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SAFEGUARDS REPORT

FOR

SAXTON CORE III

REVISION 1

6/24

TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

LIST OF ILLUSTRATIONS (Cont'd)

LIST OF TABLES

SAXTON CORE III

1.0 INTRODUCTION AND SUMMARY

The Saxton Core III configuration, similar to previous cores, contains 21 fuel assemblies. The central nine consist of seven plutonium loose-lattice assemblies which were reconstituted from the Core II assemblies and two new enriched UO_2 load-follow assemblies. The remaining twelve outer assemblies are eleven partially depleted UO_2 assemblies from Saxton Core I and one unirradiated UO_2 assembly. 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.

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 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.

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/MTM.

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 and cold-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.

1.2 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 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.

1.3 LINEAR POWER OBJECTIVES

The linear power objectives are an expected peak kw/ft of 21.2 in the loose-lattice assemblies and 17.6 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 design core power less than 28 MWt.

As in Core II 35 MWt operation, the nominal inlet temperature during full power operation is 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 water-filled tubes in alternate rod positions, which by reducing the heat addition to a coolant channel [redacted] increases the margin to DNB.

1.4 OPERATING CONDITIONS

Core III operating conditions are 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.

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 is operated from the motor-generator set to benefit from the increased pump inertia. The operating pressure for Core III is increased from 2200 psia to 2250 psia to more closely simulate conditions in current PWR's.

Approximately 50 fuel rods 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 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 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.

In addition to daily sampling and periodic operation of the letdown/charging system radiation detector for monitoring coolant activity, an experimental failed fuel monitor has been installed at Saxton. The system utilizes the pressure drop across the steam generator to circulate reactor coolant by a gamma detector. The system is described more fully in Appendix A.

1.5 GENERAL OPERATING PROCEDURES

Saxton Core III will be operated such that the lowest power associated with the following three design conditions will not be exceeded at any time during Core III steady state operation:

a) a design peak linear power of 24.0 kw/ft in the PuO_2-UO_2 loose-lattice assemblies as determined by power distribution <u>measurements</u>, or

b) a design peak linear power of 19.9 kw/ft in the unirradiated UO_2 fueled load-follow assemblies as determined by <u>measurements</u>, or

c) 28 MWt

The power associated with the above limit (a) is expected to be 24.9 MWt and is expected to be controlling from the beginning of Core III life to approximately 1200 effective full power hours. During this period the power will be slowly increased to 25.8 MWt which is the power associated with limit (b) above. After this the core power will continue to increase to a maximum of 26.5 MWt to maintain the design linear power of 19.9 kw/ft in the load-follow assemblies.

A peak pellet burnup of approximately 55,000 MWD/MTM will be achieved after 5000 equivalent full power hours (EFPH @ ~3.5 MWt) operation.

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 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.

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.

2.0 CORE DESIGN

2.1 MECHANICAL DESIGN

2.1.1 CORE LOADING

The 21 main fuel assemblies (9 x 9 rod array) in Saxton Core III are made up of seven loose-lattice assemblies, two load-follow assemblies, one peripheral unirradiated UO_2 assembly, and eleven UO_2 assemblies from Core I, arranged as illustrated in Figure 2.1-1. The seven loose-lattice assemblies (described in greater detail below) consist of four 36-rod assemblies, one 35-rod assembly, one 34-rod assembly, and one 32-rod assembly which accommodates, in addition to the 32 fuel rods, a 4 rod removable subassembly. The rods in the loose-lattice assemblies are composed of irradiated rods removed from Core II assemblies and loaded into new assemblies containing water-filled hollow tubes at every other rod location.

Each of the two load-follow assemblies (described in greater detail below) will contain 60 fuel rods with design variations in fuel pellet diameter density and internal gas pressure. These assemblies have the same pitch as the standard section 72 rod assemblies.

Fuel rod pitch and cross section of the 11 peripheral Core I assemblies remain identical to those shown in Figure 203.1 of the Saxton Final Safeguards Report.

The unirradiated peripheral UO_2 assembly contains pelletized, enriched UO_2, Zircaloy-4 clad fuel rods (described in greater detail below).

The remaining fuel loading will consist of six control rod followers which were in Cores I and II, four removable fuel subassemblies in the peripheral UO_2 fuel assemblies and the special L-shaped assemblies which were in the peripheral slot positions in Cores I and II.

2.1.2 FUEL ASSEMBLY DESIGN

1. Overall Construction

The construction of the 9 center assemblies and of the unirradiated peripheral UO_2 assembly remains essentially the same as that of Cores I and II, except that these assemblies have been reinforced as described in Appendix B.[1] No change has been made in the overall dimensions of the fuel assemblies. Two locations in each of the 21 main fuel assemblies in Core III (A-5 and E-1 in Figure 2.1-2) will be used for either flux thimbles, secondary source rods, removable fuel rods, removable cladding test specimen assemblies, removable water-filled tubes containing spot weld test specimens or removable solid Zircaloy bars. In the text, removable fuel rods, tubes, etc. are defined as those which can be removed without removing the top nozzle from the fuel assembly. A non-removable fuel rod, tube, etc. is defined as one which cannot be removed from the fuel assembly without first removing the top nozzle. 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, 0.391 inch diameter by 0.015 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 in the same manner as a fuel rod. The top end plug consists of two square sections with a threaded section at the top end. The threaded section protrudes through the top nozzle and the captive nuts are torque loaded to 100 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 can.

(1) Summary Report on Buckling of Saxton Core II Fuel Assemblies and Prevention of Buckling in Core III. (See Appendix B)

The top of the ~~enclosure~~ can assembly has a specially prepared reinforced edge which fits into, and is held by, the top nozzle plate. Each can assembly and top nozzle is a matched pair to ensure proper alignment. The fuel assemblies, with top and bottom nozzles, are depicted in Figure 2.1-3.

2. <u>Loose-Lattice Assemblies</u>

Table 2.1-1 presents the quantity of each kind of rod, tube or test, which occupies the 72 locations in each of the loose-lattice assembles. The pitch between fuel rods in the loose-lattice assemblies is 0.820 inches. Removable rod locations E-1 and A-5 (See Figure 2.1-2 for an illustration of the location identification within an assembly) in Table 2.1-1 are shown in Figure 2.1-3. The removable rod locations in Figure 2.1-3 are vacant because these locations are filled after the assembly is fabricated.

