SAFEGUARDS REPORT

FOR SAXTON CORE III

August, 1968

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SAXTON CORE III

1.0 INTRODUCTION AND SUMMARY

The proposed Saxton Core III configuration, similar to previous cores, contains 21 fuel assemblies. The central nine consist of seven plutonium "loose lattice" assemblies to be reconstituted from the Core II assemblies and two new enriched UO_2 "load-follow" assemblies. The remaining twelve outer assemblies will be partially depleted UO_2 assemblies from Saxton Core I and Core II. The increased lattice pitch of the "loose lattice" assemblies extends the burnup capability of the zircaloy clad plutonium fuel by taking advantage of the reactivity increase associated with the increased water to fuel ratio.

As a possible result of unanticipated delays in reconstituting Core II fuel into the "loose lattice" assemblies, Core III could ultimately consist of fewer than seven or even no "loose lattice" assemblies. Should this be the case, the basic ground rules and limits on linear power and fuel temperature contained herein, for each type assembly of the above described core, would be followed in any alternate. This document would be amended to account for any differences in the ultimate configuration.

1.1 DESCRIPTION OF PROGRAM

1.1.1 PROGRAM OBJECTIVE

The objectives of the Saxton Core III irradiation program are to:

1. validate fuel element design code predictions, including determination of power/burnup failure limits;

2. demonstrate performance capability of Zircaloy clad oxide fuel elements over a broad spectrum of burnups and power levels; and

3. obtain depletion characteristics and transuranic isotope generation data for high burnup, mixed oxide fuel.

Because the Saxton Core II mixed oxide fuel rods were designed for relatively low peak burnups (20,000 MWD/MTM) and operation at power densities $\leq$ 16 kw/ft, there is a significant risk of failure of certain of these rods in Core III. By careful selection and placement of these rods in the loose lattice assemblies, it is possible to control their burnup and operating power levels and thus permit power/burnup limits to be established while operating safely and in full compliance with the reactor license Technical Specifications.

1.1.2 SCOPE

The linear power objectives are achieving expected peak kw/ft of 21.2 in the "loose lattice" assemblies and 17.6 kw/ft in the load follow assemblies. The corresponding design linear power including a conservative combination of the design uncertainties are 24.0 kw/ft and 19.9 kw/ft in the two type assemblies, respectively. These design linear powers are the basis for the analysis for Core III and will be achieved at a core design power less than 28 MWt.

As in Core II 35 MWt operation, the nominal inlet temperature during full power operation will be 480°F. The 19.9 kw/ft design linear power of the two "load follow" assemblies is 4% higher than the design value for Core II 35 MWt operation. The increased linear power of the rods in the "loose lattice" assemblies results from the use of dummy rods in alternate rod positions which reduces the heat addition to a coolant channel and increases the margin to DNB.

Core III operating conditions have been selected to maintain fuel temperature below center melt and the minimum DNB ratio greater than 1.3 at the control and protection system reactor trip setpoint conditions thus protecting the fuel for anticipated transient conditions.

1.2 GENERAL PROCEDURES AND OPERATING CONDITIONS

At the beginning of Core III life the power will be increased to reach an expected peak linear power of 21.2 kw/ft in the "loose lattice" assemblies (∿24.7 MWt). The initial power level will be gradually increased to hold this linear power rating as the local power peaking decreases with burnup. Initially, the expected peak linear power of the "load follow" assemblies will be less than 17.6 kw/ft. However, the ratio of peak to average power in the load follow assemblies burns down slower than that of the "loose lattice" assemblies. Hence, the expected peak linear power in these assemblies will increase with burnup to the desired 17.6 kw/ft (∿500 EFPH) at which time further core power increases will be maintained at a rate to hold the 17.6 kw/ft in the peak "load follow" rods while the power in the "loose lattice" rods will then decrease.

A peak pellet burnup of approximately 55,000 MWD/MTM can be achieved after 5000 effective full power hours (EFPH @ 23.5 MWt) operation.

Load Follow Simulation

The load follow operation will be carried out in a manner which will not result in a higher peak power in the core than that imposed by the design limits. In order to simulate Plant Load Follow Operation from a materials standpoint cycling between approximately 100% and 40% of full power will be performed.

The mode of operation will be based on the following:

1. Achieving a minimum of 1000 cycles in Core III.

2. Existing operational limitations with regard to power loading and unloading.

3. Core III lifetime.

1.3 OPERATING CONDITIONS

The fuel and moderator temperature coefficients and kinetic parameters for Core III are intermediate to those of previous cores. Therefore, Core III represents no extrapolation from previous operation except for the increased peak linear power ratings. Operation is restricted to prevent center melt and to maintain margin to DNB. As in Core II 35 MWt Operation, the reactor coolant pump will be operated from the motor-generator set to benefit from the increased pump inertia. The operating pressure for Core III will be increased from 2200 psia to 2250 psia to more closely simulate conditions in current PWR's.

Approximately 50 fuel rods will operate within 10% of the peak linear power in the seven loose lattice assemblies (design peak linear power 24 kw/ft) and approximately 80 fuel rods will operate within 10% of the peak linear power in the load follow assemblies (design peak linear power 19.9 kw/ft). The Core II mixed oxide fuel rods to be used in the "loose lattice" assemblies were designed for lower peak burnups and lower linear power than they will experience in Core III. As a result certain of these rods may fail due to excessive clad strain and internal gas pressure at the high linear power and burnups to be achieved in Core III. The probable mode of failure will be short cracks or local blisters having no significant "ballooning" which could restrict coolant flow or affect adjacent rods. The location of the individual fuel rods in the "loose lattice" assemblies have been judiciously selected to obtain power/burnup combinations favoring the longest fuel lifetimes and those likely to fail early are located in the center removable subassembly to permit easy access.

To supplement daily sampling and periodic operation of the letdown/charging system radiation detector in monitoring coolant activity, a modification has been made utilizing a beta/gamma counter and a delay loop connected across

the steam generator. This modification will give a continuous and sensitive indication of gross coolant activity. The arrangement is described more fully in Appendix A. The estimated response time of this system is 80 seconds which will give early warning of activity release through defected fuel clad.

2.0 CORE DESIGN

2.1 INTRODUCTION

The major objective of the Saxton Loose-Lattice Program is determination of the performance capability of fuel and clad at specified linear power ratings. To meet the defined power and burnup objectives, existing Saxton Core II zirc-clad plutonium fuel assemblies will be modified to form seven loose-lattice assemblies for further irradiation in Saxton Core III. Two unirradiated, enriched UO_2 "load-follow" assemblies together with 12 partially depleted UO_2 assemblies (originally 5.7 w/o U-235) from Saxton Core I and Core II operation will also be installed in Core III.

2.1.1 BACKGROUND

The Saxton Reactor was designed to investigate areas of interest in the development of pressurized water reactors. It has an extensive in-core instrumentation system for measuring outlet temperature, flow rate, and pressure drop. In addition, fission detectors and flux wires are used in conjunction with analysis to determine nuclear power distributions throughout the core. An extensive experimental follow program was carried out during the operation of two previous cores in this reactor. C

The detailed information obtained during the design and operation of these cores provides the basis for the nuclear design of Saxton Core III.

Saxton Core I

Saxton Core I was composed of 21 UO_2 fuel assemblies with an enrichment of 5.7 w/o U-235. The core operated for a total of 8630 megawatt days and accumulated a core average burnup of 9700 MWD/MTF.

Saxton Core II

Saxton Core II contained a partial core loading of plutonium fuel. Nine plutonium fuel assemblies were installed in the center of the core with 12 uranium fuel assemblies installed on the periphery. A series of critical experiments was carried out using both fuel types before the core was installed in the Saxton reactor. Included in these measurements were critical configurations at a lattice pitch equivalent to that of the loose lattice region of Saxton Core III.

Extensive hot-zero power measurements were made at the beginning of life and periodically during Saxton Core II operation. At power measurements of various core parameters have also been made periodically and the critical boron concentration as a function of depletion has been monitored throughout operation. Comparisons of analysis with experiment show good agreement.

During the course of the operations follow work a number of basic improvements were also made in the methods of analysis of plutonium fueled system. In particular, a detailed PDQ-7 depletion calculation was carried out to analytically simulate the actual operation of Saxton Core II. These calculations are experimentally verified by the good agreement between the calculated and measured critical boron concentrations as a function of core life. The analytic results from this calculation include a detailed burnup and power distribution for every plutonium fuel rod. This detailed distribution provided the basis for the selection of fuel rods for further irradiation in Saxton Core III. The burnup range and fuel rod population from which rods could be selected for use in Core III is summarized in Figure 2.1-1.

2.1.2 NUCLEAR DESIGN PRINCIPLE

Saxton Core II operation is scheduled to continue until mid-October 1968. At that time, irradiated zircaloy-clad plutonium fuel assemblies will be modified by removing irradiated fuel rods from existing assemblies and reinstalling them in seven new assemblies at a larger pitch and interspersed with water filled rods. The resulting increase in the water-to-fuel ratio provides a significant increase in the reactivity capability of the burned

plutonium fuel. The increased reactivity capability of the seven loose lattice assemblies combined with the use of the least burned UO_2 fuel from Core I and Core II operation and the addition of two unirradiated UO_2 load follow assemblies provides, in effect, a new core permitting additional operation. The increased reactivity capability of the fuel combined with a net reduction in the total amount of fuel installed makes it possible to achieve a high linear power rating in the loose-lattice fuel at a lower core thermal power. In addition, since the loose-lattice has more favorable DNB characteristics than does the tight lattice, higher linear power ratings can be reached in these assemblies without exceeding the same imposed DNB limit.

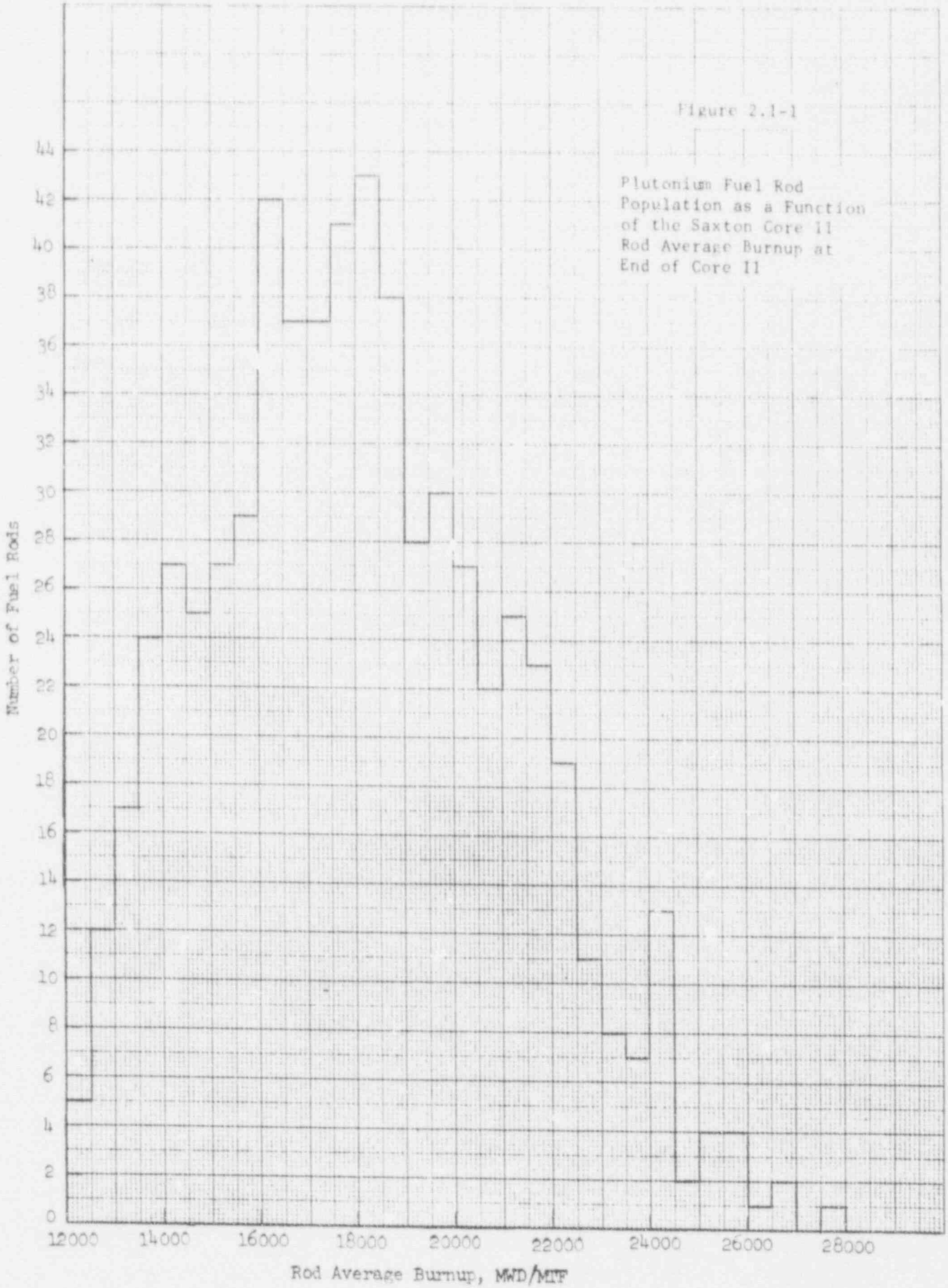


Figure 2.1-1
Plutonium Fuel Rod Population as a Function of the Saxton Core II Rod Average Burnup at End of Core II
Number of Fuel Rods
0
2
4
6
8
10
12
14
16
18
20
22
24
26
28
30
32
34
36
38
40
42
44
12000
14000
16000
18000
20000
22000
24000
26000
28000
Rod Average Burnup, MWD/MTF

2.2 NUCLEAR CHARACTERISTICS

2.2.1 Analytic Configuration

The analytic configuration used as the basis for the Saxton Core III nuclear design is shown in Figure 2.2-1.

The loose lattice plutonium fuel region which is made up of seven reconstituted assemblies is subdivided into zones distinguished by the amount of burnup the rods received during their previous operation in Core II. At intermediate positions between the loose lattice fuel rods open zircaloy tubes are installed to provide additional moderator while maintaining the desired flow characteristics. Three of these intermediate positions in each assembly contain open stainless steel tubes which are used to attach the fuel assembly nozzle.

The two UO_2 load follow assemblies are installed on the flats of the center nine assemblies (See Fig. 2.2-1) and contain variations in the U-235 enrichment. The purpose of these enrichment variations is to increase the number of rods that operate near the peak power rating. These two assemblies are to be interchanged at an intermediate point in core life. To avoid exceeding an imposed power limit when the assemblies are reversed, the enrichment patterns are identical. The additional water at positions required for structural rods in the tight lattice causes an appreciable increase in local power. To minimize this local power perturbation the structural tubes in the load follow assemblies are sealed from the coolant and contain a solid rod of either stainless steel or Inconel. To improve the power distribution characteristics of these assemblies five fuel rod locations contain open zircaloy or stainless steel tubes to provide additional moderator. In addition, the fuel density and pellet dimensional variations, Section 2.4, are also used to flatten power. Two stainless steel clad UO_2 rods with an enrichment of 5.7 W% U-235 are installed at specified corners in each assembly (See Figure 2.2-1).

2.2.2 Power Characteristics

Power distributions throughout the expected life of Saxton Core III were determined using a LEOPARD-PDQ-7 analysis sequence. The calculations show

that the peak power in both load-follow and loose-lattice assemblies will decrease with burnup with the loose-lattice rods burning down at a faster rate than the load-follow rods. Figure 2.2-2 shows the relative change in power calculated for each fuel type. Therefore, in order to maintain the peak power rating for as long as possible, the reactor power will be increased as the peaks burn down. Figure 2.2-3 summarizes the anticipated power operation and the resulting effect on the peak kw/ft power in each region. As shown in this figure the peak power is maintained constant for approximately the first 1500 hours operation by increasing the core power from 24.7 to 25.8 MWt. At that time the peak power in the load follow fuel has reached its design limit. Thereafter, the core power is increased at a reduced rate holding the peak power in load follow rods at their design limit. Because of the difference in burnup rate the power in the loose lattice rods reduces with increased burnup. At the end of the design life the core power reaches approximately 26.5 MWt and both load follow and loose lattice rods have operated at the peak power rating for a significant part of the total operating period. The actual power operation sequence will be based on power measurements at the beginning of life and periodically during operation.

It is apparent from the operating sequence summarized in Figure 2.2-3 that the expected values of core power (26.5 MWt), loose-lattice linear power (21.2 kw/ft), and load-follow linear power (17.6 kw/ft) will not all occur simultaneously at any point during the operation of Saxton Core III.

In addition, a small margin in total core power was included to insure that the objective linear power ratings in each region could be reached even if the local peaking factors were less than those anticipated. Therefore, the thermal-hydraulic and transient evaluations were made at a reactor power level of 28 MWt and with the design linear power ratings in each fuel type i.e. the design values of 24 kw/ft in the loose lattice assemblies and 19.9 kw/ft in the load follow assemblies. The design core and assembly power distribution is given in Figure 2.3-1. The power characteristics of the fuel rods in the loose lattice and load follow assemblies is shown in Figures 2.2-7 and 2.2-8.

2.2.3 LIFETIME CHARACTERISTICS

The lifetime available in Saxton Core III was determined using the LEOPARD-PDQ-7 analysis sequence and supporting one-dimensional PANDA calculations. The anticipated boron concentration requirement as a function of lifetime is summarized in Figure 2.2-4.

2.2.4 REACTIVITY CHARACTERISTICS

The reactivity characteristics of Saxton Core III were determined by means of both one-and two-dimensional calculations. The analytic procedure used was consistent with that producing good agreement between analysis and experiment in two previous Saxton cores. For purposes of the transient evaluation, conservative values were used within the range of parameters listed in Table 2.2-1.

Moderator Temperature Coefficient

The moderator temperature coefficient was determined in a series of one-dimensional radial PANDA calculations. The calculations were made at the beginning and end of the expected core life with an appropriate range of boron concentration. The results are summarized in Figures 2.2-5 and 2.2-6. As shown in these figures the results are within the defined parameter range.

Doppler Coefficient

Doppler and power coefficient calculations were carried out using one dimensional radial PANDA calculation. The latest methods for calculating fuel temperature were used and an effective resonance temperature was determined by multiplying this calculated temperature by an empirical factor determined from a correlation of power coefficient data from previous Saxton cores. Fuel temperature variations resulting from differences in power, fuel r[illegible] design and fuel type were included.

The doppler coefficient is governed primarily by the amount of U-238 present. However, since Core II also certained a small quantity of Pu-240, the doppler coefficient in this core was slightly more negative than that of Core I. Fuel (including both U-238 and Pu-240) will be removed to form the loose lattice region in Saxton Core III. However, the amount of Pu-240 present will actually be higher than that initially installed in Core II because of the buildup of this isotope during Core II operation. The end result is that the doppler and power coefficient as calculated for Core III is within the range of that determined for Core II.

Control Rod Worth

The worth of control rods in Saxton Core III was determined by using PDQ-7 two-dimensional calculations. A total bank worth of 18.4%Δk/k was determined which is an intermediate value to that of the two previous cores. The most reactive stuck rod was calculated to be worth 6.3%Δk/k. The bank worth with the maximum worth rod stuck withdrawn is thus 12.1% for Core III as compared to 11.7% for Core II. The worth of boron as a function of concentration was determined using one dimensional PANDA calculations. The results are summarized in Figure 2.2-9.

Steam Break Calculations

Adequate boron concentration will be maintained during Core III lifetime to insure at least a 1% shutdown by rods for the worst possible primary system cooldown that could result from a steam break. Hence the core will not return to critical as a result of this accident. Detailed calculations were performed to determine the minimum allowable boron concentration versus core lifetime. Table 2.2-2 lists the installed reactivity for various Saxton Core III conditions and specifies the minimum boron requirement to prevent return to critical as a result of a steam break cooldown.

Conclusions

The calculated response of Saxton Core III is intermediate to that of two previous cores. It represents no extrapolation from previous operation except for the higher linear power ratings. Even then, this higher linear power is reached at a lower total core power (28 MWt) than that reached in Saxton Core II (32 MWt).

TABLE 2.2-1

SUMMARY OF SAXTON REACTOR PHYSICS PARAMETERS

Parameter	Core I	Core II	Core III
Delayed Neutron Fraction, β_{eff}	0.0063	0.0049	0.0049
Prompt Neutron Life	1.6×10^{-5} sec	1.15×10^{-5} sec	1.5×10^{-5} sec
Doppler Coefficients	-1.6×10^{-5} Δk/k°F	-1.25×10^{-5} Δk/k°F	-1.6×10^{-5} Δk/k°F
	-0.4×10^{-5} Δk/k/°F	-1.0×10^{-5} Δk/k/°F	-0.4×10^{-5} Δk/k/°F
Moderator Temperature Coefficient Beginning of Life at 80°F	$+1.0 \times 10^{-4}$ Δk/k/°F Borated	$+0.3 \times 10^{-4}$ Δk/k/°F	$+1.0 \times 10^{-4}$ Δk/k/°F Borated
	-1.0×10^{-4} Δk/k/°F Rodded		-0.5×10^{-4} Δk/k/°F Rodded
at hot operating conditions*	-5.3×10^{-4} Δk/k/°F Borated -2.0×10^{-4} Δk/k/°F Rodded	-2.7×10^{-4} Δk/k/°F (1000 ppm)	-1.0×10^{-4} Δk/k/°F (BOL)
End of Life at Hot Conditions	-4.6×10^{-4} Δk/k/°F	-4.1×10^{-4} Δk/k/°F	-4.6×10^{-4} Δk/k/°F
Pressure Coefficient at 80°F	-1.0×10^{-6} to $+2.7 \times 10^{-6}$ Δk/k/psi		-1.0×10^{-6} Δk/k/psi to 2.7×10^{-6} Δk/k/psi
at Hot Conditions	1.9×10^{-6} Δk/k/psi to 5.1×10^{-6} Δk/k/psi	$+3.5 \times 10^{-6}$ Δk/k/psi (1000 ppm)	1.9×10^{-6} Δk/k/psi to 5.1×10^{-6} Δk/k/psi

* The moderator temperature was 530°F for Core I operation. This value was also used in Core II design analysis.

