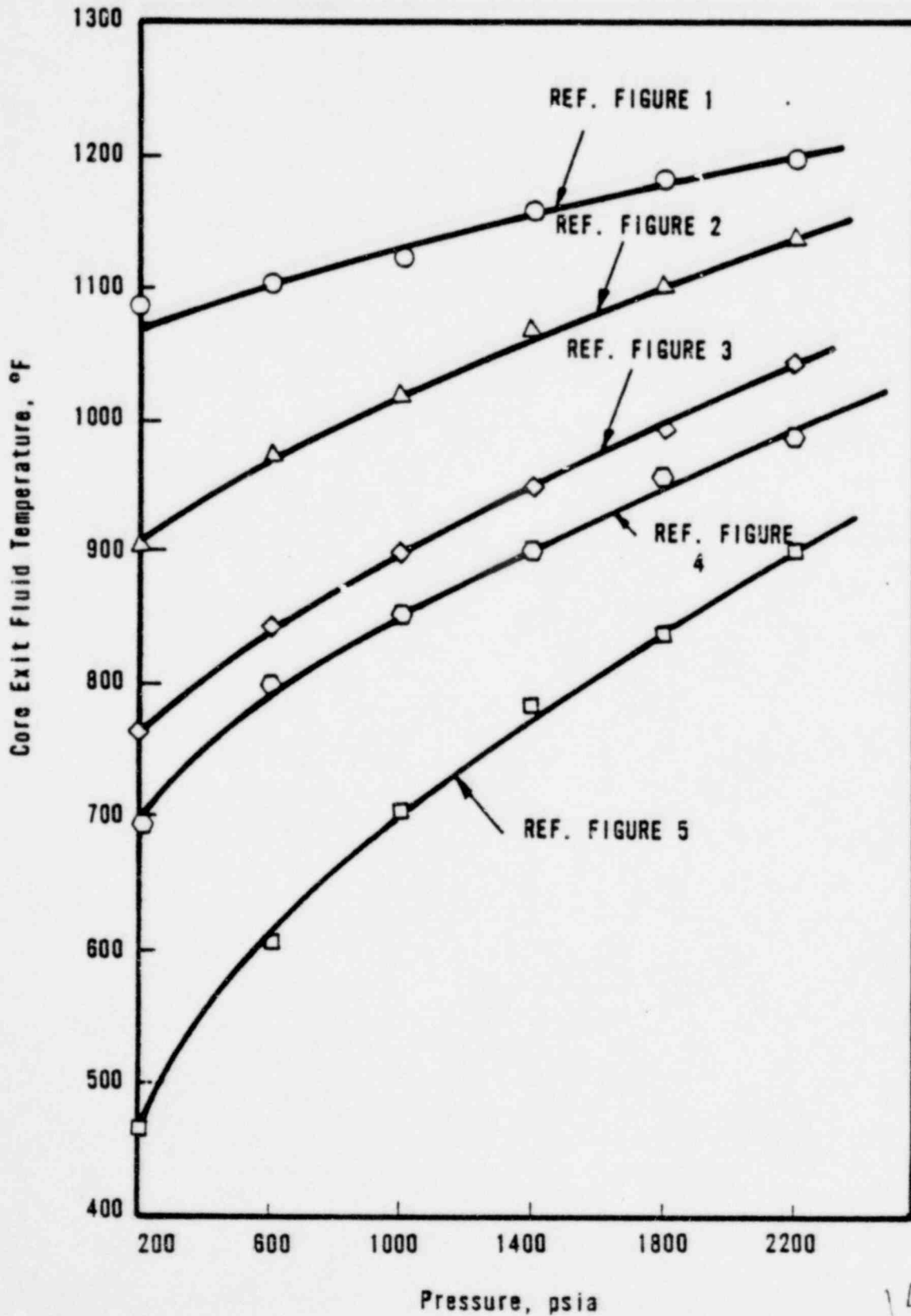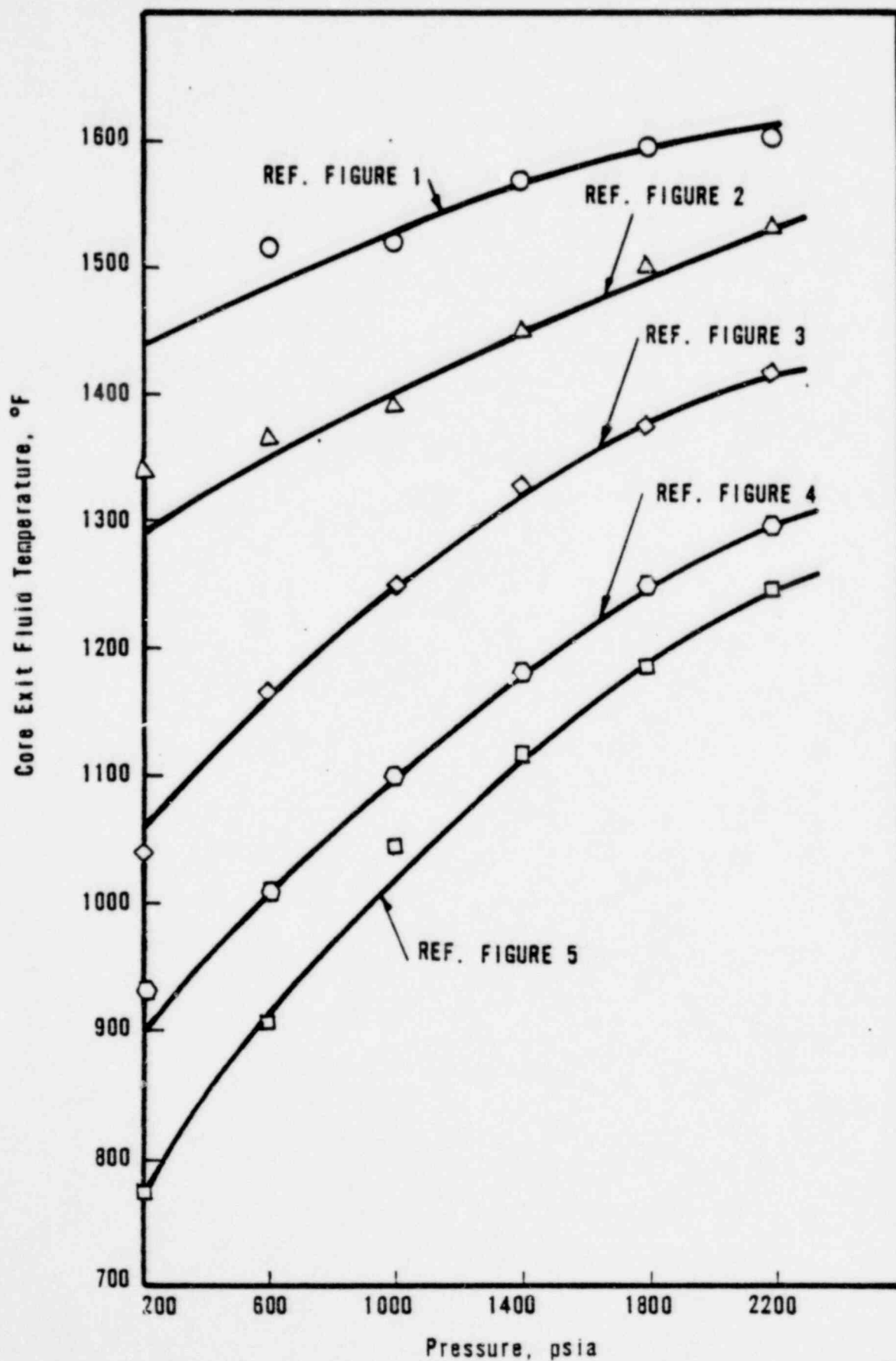FIGURE 4.   LOCA LIMITS POWER SHAPE - 10 FT PEAK