The loose-lattice tie rods referred to in a foot note in Table 2.1-1 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 2.1-4.

There are two arrangements of tie rods in the loose-lattice assemblies, a Type A and Type B loose-lattice assembly as shown in Figure 2.1.3. In the Type A assembly the tie rods are located in the B-2, B-8 and H-2 locations within the assembly. In the Type B arrangement, the tie rods are located in the A-2, D-9 and J-4 assembly rod locations.

The locations of the fuel rods and water-filled tubes in the Type A assemblies are the reverse of those in the Type B assemblies in order to maintain a uniform fuel rod pitch between adjacent loose-lattice assemblies. As a result the tie rods are in different locations in the two types of assemblies.

The 7 loose-lattice assemblies consist of four 36-rod Type A, one 35-rod Type B, one 34-rod Type B, and one 32-rod Type A assembly.

The 32-rod [redacted] Type A assembly contains, in addition to the 32 rods, a removable subassembly with 4 fuel rods arranged on a 0.758 inch pitch.

The loose-lattice assemblies have been strenghtened by spot welding 0.028 inch thick angles between the can and 6 stainless steel water-filled tubes as illustrated in the cross section of the assembly shown in Figure 2.1-3. The one inch long clips previously used to fasten the halves of the can have been replaced by full length clips in the spans between the grids.

The grid springs have been reset to give a nominal 6.5 lbs. contact force compared to the 15.5 lbs. force previously used. The combined effect of the charges produces a safety factor greater than 1.5 between the forces generated in the assembly and the buckling strength of the cans. A detailed explanation of the buckling is given in Appendix B.

The loose-lattice fuel rods, which are described in the Saxton Core II Plutonium Project Safeguards Report, were selected from previously irradiated plutonium rods removed from Core II fuel assemblies. Figure 2.1-5 depicts a typical plutonium rod. The rods were inspected prior to being loaded into the new assemblies.

The general basis for acceptance of irradiated PuO_2-UO_2 rods was 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 reconstituted loose-lattice assemblies served as a further check on fuel rod integrity.

The water-filled tubes, referred to in columns 4 and 5 of Table 2.1-1 and which occupy approximately every other rod location in the loose-lattice assemblies, are shown in Figure 2.1-6. They are fabricated from Zircaloy-4 tubing having the same physical properties as Zr-4 cladding and are filled with end plugs. Each water-filled tube has two 1/16 inch diameter holes at the bottom and two 1/8 inch diameter holes at the top of the tubing to allow the flow of primary coolant through the rod. Forty-two of the water-filled tubes will be hydrided to various levels as described in Change Report No. 21 in Appendix C.

The loose-lattice region of the core contains one L-assembly at core location E-2. This L-assembly contains 9 pelletized UO_2 fuel rods clad with type 348 stainless steel.

3. Load-Follow Assemblies

At the beginning of Core III life, core locations E-3 and C-3 will be occupied by load-follow assemblies 503-18-3 and 503-18-1 respectively; however, at mid-life the location of these assemblies will be reversed.

These two assemblies each contain 60 fuel rods arranged on a nominal pitch of 0.580 inches. They have the same overall dimensions as Core II type fuel rods and contain pelletized UO_2 fuel, of two enrichments (9.5 or 12.5 w/o U-235). The pellet densities range between 8[illegible].5 and 94.5 percent of theoretical density and have nominal pellet to clad gaps of 5.5 to 9.5 mils. The rods in these assemblies are described in greater detail in Reference (2).

Each of the two load-follow assemblies also contain 4 water-filled Zircaloy tubes (item 5 in Figure 2.1-3) and one stainless steel water-filled tube (item 9 in Figure 2.1-3). These tubes are illustrated in Figure 2.1-6. The tie rods in each load-follow assembly contain filler material. Two contain Inconel (item 8 in Figure 2.1-3) and one contains stainless steel filler (item 7 in Figure 2.1-3). The purpose of filler material is to control local power perturbations in the assemblies. The tie rods are illustrated in Figure 2.1-7.

The load-follow assemblies were also strengthened as described in Appendix B. A solid square stainless steel bar, with two angles spot welded between the bar and the can, is used in place of two stainless steel clad fuel rods in opposite corners of the assembly. An angle, 0.05 inches thick, has been spot welded to the inside of the can at the center of each long span between grids. Full length angle clips are also used between the edges of the can halves as was done with the loose-lattice assemblies.

In each assembly the removable rod location A-5 (See Figure 2.1-2) will be occupied by a solid Zircaloy bar (as described in Change Report 23, in Appendix C). Removable rod location E-1 will be occupied by a flux thimble tube in Assembly 503-18-1 at core location C-3 and by a cladding test specimen assembly (as described in Change Report No. 22 in Appendix C) in Assembly 503-18-3 at core location E-3. As with the loose-lattice assemblies, locations A-5 and E-1 are blank in Figure 2.1-3 since these locations are filled after the assembly is fabricated.

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(2) WCAP-7219-L Rev. 1, Addendum to Saxton Core III License Application (Westinghouse Confidential), April 29, 1969.

4. Core I Assemblies

Eleven previously irradiated assemblies from Core I are loaded into Core III peripheral locations. The contents and core locations of these assemblies is given in Table 2.1-2. The pitch and cross section of these assemblies remains identical to that shown on Figure 203.1 of the Saxton Final Safeguards Report.

An examination of the thermal gradients across these assemblies shows that, the worst thermal gradient would produce a bow of approximately 0.015 inches over the length of the assembly. This bow would always be of an elastic nature and therefore no buckling of the assembly would occur.

5. Unirradiated Peripheral Enriched UO_2 Assembly

One of the 12 peripheral assemblies will be unirradiated. The assembly is 503-10-7 and is located at the B-2 core location. This assembly contains 68 Zircaloy-4 clad fuel rods containing pelletized UO_2 fuel arranged on a 0.580 inch pitch. As explained in Appendix B Reference (1), this Assembly was strengthened similarly to the load-follow assemblies. However, this assembly contains removable solid Zircaloy bars at both the E-1 and A-5 removable rod locations; whereas, the load-follow assemblies each contain only one solid Zircaloy bar.

