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ANALYSIS SUMMARY IN SUPPORT OF
INADEQUATE CORE COOLING GUIDELINES

BABCOCK AND WILCOX

CONTENTS

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 inadequate core cooling conditions described in Item 1 above.

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 Part I and II of the Small Break Operating Guidelines (References 3, 4 and 5) in Part I and II of the Operating Instruction for a Loss of Decay Heat Removal System and in Part I and II of the Operating Instruction for Normal Power Operation.

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.

2.0 ANALYSIS SUMMARY FOR LOSS OF RCS INVENTORY

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.

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 guidelines 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 inadequate 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 decreased 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 consistent 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 2-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.

8. Time 0654 to 0724 - 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. Time 0724 - 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. After 0724 - 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

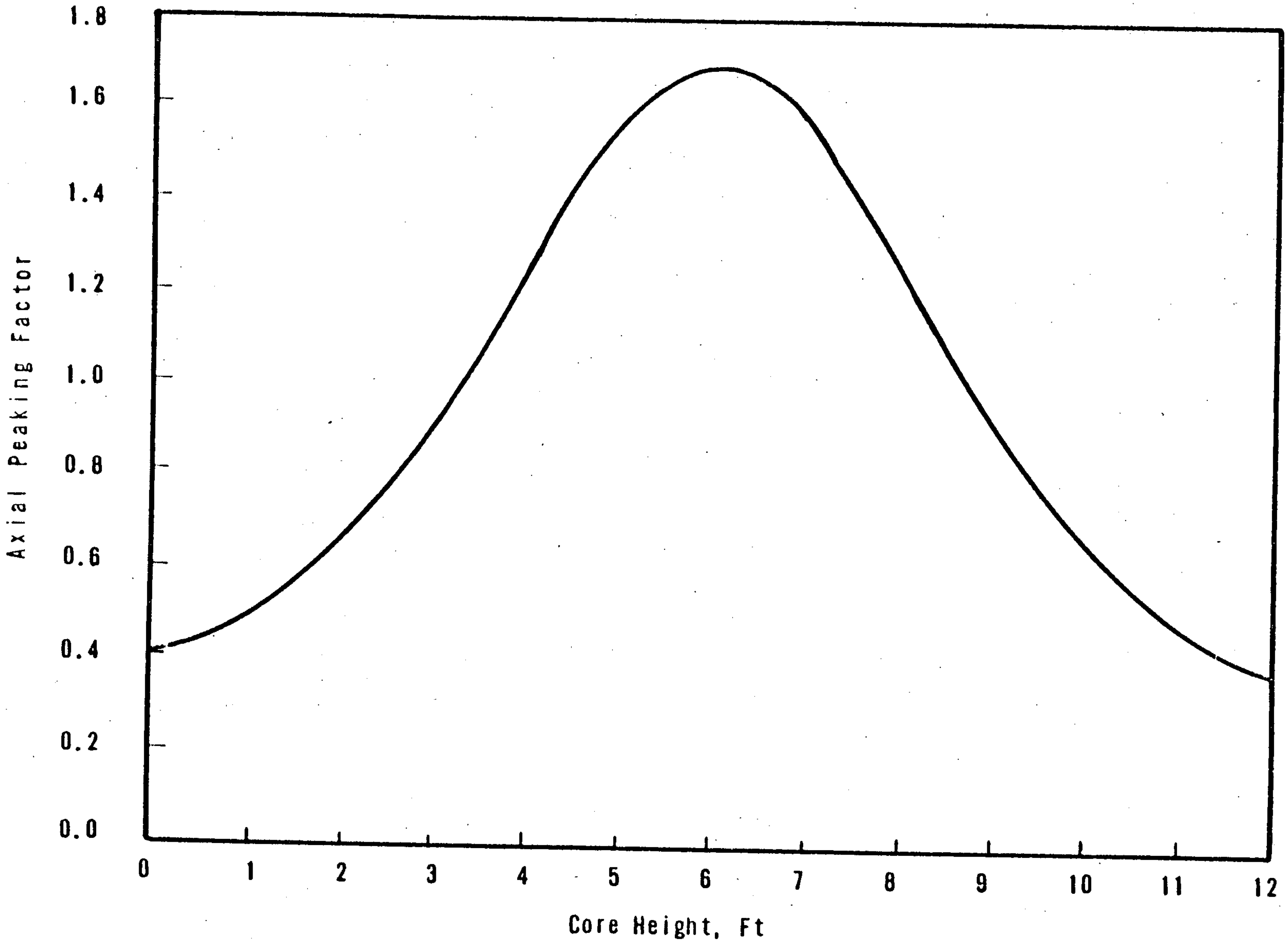


FIGURE 2. SADDLE SHAPE POWER CURVE-UNEQUAL PEAKS

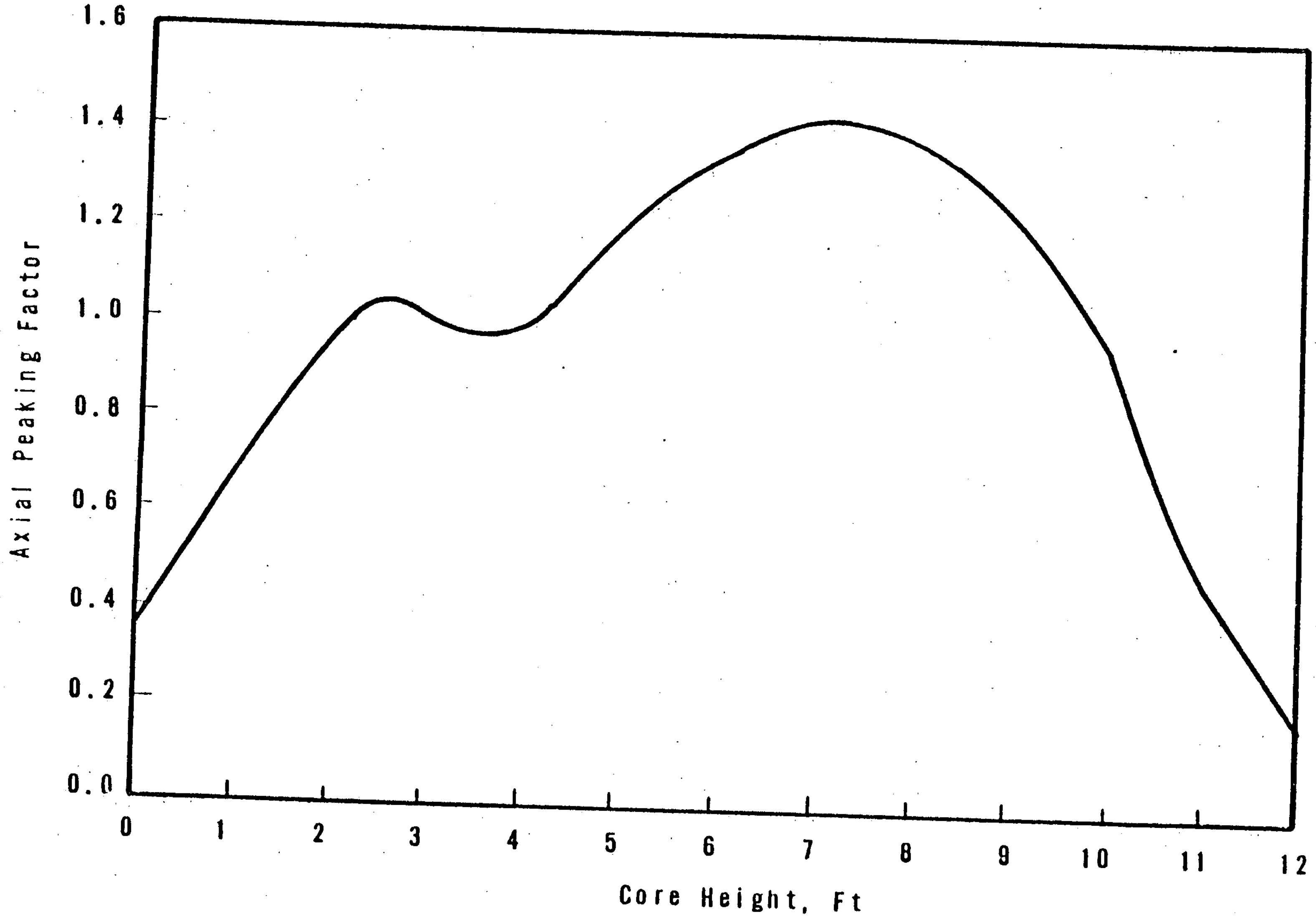


FIGURE 3. SADDLE SHAPE POWER CURVE-EQUAL PEAKS

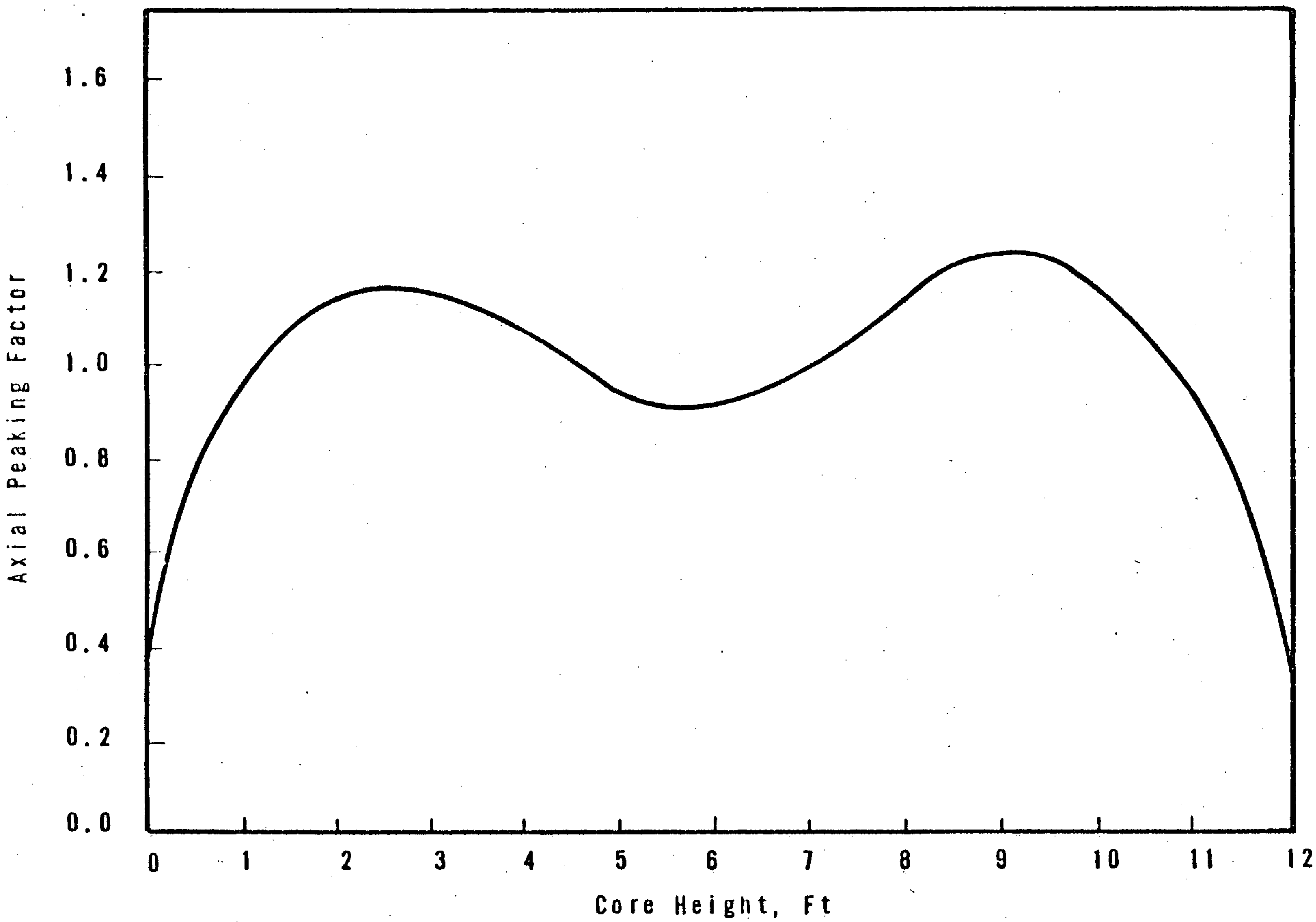


FIGURE 4. LOCA LIMITS POWER SHAPE - 10 FT PEAK

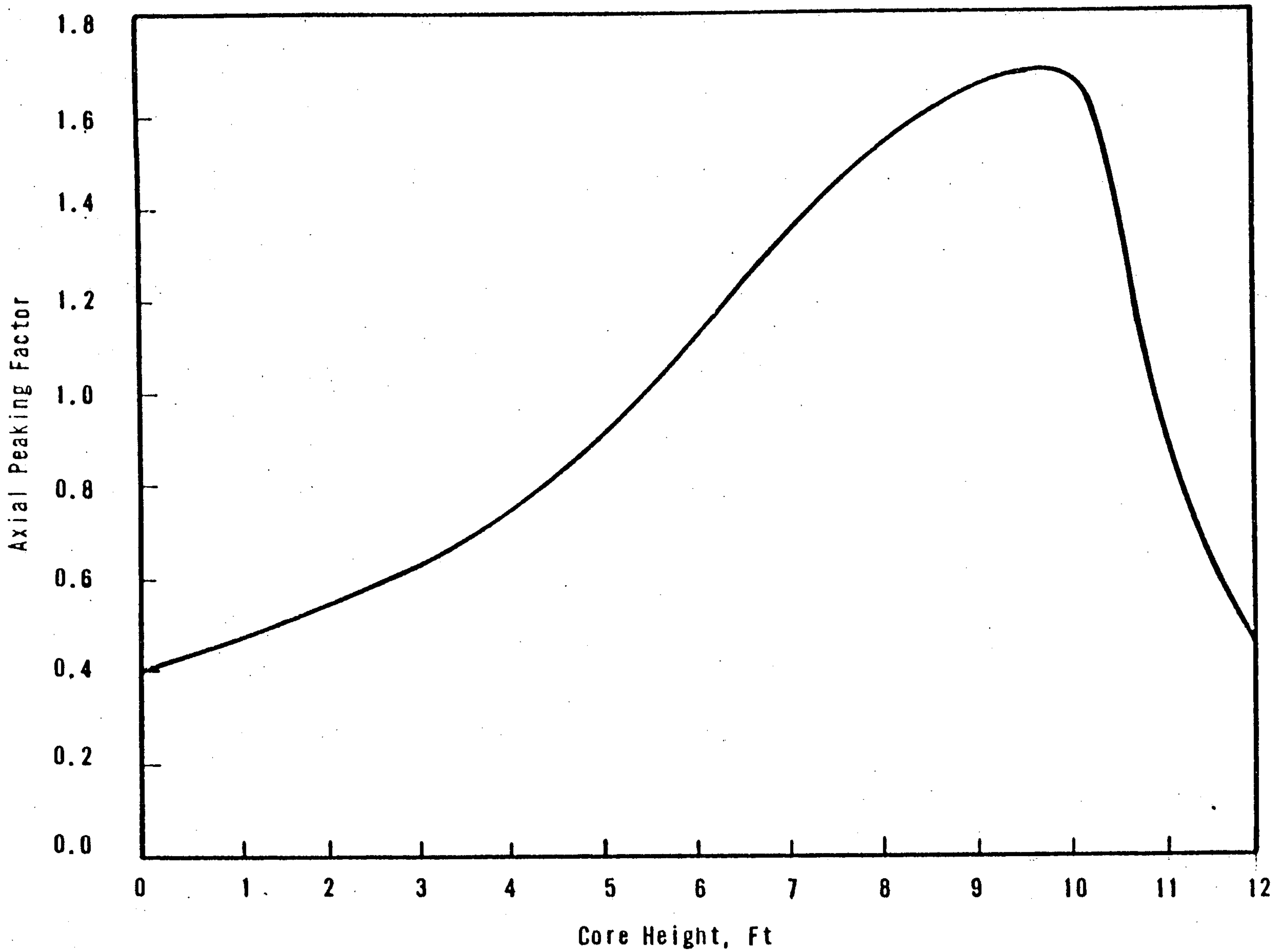