TABLE 2.2-1 (CONT'D)

SUMMARY OF SAXTON REACTOR PHYSICS PARAMETERS

Parameter	Core I	Core II	Core III
Void Coefficients at 80°F	-0.16 to +0.08% Δk/k/% Void		-0.16 to +0.08% Δk/k/% Void
hot operating	-0.39 to -0.14% Δk/k/% Void	-0.27% Δk/k/% Void	-0.39 to -0.14% Δk/k/% Void
Control Rod Worths All Rods		16.9% Δk/k	16.9% Δk/k
Banks worth with most Reactive Rod Stuck		11.7% Δk/k	11.7% Δk/k

TABLE 2.2-2

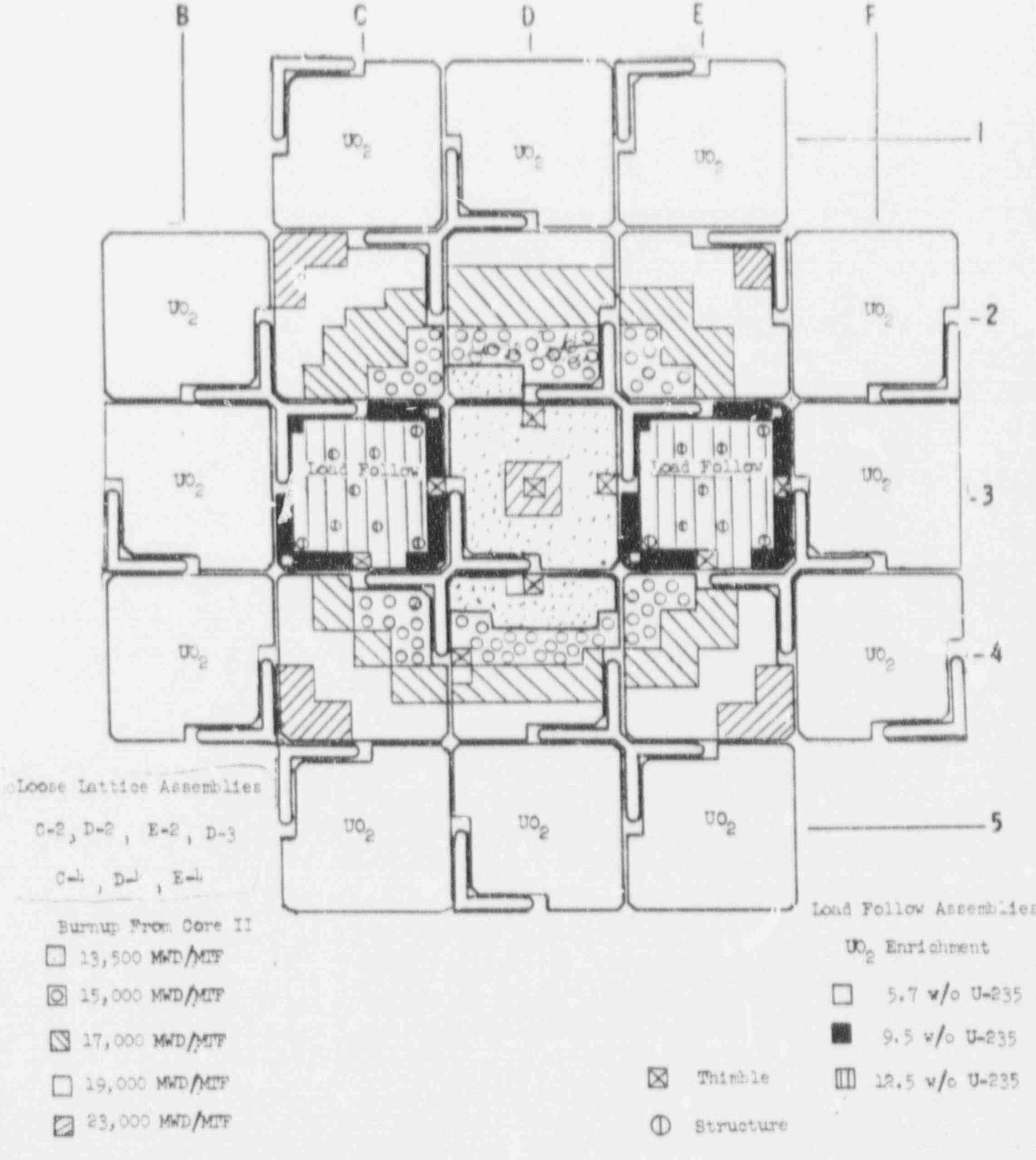
INSTALLED REACTIVITY AND MINIMUM BORON REQUIREMENTS FOR CORE III

Core Condition	Installed Reactivity % ΔK/K	Worth of Control Bank With Rod Stuck % ΔK/K	Uncontrolled Reactivity K_{eff} = 0.99 % ΔK/K	Maximum Reactivity Addition on Cooldown* % ΔK/K	Minimum Boron Concentration Requirement for:	
					Stuck Rod	Stuck Rod Steam Break
Hot Zero Power (BOL)	14.8	12.1	3.7	1.7 (600 ppm)	340	500
Hot Full Power (BOL) Equi Xe	10.6	12.1	-0.5	3.2 (600 ppm)	0	less than above
Hot Zero Power 5000 Hrs (EOL)	4.8	12.1	-6.3	2.2 (300 ppm)	0	0
Hot Zero Power 5000 Hrs (EOL)	4.8	12.1	-6.3	3.4 (0 ppm)	0	0

* This represents the maximum reactivity addition achieved in cooling from hot operating conditions to 212°F or to the temperature at which the moderator temperature coefficient changes from negative to positive whichever results in the greatest insertion.

Figure 2.2-1

Analytic Configuration Used in the Saxton Core III
Nuclear Design



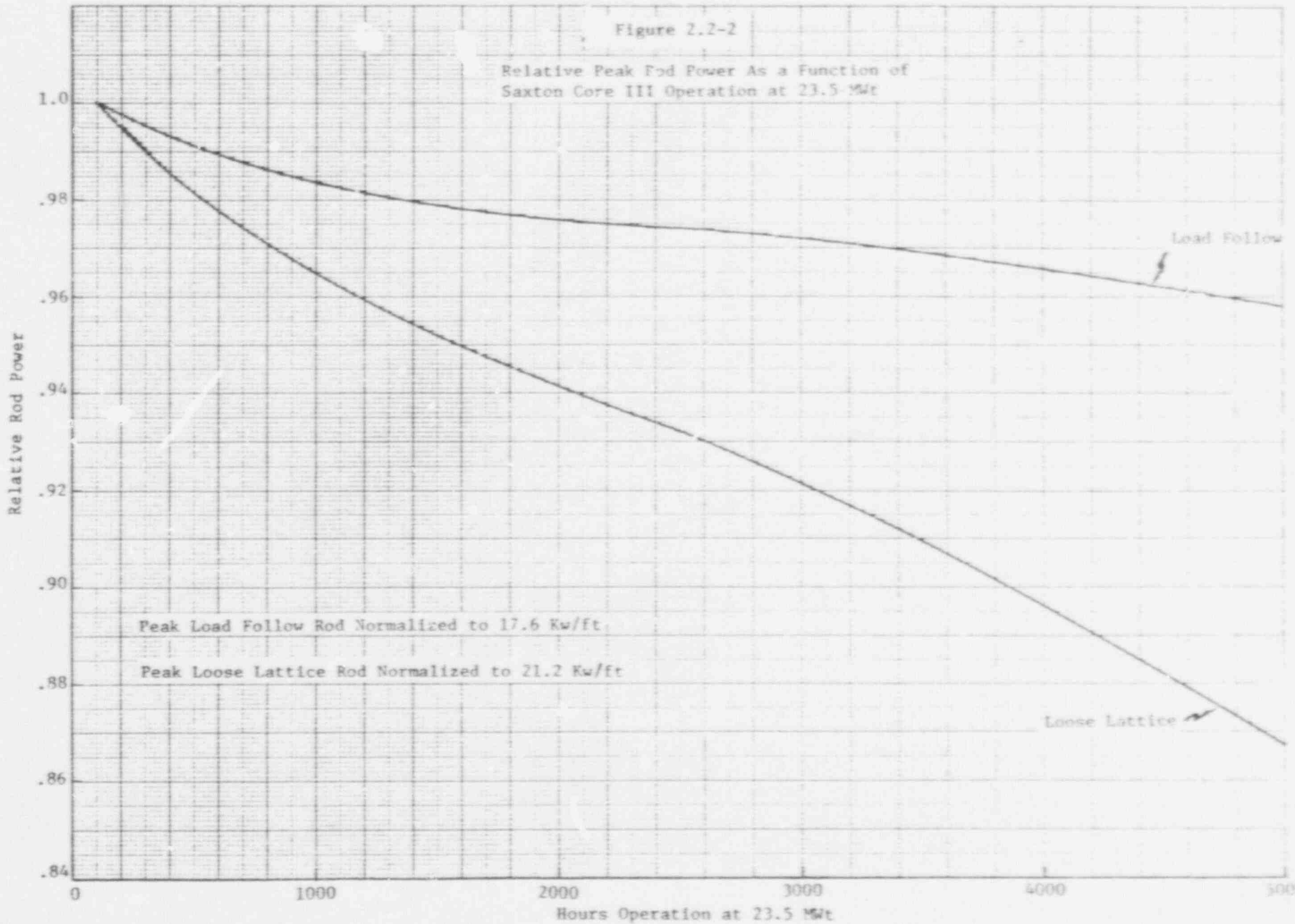


Figure 2.2-2
Relative Peak Fod Power As a Function of
Saxton Core III Operation at 23.5 MWt
1.0
.98
.96
.94
.92
.90
.88
.86
.84
Relative Rod Power
Load Follow
Loose Lattice
Peak Load Follow Rod Normalized to 17.6 Kw/ft
Peak Loose Lattice Rod Normalized to 21.2 Kw/ft
0
1000
2000
3000
4000
5000
Hours Operation at 23.5 MWt

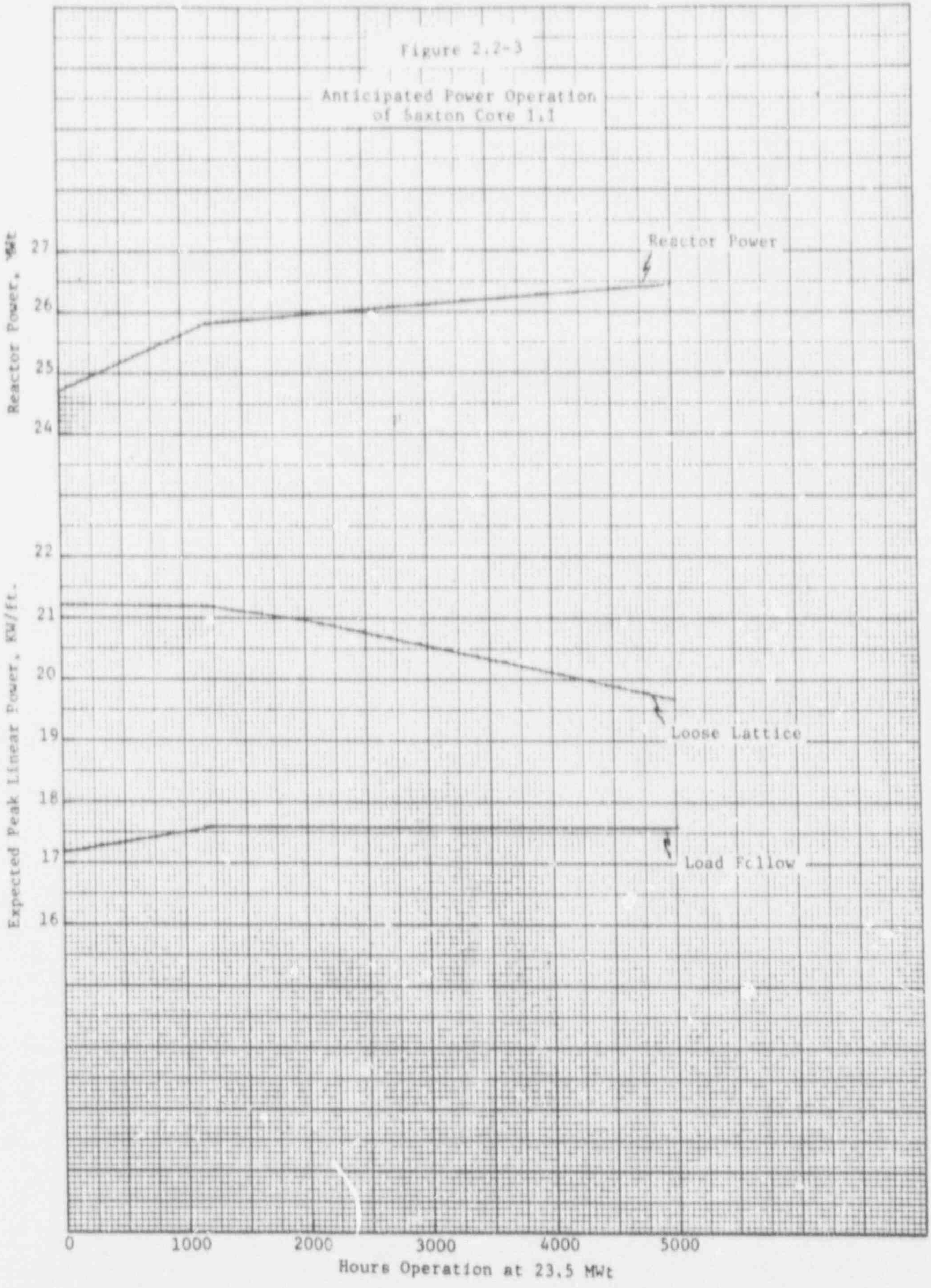


Figure 2.2-3

Anticipated Power Operation of Saxton Core 1.1

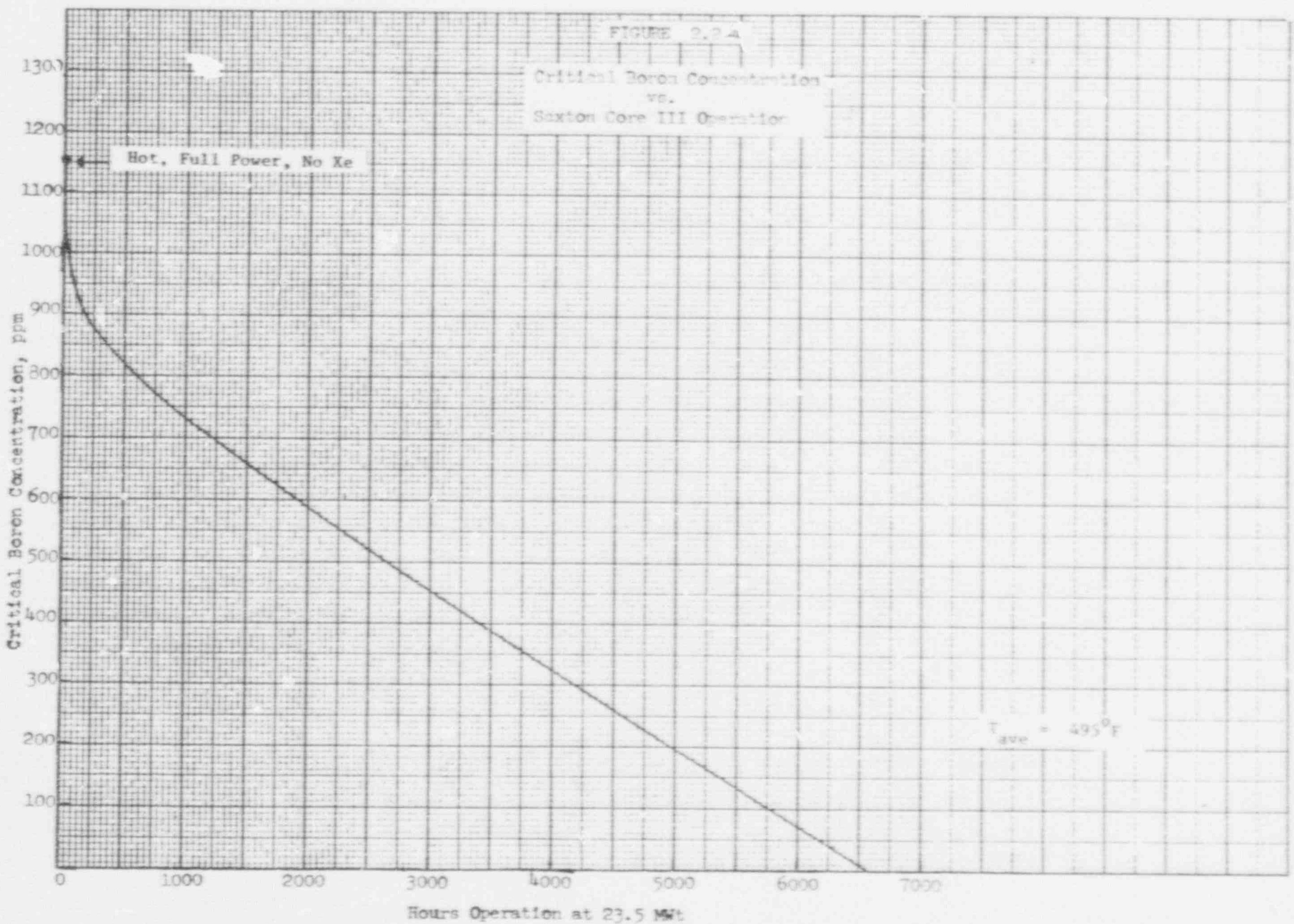


FIGURE 2.2-A
Critical Boron Concentration
vs.
Saxton Core III Operation
Hot, Full Power, No Xe
T_{ave} = 495°F
Critical Boron Concentration, ppm
1300
1200
1100
1000
900
800
700
600
500
400
300
200
100
0
1000
2000
3000
4000
5000
6000
7000
Hours Operation at 23.5 MWt