6. Mid-life Inspection

At approximately mid-life of Core III, a detailed fuel inspection will be carried out. A minimum of 4 assemblies (two load-follow and two loose-lattice) will be removed from the core and given a visual inspection with a periscope and T.V. camera. The top nozzles will be removed and a sample of both fuel rods and water tubes will be examined. The sample size will be dependent upon the results obtained by examination of a minimum of 5 fuel rods and 5 water tubes. In addition, some fuel rods and water tubes will be removed for a more extensive examination.

2.1.3 SUBASSEMBLY DESIGN

In general, the removable subassemblies are similar in construction to the main fuel assemblies except that the can is of 0.019 inch thick perforated stainless steel and the rods are arranged on a 0.536 inches pitch. On insertion of a subassembly into a fuel assembly, the end plates of the subassembly are compatible with the surrounding nozzle plates so that the coolant encounters a flow path similar to that in a fuel assembly.

The location of the subassemblies in Core III at initial start-up and the individual subassembly composition is given in Table 2.1-3. The location of the peripheral subassemblies in the core may be switched during Core III life. The arrangement of fuel rods, water-filled tubes, flux thimble tubes, etc. within each subassembly is illustrated in Figure 2.1-8 to 2.1-12.

Subassembly 503-4-31 which is initially in the N-1 location is illustrated in Figure 2.1-8. This subassembly contains four previously irradiated PuO_2 - UO_2 Zr-4 clad fuel rods, a flux thimble in the center rod location and 4 water-filled Zr-4 tubes in the four corner locations.

Subassembly 503-4-33 is initially in the N-2 location and is illustrated in Figure 2.1-9. This subassembly contains 2 hydriding effects test fuel rods which contain fuel having an average D_2O content of 120 ppm (see Change Report 18 in Appendix C) and 2 high pressure creep rods at 1915 psia initial pressure (see Change Report 19 in Appendix C). This subassembly will also contain a flux thimble in the center location and 4 water-filled Zr-4 tubes at the four corner locations.

Subassembly 503-4-25, which is initially in the N-3 location, is illustrated in Figure 2.1-10. This subassembly contains 5 non-removable fuel rods clad with 304 s.s., enriched to 5.7 w/o U-235. In addition, this subassembly contains 4 removable fuel rods clad with Zr-4, enriched to 5.7 w/o U-235 with various internal pressures.

Subassembly 503-4-32 which is initially in the N-4 location is illustrated in Figure 2.1-11. This subassembly contains 4 non-removable fuel rods clad with Zr-4, enriched to 12.5 w/o U-235; 2 low pressure creep test rods (see Change Report 19 in Appendix C); a flux thimble, and 2 irradiated PuO_2 - UO_2, Zr-4 clad rods.

Subassembly 503-4-34 which is initially in the [illegible]-5 location and described in Change Report 17 in Appendix C is illustrated in Figure 2.1-12. This subassembly contains 2 hydriding effects test fuel rods (described in Change Report 18 in Appendix C) and 2 materials compatibility test rods (described in Change Report 17 in Appendix C).

TABLE 2.1-1

CORE LOCATION AND COMPOSITION OF LOOSE-LATTICE ASSEMBLIES *

Assembly Serial Number	Assembly Core Location	Number of PuO_2-UO_2 Zr-4 Clad Fuel Rods	Number of Zr Clad** Water-Filled Tubes: Not Hydrided	Number of Zr Clad** Water-Filled Tubes: Hydrided	Assembly Type	Sub-Assembly Location	Associated L-Assembly Serial Number	Removable Rod Location E-1 and A-5****: Flux Thimble	Spot Weld Test Specimens	Cladding Test Specimens	PuO_2-UO_2 Zr-4 Clad Fuel Rods	Secondary Nuclear Source
503-17-2	D-2	34(1-VP)†	2	25	B	none	none	-	-	-	1 (E-1)	1 (A-5)
503-17-8	C-2	36(2-VP)	27		A	none	none	-	2	-	-	-
503-17-5	C-4	36(4-VP)	25		A	none	none	1 (E-1)	-	1 (A-5)	-	-
503-17-4	D-3	32	20		A	E-1***	none	2	-	-	-	-
503-17-6	D-4	34(2-VP)	10	17	B	none	none	2	-	-	-	-
503-17-3	E-2	36	25		A	none	503-3-2	1 (E-1)	-	1 (A-5)	-	-
503-17-9	E-4	36(1-VP)	25		A	none	none	-	2	-	-	-

* All of these assemblies contain 3 stainless steel water-filled tie rods, 6 water-filled stainless steel tubes to which 0.028 inch thick angles are spot welded (a total of 72 rod locations).

** See Change Report 21 in Appendix C for detailed explanation

*** This subassembly location contains subassembly 503-4-31 which is composed of 4 water-filled tubes; 4 PuO_2 - UO_2, Zr-4 clad fuel rods and a flux thimble.

**** See Figure 2.1.4 for explanation of alphanumeric location designation E-1 and A-5.

† VP denotes number of PuO_2 - UO_2, Zr-4 clad fuel rods which have vipac fuel.

TABLE 2.1-2

CORE LOCATION AND COMPOSITION OF PERIPHERAL ASSEMBLIES (CORE I)

Assembly Serial Number	Assembly Core Location	Number of Pelletized UO_2 Stainless Steel Clad Fuel Rods	Associated* L-Assembly Serial Number	Subassembly Location	Removable Rod Location E-1 and A-5: Pelletized UO_2 Stainless Steel Clad Fuel Rods	Flux Thimble	Secondary Nuclear Source
503-2-4	B-3	61	503-3-5	N-2		2	
503-1-12	C-1	70	503-3-1	none	1(A-5)	1	
503-2-2	C-5	61	503-3-3	N-3		2	
503-1-16	D-1	70	none	none	1(E-1)	1	
503-1-11	D-5	70	503-3-4	none		1	1(E-1)
503-2-6	E-1	61	503-3-7	N-4		2	
503-1-5	E-5	70	none	none	2		
503-1-9	F-2	70	503-3-10	none	2		
503-1-17	F-3	70	none	none	1(A-5)	1	
503-2-5	F-4	61	503-3-9	N-5		2	
503-1-8	B-4	70	503-3-6	none	1(E-1)	1	

* Each L-Assembly contains 9 pelletized UO_2 348 stainless steel clad fuel rods.