FIGURE 5.   SMALL BREAK POWER SHAPE

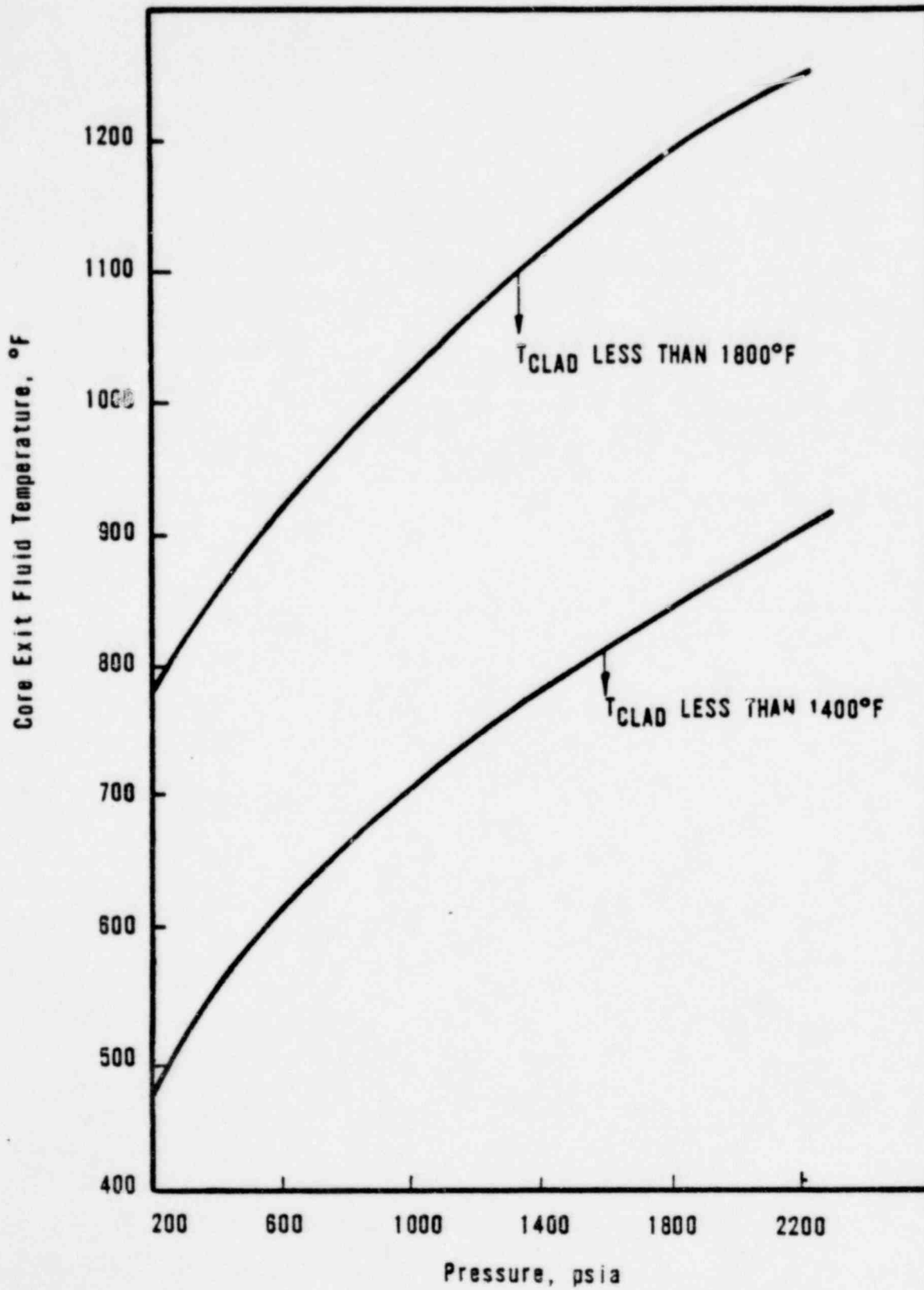Figure 6 RCS PRESSURE VS CORE EXIT FLUID TEMPERATURE FOR 1400°F CLAD TEMPERATURE LIMIT