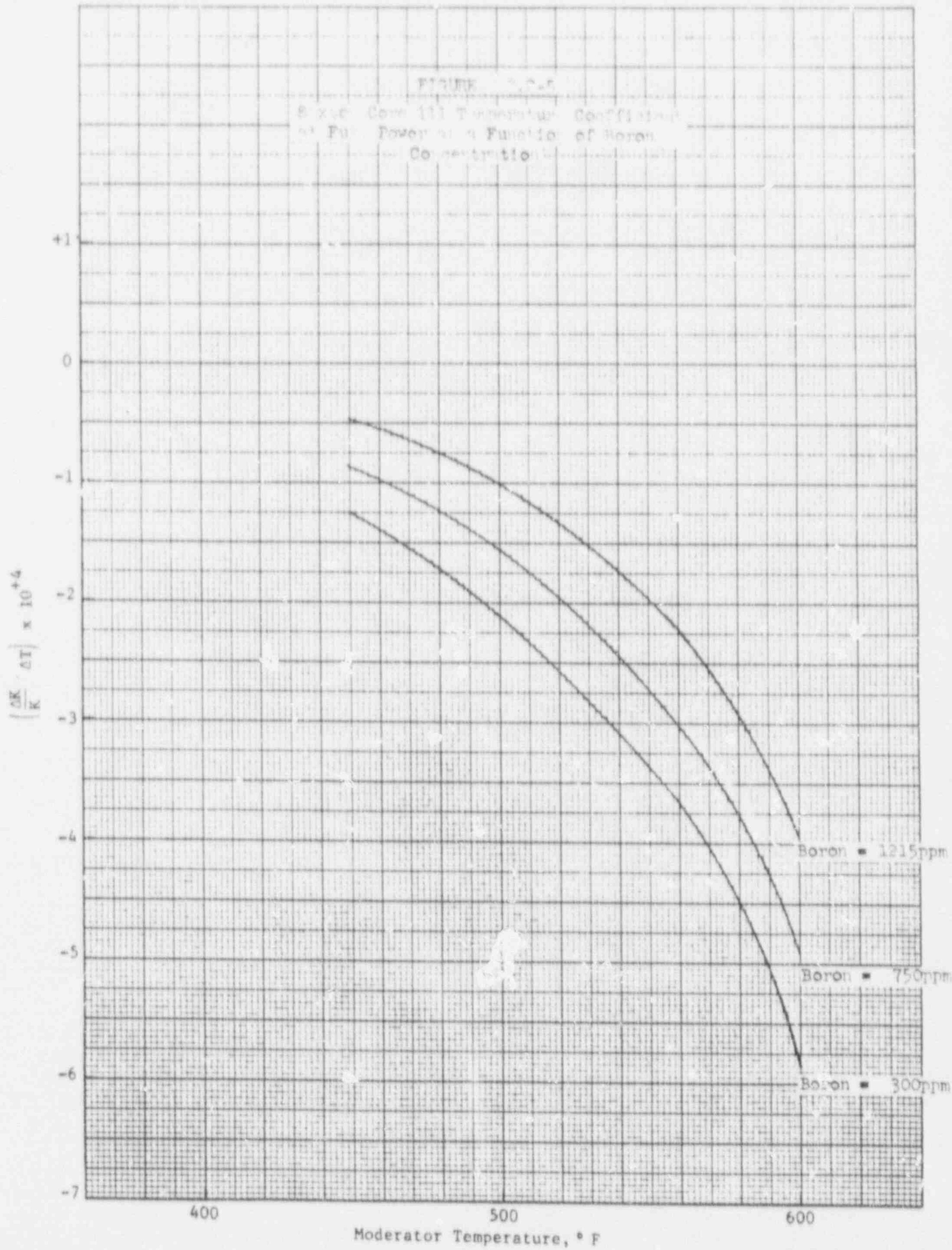
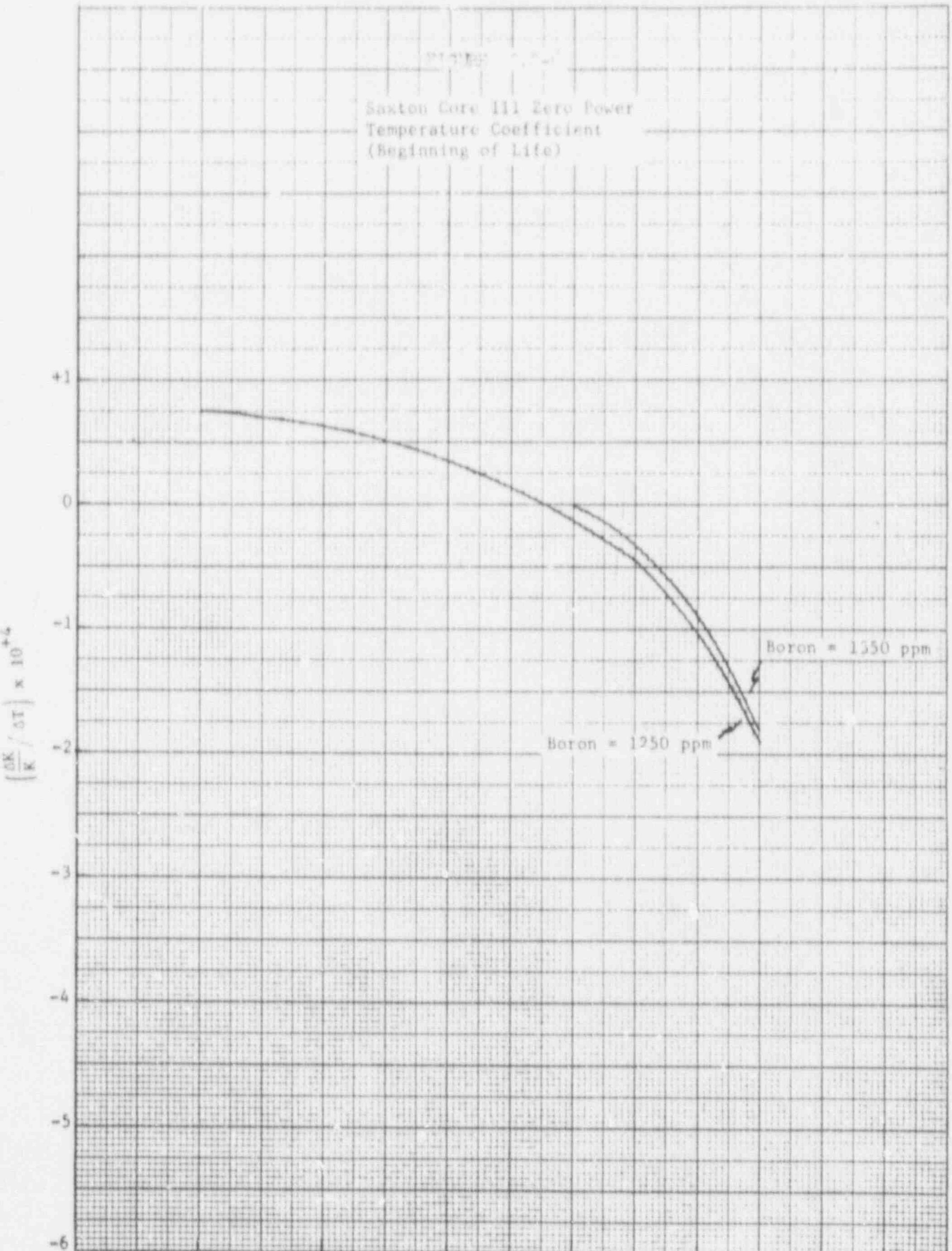
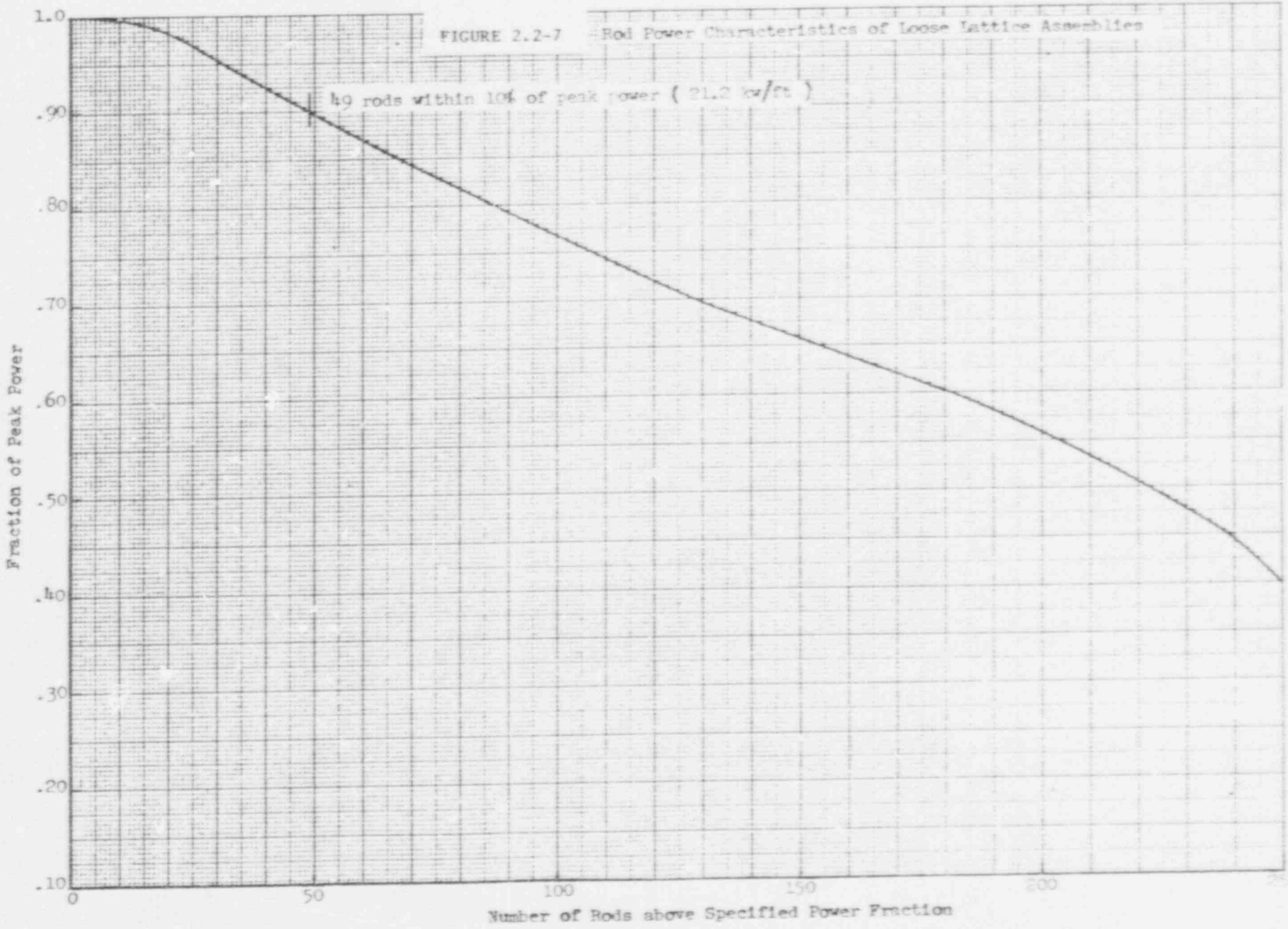


FIGURE
Core III
Power
of Boron
+1
0
-1
-2
-3
-4
-5
-6
-7
$\left(\frac{\Delta K}{K} \Big/ \Delta T\right) \times 10^{+4}$
400
500
600
Moderator Temperature, ° F
Boron = 1215ppm
Boron = 750ppm
Boron = 300ppm



Saxton Core III Zero Power
Temperature Coefficient
(Beginning of Life)
Boron = 1550 ppm
Boron = 1250 ppm
$\left(\frac{\Delta K}{K} / \Delta T\right) \times 10^{+4}$
+1
0
-1
-2
-3
-4
-5
-6
0
100
200
300
400
500
600
Moderator Temperature, °F

FIGURE 2.2-7 Rod Power Characteristics of Loose Lattice Assemblies



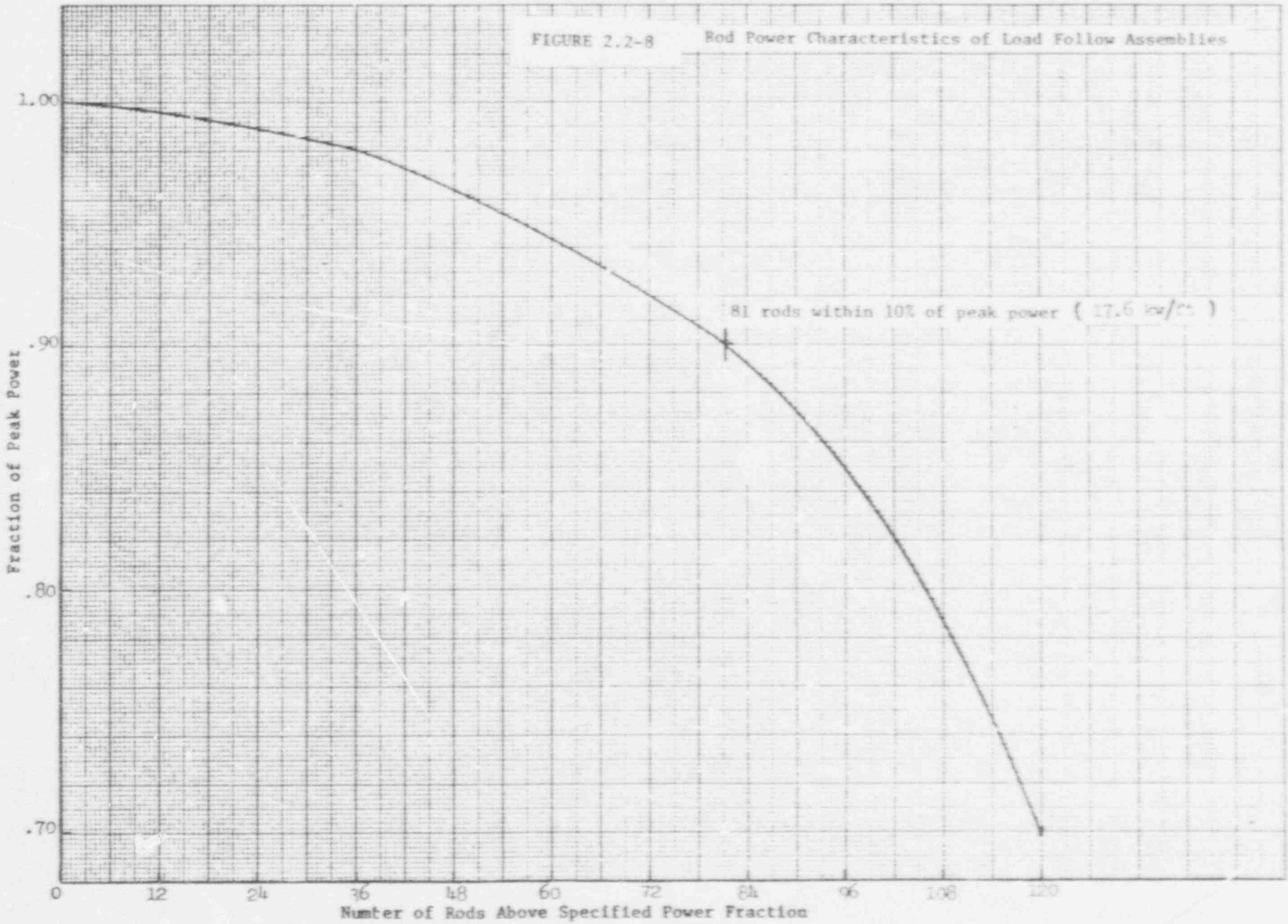


FIGURE 2.2-8 Rod Power Characteristics of Load Follow Assemblies

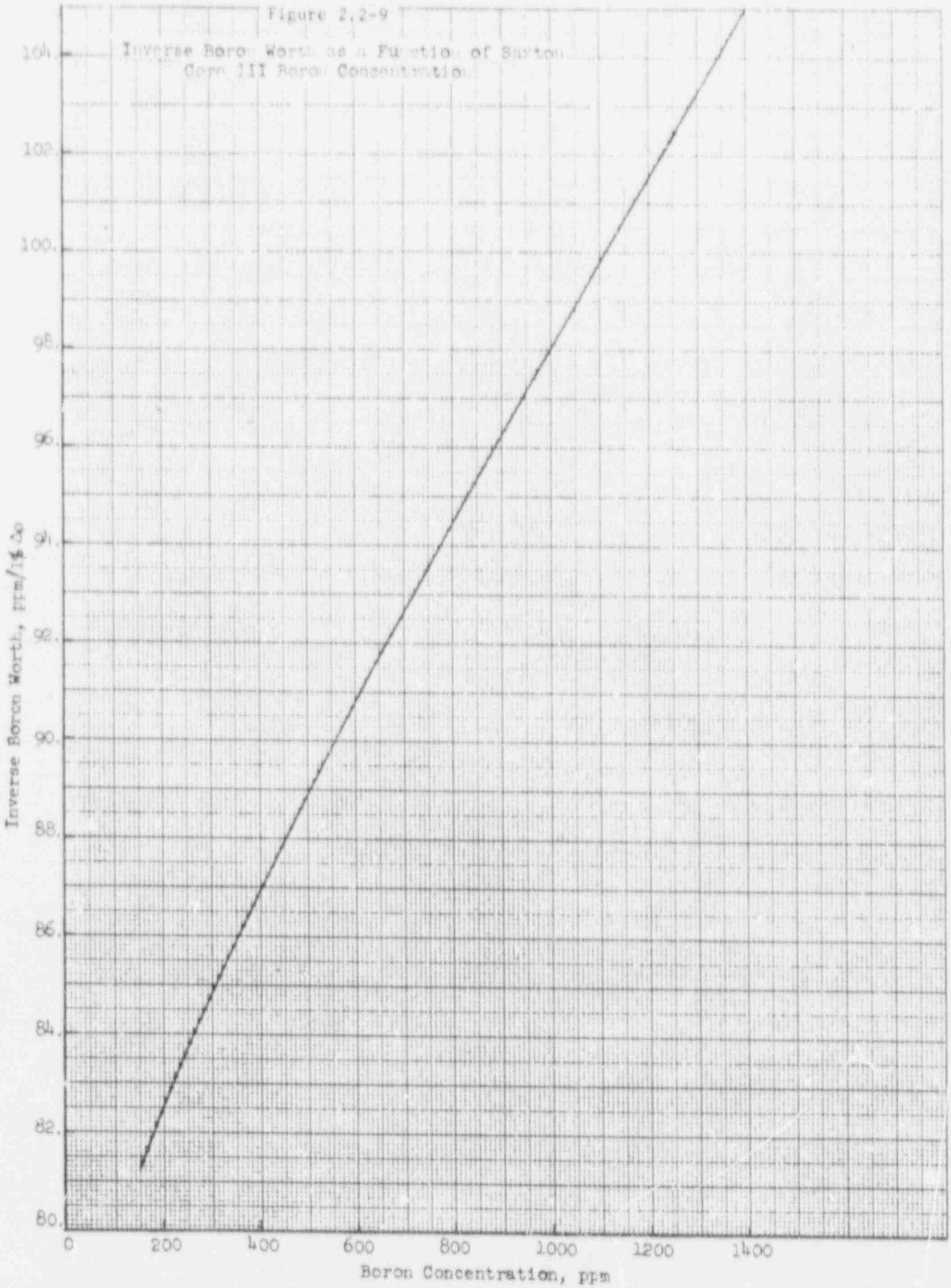


Figure 2.2-9

Inverse Boron Worth as a Function of Saxton Core III Boron Concentration

2.3 THERMAL-HYDRAULIC DESIGN

2.3.1 GENERAL

A detailed evaulation of the thermal and hydraulic characteristics of the proposed Saxton Core III configuration has been conducted. The thermal and hydraulic margins have been identified. It has been demonstrated that sufficient margins exist such that safety limits as defined below, are not exceeded during steady state operation and reactor transients.

The maximum fuel temperatures have been conservatively estimated for both the highest power load follow fuel rod and highest power loose lattice fuel rod. At no time during reactor operation will center melting occur in the core.

Extensive calculations have been performed to investigate the hydraulic conditions to be expected during the Saxton Core III operation. The minimum DNB ratio within the core will be greater than 1.30. At the maximum reactor control set point conditions adequate design margins were employed for the evaluation of the DNB ratio. The reactor system pressure and inlet coolant temperature have been conservatively selected to preclude the possibility of any adverse thermal and hydraulic conditions occuring within the core.

2.3.2 THERMAL AND HYDRAULIC DESIGN CRITERIA

To assess the acceptability of the Saxton Core III operating conditions, several thermal and hydraulic criteria must be met. The design criteria and design methods which were imposed and employed were the same as used in the analysis of Core II 35 MWT operation. They include:

(1) A DNB ratio for all fuel rod channels of greater than 1.30 at the reactor limiting control set point conditions. (See page 3.1-2).
(2) Center melting of the fuel is not permitted during normal steady state operation or anticipated transient conditions.

Evaluations were conducted primarily at the upper core protection limits employing the concept of hot channel factors. The results are, therefore, felt to be conservative. There exists a high degree of assurrance that actual conditions will not be nearly as severe as predicted in this analysis.

2.3.3 THERMAL AND HYDRAULIC ANALYSIS

The best estimate radial assembly power distribution which was employed for this Saxton Core III analysis is shown in Figure 2.3-1. This core configuration consists of seven loose lattice assemblies and two load follow assemblies. In Figure 2.3-2 the detailed rod by rod power values presented for the assemblies D-3 and E-3. These two assemblies contain the peak fuel rods of both the loose lattice rod and load follow rod types. In the analyses for these assemblies an 8% nuclear uncertainty factor was applied.

(1) Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratio of these quantities. The heat flux factors consider the local maximum at a point (the "hot spot"), and the enthalpy rise factors involve the maximum integrated value along a channel (the "hot channel").

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances and are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density, and enrichment; fuel rod diameter; pitch and bowing; inlet flow distribution; flow redistribution; and flow mixing.

(a) Engineering Hot Channel Factors

The engineering hot channel factors are essentially identical to those used in the evaluation of Saxton 35 MWt operation. The heat flux engineering hot channel factor, F_q^E, remains 1.045 and the $F_{\Delta H}^E$ is reduced to 1.12 due to the benefit of coolant mixing obtained by the use of the THINC code [1]. Table 2.3-1 is a tabulation of the design engineering hot channel factors.

(2) Departure From Nucleate Boiling

The evaluation of the Saxton Core III, DNB conditions have been made using the W-3 correlations[2]. The W-3 DNB design minimum value of 1.30 has been choosen statistically to insure a 95% probability that DNB will not occur with a confidence level of 95%. For fuel channels adjacent to the assembly enclosure, the prescence of the unheated wall effects the amount and degree of coolant mixing inside the channel. The effect of the unheated wall on the DNB ratio has been considered (modified W-3 correlation used)[3] in the design analysis using experimental data.

(3) Thermal and Hydraulic Design Paramenters

Table 2.3-2 presents the thermal and hydraulic characteristics for operation of Core III at its peak power level of 28 MWT. Shown are the thermal and hydraulic characteristics for both the most thermal limiting loose lattice type fuel rod and load follow fuel rod. These parameters have been calculated using the nuclear radial power distribution and hot channel factors previously presented.

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(1) Chelemer, H., Weisman, J., Tong, L.S. "Subchannel Thermal Analysis of Rod Bundle Cores", WCAP-7015, January 1967.

(2) L.S. Tong. "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution", J. of Nuc. Energy Vol. 21 pp 241-248, (1967).

(3) L.S. Tong, et.al., "Critical Heat Flux on a Heater Rod on the Center of Smooth and Rough Squares Sleeves, and in Line Contact with an Unheated Wall", ASME 67-WA/HT-29 (1967).