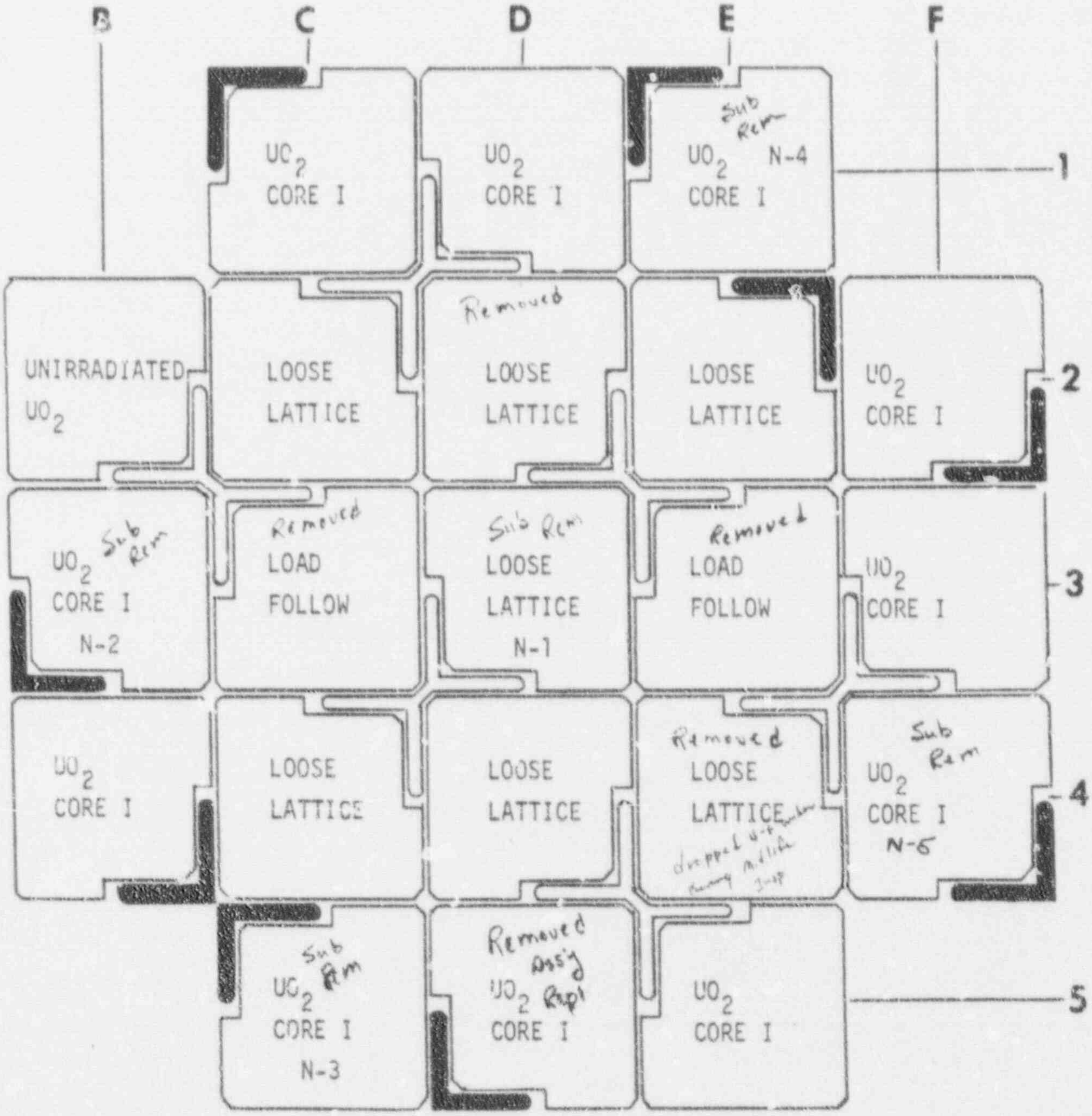
TABLE 2.1-3

INITIAL CORE LOCATION AND COMPOSITION OF SAXTON CORE III SUBASSEMBLIES

Subassembly Serial Number	Initial Subassembly Core Location	Flux Thimble at Center	Number of Water-Filled Tubes	Number of Pelletized 5.7w/oUO_2 Zr Clad Rods	Number of Pelletized Rods 5.7 w/o UO_2 S.S. Clad	Number of Pelletized 12.5w/oUO_2 Zr Clad Rods	Number of Pelletized PuO_2-UO_2 Rods from Core II	Miscellaneous
503-4-31	N-1	Yes	4				4 (high burnup)	
503-4-33	N-2	Yes	4			2 hydride-120ppm* D_2O 2 high pressure** creep		
503-4-25	N-3	none		4 internally pressurized	5 non-removable			
503-4-32	N-4	Yes				4 non-removable 2 low pressure** creep	2	
503-4-34	N-5	none				2 hydride-25* ppm D_2O		2 Material*** Compatibility Rods