FIGURE 5. SMALL BREAK POWER SHAPE

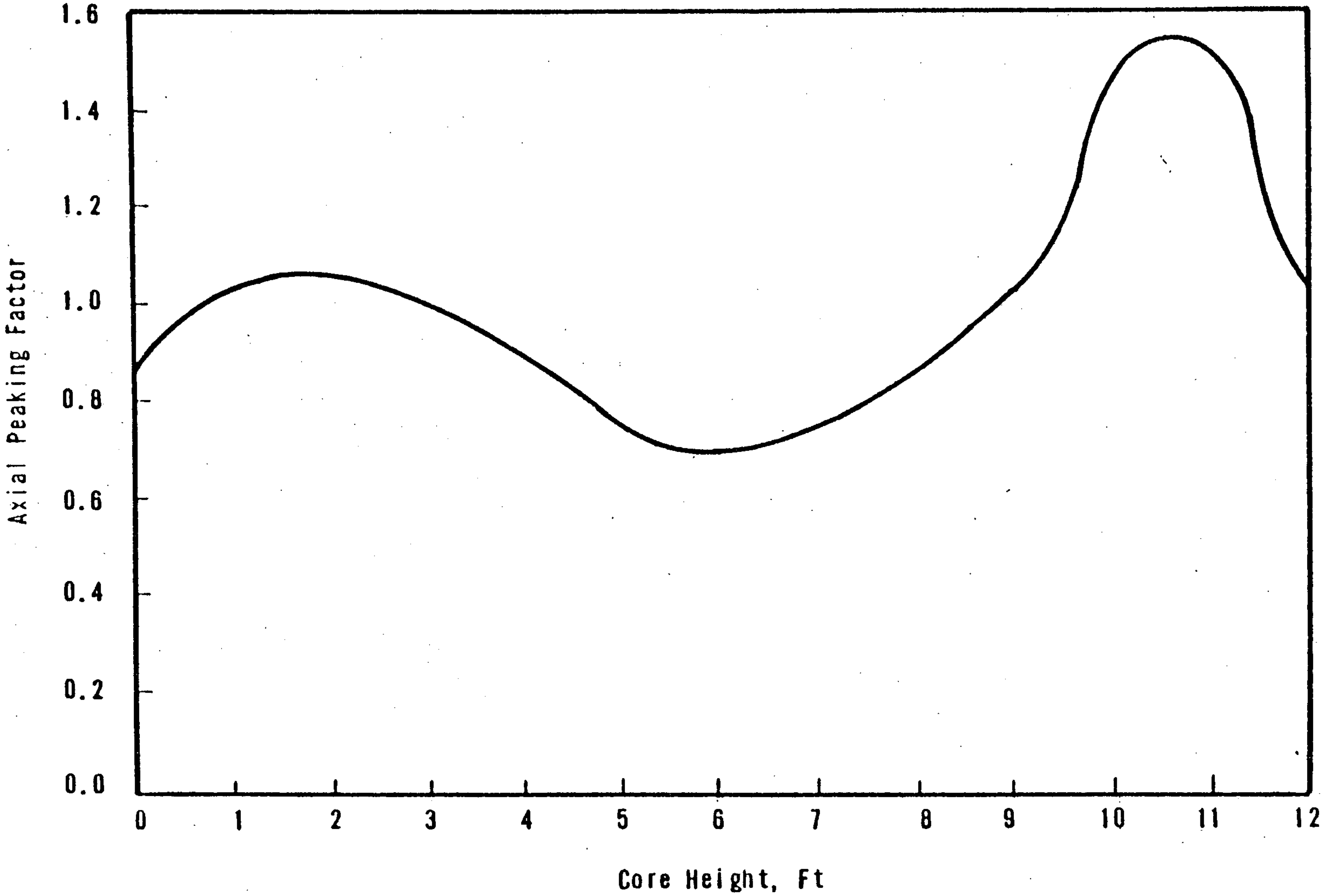


Figure 6 RCS PRESSURE VS CORE EXIT FLUID TEMPERATURE FOR 1400°F CLAD TEMPERATURE LIMIT

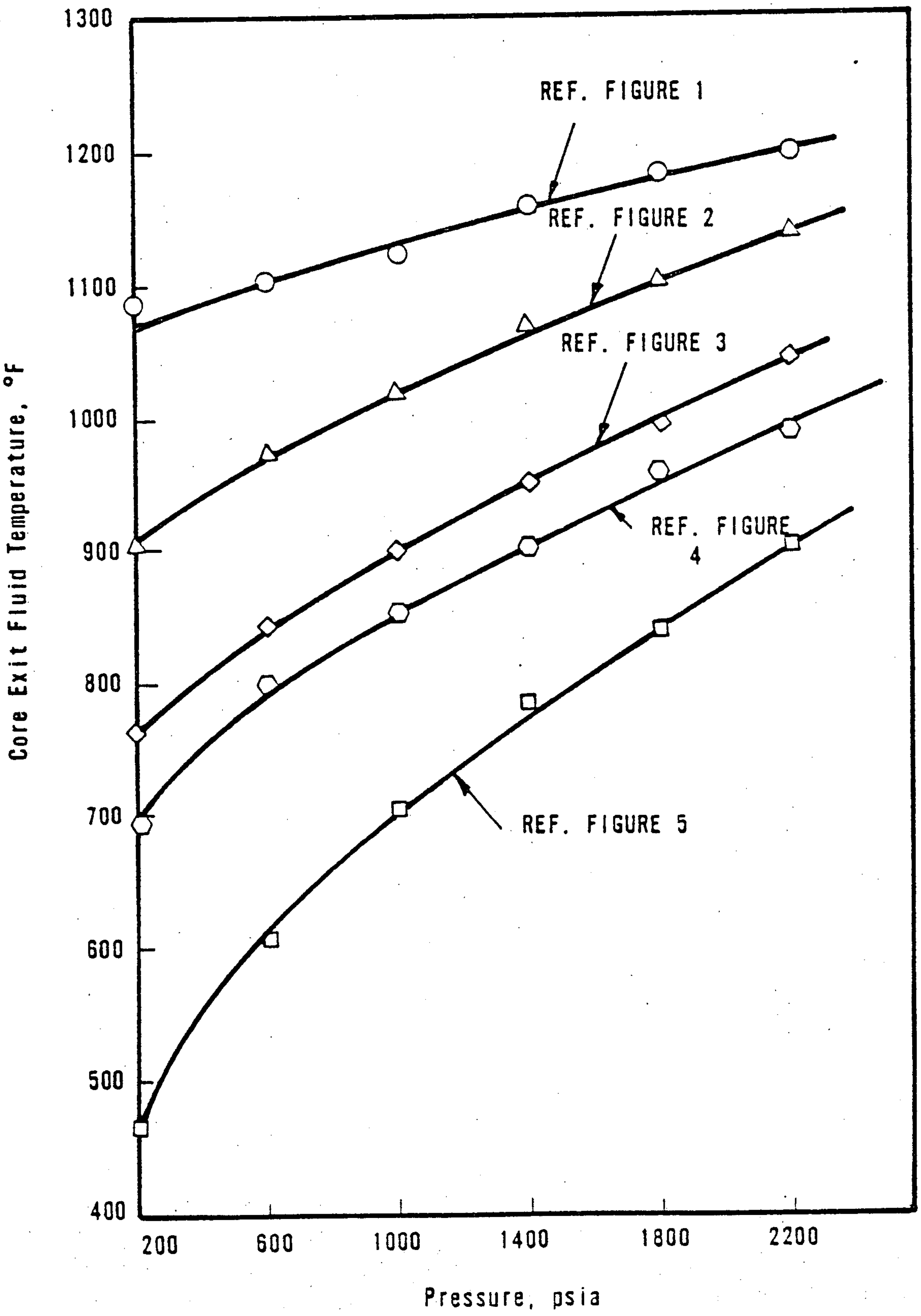


Figure 7 RCS PRESSURE VS CORE EXIT FLUID TEMPERATURE FOR 1800°F CLAD TEMPERATURE LIMIT

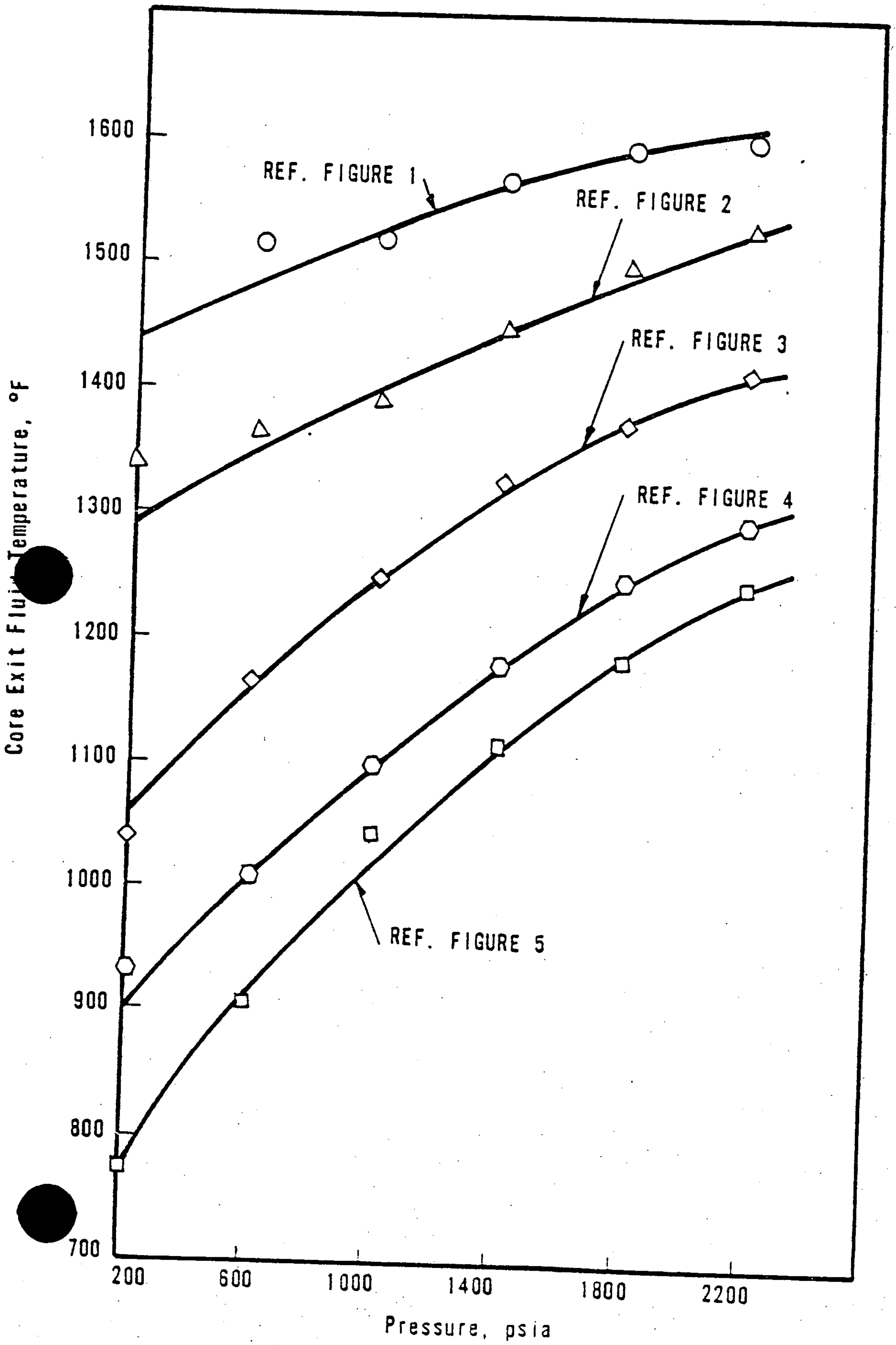


Figure 8 CORE EXIT FLUID TEMPERATURE INDICATION TO LIMIT CLAD TEMPERATURE

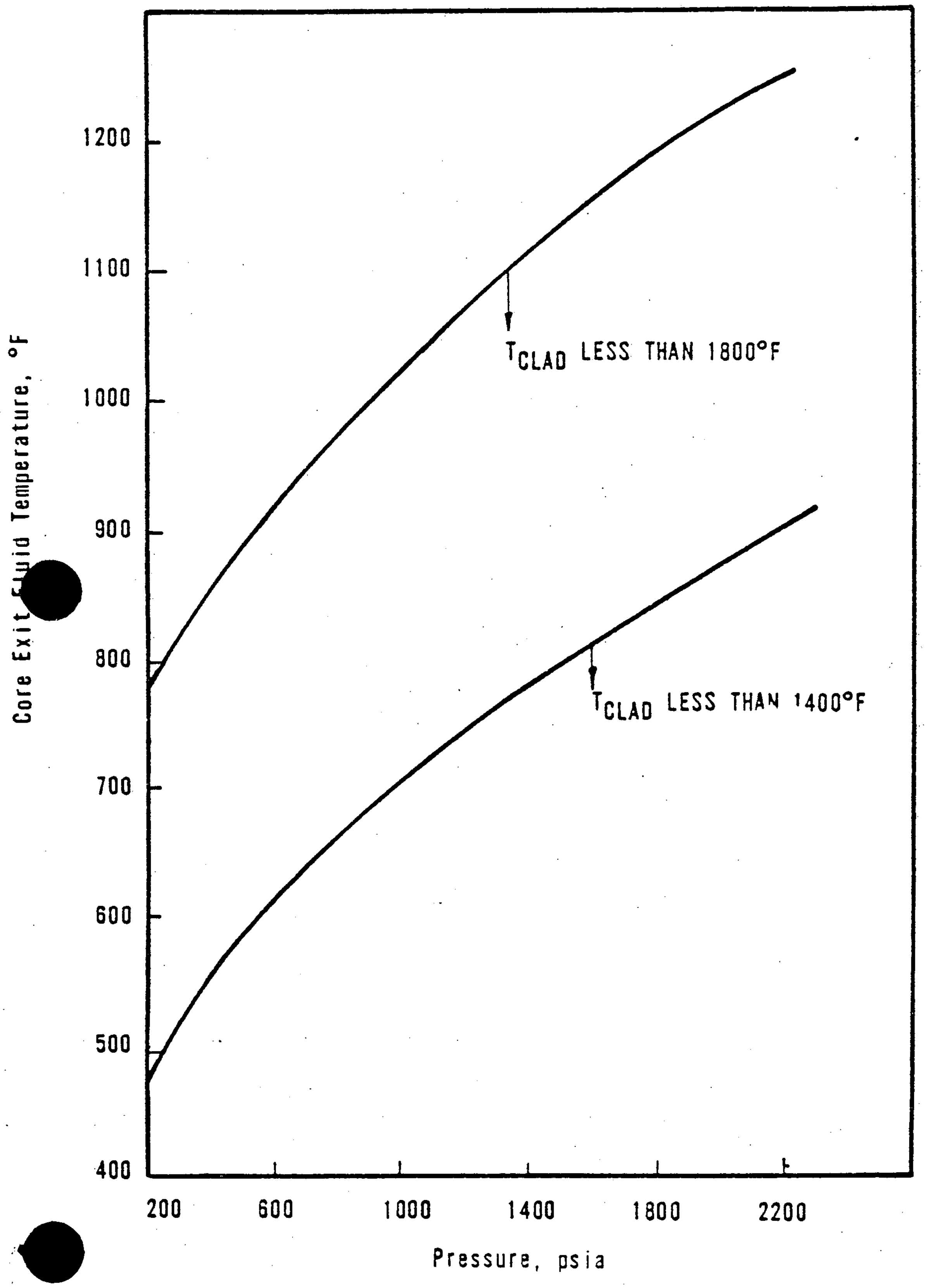
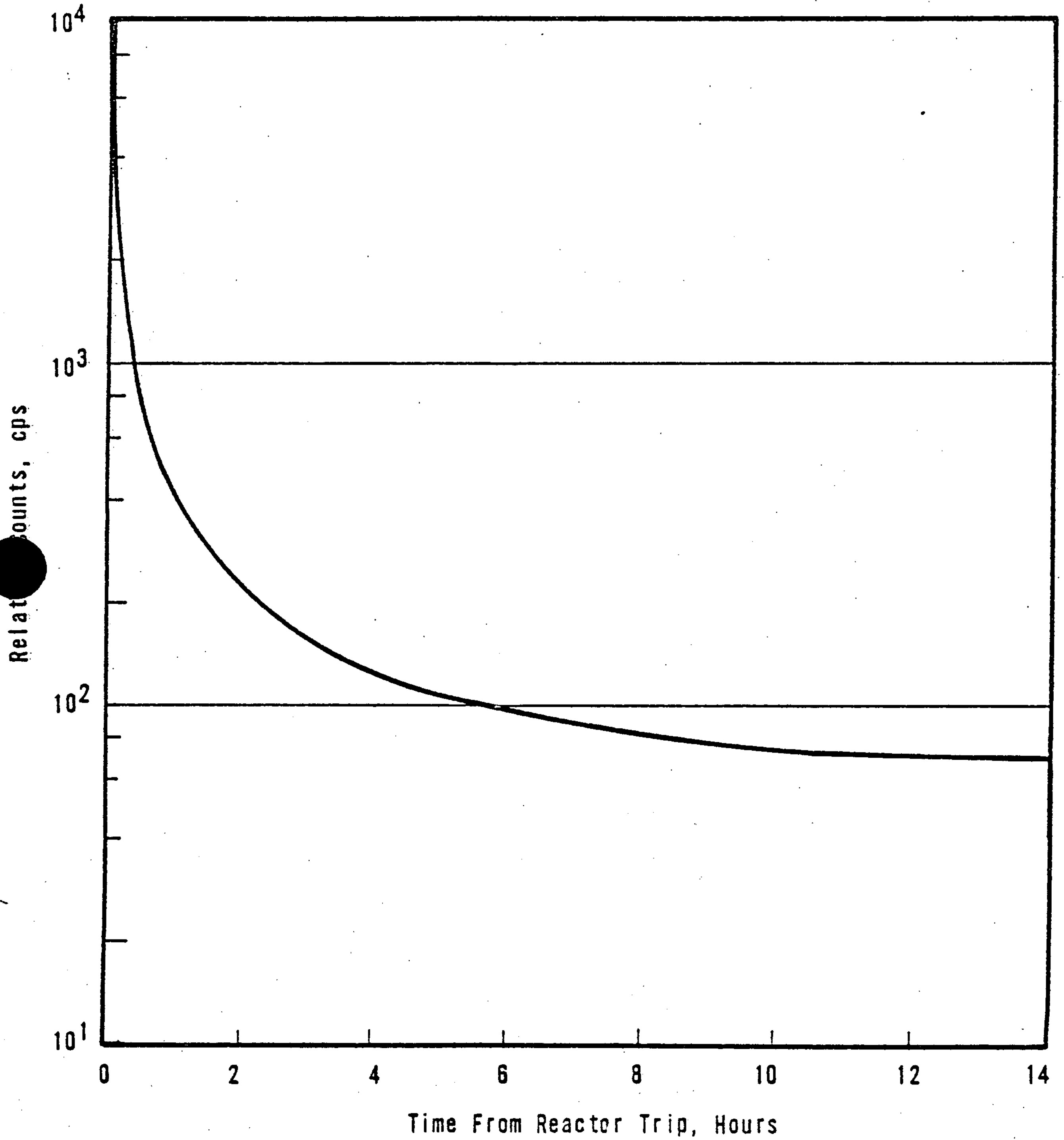
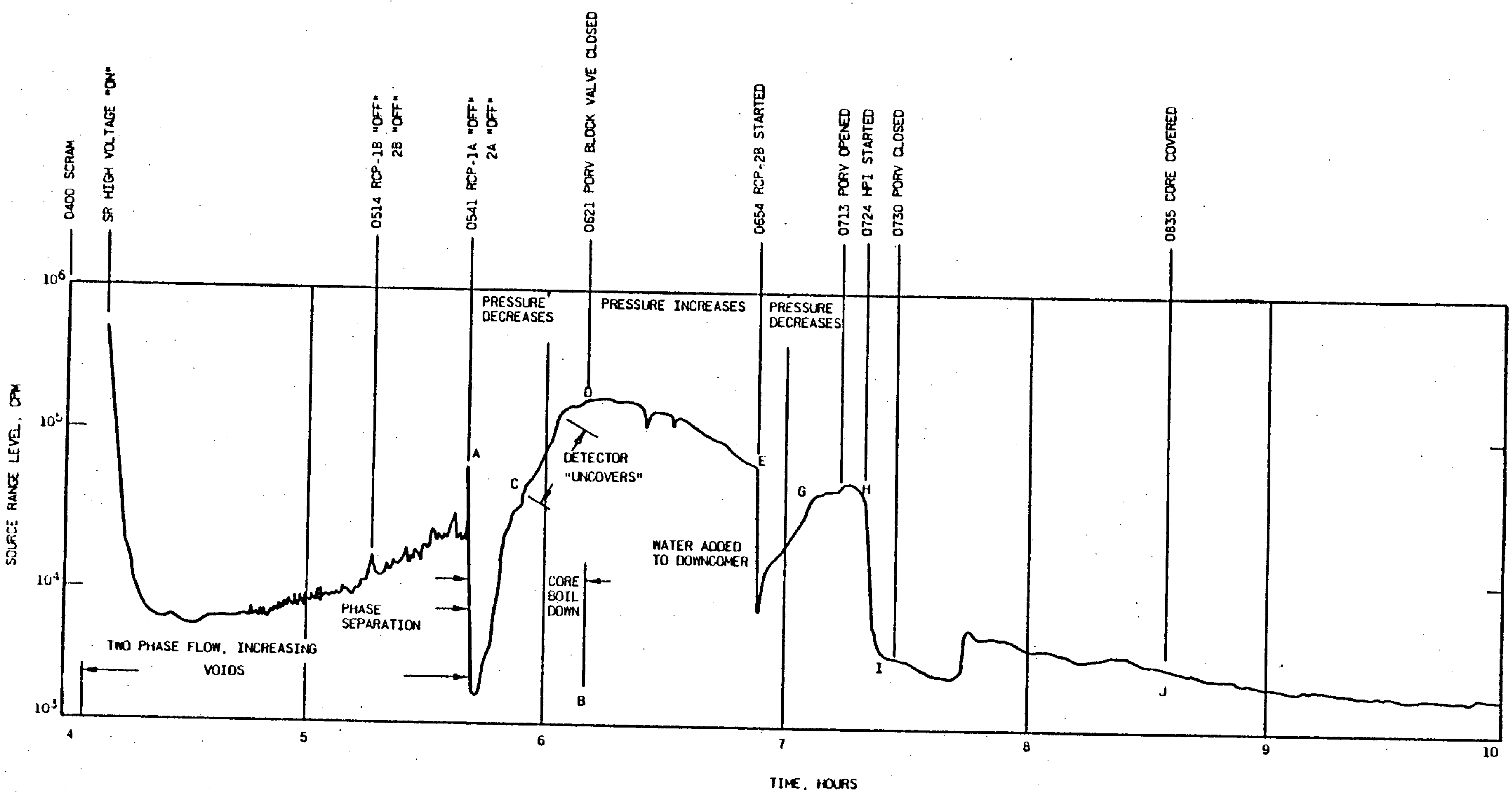