2.3.4 THERMAL ANALYSIS OF FUEL ROD

The characteristics of the two basic types of fuel rods which will be exposed to the highest thermal conditions during Core III operation have been established. The power and burnup history which were employed for Core III operation are shown in Figure 2.2-3. The thermal conditions and characteristics are summarized in Table 2.3-2 with the temperature estimates being provided below.

Fuel Central Temperature

The maximum fuel central temperatures for the core at the design power level of 28 MWt has been estimated to be 4540°F for the loose lattice rods and 4650°F for the load follow rods. At the maximum overpower condition of 112% power (31.4 MWt) a maximum central temperature of 4880°F is predicted for the loose lattice assemblies. This is approximately 120°F below melt @ 500 the fuel melting temperature predicted for this mixed oxide fuel. At this 112% overpower condition, the highest power load follow fuel rod (22.2 kw/ft) is expected to have a maximum central temperature of 4900°F. This is also below the melting temperature for UO_2 at the equivalent fuel burnup.

Fuel Clad Temperature

At the design full power condition of 28 MWt about 150 of the loose lattice fuel rods (60% of total loose lattice rods) can be expected to operate with a mean cladding temperature at some point on the fuel rod greater than 700°F. All of the 134 load follow fuel rods are expected to operate with some small fraction of their cladding temperature greater than 700°F at the maximum 28 MWt reactor operating power level. At the design basis power condition (all of the uncertainties in the power calculation considered to be maximum at the same time) the mean cladding temperature at the peak loose lattice power condition (24 kw/ft) would be approximately 725°F. At the comparable load follow condition (power of 19.9 kw/ft) the maximum mean cladding temperature would be 720°F.

TABLE 2.3-1

Engineering Hot Channel Factors

F_q^E	Pellet Diameter, Density Enrichment, and Eccentricity Rod Diameter, (Pitch and Bowing)	1.045
$F_{\Delta H}^E$	Pellet Diameter, Density Enrichment Rod Diameter, Pitch and Bowing	1.08
	Inlet Flow Maldistribution	1.07
	Flow Redistribution	1.02
	Flow Mixing	.95*
	Resulting $F_{\Delta H}^E$	1.12

* To the point of Minimum DNB ratio

TABLE 2.3-2

Thermal and Hydraulic Design Parameters

Total Core

Total Heat Output	28.0 MWt
Total Heat Output	95.56×10^6 Btu/hr
Heat Generated in Fuel	97.4%
System Pressure - Nominal	2250 psia
System Pressure - Minimum - Steady State	2200 psia
Total Flow Rate *	3.21×10^6 lb/hr
Effective Flow Rate for Heat Transfer	2.73×10^6 lb/hr
Flow area for Heat Transfer Flow (unit cell)	2.2 ft^2
Average Velocity Along Fuel Rods	6.83 ft/sec

Coolant Temperatures

Nominal Inlet	480 F
Maximum Inlet Including Instrument Errors and Deadband	485 F
Average Rise in Vessel	26.0 F
Average Rise in Core	30.5 F
Average in Vessel	493.0 F
Average in Core	495.2 F

Heat Transfer

Active Heat Transfer Surface Area of Fuel Rods	376.2 ft^2
Average Heat Flux	220,400 Btu/hr-ft^2
Average Thermal Output	6.62 kw/ft
Maximum Clad Surface Temperature at Nominal Pressure	657.4 F

* At 63 cycles.

TABLE 2.3-2 (Cont'd)

	Loose Lattice Assembly	Load Follow Assembly
Center Core Region	UO_2-PUO_2	UO_2
F_q Heat Flux Hot Channel Factor	3.65	3.01
$F_{\Delta H}^{NUC}$ Nuclear Radial Factor	2.62	2.10
$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor	2.94	2.35
F_Z^{NUC} Nuclear Axial Factor	1.33	1.38
Nominal Outlet Enthalpy Hot Channel	528.3 Btu/lb	540.9 Btu/lb
Saturation Enthalpy at Minimum Steady State Pressure	695.0 Btu/lb	695.0 Btu/lb
Maximum Heat Flux	799,800 Btu/hr-ft^2	662,300 Btu/hr-ft^2
Maximum Thermal Output	24.0 kw/ft	19.9 kw/ft
W-3 DNB Ratio at 100% Power Nominal Conditions	2.0	1.75

TOTAL CORE POWER = 28 MWT

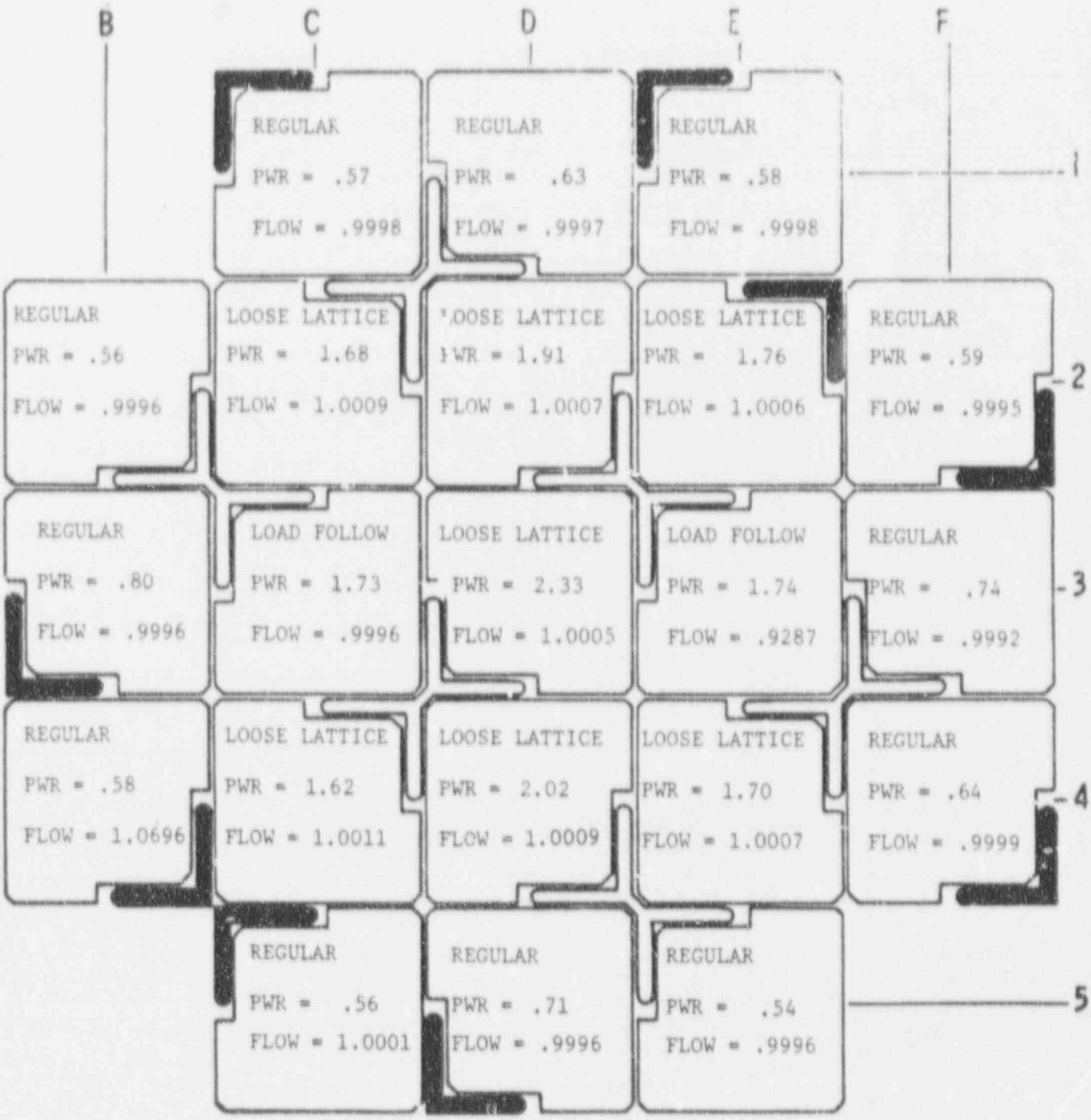
FUEL ROD POWER = 24.79 MWT (AVG. KW/FT = 6.62)

FOLLOWERS AND ELEMENT ROD POWER = 11.5% OF TOTAL

POWER GENERATED IN FUEL = 97.4%

ASSEMBLY POWER IS NORMALIZED TO CORE AVG. = 6.62 KW/FT

ASSEMBLY FLOW IS NORMALIZED TO CORE AVG. = 1.23×10^6 LB/HR-FT^2



THERMAL HYDRAULIC ANALYSIS - ASSEMBLYWISE
POWER AND COOLANT FLOW DISTRIBUTION

Fig. 2.3-1

BEST ESTIMATE NUCLEAR RADIAL POWER FACTORS IN

THERMALLY CONTROLLING CORE III ASSEMBLIES

(Normalized to core average of 6.62 $\frac{kw}{ft}$)

(Values do not include 1.08 uncertainity factor)

	2.38		2.41	X	2.38		2.37	
2.25		2.35		2.38		2.35		2.35
	2.30		2.39		2.39		2.35	
2.18		2.36		2.29		2.40		2.38
	2.20		2.27	X	2.30		2.40	X
		2.30		2.27		2.40		2.38
	2.07		2.31		2.36		2.34	
		2.11		2.24		2.34		2.35
					2.41		2.40	

EXPECTED PEAK
KW/FT = 21.2

ASSEMBLY D-3 LOOSE LATTICE

					1.85	1.55	1.56	1.11
	1.66	1.92	1.92	1.88	1.84	1.81	X	1.38
	1.92	1.89	X	1.90	X	1.76	1.61	1.36
	1.92	1.82	1.91	1.91	1.84	1.61	1.56	1.40
	1.90	1.80	1.86	X	1.82	1.56	1.60	X
1.90	1.90	1.82	1.87	1.90	1.83	1.60	1.56	1.40
1.91	1.91	1.91	X	1.89	X	1.76	1.61	1.38
1.94	X	1.92	1.91	1.91	1.84	1.77	X	1.41
1.56	1.94	1.82	1.79	X	1.89	1.83	1.43	1.42

EXPECTED PEAK
KW/FT = 17.6

ASSEMBLY E-3 LOAD FOLLOW

Figure 2.3-2

2.4 MECHANICAL DESIGN

2.4.1 CORE LOADING

The 21 main fuel assemblies (9 x 9 rod array) in Saxton Core III will be made up of seven loose lattice assemblies, two load follow assemblies, and twelve UO_2 assemblies from Cores I and II.

The seven loose lattice assemblies will consist of six 36-rod assemblies and one 32-rod assembly to accommodate a four rod removable sub-assembly. The rods in the loose lattice assemblies will be composed of irradiated rods removed from Core II assemblies and loaded into new assemblies containing Zircaloy tubes which operate filled with coolant at every other rod position.

[illegible] 34 [illegible]

The two load follow assemblies will contain 62 removable fuel rods with design variations in fuel pellet diameter and density. These rods are installed at the same pitch as previous 72 rod assemblies. All nine assemblies (seven loose lattice and two load follow) in the center of the core will have removable top nozzles.

As with the present core in Saxton, two rod positions in each of the main fuel assemblies in Core III will be used for either in-core instrumentation, source rods, or removable fuel rods, depending on the location of the fuel assembly in the core. In addition, the special L-shaped assemblies presently installed in corner slots in the peripheral fuel assemblies will be reused with the fuel assemblies in Core III.

The remaining fuel loading will consist of six control rod followers presently in the reactor, four removable 3 x 3 type fuel sub-assemblies in the peripheral UO_2 fuel assemblies and the special L-shaped assemblies at present in the peripheral slot positions.

A breakdown of the number of plutonium and UO_2 fuel assemblies in Core III is given in Table 2.4-1.

TABLE 2.4-1

FUEL ASSEMBLIES IN CORE III

	Number of Assemblies			
	72-Rod	63-Rod	36-Rod	32-Rod
Loose lattice assemblies mixed oxide	-	-	6	1
Load follow assemblies UO_2	2	-	-	-
Cores I and II UO_2 assemblies	8	4	-	-

2.4.2 FUEL ASSEMBLY DESIGN

1. Overall Construction

The construction of the fuel assemblies remains essentially the same as for Cores I and II except that the top nozzles of the 9 center assemblies are removable. No change has been made in the overall dimensions of the fuel assemblies. The fuel rod pitch and cross section of the peripheral 12 assemblies remains identical to that shown on Fig. 203.1 of the original Saxton Final Safeguards Report.

2. Removable Top Nozzles

As discussed previously, the center nine assemblies differ from previous assemblies in that the top nozzle is removable. Each removable top nozzle is fastened to the assembly by three stainless steel tie rods and held by captive nuts with integral locking cup washers.

The tie rods consist of Type 304 stainless steel tubing, .391 inch diameter by .015 inch wall thickness and Type 304 stainless steel special end plugs. The bottom end plug of each tie rod is welded into the bottom nozzle and the rod passes through the grids as a normal fuel rod. The top end plug contains two square sections with a threaded section at the lead end. The threaded section protrudes

through the top nozzle and the captive nuts are installed and torque loaded to 75 in-lbs. One of the square sections fits into the nozzle end plate and prevents rotation under torquing. The other square section shoulders on the nozzle end plate thus fixing the fuel assembly overall length and preventing overloading the enclosure.

The top of the enclosure assembly has a specially prepared reinforced edge which fits into, and is held by, the top nozzle plate. Each enclosure assembly and top nozzle is a matched pair to ensure proper alignment. The fuel assemblies, with top and bottom nozzles, are depicted in Figure 2.4-1.

3. Load Follow Assemblies

These two assemblies are of similar construction and cross-section as the previous Core II assemblies except for the removable top nozzle. The assemblies contain 62 removable fuel rods of various types. The rods in these assemblies are held at the regular Saxton nominal pitch of .580 inches. The cross-section of these assembles and positions of each type of rod are shown in Figure 2.4-1. Four of the load follow tie rods (Two per assembly) will be filled with Inconel and two of the tie rods (one per assembly) will be filled with stainless steel to minimize local power perturbations at the boundary between the load follow and loose lattice assemblies. The load follow tie rods are depicted in Figure 2.4-2. Five fuel rod locations in each assembly will contain stainless steel or zircaloy tubes open to the coolant.

4. Loose Lattice Assemblies

The 7 loose lattice assemblies consist of four 36 rod Type "A", two 36 rod type "B" and one 32 rod assembly.

The 32 rod assembly is similar to the type "A" assembly except that it contains a 3 x 3 removable sub-assembly with 4 fuel rods on th loose lattice pitch. This sub-assembly will contain rods with the highest burnups in the core.

The only difference between Type "A" and Type "B" loose lattice assemblies is that the position of the irradiated rods and dummy rods are reversed to maintain uniform loose lattice pitch across the core. This means that the position of the tie rods holding the top nozzle is also changed, but otherwise the tie rods are identical. The loose lattice tie rods will be open to the primary coolant permitting a ready entry and exit of water via two 1/16 inch diameter holes near the bottom of the clad and two 1/8 inch diameter holes near the top of the cladding. The loose lattice tie rods are shown in Figure 2.4-3.

The loose lattice assemblies are the same as the load follow assemblies in construction but the fuel rod effective pitch is increased. The assemblies will contain previously irradiated rods in every other rod position of the Core III assemblies on a staggered pitch producing an effective fuel rod pitch equal to $\sqrt{2}$ times the previous pitch, i.e., .820 inches.

The remaining rod positions will be occupied by tubes fabricated from Zircaloy fuel cladding and containing primary coolant. The cross section of these assemblies is shown on Figure 2.4-1.

2.4.3 FUEL ROD DESIGN

1. <u>Loose Lattice Fuel Rods</u>

These fuel rods, which are described in the Saxton Core II Plutonium Project Safeguards Report, will be selected from previously irradiated plutonium rods removed from Core II fuel assemblies. Figure 2.4-4

depicts a typical plutonium rod. The rods will be inspected prior to being loaded into the new assemblies.

a. Acceptance Criteria

The general basis for acceptance of irradiated PuO_2-UO_2 rods will be satisfactory performance of a rod in Core II and an absence of anomalous conditions as determined by visual and dimensional inspections at the time of reconstitution. Leak testing of the rods will serve as a further check on fuel rod integrity. Rods with cracks, blisters, or collapse in the unsupported area of the plenum will not be loaded into loose lattice assemblies.

b. Water Rods

The water rods are basically dummy fuel rods which are utilized in the alternate fuel assembly lattice positions to provide the desired flow characteristics. These rods shown on Figure 2.4-5 are fabricated from Zircaloy-4 cladding (nominally .391 inch diameter by .023 inch wall) and fitted with plain end plugs. Each rod has two 1/8 inch diameter holes at the bottom and two 1/16 inch diameter holes at the top of the cladding to allow the flow of primary coolant through the rods.