Figure 7   RCS PRESSURE VS CORE EXIT FLUID TEMPERATURE FOR 1800°F CLAD
TEMPERATURE LIMIT

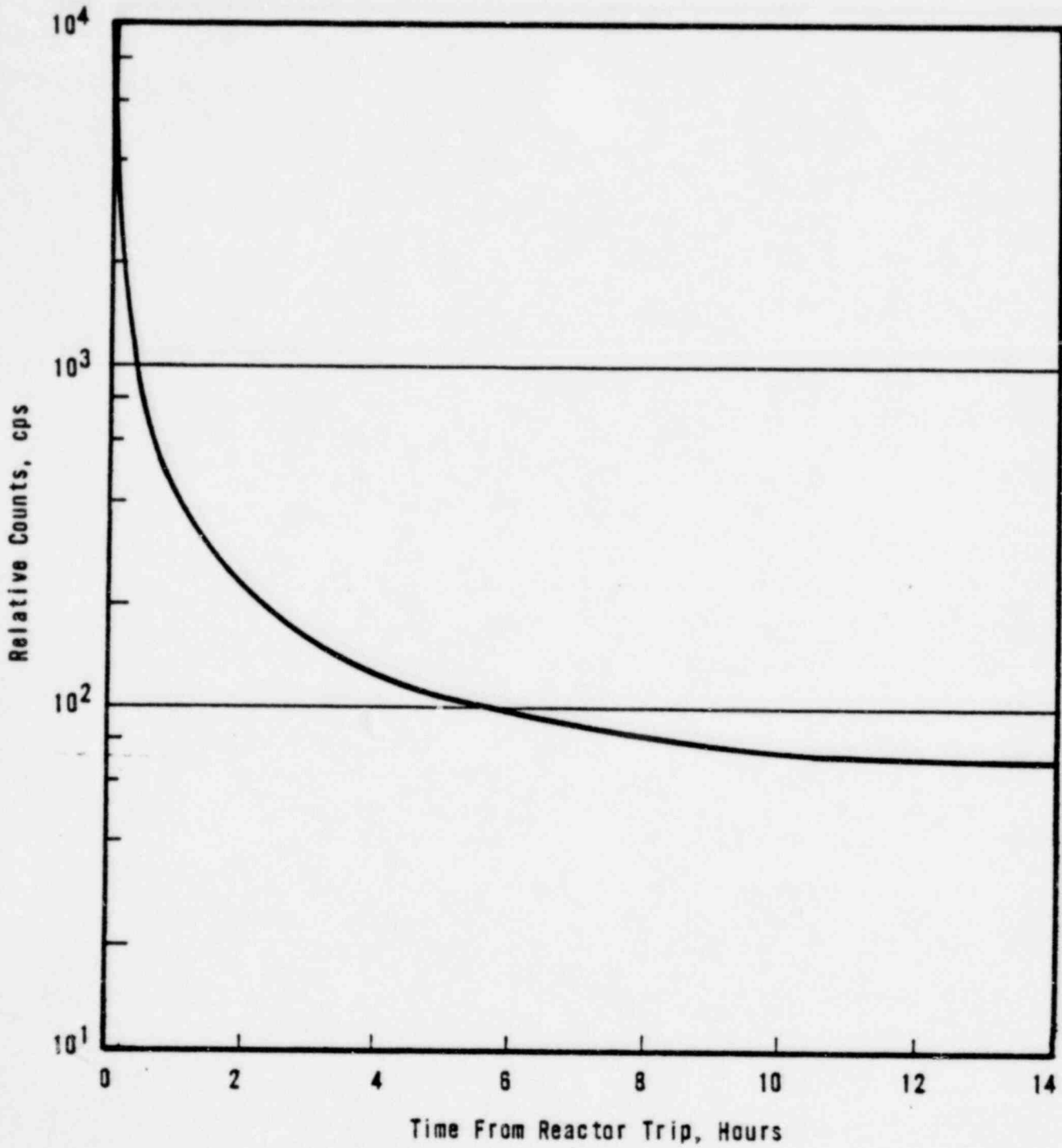