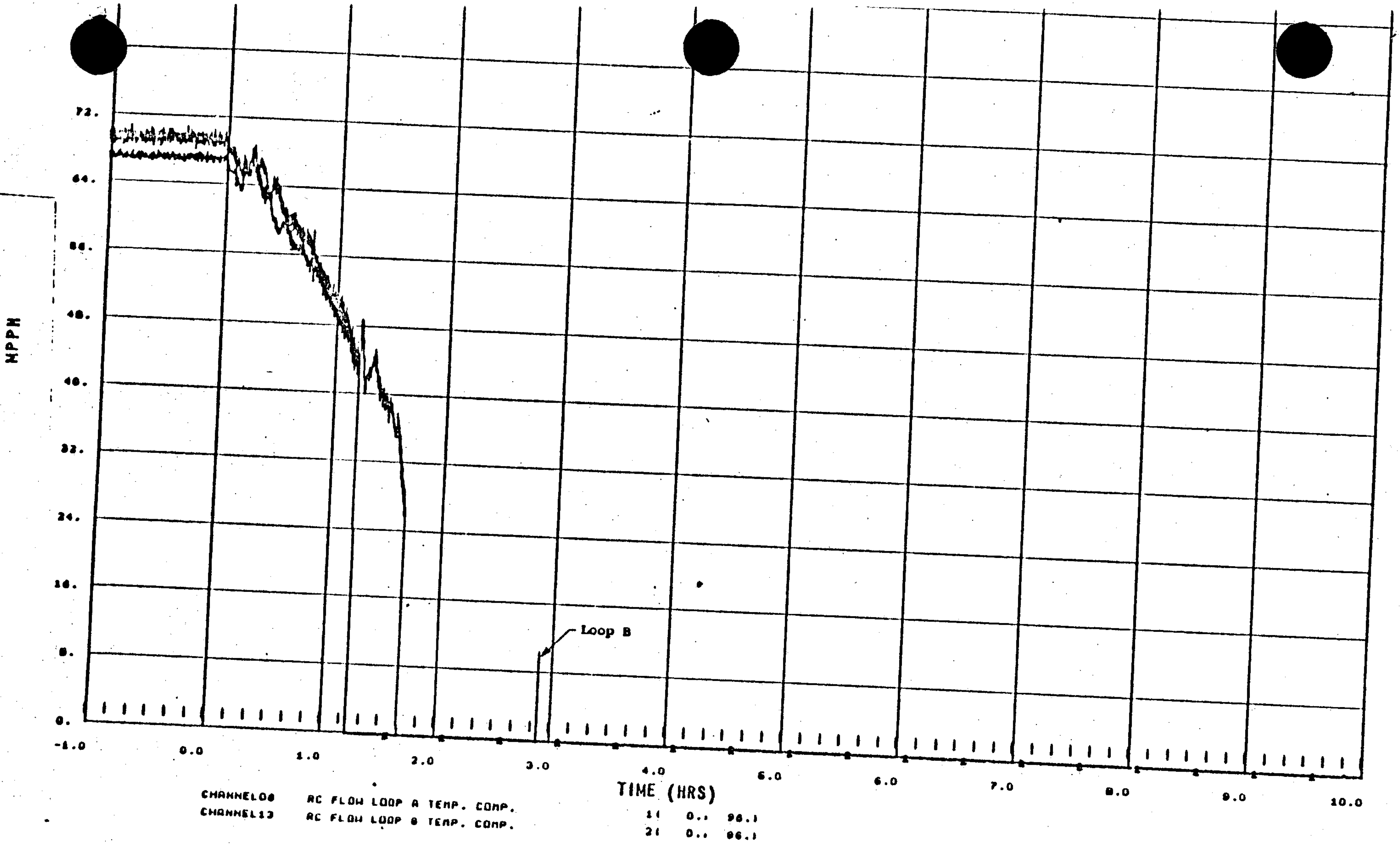
FIGURE 9. SOURCE RANGE TRACE FOLLOWING REACTOR TRIP (TYPICAL)





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

Figure 10



PRIMARY SYSTEM FLOW

Figure 11

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

3.0 ANALYSIS SUMMARY FOR REFUELING CONDITIONS

Parts I and II of the Inadequate Core Cooling Operating Guideline provide direction to the operator in the event of inadequate core cooling during refueling. The guidelines are based, in part, on the reaction time of the operators. Section 3.1 describes the basis and results of the calculation to determine the maximum time the operator has to act and the minimum flowrate of water to recover the core in the event that the decay heat system fails. Section 3.2 describes the basis and results of the calculation to determine the maximum time the operator has to act and the minimum flowrate of water to recover the core in the event that there is a loss of reactor coolant system (RCS) inventory.

3.1 LOSS OF DECAY HEAT SYSTEM

After a failure of the decay heat removal system (DHR), the water in the reactor vessel will begin to heatup. If the DHR system cannot be regained for a period of time the coolant will begin to boil and the liquid volume above the core will slowly decrease. The time from the failure of the decay heat system until the liquid level in the reactor vessel reaches the top of the core was calculated. That liquid level is considered to provide inadequate core cooling. The assumptions made in the calculation were:

1. Decay heat level is 1.0 times ANS Standard
2. Time from last shutdown is 48 hours
3. Decay heat level is constant for the duration of the transient.

These assumptions provide a heat input to the coolant of 0.42% of full reactor power.

Initial plant conditions were defined as follows:

1. Reactor vessel head is removed

2. Reactor vessel water level is at the RV ledge
3. Reactor coolant temperature is $140^{o}F$
4. Reactor vessel refueling seal is in place.

These plant conditions define the heat removal capability of the coolant in the reactor vessel. The heat input will cause the coolant to boil in 26.7 minutes. It will take another 107.8 minutes for the coolant to boil down to the top of the core. The total time from the loss of the DHR system until the core is inadequately cooled is 2¼ hours.

The rate that coolant is boiled off in the reactor vessel will set the minimum flowrate required to recover the system. The coolant will boiloff at 81.6 gallons per minute. The decay heat pumps can each supply a nominal flowrate of 3000 gpm as makeup from theBWST. As shown in Figure 1, this flowrate will reflood the reactor vessel is about 3 minutes. An absolute minimum flowrate of 250 gpm should be provided in order to reflood the core within 1 hour.

3.2 LOSS OF RCS INVENTORY

Prior to removing the reactor vessel head, a loss of RCS inventory is covered in the small break operating guidelines. After the reactor vessel head has been removed, the RCS inventory is assumed to be reduced non-mechanistically. The RCS inventory will be reduced as if there were a break of equivalent area to the cross sectional area of one incore nozzle. The time from the initiation of the loss of RCS inventory until the liquid level in the reactor vessel reaches the top of the core was calculated. That liquid level is considered to provide inadequate core cooling.

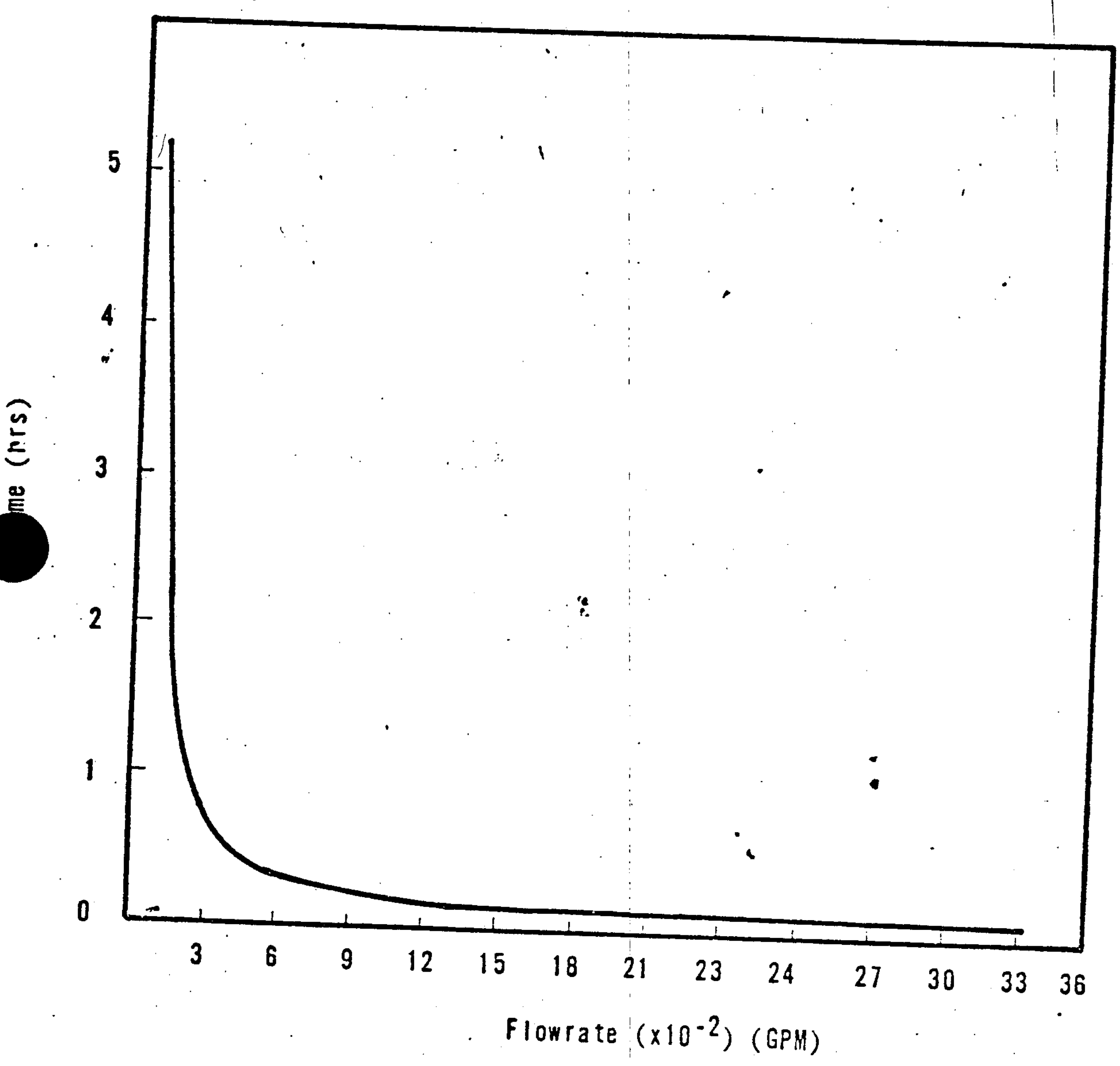
Initial plant conditions were defined as follows:

1. Reactor vessel head is removed
2. Reactor vessel water level is at the RV ledge.
3. Reactor coolant temperature is 140^{o}F
4. Reactor vessel refueling seal is in place

Applying these initial plant conditions to the assumed break area generates a volumentric flowrate out of the break of 0.094 ft.^{3}/sec. Knowing the diameter of the reactor vessel, this flowrate is transformed to a reactor vessel level decrease rate of 0.456 inches per minute. At that rate the top of the core will be uncovered in 3.6 hours.