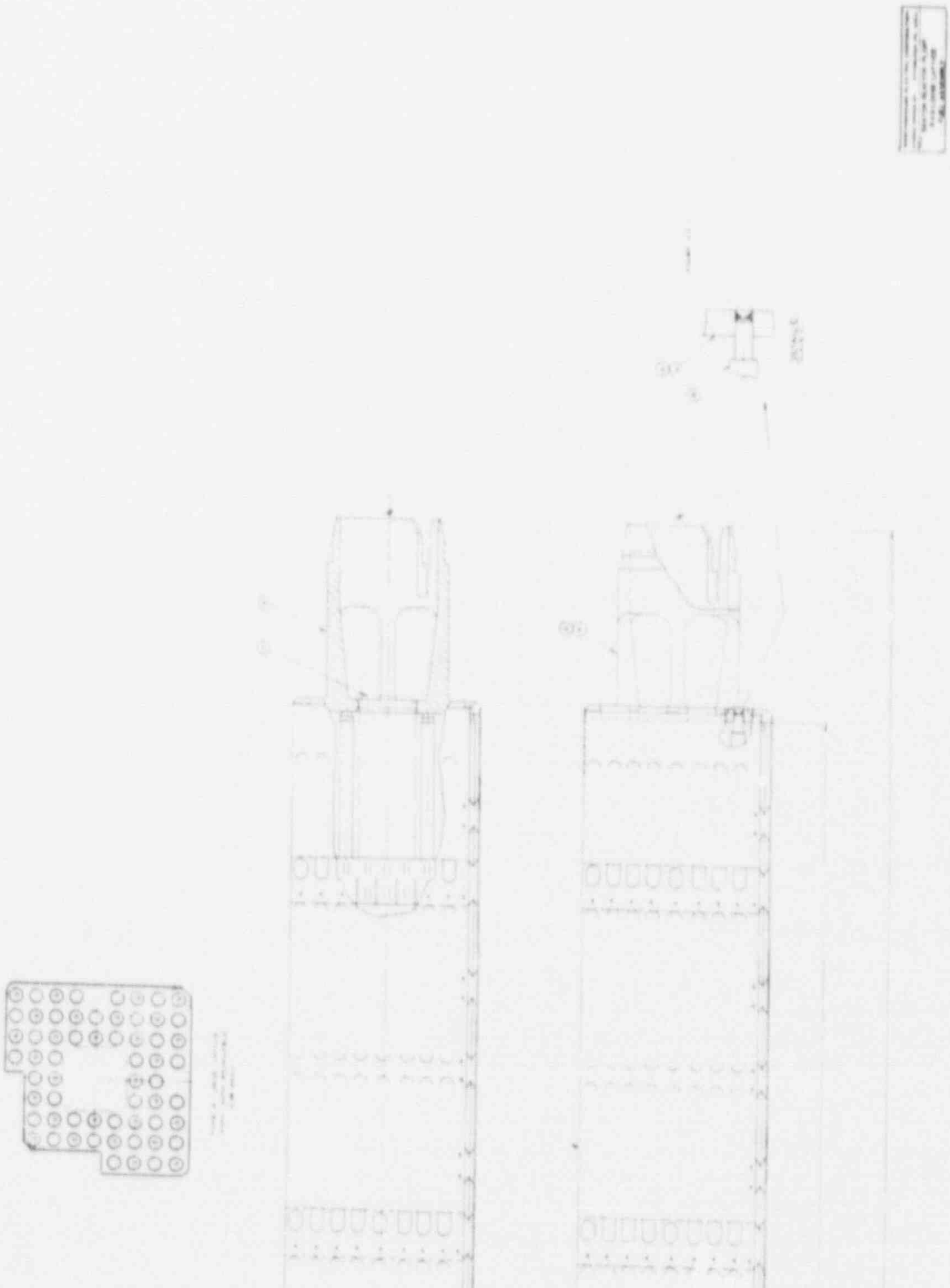
2. Load Follow Fuel Rods

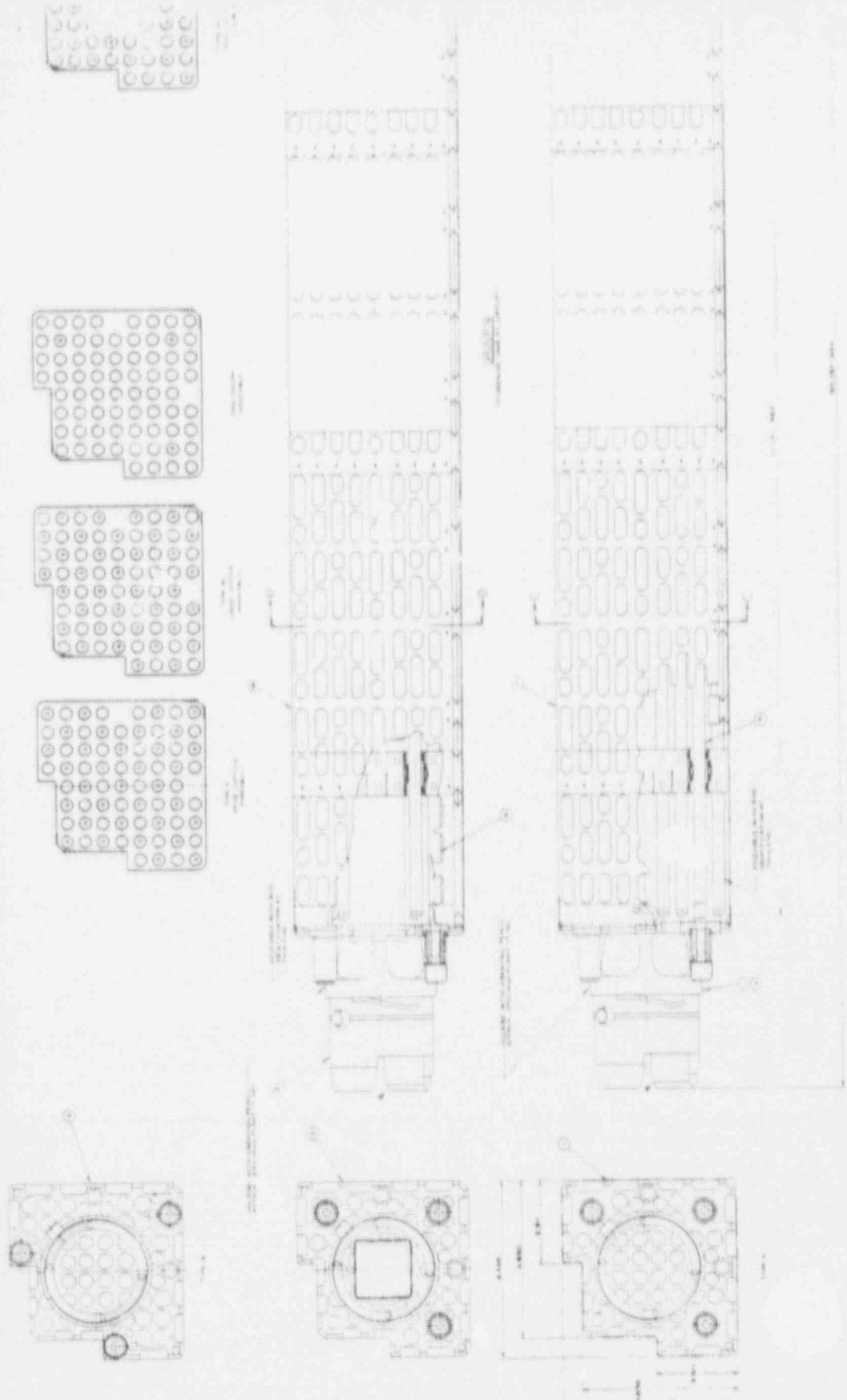
These are all new rods of the pelletized UO_2 type. All of the rods have the same overall dimensions and are removable. There are 124 rods in the two assemblies. The rods are made up of one of three enrichments: 5.7, 9.5, and 12.5 w/o U-235, for power distribution control. The other main variables are: (a) pellet densities of 89.5 to 94.5 percent theoretical; (b) fuel rod internal atmosphere;

(c) nominal pellet to clad gaps of 5.5 to 9.5 mils.(1)

The majority of the rods are clad (0.023 inch wall) with cold worked and stress relieved Zircaloy-4 tubing. A small number of rods are fabricated from annealed Zircaloy-4 tubing. Sixteen rods are clad with stainless tubing, Type 304, with a 0.015 inch wall.

(1) WCAP- 7219, Addendum to Saxton Core III License Application (Westinghouse Confidential), July 1968





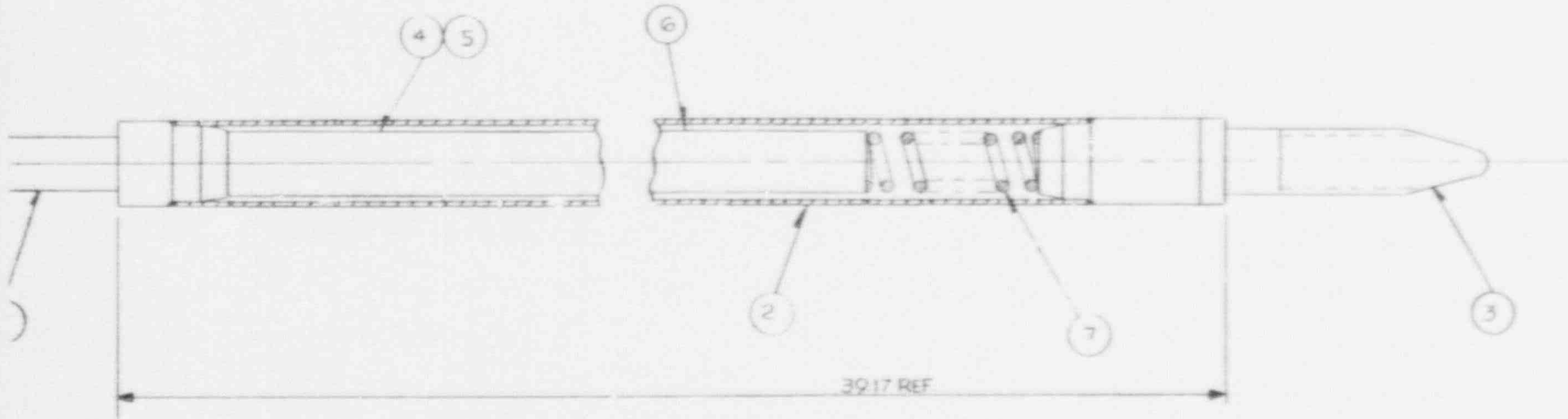


Figure 24-2

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.

TITLE:
SAXTON REACTOR PLANT
9X9 LOAD FOLLOW
TIE ROD ASSEMBLY

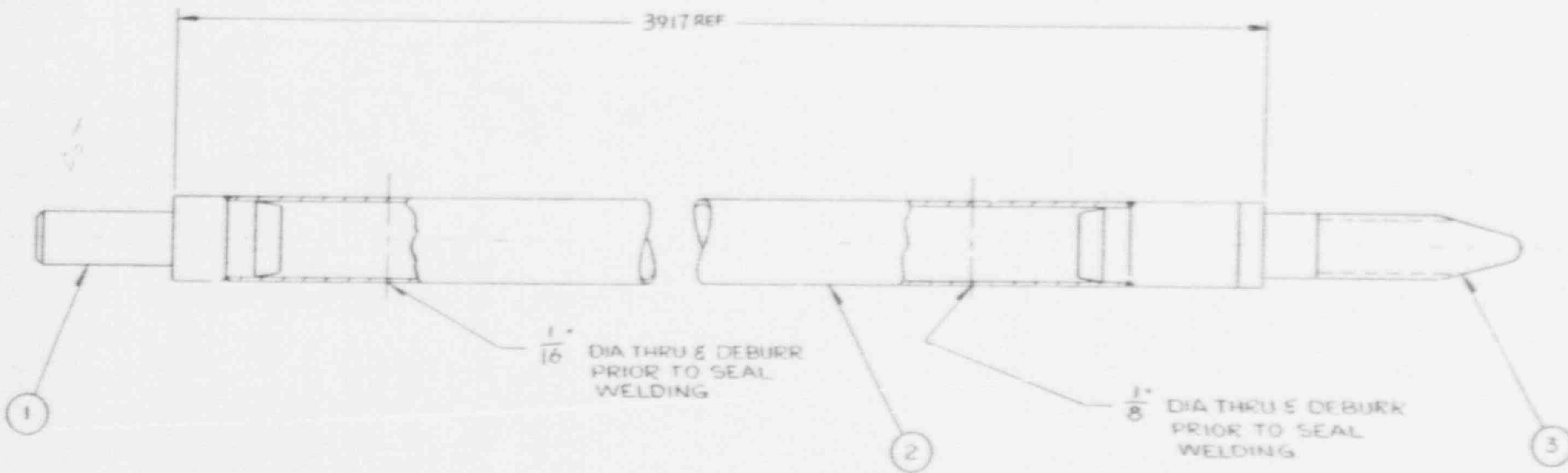
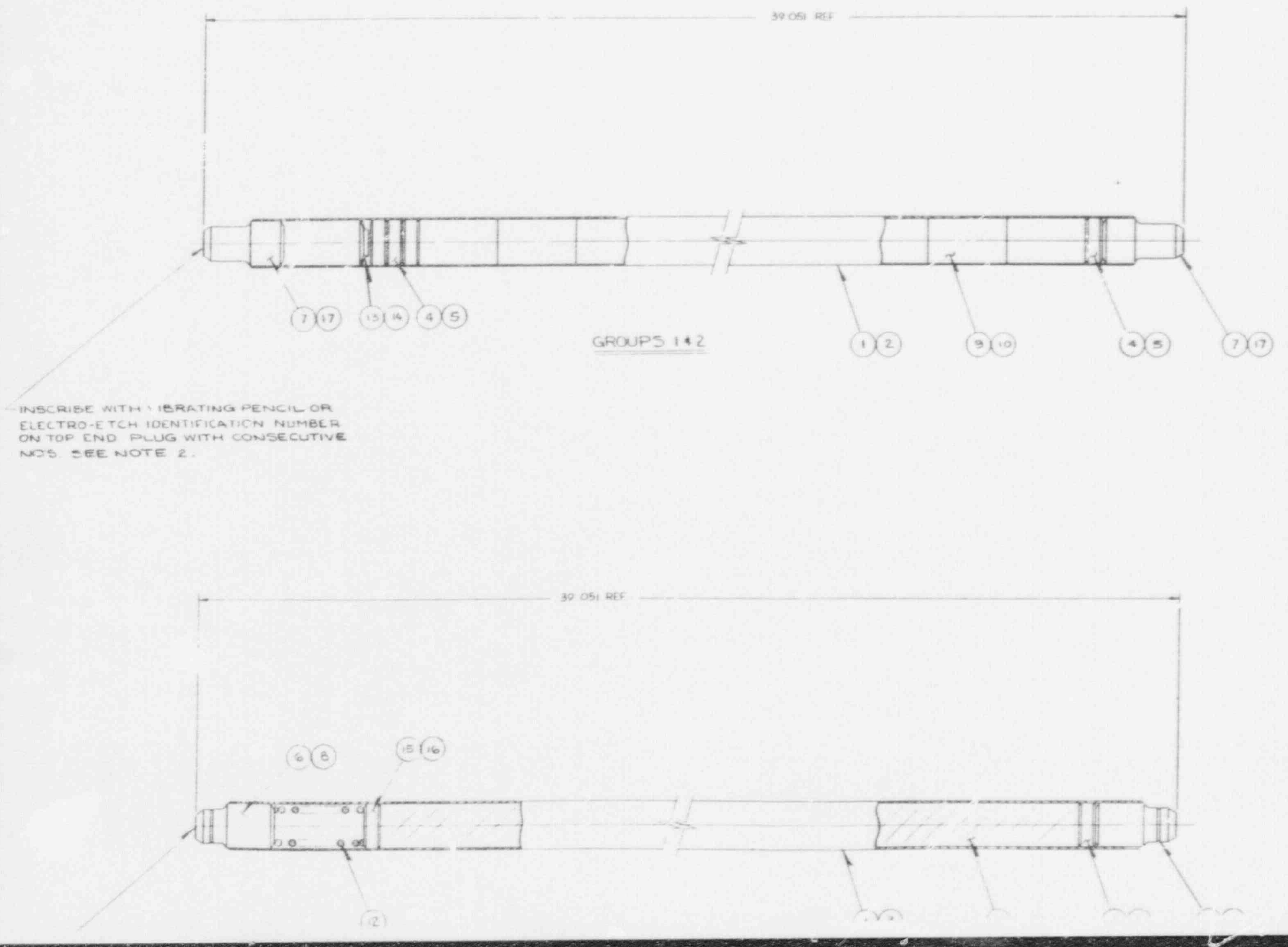


Figure 24-3

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIV. PITTSBURGH, PA. U.S.A.
TITLE: SAXTON REACTOR PLANT 9x9 LOOSE LATTICE TIE ROD ASSEMBLY



39.051 REF
7 17
13 14
4 5
GROUPS 1 & 2
1 2
9 10
4 5
7 17
INSCRIBE WITH VIBRATING PENCIL OR
ELECTRO-ETCH IDENTIFICATION NUMBER
ON TOP END PLUG WITH CONSECUTIVE
NOS. SEE NOTE 2.
39.051 REF
6 8
15 16
12

2.5 FUEL PERFORMANCE EVALUATION

2.5.1 BACKGROUND

The objectives of the Saxton Core III irradiation program are to:

1. validate fuel element performance predictions, including determination of power/burnup failure limits;

2. demonstrate performance capability of Zircaloy clad oxide fuel elements over a broad spectrum of burnups and power levels; and

3. obtain depletion characteristics and transuranic isotope generation data for high burnup, mixed oxide fuel.

Because the Saxton Core II mixed oxide fuel rods were designed for relatively low peak burnups (20,000 MWD/MTF) and operation at power densities ≤ 16 kw/ft, there is a significant risk of failure of certain of these rods in Core III. By careful selection and placement of these rods in the loose lattice assemblies, it is possible to control their burnup and operating power levels and thus permit power/burnup limits to be established while operating safely and in full compliance with the reactor license Technical Specifications.

It is planned to operate the lead rods in Saxton Core III loose lattice at a peak expected linear power rating of 21.2 kw/ft. Design analyses predict clad strains and internal gas pressures which exceed design limits early in Core III if rods with highest prior burnup are operated within 10% of the peak power. Therefore, it is necessary to restrict the highest power rods in Core III to those of relatively low prior burnup (≤ 18,000 MWD/MTF). Similarly, the high burnup rods will be limited to operation at lower power ratings in order to achieve acceptable lifetimes. The high burnup rods which are likely to fail first will be located in the center removable subassembly to permit easy access when removal of these rods becomes necessary.

2.5.2 EXPECTED FUEL PERFORMANCE OF SAXTON CORE III RODS

The operation of all rods in Saxton Core III has been analyzed and the results compared with the following PWR fuel element design criteria:

1. Cladding strains less than 1%.
2. Internal pressure less than external (system) pressure.
3. Clad stresses less than 0.2% yield strength at operating conditions.
4. No center melting of fuel
5. DNB ratio of at least 1.3.

The latter two limits are considered in Section 2.3 of this document.

Table 2.5-1 summarizes the projected power-burnup combinations for all rods to be irradiated in the center nine assemblies of Saxton Core III. It includes both beginning-of-life and end-of-life burnups as well as peak power densities expected for fuel rods in the seven loose lattice assemblies. The table also shows the predicted peak power and burnup for the rods in the two load follow assemblies.

The analysis of expected fuel performance is based on current design procedures and material properties data. Among the uncertainties considered in the hot channel interpretation of the code predictions were the nuclear and engineering factors, Core II power history, projected Core III power levels, and the dimensional uncertainties (diametral gap, fuel density and plenum size). The fuel rod performance was generally analyzed on a "most probable" basis; however, the studies also evaluated the effect of "worst combinations" of dimensions, and projected power history.

Analysis of the four rods to be located in the center removable subassembly indicates a high probability of failures early in Core III because of large expected cladding strains (> 2%) relative to the 1% strain design criterion and high internal gas pressure. Any failures in these rods would provide a better indication of performance capabilities of other rods in the experiment, i.e., help predict when failures are likely to occur and confirm the mode of failures. The design codes also predict failures in some of the loose

lattice rods outside the subassembly due to excessive internal pressure at moderately high burnups. However, such failures are expected to be statistical, i.e., will not all occur at one time, because of the broad spectrum of burnups and power levels represented and statistical variations in diametral gaps, densities, etc. Irradiation of these rods will continue as long as operation is judged safe and technical specification requirements are met.

Previous observations of high power CVTR and Saxton test rods indicate that failures will occur as either:

1. short circumferential cracks associated with clad ridging at pellet interfaces, or

2. local clad blistering with short, randomly oriented cracks.

In either case, cladding strains will be small with no significant "ballooning." Thus, any such failures will not restrict coolant flow or have significant effects on adjacent fuel rods. In addition, evidence to date on defected rods shows that the cracks will not propagate to produce catastrophic fuel rod failures. Furthermore, the nature of the defects (short ruptures) would limit the activity release and thus permit continued operation of the reactor with a limited number of failed rods. To further assess the effects of cladding strain, thermal-hydraulic analyses of the flow blockage required to reduce the steady-state DNBR to the lower limit of 1.3 have been performed for the loose lattice fuel assemblies, since intentional operation to failure is planned for this type of fuel. Figure 2.5-1 summarizes these results and indicates that diametral strains would have to be (>35%). It is concluded that loose lattice cladding failures are unlikely to result in violation of the DNBR design limit of 1.3, since cladding strains in failed rods are generally found to be small (<5%) compared to the 18-35% values required.(1) Further experimental evidence supporting this conclusion is obtained from out-of-pile elevated temperature burst tests of unirradiated Zircaloy tubing

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(1) WCAP-3850-1, October 1967

in which failure strains in the range 10-19% have been observed [2, 3]. Failure strains measured in elevated temperature burst tests of irradiated Zircaloy tubing samples have also generally ranged from 5-20% [4].

In summary, the Saxton Core III irradiation testing is expected to provide valuable information in the validation of fuel element design procedures; although failures are expected in certain rods of the loose lattice assemblies, all requirements of the reactor Technical Specifications will be met.

(2) WCAP-3850-2, March 1968
(3) WCAP-7163, January 1968
(4) WCAP-3850-3, to be published

TABLE 2.5-1

SUMMARY OF POWER RATINGS AND BURNUPS EXPECTED

FOR SAXTON CORE III

Group No.	Nominal Peak Linear Heat Rating, kw/ft	No. of Rods	Peak Initial Burnup, MWD/MTF	Peak Expected EOL Burnup, MWD/MTF
LL-1*	20	4	> 32,000	55,000
LL-2	21.2-18.7	47	16,000-19,000	39,000-45,000
LL-3	18.7-16.3	43	19,000-21,000	39,000-44,000
LL-4	17.4-14.3	58	21,000-23,000	38,000-44,000
LL-5	14.3-10.6	78	23,000-27,000	36,000-44,000
LL-6	10.6-8.5	20	27,000-32,000	36,000-45,000
LF-1	17.6-15.8	80	0	19,000-21,000**
LF-2	15.8-14.1	26	0	16,800-19,000**
LF-3	14.1-12.3	12	0	14,700-16,800**
LF-4	12.3-8.8	6	0	10,500-14,700**

* These four rods are in the center removable subassembly.

** These values assume constant exposure at the peak linear power level indicated. Since the two load follow assemblies are to be interchanged midway through Core III life, the peak burnup values for the high linear power rods will be slightly reduced from the tabulated values.

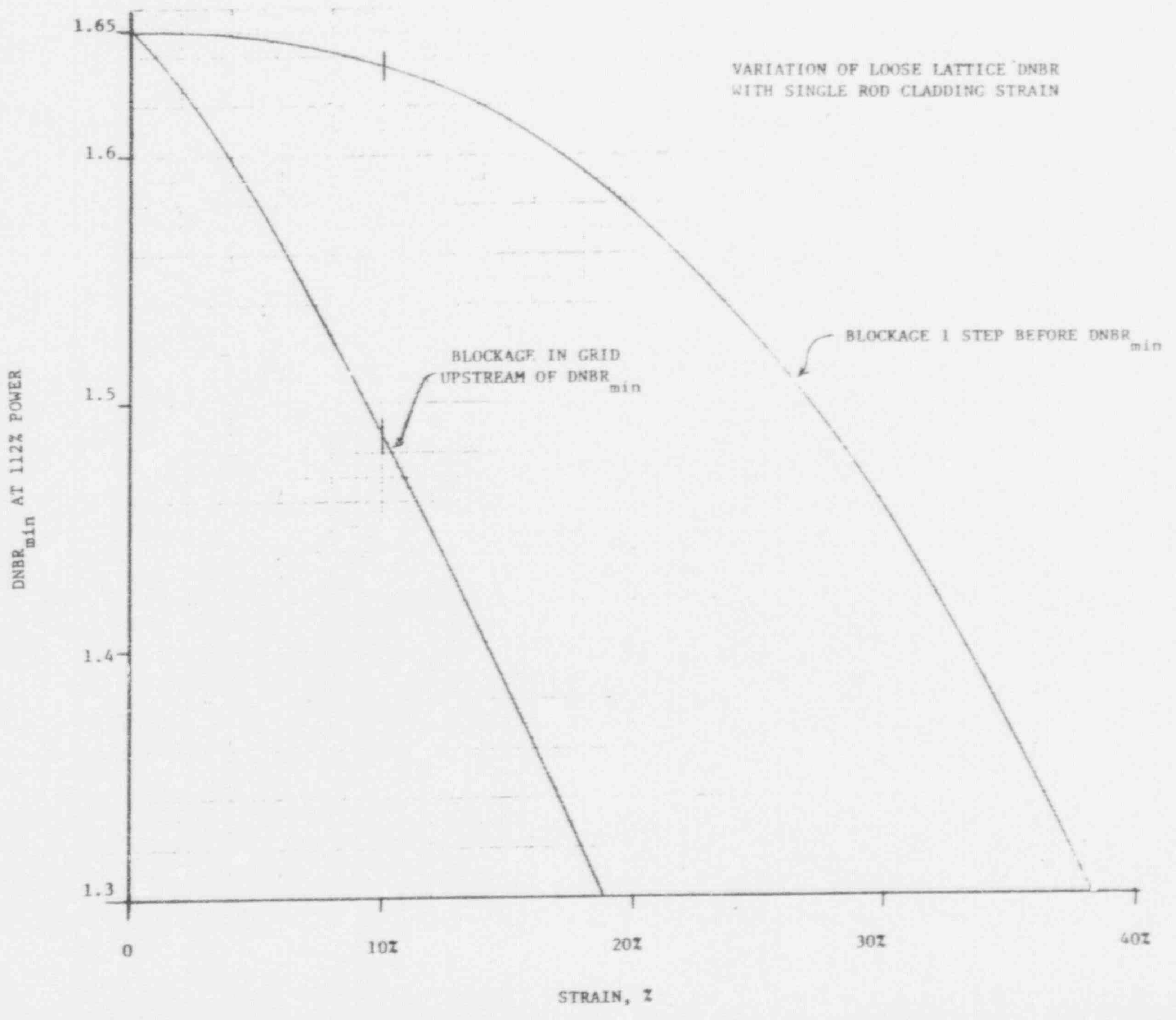
VARIATION OF LOOSE LATTICE DNBR
WITH SINGLE ROD CLADDING STRAIN
BLOCKAGE 1 STEP BEFORE $DNBR_{min}$
BLOCKAGE IN GRID
UPSTREAM OF $DNBR_{min}$
$DNBR_{min}$ AT 112% POWER
1.65
1.6
1.5
1.4
1.3
0
10%
20%
30%
40%
STRAIN, %

FIGURE 2.5-1

3.0 SAFETY ANALYSIS

3.1 GENERAL

A no fuel melt limit has been imposed for the high linear power "loose lattice" assemblies during Core III operation. The effect of this limit on the permissible fuel rod operating conditions is illustrated in Figure 3.1-1. The figure shows the calculated center fuel melt limit in terms of linear heat rating and peak fuel pellet burnup. This relationship has been calculated for the Saxton fuel rods considering:

1) The dependence of the mixed oxides melt temperature with fuel burnup.