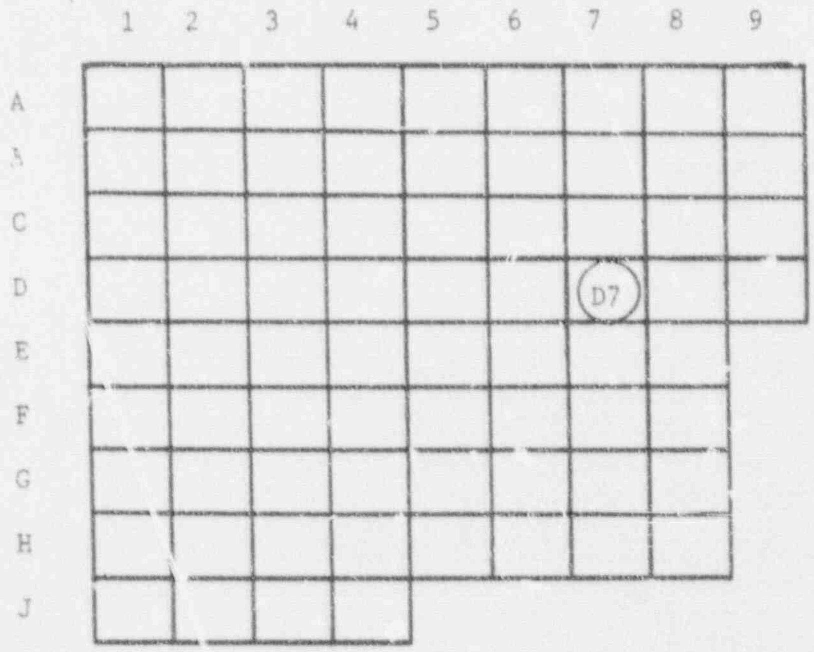
*   See Change Report 18 in Appendix C
**   See Change Report 19 in Appendix C
***   See Change Report 17 in Appendix C