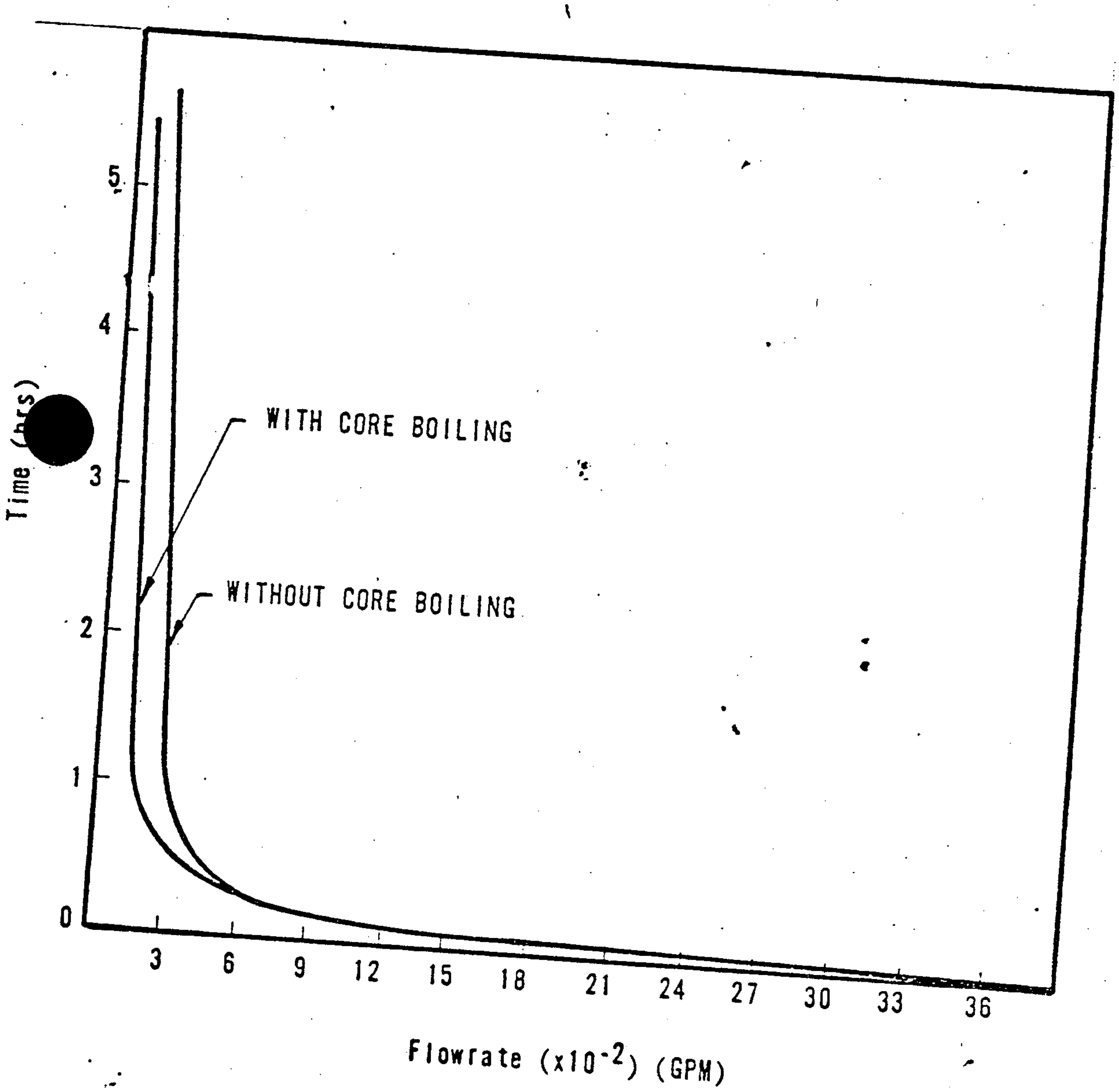
The minimum flowrate required to recover the core will have two components. They are the leak rate and the boiloff rate. The leak rate will have to be overcome regardless of the status of the DHR system. The DHR system will continue to maintain system temperature as long as there is still flow through the DHR system. However, when RCS inventory reaches the bottom of the hot leg nozzles, the DHR system will no longer function, and reactor coolant temperature will begin to rise. It will take a finite amount of time for the coolant to boil, but that time has been assumed to be zero. It has also been assumed that the coolant will boil off at the same rate as in the loss of DHR system case. The total coolant loss will be 123.8 gpm (42.2 + 81.6 gpm) with no DHR system. The decay heat pumps can each supply a nominal flowrate of 3000 gpm as makeup from the BWST. As shown in Figure 2, this flowrate will reflood the reactor vessel in about 3 minutes, regardless of the status of the remainder of the DHR system.

If the DHR system has also been lost, an absolute minimum flowrate of 300 gpm should be provided in order to reflood the core within 1 hour. If operator actions are initiated so that the DHR system is not lost, an absolute minimum flowrate of 200 gpm should be provided in order to reflood the core within 1 hour.



REFLOOD TIMES FOR A
LOSS OF DHR SYSTEM

Figure 1



REFLOOD TIMES FOR A LOSS OF RCS INVENTORY

Figure 2

4.0 ANALYSIS SUMMARY FOR POWER OPERATION - DNB CONDITION

The core thermal-hydraulic study was performed to investigate the DNBR response to two postulated conditions which could result in inadequate core cooling. These conditions are:

1) An undetected increase in the core radial peaking (increase in hot assembly power relative to core average), and

2) an undetected reduction in core coolant flow.

While either or both of these conditions can be postulated to occur, to a limited extent, no credible mechanism has been postulated which could cause either of these events to proceed undetected to an extent such that core cooling effectiveness would be reduced sufficiently to result in a departure from nucleate boiling (DNB). However, in order to assess the potential for inadequate core cooling resulting from these postulated events the following study has been performed independently of any evaluations regarding the credibility of the occurrences.

4.1 Analysis Assumptions and Initial Conditions

These analyses represent a "best estimate" calculation of the minimum departure from nucleate boiling ratio (DNBR), with the exception that the initial core power distribution selected for the analysis was a design distribution without uncertainties. This design peaking distribution bounds (in terms of maximum assembly relative power) typical core power distributions in operating plants. The power distribution was held constant for the reduction in coolant flow study, without consideration of moderator feedback effects which would tend to flatten assembly peaks as coolant voiding occurs. A symmetrical axial power distribution was assumed as a conservative (for DNBR calculations) representation of actual core axial distributions. Initial reactor operating conditions were selected to [illegible]

the nominal operation of a typical Babcock & Wilcox reactor rated at 2772 MWt core power. The initial reactor coolant system (RCS) flowrate selected for analysis was 108% of design (where design flow is 88,000 GPM/pump), which is representative of the minimum measured flowrate in operating B&W reactors. The RCS pressure and inlet temperature (2200 PSIA and 557.3^{0}F, respectively) represent nominal operation at this flowrate.

Core and fuel assembly models used for this evaluation were representative of the nominal geometry, without application of tolerances and engineering hot channel factors.

Computer codes used for this analysis included LYNX1 (1) and LYNX2 (2). LYNX1 used to model the core, on an assembly by assembly basis, while LYNX2 was used to model an individual assembly on a subchannel basis.

2 METHOD OF ANALYSIS AND RESULTS

1) Increase in Core Radial Peaking

Various power increases in the hot pin were investigated by increasing the hot bundle relative power while maintaining normalization of the power distribution across the core. The power gradient around the hot bundle was also maintained by increasing the bundle radial power factors for the seven surrounding bundles in an eighth-core symmetric model.

A symmetrical axial power distribution (1.4 cosine) was selected as a conservative representation of actual core axial power shapes.

re 1 shows the effect of increasing the bundle radial power on the minimum DNBR response of the hot pin. A bundle radial peak greater than 2.3 is required

to achieve the BAW-2 CHF design limit of 1.30 and a bundle radial peak of 2.45 is required to achieve a DNBR of 1.00. Figures 2 through 5 provide bundle radial power distributions and coolant exit temperature distributions for core models with hot bundle radial powers of 2.3 and 1.8. It can be noted the hot bundle and its surrounding bundles have coolant exit temperature of T_{sat} for the case with a hot bundle radial power of 2.3.

2) Reduction in Core Coolant Flow

Various reductions in the nominal flowrate were studied in conjunction with the "best estimate" bundle radial power distribution in Figure 6. This distribution is designated "best estimate" but still possesses higher bundle peaks than a typical B&W 2772 MWt core.