2) The reduction in fuel center temperature, as estimated using the Laser Computer program, due to flux depression.(1)

3) The burnup dependent fuel rod center temperature due to irradiation and time induced changes in the thermal and physical characteristics (i.e., pellet-clad gap conductance, fuel swelling, clad creep, fission gas release, etc.).

Figure 3.1-2 provides the fuel melt temperature and flux depression fuel burnup relationships which were employed to generate the Saxton fuel melt limit curve shown in Figure 3.1-1.

Also included in Figure 3.1-1 are three representative curves showing the design-overpower limits for different initial peak pellet burnups. These curves in combination with the proposed power schedule (Figure 2.2-3) have been used in selecting power density/burnup combinations and hence, the fuel rod locations in the loose lattice assemblies to prevent fuel center melt during Core III operation.

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[1] Poncelet, C.G., "Laser - A Depletion Program for Lattice Calculations Based Upon MUFT and THERMOS", WCAP-6073, April 1966.

It should be noted that these curves contain 12.5% nuclear and engineering uncertainty as well as 5% allowance for the overpower trip setpoint and 7% for trip accuracy. There is a high degree of assurance that actual fuel temperature would be less than indicated by the design overpower limit curves because in the above approach the nuclear uncertainty, the engineering uncertainty, and the trip accuracy uncertainty have been combined additively rather than statistically.

The overpower trip setpoint for Core III operation will be set for a power no greater than 5% above the lowest power associated with the following three conditions:

a) a design peak of 24 kw/ft in the Pu-UO_2 fueled "loose lattice" assemblies as determined by power distribution measurements, or

b) a design peak of 19.9 kw/ft in the unirradiated UO_2 fueled "load follow" assemblies as determined by power distribution measurements, or

c) 28 MWt.

Other trip setpoints are as follows:

a)	Hot leg trip setpoint	511°F
b)	Low pressure trip setpoint	2125 psia
c)	Low flow trip setpoint	3.05×10^6 lbs/hr
d)	Low M-G set frequency trip setpoint	60cycles/sec

The minimum DNB ratio occurs in the "load follow" assemblies. The above reactor trip setpoints ensure that reactor trip will be initiated prior to reaching a minimum DNB ratio of 1.30. The transient resulting in the minimum DNB ratio is the loss of flow incident which is re-analyzed in detail in the next section for Core III operating conditions.

Reactivity insertion rates for a rod withdrawal accident during Core III operation are within those previously analyzed and the transient would be essentially the same as presented in the Core II 35 MWt operation safety report. The consequence would be reactor trip resulting in a minimum DNB ratio of $\geq$ 1.30, hence no core damage.

The loss of coolant accident is re-analyzed since the higher fuel temperature and heat flux in the peak power density region of the core during Core III operation would result in increased clad temperature transients.

The steam line break accident has been analyzed previously in the Core II Safety Analysis Report. Although the linear heat ratings of the peak rods are considerably higher in Core III, the control and protection system would cause reactor trip and prevent the minimum DNB ratio from decreasing below 1.30. The principle concern for this accident is the possibility of the core returning to significant power as a result of loss of shutdown margin resulting from the primary cooldown and the negative moderator coefficient. As discussed in Section 2.2, the worth of the Saxton control bank is sufficient to maintain a 1% shutdown margin considering the maximum possible reactivity addition from cooldown and the highest worth control rod stuck out of the core (see Table 2.2-2). Hence, there would be no return to power.

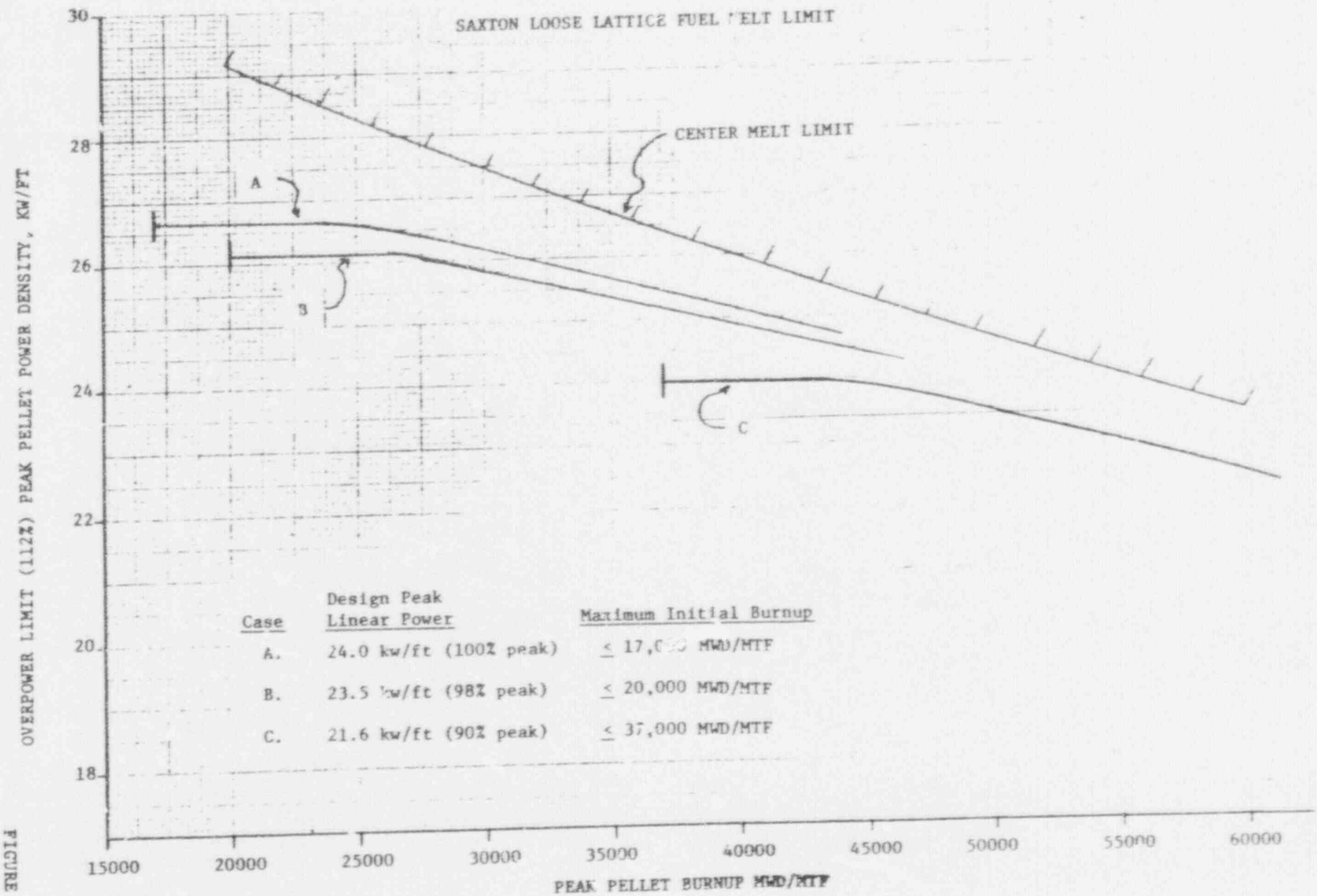

SAXTON LOOSE LATTICE FUEL MELT LIMIT
CENTER MELT LIMIT
A
B
C
OVERPOWER LIMIT (112%) PEAK PELLET POWER DENSITY, KW/FT
PEAK PELLET BURNUP MWD/MTF
30
28
26
24
22
20
18
15000
20000
25000
30000
35000
40000
45000
50000
55000
60000
Case
Design Peak Linear Power
Maximum Initial Burnup
A. 24.0 kw/ft (100% peak) ≤ 17,000 MWD/MTF
B. 23.5 kw/ft (98% peak) ≤ 20,000 MWD/MTF
C. 21.6 kw/ft (90% peak) ≤ 37,000 MWD/MTF

FIGURE 3.1-?

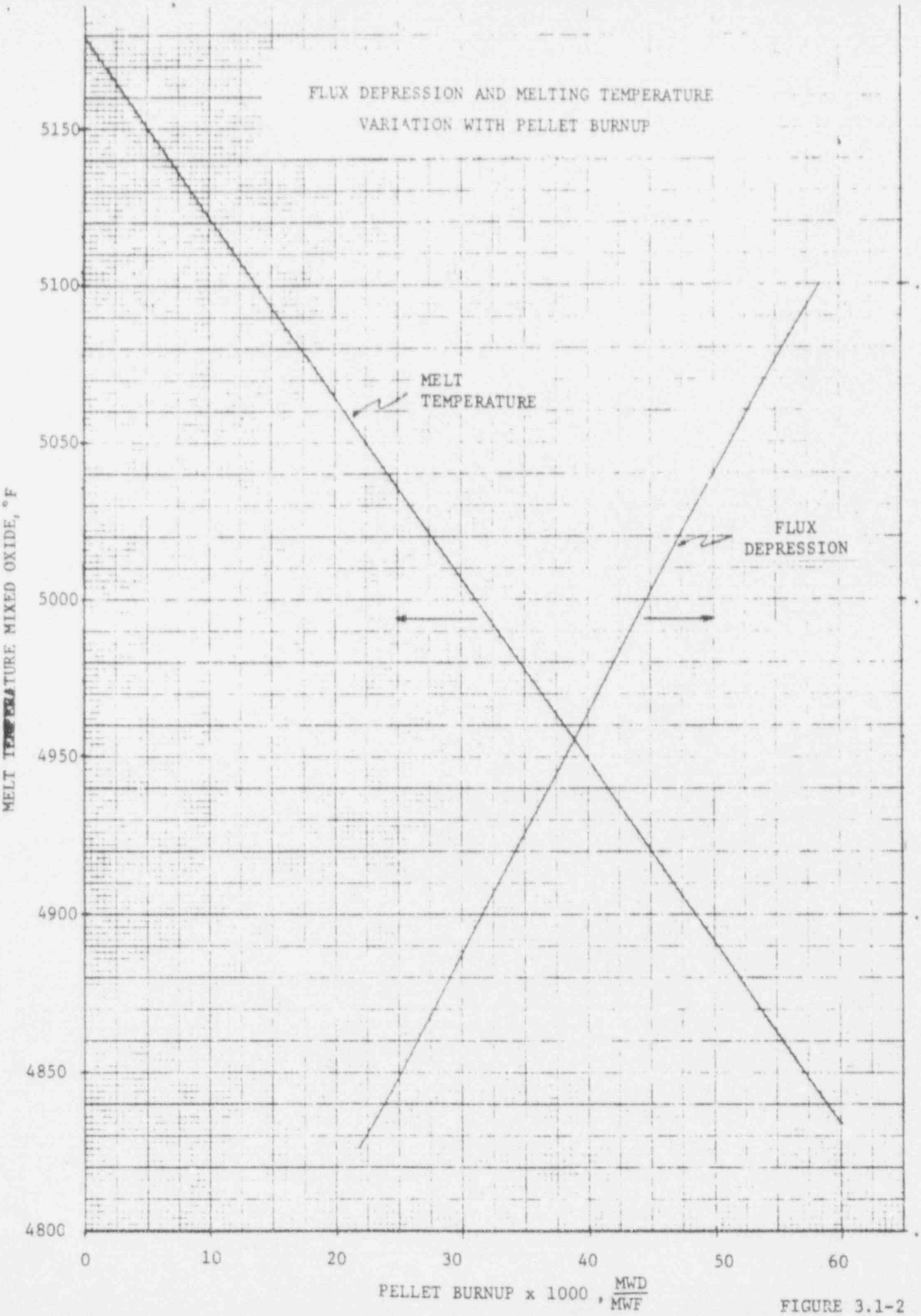

FLUX DEPRESSION AND MELTING TEMPERATURE
VARIATION WITH PELLET BURNUP
MELT TEMPERATURE
FLUX DEPRESSION
MELT TEMPERATURE MIXED OXIDE, °F
5150
5100
5050
5000
4950
4900
4850
4800
.90
.85
.80
.75
0
10
20
30
40
50
60
PELLET BURNUP x 1000 , $\frac{MWD}{MWF}$

FIGURE 3.1-2

3.2 LOSS OF COOLANT FLOW

The loss of flow analysis presented in the Core II 35 MWt Safety Report has been repeated for Core III operating conditions. The transient W-3 DNB ratio was evaluated only for the UO_2 assemblies since the loose lattice assemblies were not DNB limiting. Other assumptions and initial conditions were as follows:

a) Reactor power level is at 103% of nominal rating (28 MWt) as result of possible calorimetric errors.

b) Inlet temperature is 485°F allowing 5°F for instrumentation error.

c) Primary pressure is at 2200 psia allowing 50 psi for instrumentation error.

d) Maximum expected absolute value of the fuel temperature coefficient (Doppler) is: -1.0×10^{-5} $\Delta k/°F$.

e) Minimum expected absolute value of the moderator temperature coefficient is: -1.0×10^{-4} $\Delta k/°F$.

f) Scram delay of 1.1 sec (0.5 seconds due to instrumentation and 0.6 seconds due to rod motion in a region of small effectiveness).

g) The hot spot heat flux is evaluated for the maximum fuel gap (9.5 mils, cold). As shown in Figure 3.2-2, the largest fuel to clad gap results in the highest heat flux response after reactor scram.

h) A negative reactivity insertion rate of 2.22×10^{-4} Δk/seconds (reactor trip).

The flow coastdown curves with and without MG set inertia are shown in Figure 3.2-1. The flow coastdown with inertia was obtained from recent experimental measurements at the Saxton Plant. The flow coastdown without inertia is given in the Saxton Core II Final Safety Report.

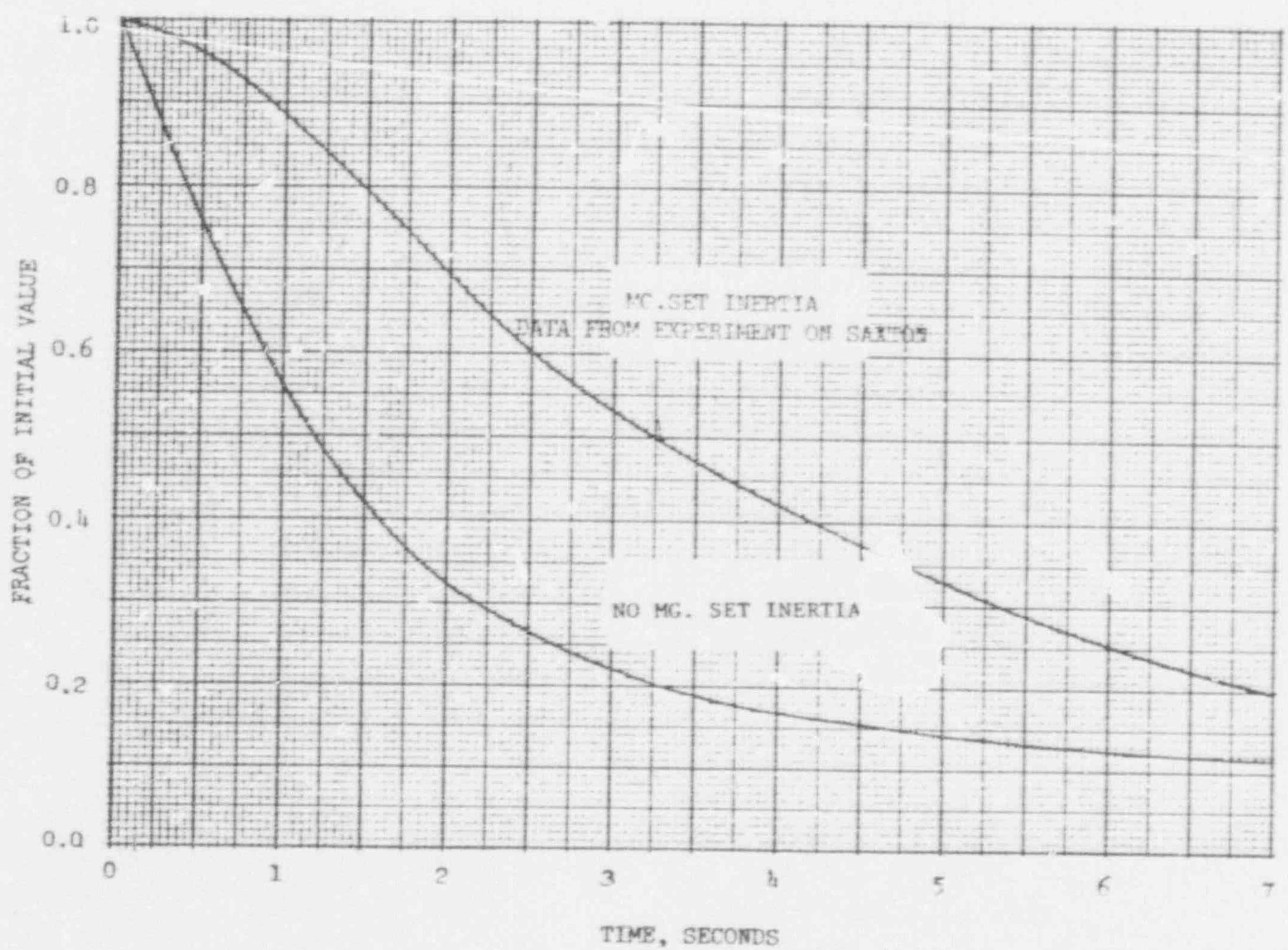
Figures 3.2-2 and 3.2-3 show the neutron flux response, the average and the hot spot heat flux responses with and without MG set inertia.

With MG set inertia a minimum DNB ratio of 1.52 occurs at approximately 1.5 seconds as shown in Figure 3.2-4. Therefore, there is adequate margin to DNB and fuel damage will not result from this accident.

Without inertia, the minimum DNB ratio is about 1.14 and clad damage could be expected in the hot channel. However, as shown in Figure 3.2-5, the W-3 DNB ratio decreases below 1.3 only for channels within 10% of the hot channel power. For the Saxton Core III power distribution, this amounts to about 80 fuel rods of the load follow assemblies. Hence, for the very unlikely conditions that no inertia is available, the core damage will be limited to approximately 80 fuel rods. This is slightly better than similar conditions analyzed for Saxton Core II at 35 MWt, where approximately 107 rods were found to reach DNB.

SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT

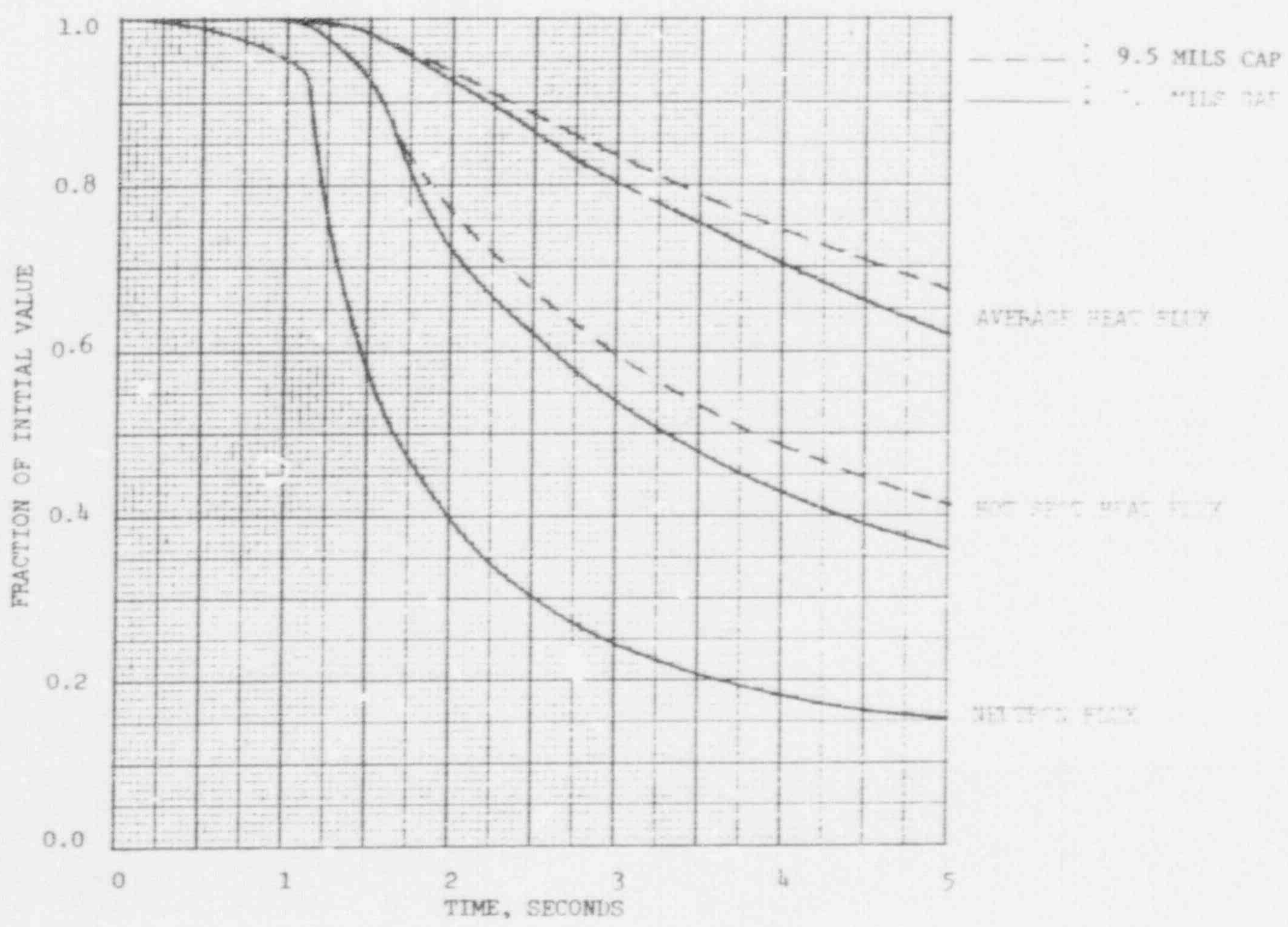
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: -1. x 10^{-5} $\Delta K/K/^{\circ}F$
MODERATOR REACTIVITY COEFFICIENT: -1. x 10^{-4} $\Delta K/K/^{\circ}F$
TRIP REACTIVITY: 3%



FLOW COASTDOWN CURVES

FIGURE 3.2-1

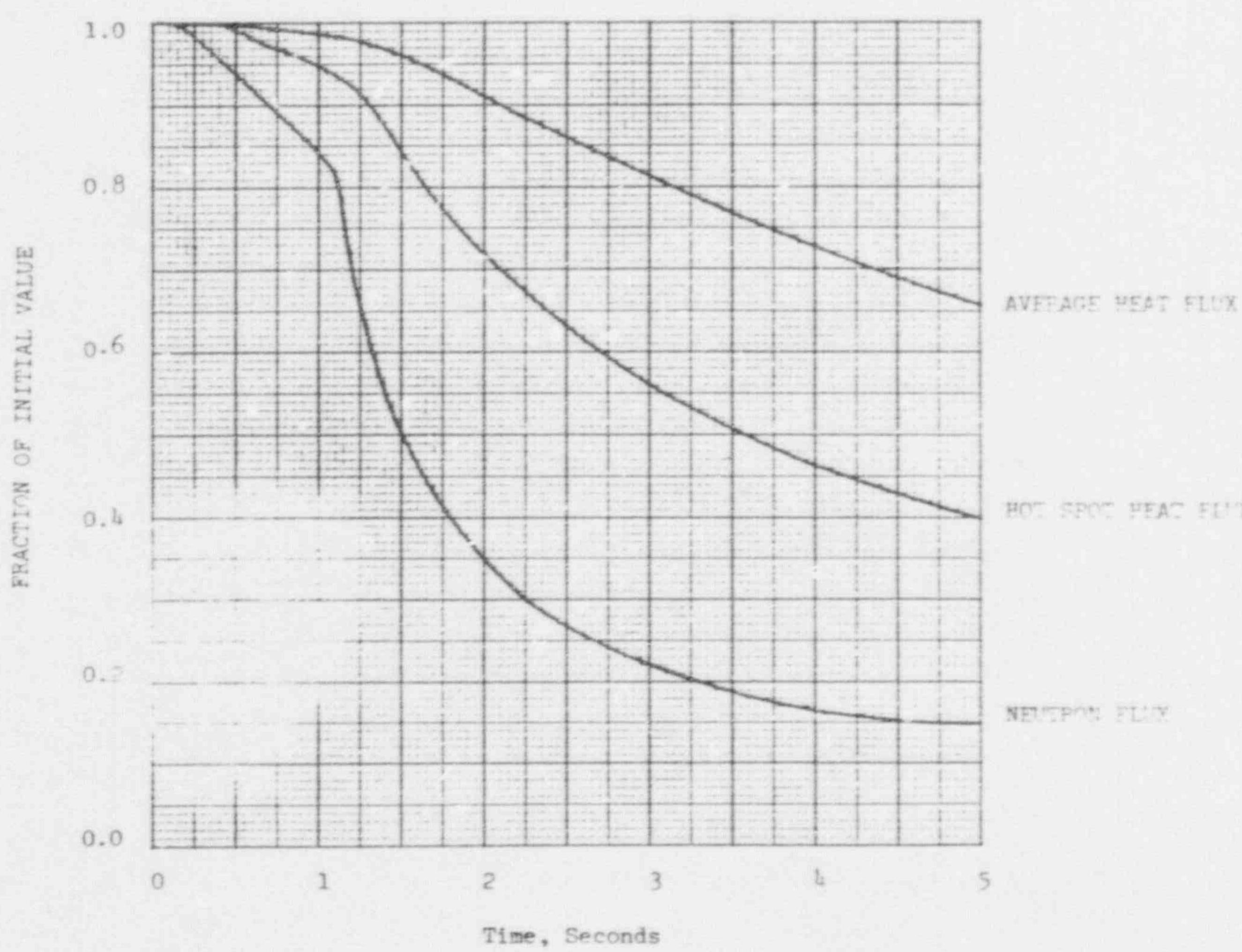
SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5}\Delta K/K/^{\circ}F$
MODERATOR REACTIVITY COEFFICIENT: $-1. \times 10^{-4}\Delta K/K/^{\circ}F$
TRIP REACTIVITY: 3%



MG SET INERTIA

FIGURE 3.2-2

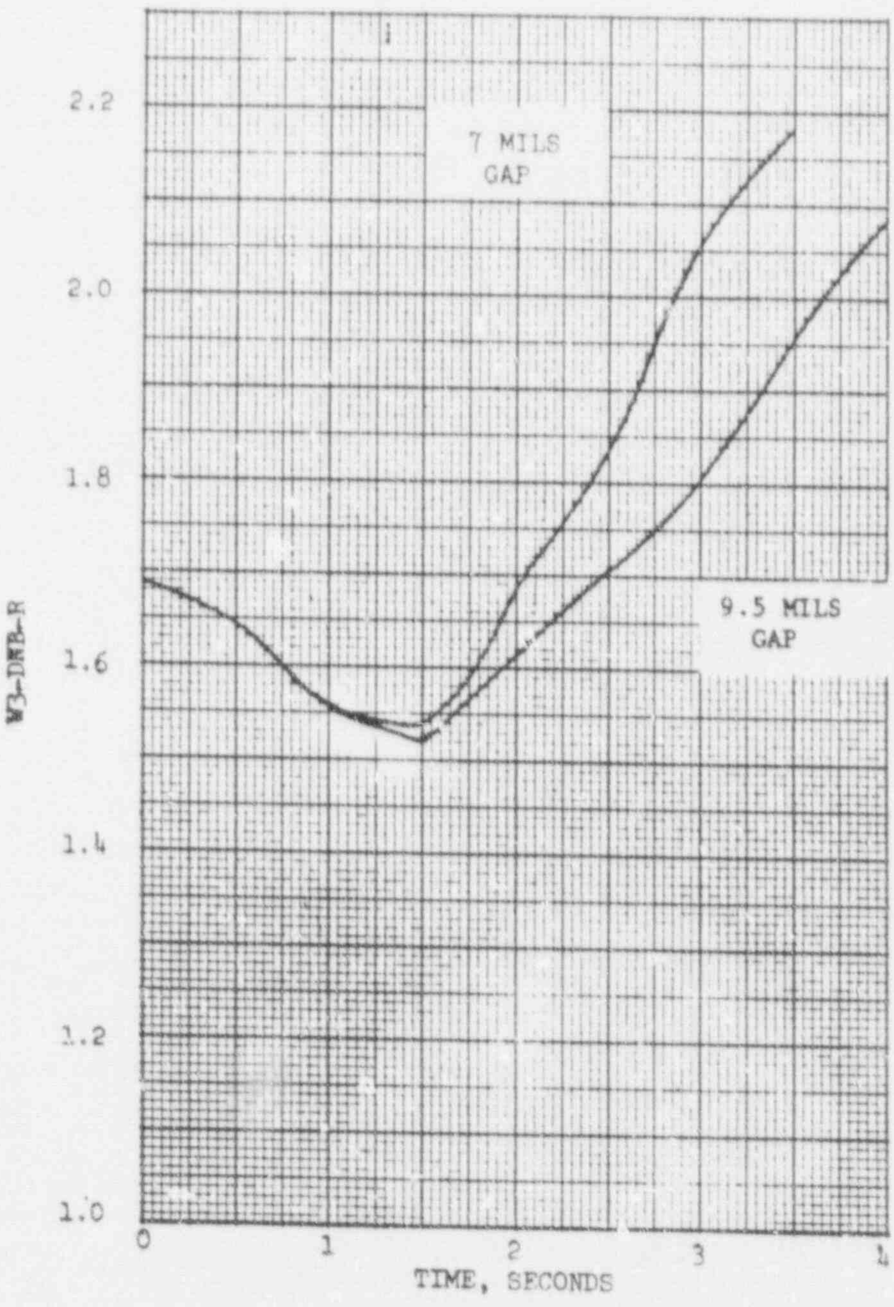
SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5} \Delta K/K/^{\circ}F$
MODERATOR REACTIVITY COEFFICIENT: $-1. \times 10^{-4} \Delta K/K/^{\circ}F$
TRIP REACTIVITY: 3%
FUEL GAP: 9.5 MILS



NO MG. SET INERTIA

FIGURE 3.2-3

SAXTON 28 MW_t OPERATION
LOSS OF FLOW ACCIDENT
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5} \Delta K/K/°F$
MODERATOR REACTIVITY COEFFICIENT: $-1. \times 10^{-4} \Delta K/K/°F$
TRIP REACTIVITY: 3%

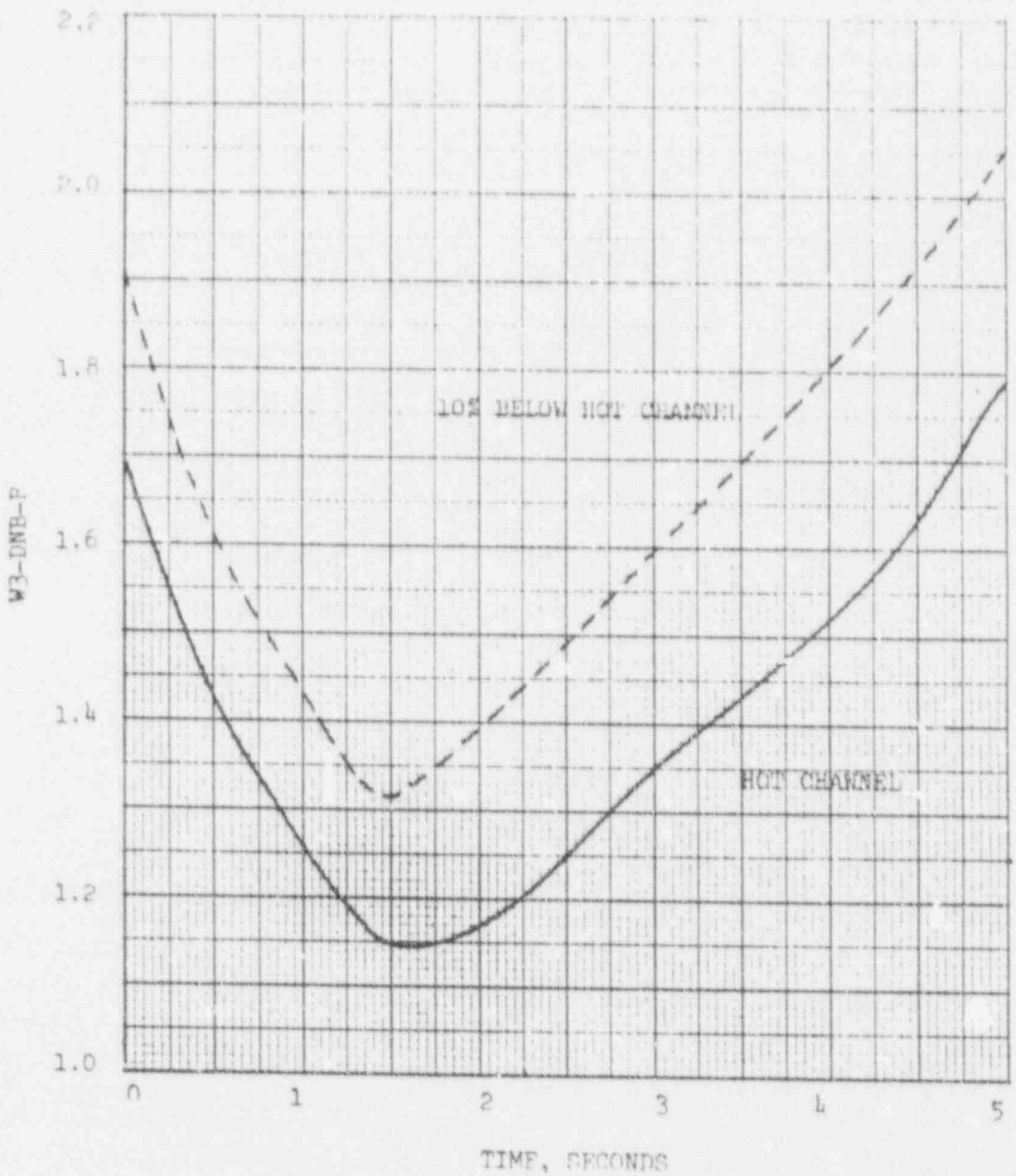


MG SET INERTIA

FIGURE 3.2-4

SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT

TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5}$ $\Delta K/K/°F$
MODERATOR REACTIVITY COEFFICIENT: $-1. \times 10^{-4}$ $\Delta K/K/°F$
TRIP REACTIVITY: 3%
FUEL GAP: 9.5 MILS



NO MG.SET INERTIA

Figure 5

FIGURE 3.2-5

3.3 LOSS OF COOLANT

The loss of coolant accident has been re-analyzed for Core III operating conditions. The assumptions and analysis techniques used are as described in Section 4 of the report "Saxton Loss of Coolant Accident Prevention and Protection." The specific cases analyzed and a summary of results follows:

Break Size	Total % Core Clad Melt	Total % $Zr-H_2O$ Reaction
Doubled Ended Severance - 1.28 ft^2	9.1	12.3
Intermediate - 0.173 ft^2	6.8	9.7
Surge Line 0.0375 ft^2	1.5	3.1

Figures 3.3-1 through 3.3-6 show the clad temperature transients for both the zircaloy clad and stainless steel clad fuel rods in terms of rods operating at various fractions of the peak linear heat rate (24 kw/ft including 12.5% design uncertainty). The values given for stainless steel clad rods are those for the assemblies on the core periphery which operate at 33.1% of the peak 24 kw/ft. The per cent clad melt and per cent $Zr-H_2O$ reaction are greater than those calculated for Core II 35 MWt operation because of the increased power density. The clad melt is limited to 9.1% for the double ended coolant loop rupture and 1.5% for the largest connecting line to the reactor coolant system.

The containment pressure transient would be the same as presented for Core II 35 MWt, well within the 30 psig design pressure. Offsite dose would be less than presented previously in the Core II 35 MWt report because of the 20% lower total core power.

Design and fabrication standards of the Saxton Reactor Coolant System have been reviewed and were found comparable to those currently in use.[1] An inspection program was proposed supplementing the existing program to give increased assurance of system integrity.[1] The safety injection system design and operation were reviewed and a post-accident recirculation system proposed[1]. It is our opinion that the system is adequate to protect the public.

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(1)"Saxton - Loss of Coolant Accident Prevention and Protection."