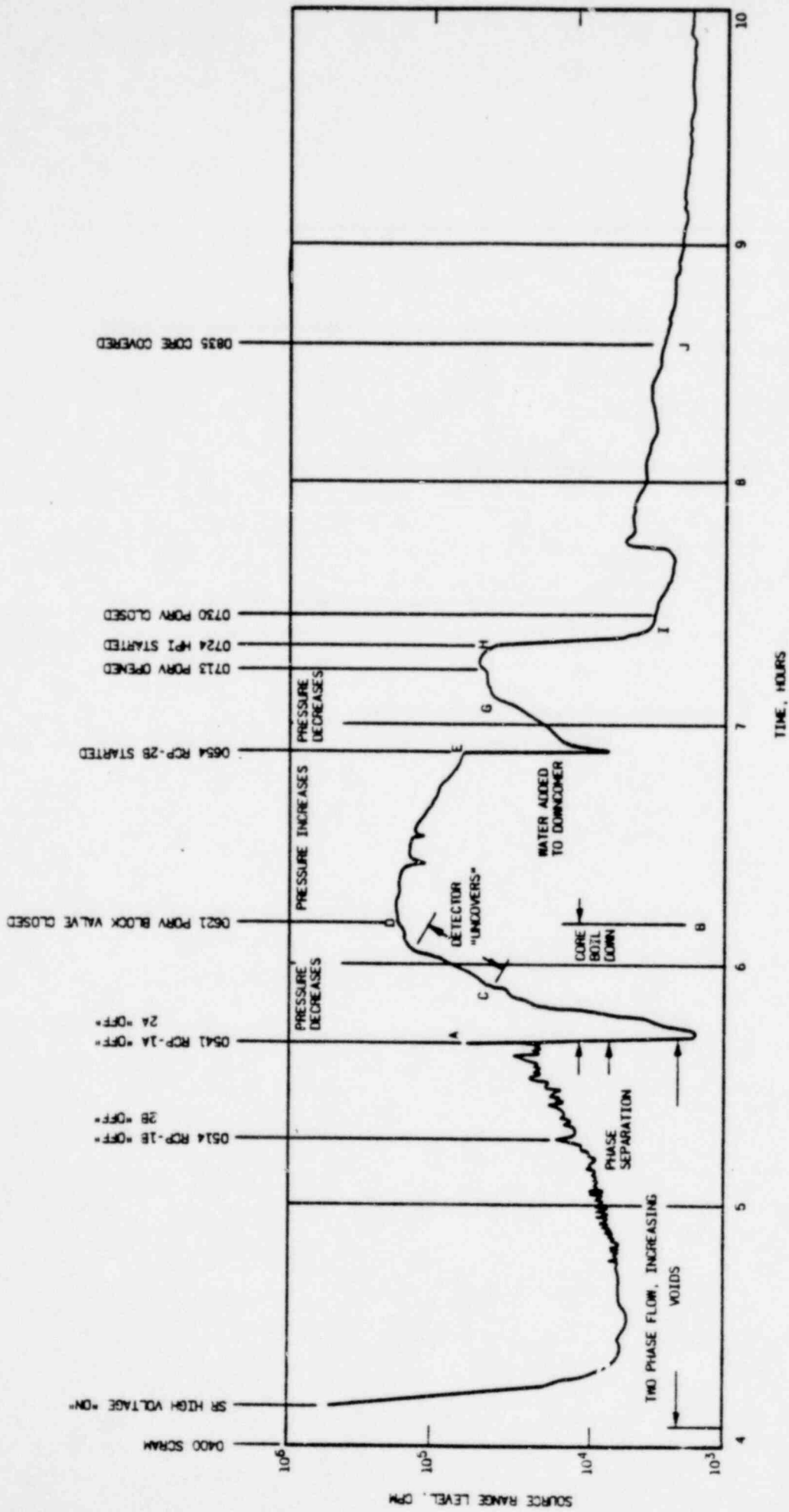# Figure 8 CORE EXIT FLUID TEMPERATURE INDICATION TO LIMIT CLAD TEMPERATURE