mid life



SAXTON CORE III ASSEMBLY CONFIGURATION

Figure 2.1-1



ALPHA NUMERIC LOCATION IDENTIFICATION WITHIN ASSEMBLY

FIGURE 2.1-2

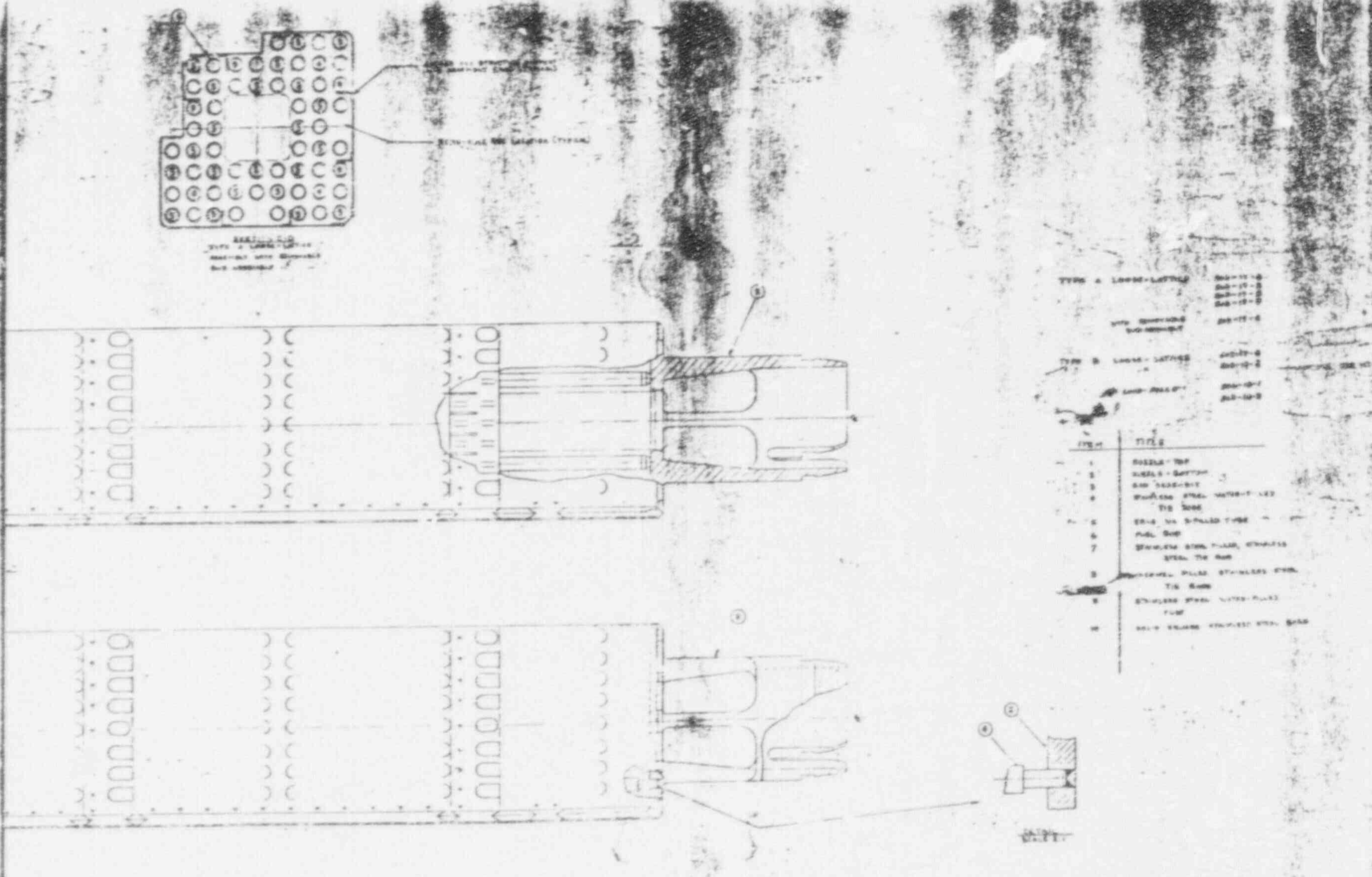
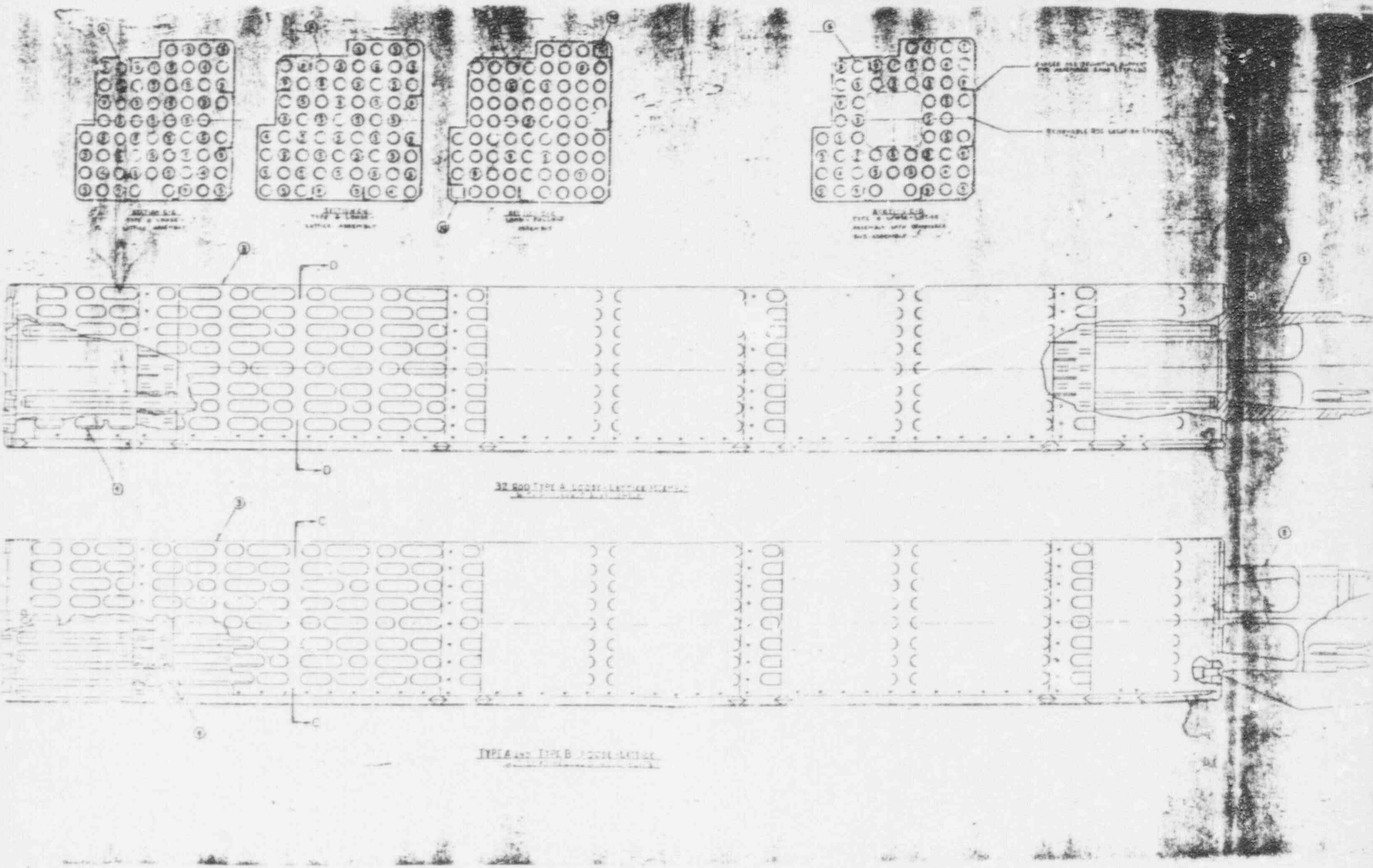
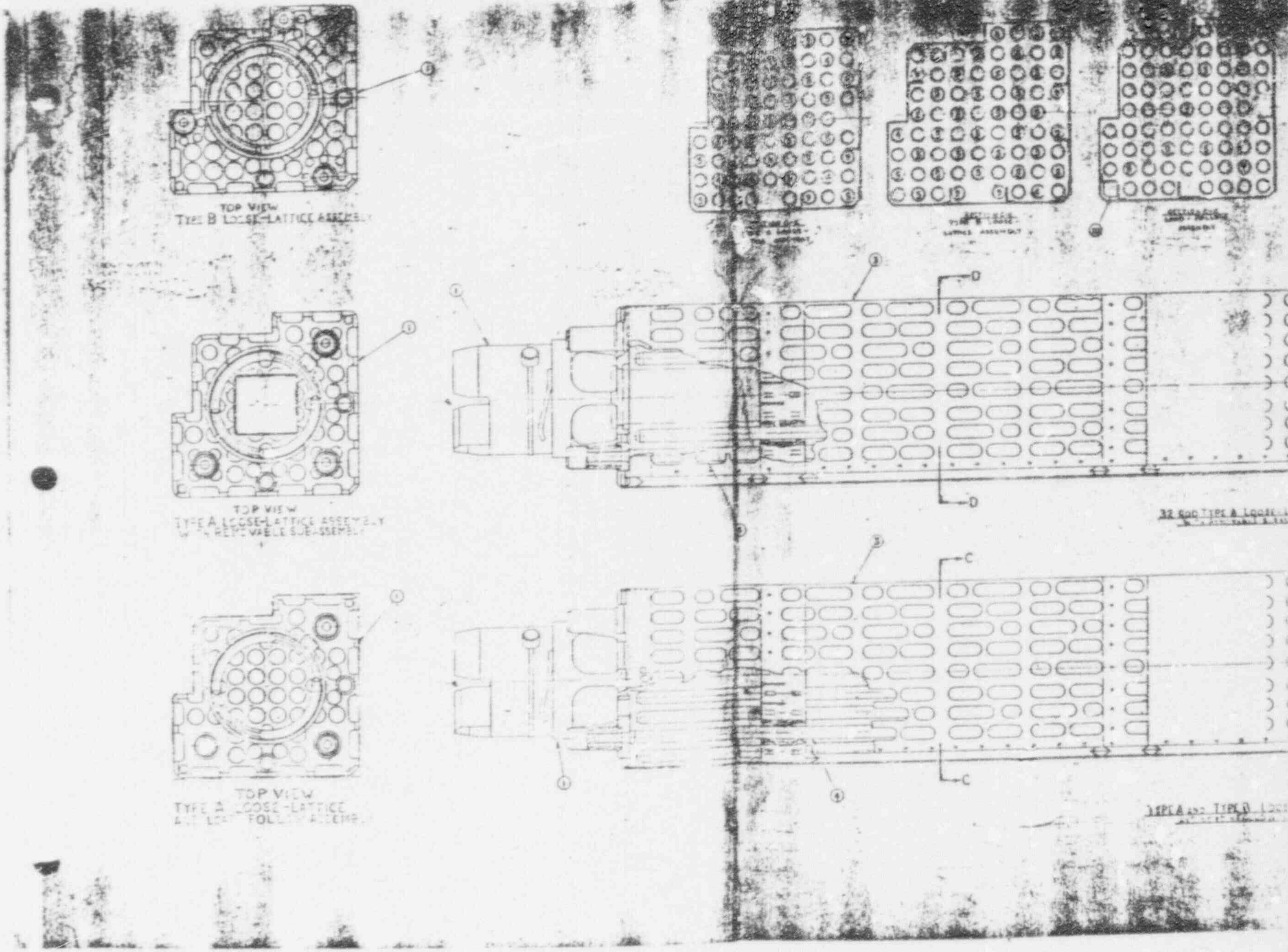
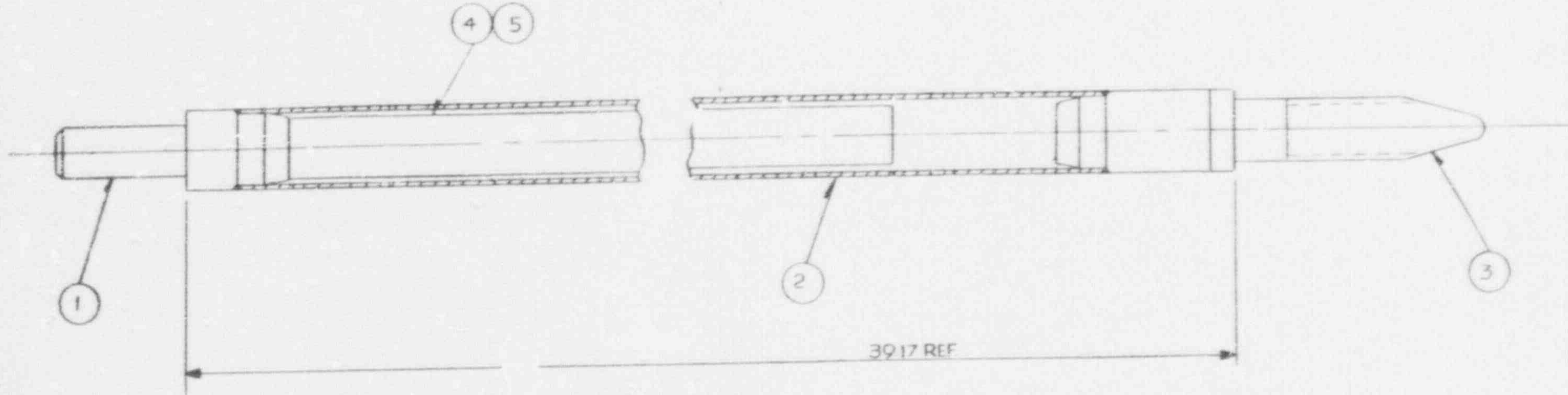