Figure 7 shows the effect of reducing the nominal flowrate on the minimum DNBR response of the hot pin. A flow reduction greater than 42% is necessary to obtain minimum DNBR values below the BAW-2 CHF design limit of 1.30 and a flow reduction greater than 55% to exceed a minimum DNBR of 1.00. Figures 8 and 9 provide coolant exit temperature distributions for two flowrates, 83% and 55% of nominal system flow. It can be seen that a majority of the bundles in the core will experience exit coolant temperatures of T_{sat} for an inadequate core cooling condition of 55% of nominal system flow.

Figure 10 has been provided to show the effect of bundle radial power on coolant exit temperature for various flowrates. Bundle exit coolant temperatures are at T_{sat} for bundles with radial powers as low as 1.10. Since the thermocouples may have measurement errors of $10^{0}F$ or more, the significant trend is the flat temperature response when T_{sat} is reached.

MINIMUM DNBR (BAW-2 CHF CORRELATION) AS A FUNCTION OF THE HOT BUNDLE RADIAL PEAK

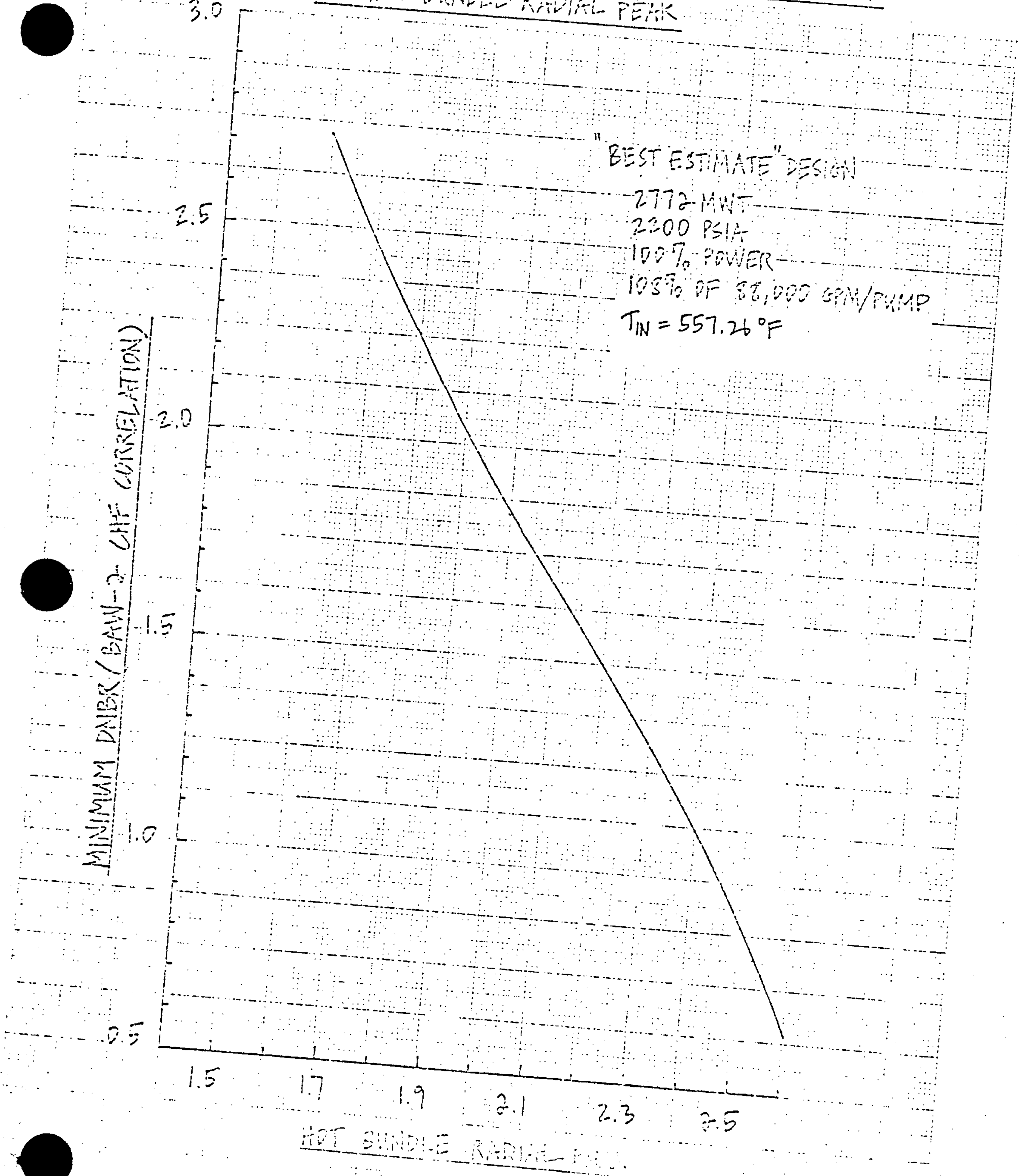


Figure 1

1.6526	2.0450	2.2915	2.2434	1.4069	0.8344	0.3194	0.2090
	2.1930	2.3000 *	2.2199	1.3252	0.8430	0.5349	0.2335
		2.2644	2.1169	1.0274	0.7067	0.5291	0.1774
			1.3337	0.8092	0.2648	0.2872	
				0.6009	0.4198	0.1412	
					0.2145		

* HOT BUNDLE

1/8TH CORE SYMMETRY

BUNDLE RADIAL POWER DISTRIBUTION

FIGURE 2

638.4	649.4	649.4	649.4	638.4	609.4	579.6	569.7
	649.4	649.4	649.4	629.7	604.7	587.9	571.5
		649.4	649.4	617.4	597.9	587.8	568.2
			632.6	604.8	574.8	573.7	
				591.9	581.0	565.9	
					569.7		

1/8TH CORE SYMMETRY

: T_{SAT}

COOLANT TEMPERATURE DISTRIBUTION AT EXIT

2.3 HOT BUNDLE RADIAL

MINIMUM DNBR ~1.0

FIGURE 3

1.2933	1.6005	1.7933	1.7557	1.4069	1.0197	0.5047	0.3943
	1.7163	1.8000 *	1.7373	1.3252	1.0283	0.7202	0.4238
		1.7722	1.6567	1.2127	0.8920	0.7144	0.3627
			1.3337	0.9945	0.4501	0.4725	
				0.7862	0.6051	0.3265	
					0.3998		

* HOT BUNDLE

1/8 TH CORE SYMMETRY

<u>BUNDLE RADIAL POWER DISTRIBUTION</u>

FIGURE 4

622.1	635.9	644.7	645.0	630.9	614.5	588.2	579.8
	641.3	645.7	643.1	625.7	612.4	597.5	581.6
		644.5	640.6	621.1	606.6	597.5	578.5
			626.7	611.9	584.0	583.8	
				600.9	590.9	576.1	
					579.9		

1/8 TH CORE SYMMETRY

COOLANT TEMPERATURE DISTRIBUTION AT EXIT

1,800 HOT BUNDLE RADIAL

FIGURE 5

1.0830	1.3402	1.5017	1.4702	1.4069	1.1282	0.6132	0.5028
	1.4372	* 1.5073	1.4548	1.3252	1.1368	0.8287	0.5323
		1.4840	1.3873	1.3212	1.0005	0.8229	0.4712
			1.3337	1.1030	0.5586	0.5810	
				0.8947	0.7136	0.4350	
					0.5033		