Core Exit Fluid Temperature, °F

T_{CLAD} LESS THAN 1800°F

T_{CLAD} LESS THAN 1400°F

Pressure, psia

FIGURE 9. SOURCE RANGE TRACE FOLLOWING
REACTOR TRIP (TYPICAL)

CORRELATION BETWEEN SOURCE RANGE LEVEL
AND EVENTS FOLLOWING TMI-2 SCRAM 3/28/79

Figure 10

TIME, HOURS

SOURCE RANGE LEVEL, CPM

0400 SCRAM

SR HIGH VOLTAGE "ON"

0514 RCP-1B "OFF"
2B "OFF"

0541 RCP-1A "OFF"
2A "OFF"

0621 PORV BLOCK VALVE CLOSED

0654 RCP-2B STARTED

0713 PORV OPENED
0724 HPI STARTED
0730 PORV CLOSED

0835 CORE COVERED

PRESSURE DECREASES

PRESSURE INCREASES

PRESSURE DECREASES

DETECTOR "UNCOVERS"

WATER ADDED TO DOWNCOMER

CORE BOIL DOWN

PHASE SEPARATION

TWO PHASE FLOW, INCREASING VOIDS

PRIMARY SYSTEM FLOW

Figure 11

CHANNEL06   RC FLOW LOOP A TEMP. COMP.
CHANNEL13   RC FLOW LOOP B TEMP. COMP.

Loop B

TIME (HRS)

MPPH

Table 1. Summary of FOAM Input

Input parameter	Description
Core power	1.2 X ANS at 200 sec for 2772 MWt
Core hydraulics	177 FA core with 15 X 15 fuel
Axial power shapes	5 shapes (Ref. Figures 1 through 5)
Initial core water level	2 through 10 feet
Core pressures	600, 1000, 1400, 1800, 2200 psia
Core inlet enthalpy	h_{sat}
Radial peaking factor	Based on LOCA limit for maximum local power

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2. Babcock & Wilcox Revisions to THETA1-B, a Computer Code for Nuclear Reactor Core Thermal Analysis (IN-1445), BAW-10094, Babcock & Wilcox, Lynchburg, Virginia, April 1975.

3. Operating Guidelines for Small Breaks for Oconee 1, 2, 3; Three Mile Island-1, 2; Crystal River-3; and Rancho Seco, Emergency Operating Specification 69-1106001-00.

4. Operating Guidelines for Small Breaks for Arkansas Nuclear One-1, Emergency Operating Specification 69-1106002-00.

5. Operating Guidelines for Small Breaks for Davis Besse-1, Emergency Operating Specification 69-1106003-00.

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