FIGURE 2.1-3
SAXTON REACTOR PLANT
9X9 LOOSE-LATTICE AND
LOAD-FOLLOW FUEL ASSEMBLY



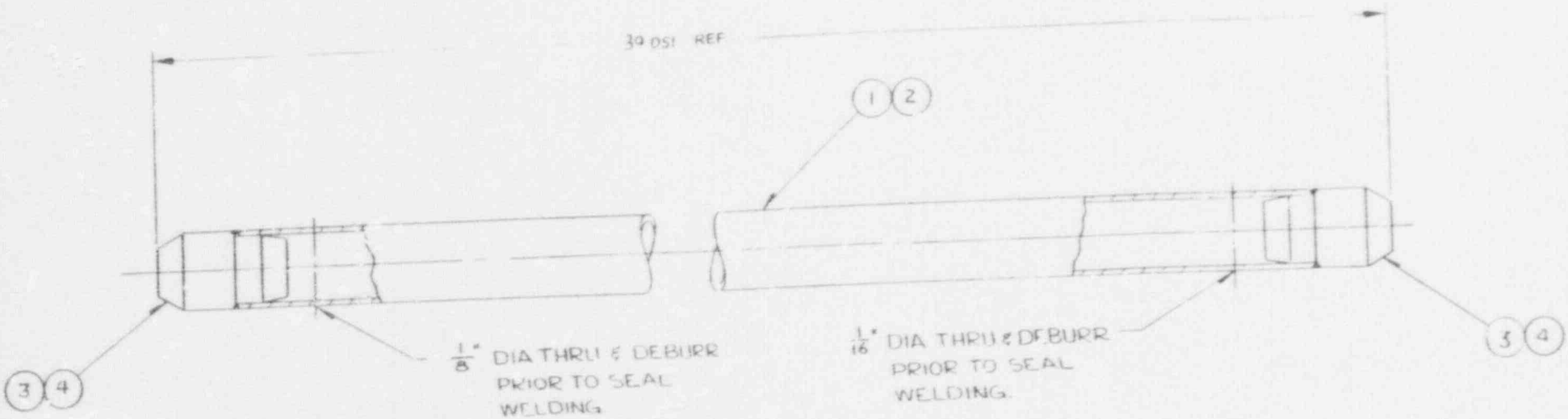

TOP VIEW
TYPE B LOOSE-LATTICE ASSEMBLY
TOP VIEW
TYPE A LOOSE-LATTICE ASSEMBLY
WITH REMOVABLE SUBASSEMBLY
TOP VIEW
TYPE A LOOSE-LATTICE
D
D
C
C