SAXTON - DOUBLE ENDED BREAK - PEAK POWER = 24.0 kw/ft -

ZIRCALOY CLAD TEMPERATURE -vs- TIME

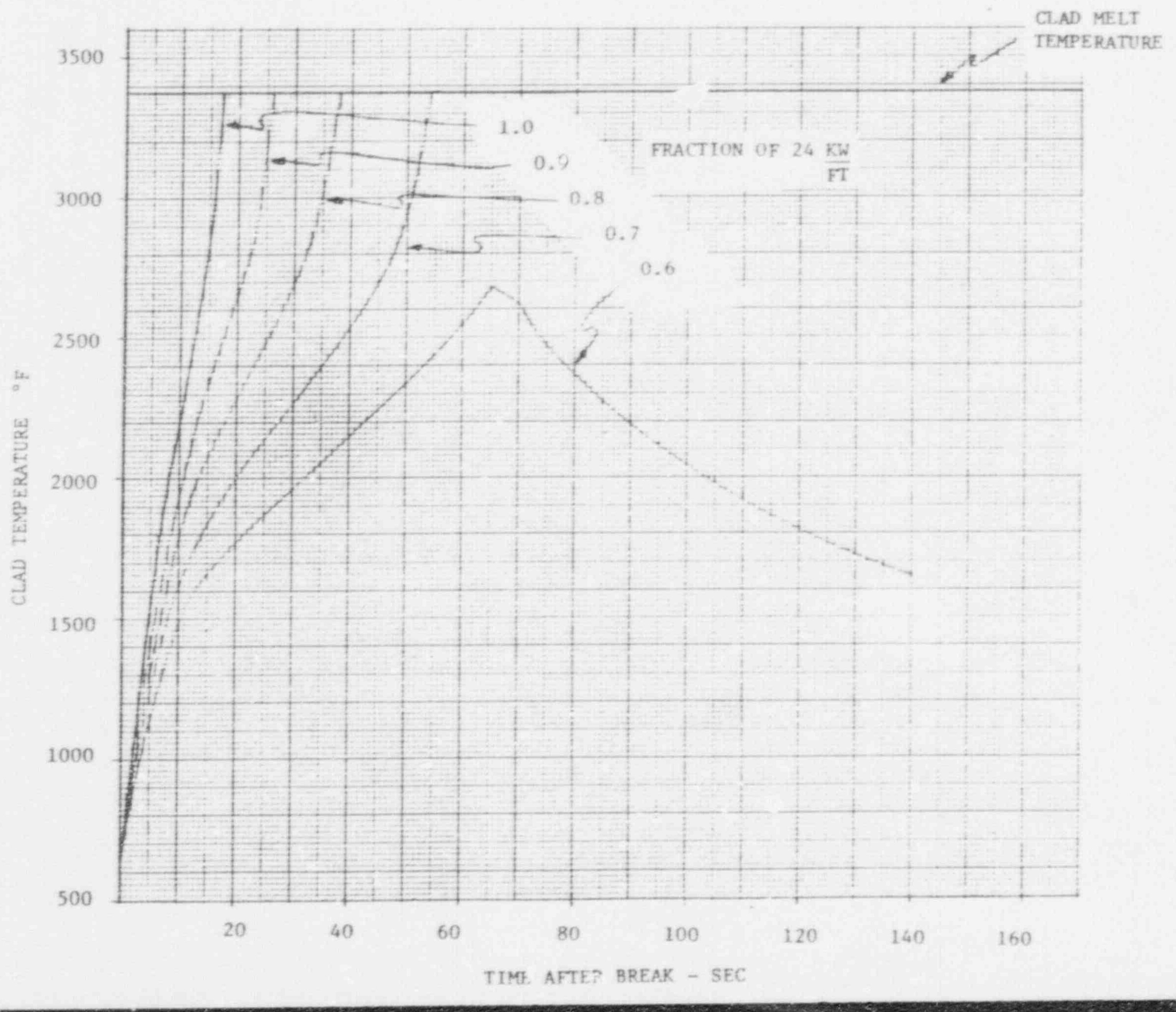

FIGURE 3.3-1

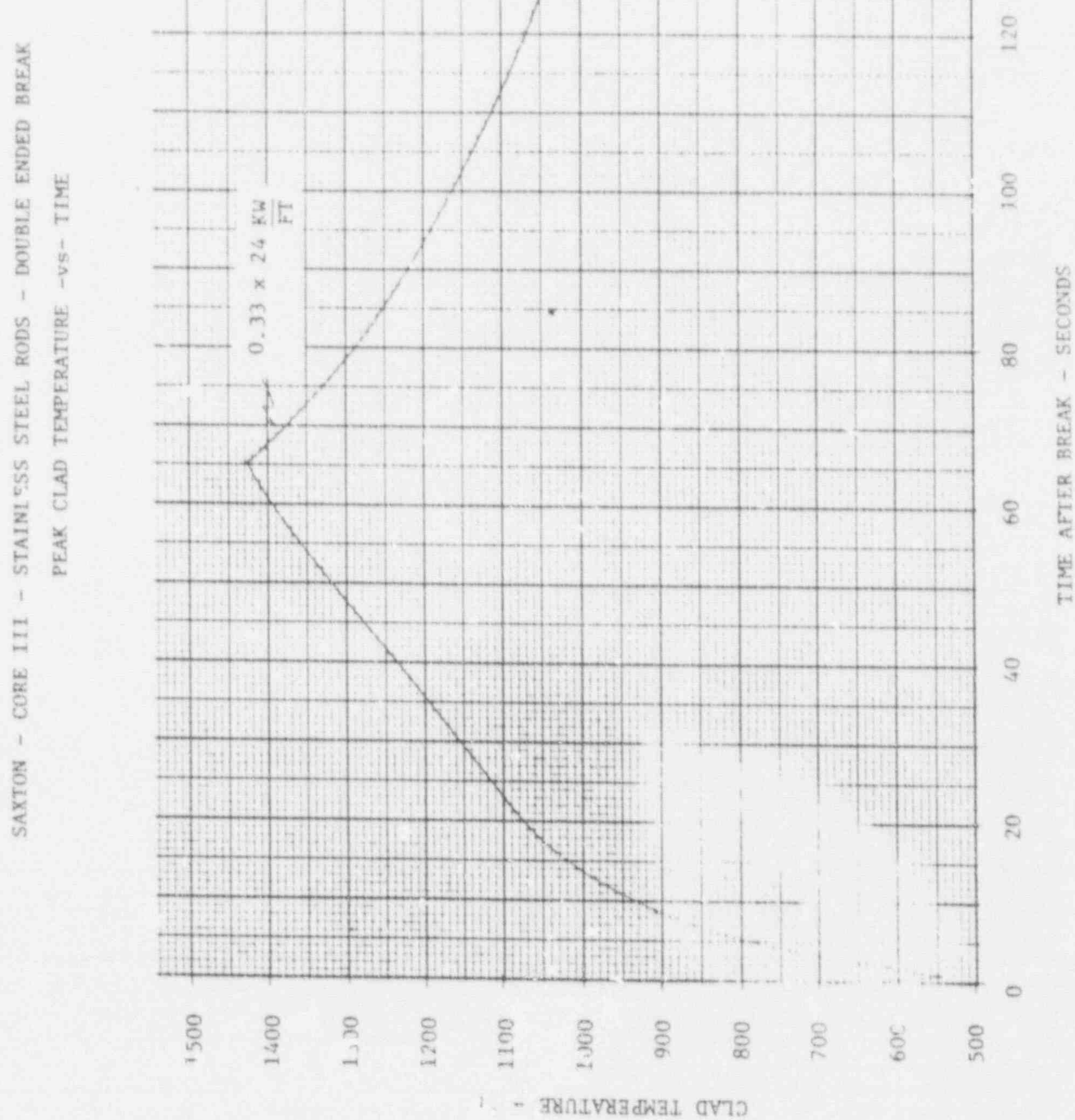
SAXTON - CORE III - STAINLESS STEEL RODS - DOUBLE ENDED BREAK
PEAK CLAD TEMPERATURE -vs- TIME
0.33 x 24 KW/FT
CLAD TEMPERATURE
1500
1400
1300
1200
1100
1000
900
800
700
600
500
0
20
40
60
80
100
120
140
TIME AFTER BREAK - SECONDS

FIGURE 3.3-2

SAXTON - 0.173 FT^2 BREAK - PEAK POWER = 24.0 kw/ft

ZIRCALOY CLAD TEMPERATURE -vs- TIME

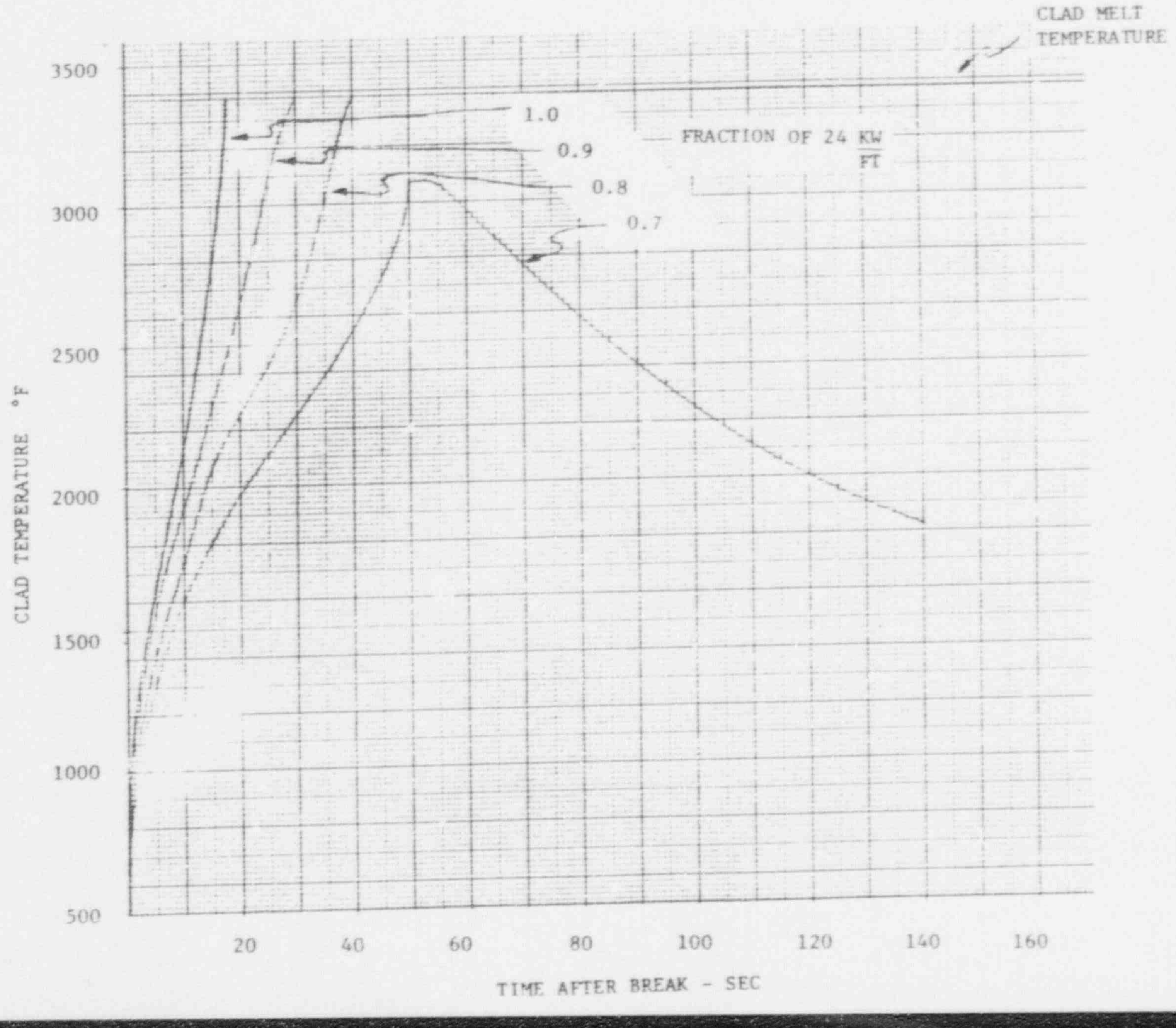

FIGURE 3.3-3

SAXTON - CORE III - STAINLESS STEEL RODS - 0.173 FT^2 BREAK

PEAK CLAD TEMPERATURE -vs- TIME

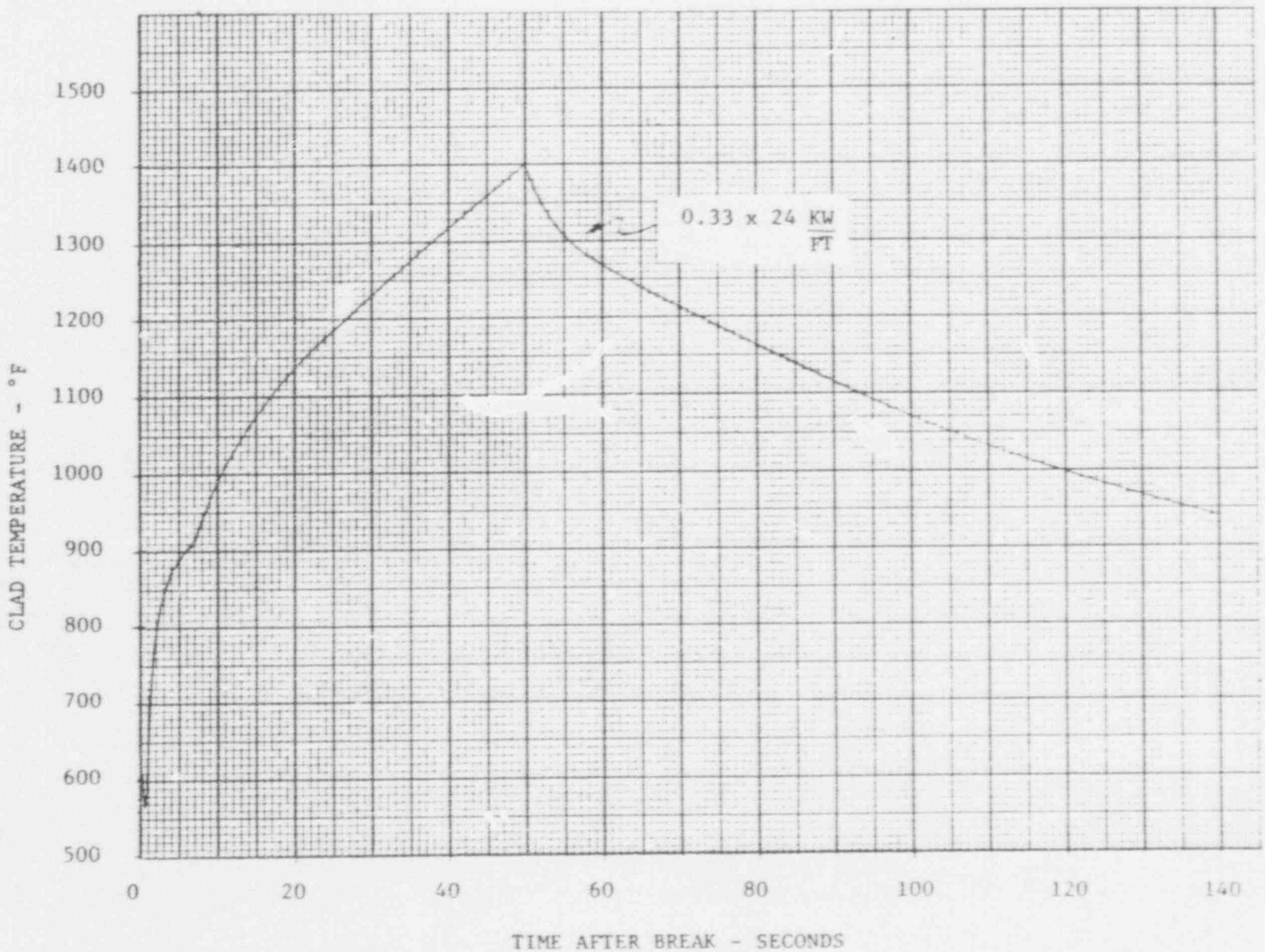

FIGURE 3.3-4

SAXTON - .0375 FT^2 BREAK - PEAK POWER = 24.0 kw/ft

ZIRCALOY RODS - PEAK CLAD TEMPERATURE -vs- TIME

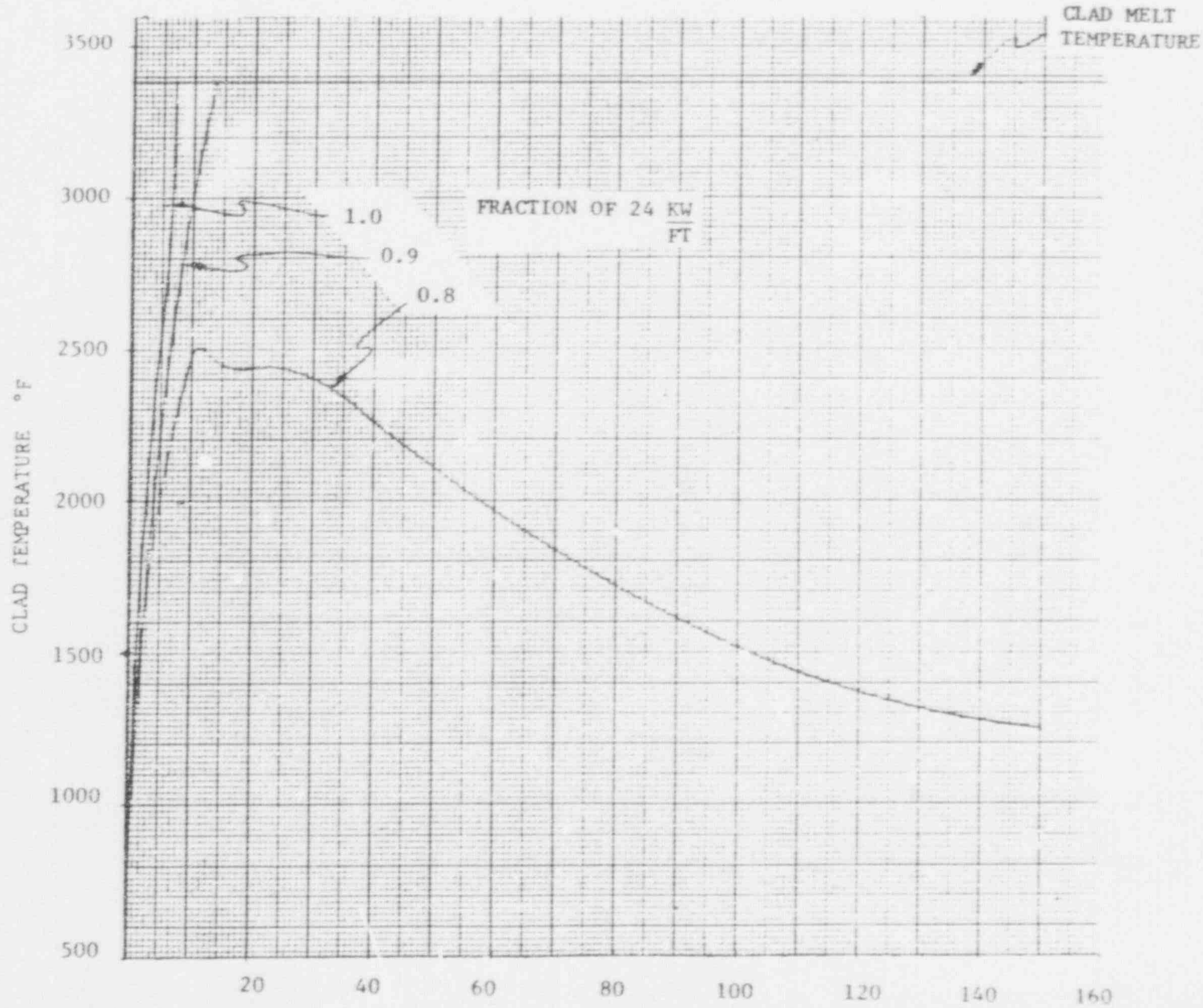

FIGURE 3.3-5

SAXTON - CORE III - STAINLESS STEEL RODS - 0.0375 FT^2 BREAK

PEAK CLAD TEMPERATURE -vs- TIME

FIGURE 3.3-6

APPENDIX A

COOLANT ACTIVITY MONITOR MODIFICATION

In order to provide a continuous, rapid response failed fuel detector, the monitoring system shown in Figure A-1 has been added.

The system utilizes the pressure differential across the steam generator to circulate reactor coolant through a 100 ft section of 3/8" stainless steel tubing. A section of the tubing is coiled around a beta gamma counter. A remote operated flow control valve and a remote reading flow meter are included in the line.

The valve is provided to give a transport time of 45 seconds from the coolant to the detector. This delay time will provide sufficient decay of the reactor coolant N-16 activity to ensure a detector sensitivity of 0.1% defected fuel (1/10 of design value). The detector is highly sensitive to the gamma emitting fission and corrosion products and has a range of .01 mr/hr to 10 r/hr. The sensitivity in determining activity release from failed fuel clad is dependent on the activity release rate as well as the background coolant corrosion product and defected fuel activity. The following indicates the expected detector sensitivity to these contributions.

Source	γ Mev/cc sec	Detector Reading*
Containment Background		15 mr/hr Normal
Normal corrosion product activity level	6.7×10^3	25 mr/hr
Activity with 0.1% fuel defects	4.9×10^4	60 mr/hr
Activity with 1% fuel defects	4.9×10^5	500 mr/hr
Activity released from gap of one rod	4.5×10^5	450 mr/hr

* The detector will be calibrated after installation.

FAILED FUEL DETECTION SYSTEM

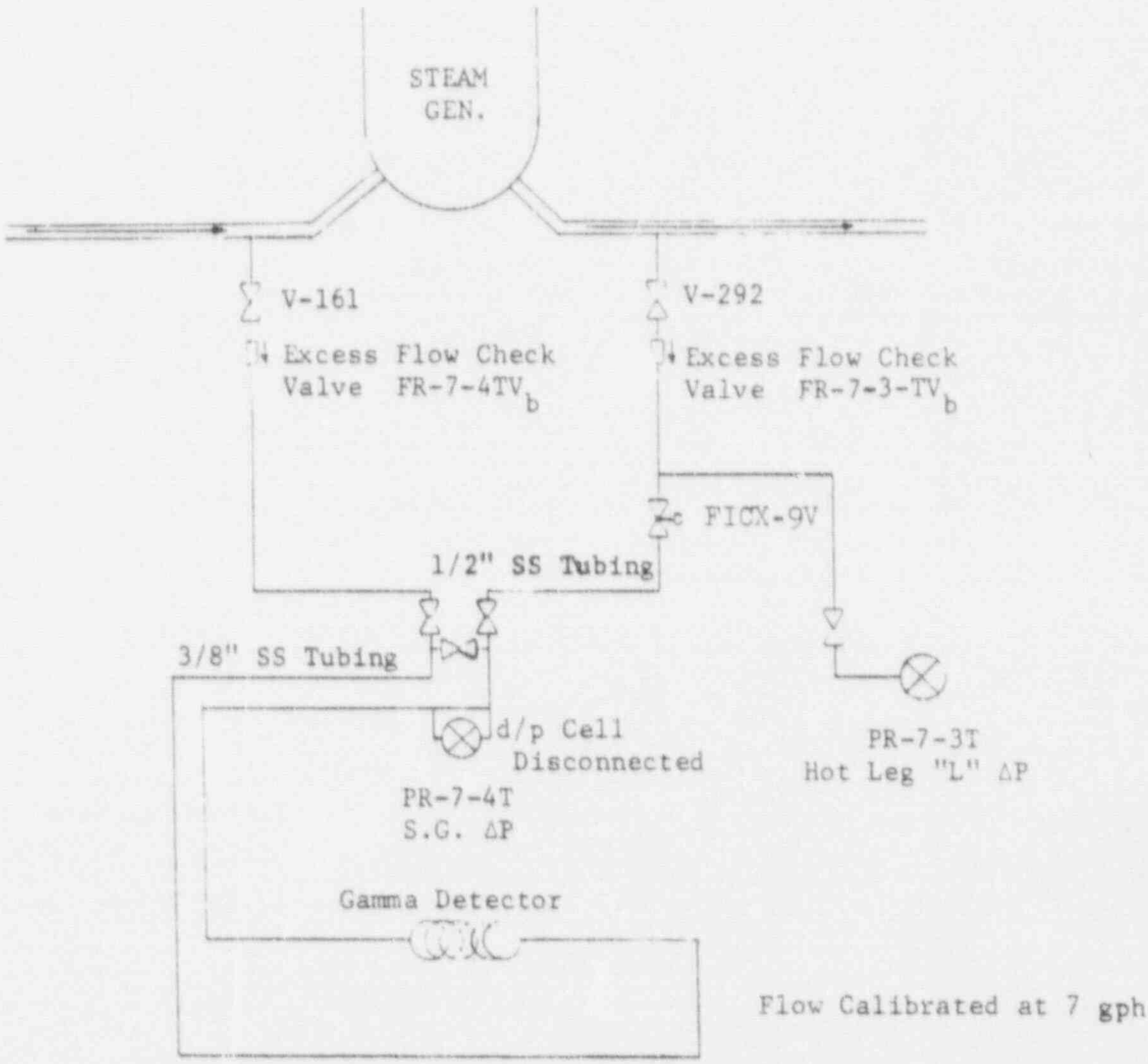

FIGURE A-1