1/8 TH CORE SYMMETRY

* DENOTES THE HOT BUNDLE

"BEST ESTIMATE" BUNDLE POWER RADIAL DISTRIBUTION

FIGURE 6

MINIMUM DNBR (BAW-2 CHF CORRELATION) AS A FUNCTION OF PERCENT NOMINAL FLOW

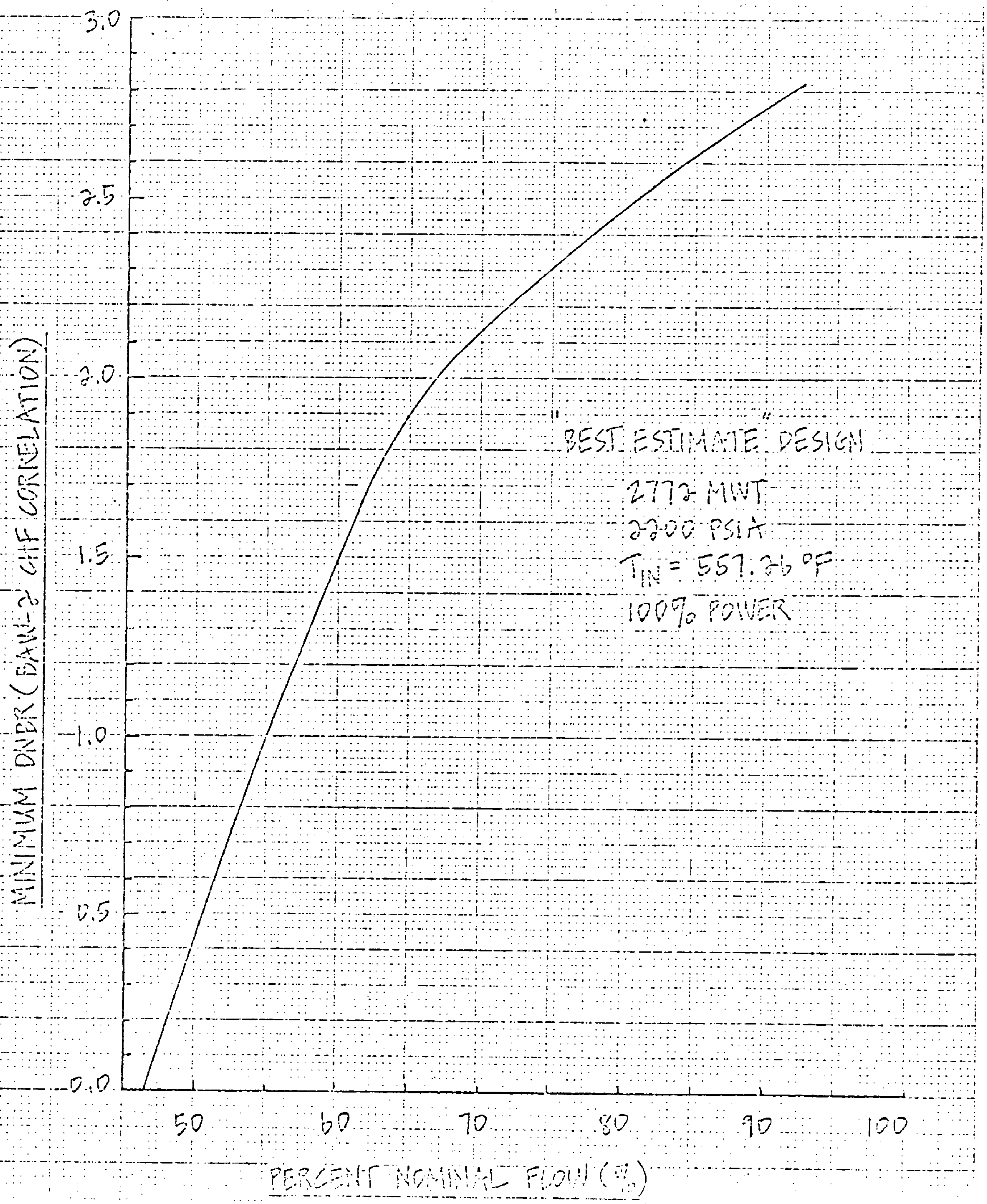


FIGURE 7

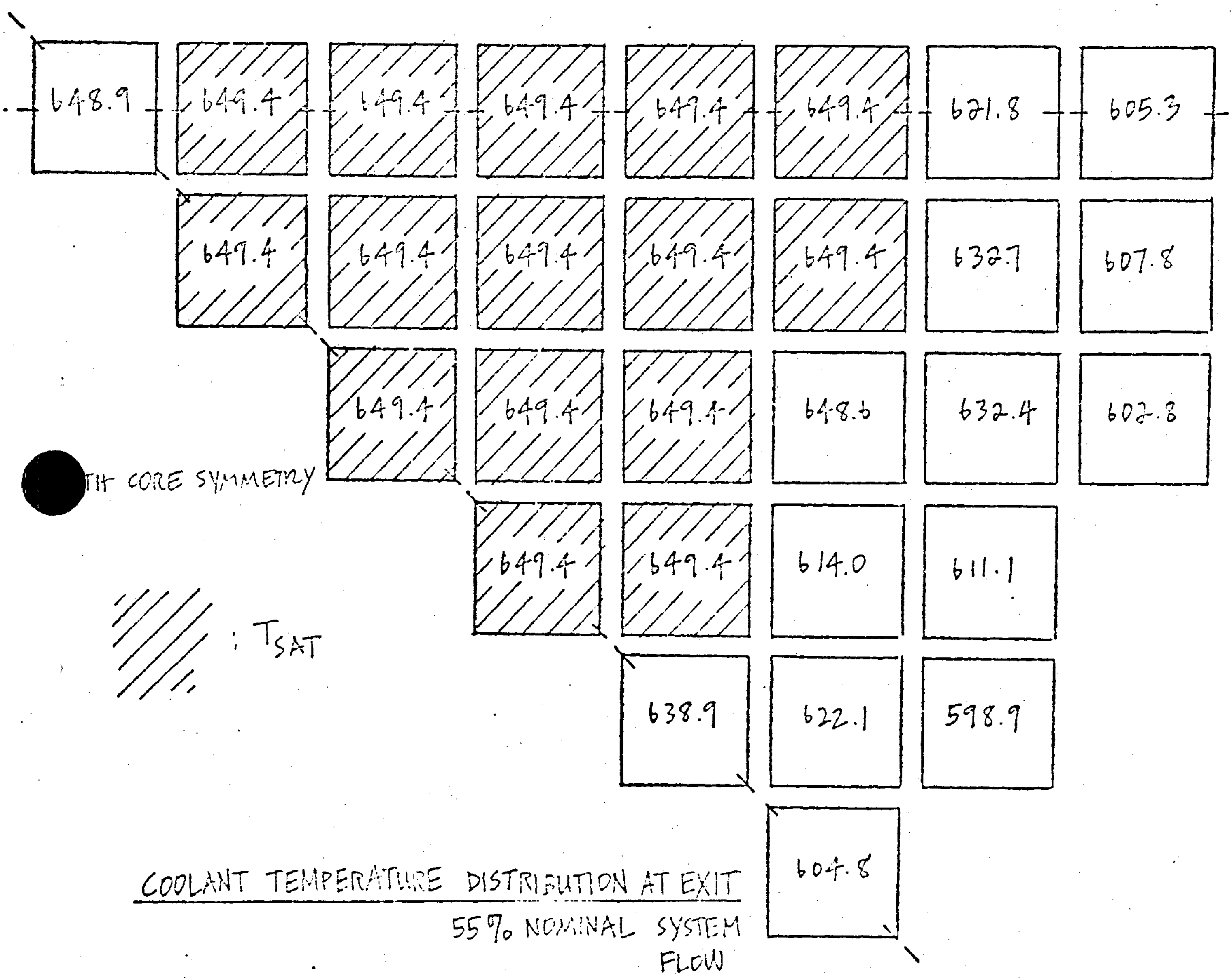
621.9	635.6	644.6	643.8	642.0	629.5	600.5	591.0
	641.0	645.5	642.5	636.5	627.5	611.2	593.1
		644.4	639.4	636.7	621.3	611.3	589.5
			637.1	626.9	595.7	595.6	
				614.9	603.7	586.7	
					591.0		

1/8TH CORE SYMMETRY

COOLANT TEMPERATURE DISTRIBUTION AT EXIT

83% NOMINAL SYSTEM FLOW

Figure 8



COOLANT TEMPERATURE DISTRIBUTION AT EXIT

55% NOMINAL SYSTEM FLOW

MINIMUM DNBR ~ 1.0

COOLANT EXIT TEMPERATURE AS A FUNCTION OF PERCENT NOMINAL FLOW

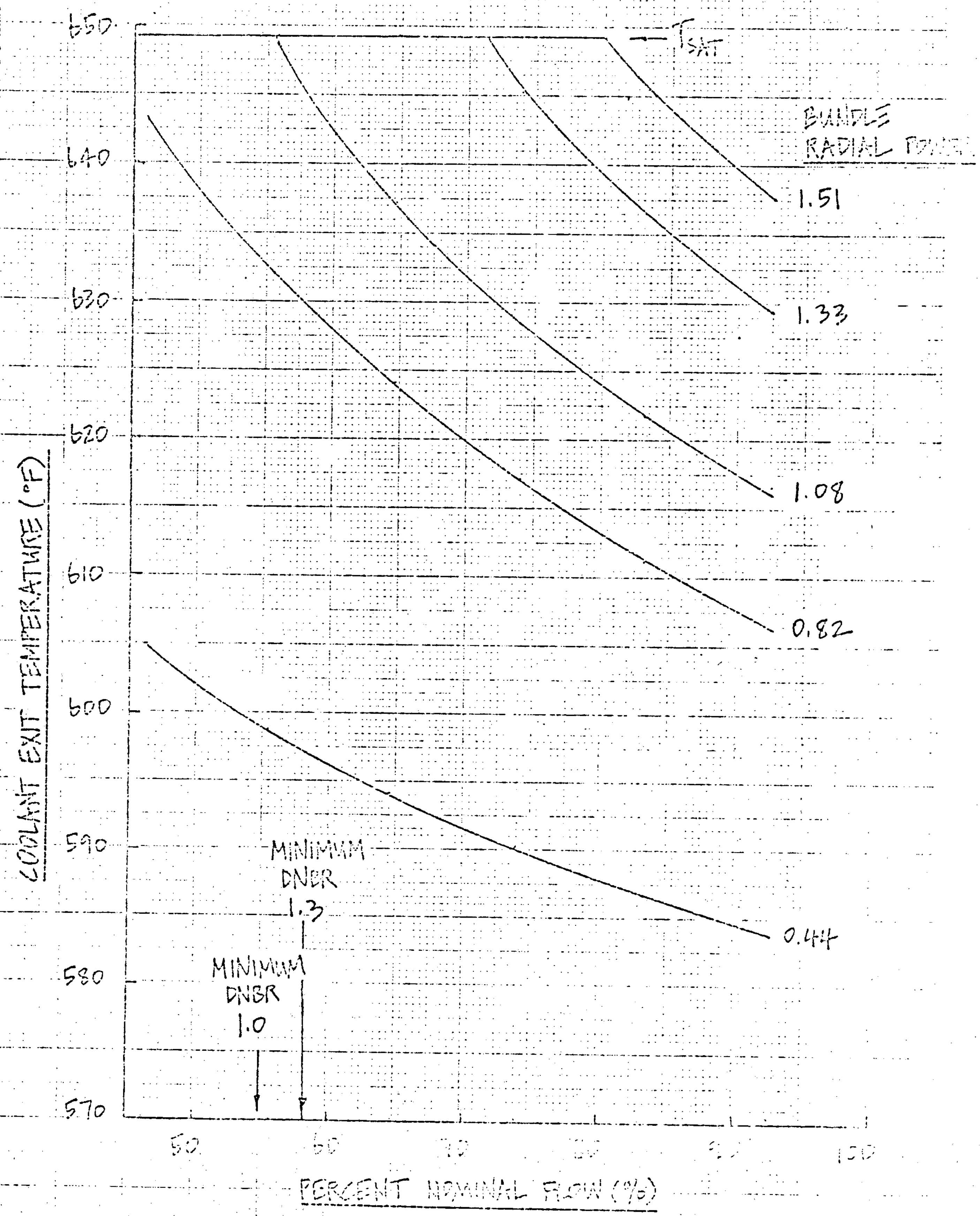


FIGURE 10

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