1. BOTTOM END PLUG
2. STAINLESS STEEL TUBE
3. TOP END PLUG
4. STAINLESS STEEL FILLER
5. INCONEL FILLER

SAXTON REACTOR PLANT
9X9 LOAD FOLLOW
TIE ROD ASSEMBLY

FIGURE 2.1-7



1. Zr-4 TUBE
2. STAINLESS STEEL TUBE
3. Zr-4 END PLUG
4. STAINLESS STEEL END PLUG

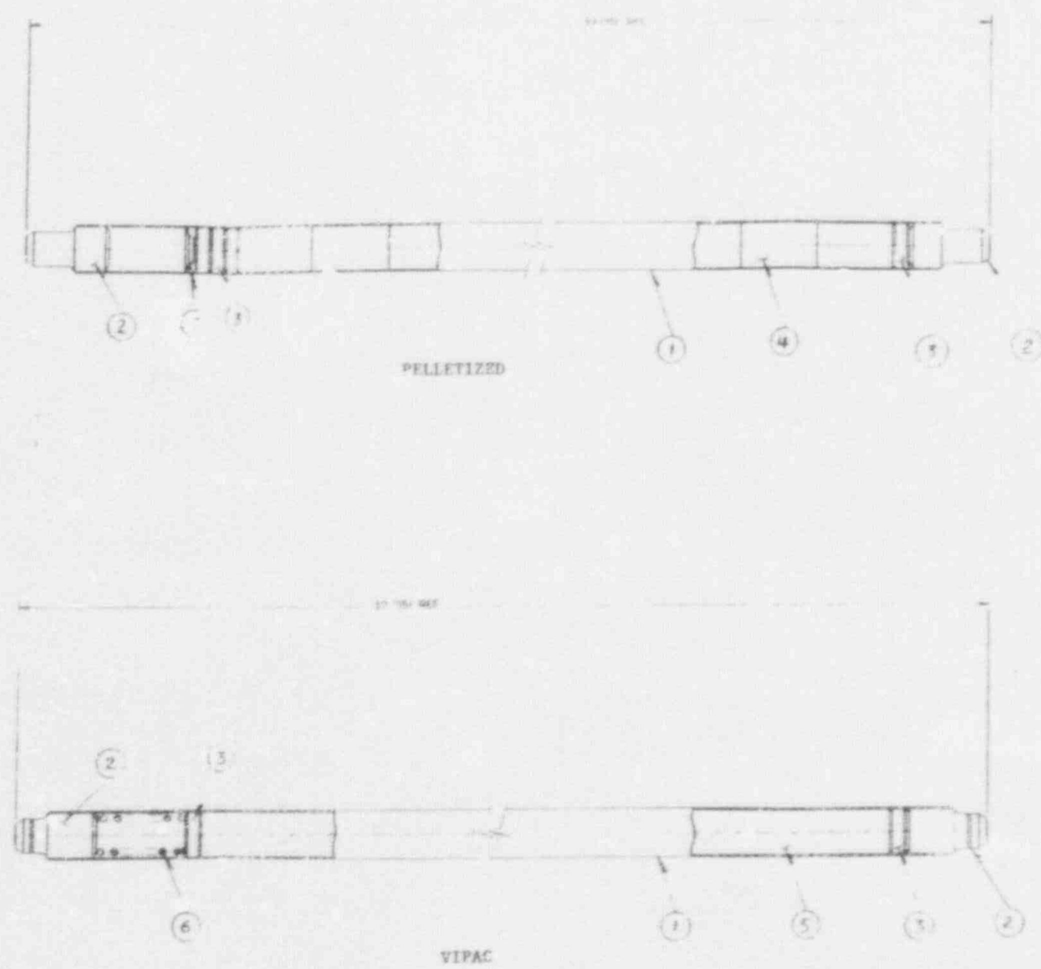
SAXTON REACTOR PLANT

WATER-FILLED

TUBE ASSEMBLY

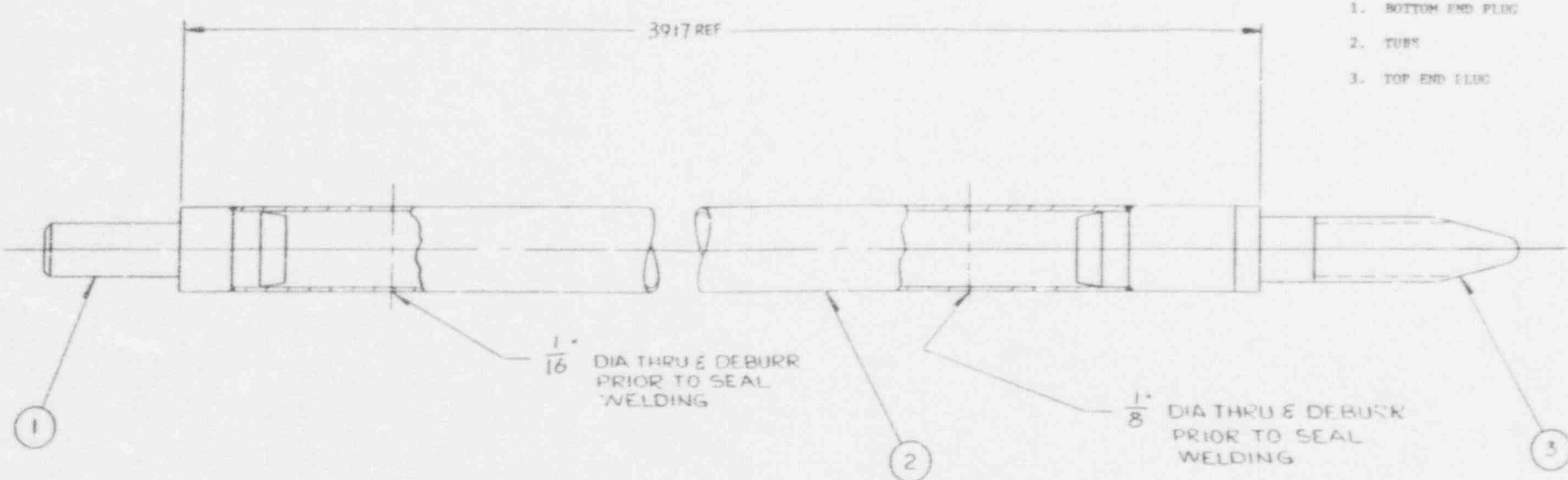
FIGURE 2.1-6

1. Zr-4 FUEL TUBE
2. NON-REMOVABLE TYPE END PLUG
3. Filler Al_2O_3
4. FUEL PELLET
5. VIPAC FUEL COLUMN
6. SPRING
7. BELLEVILLE SPRING



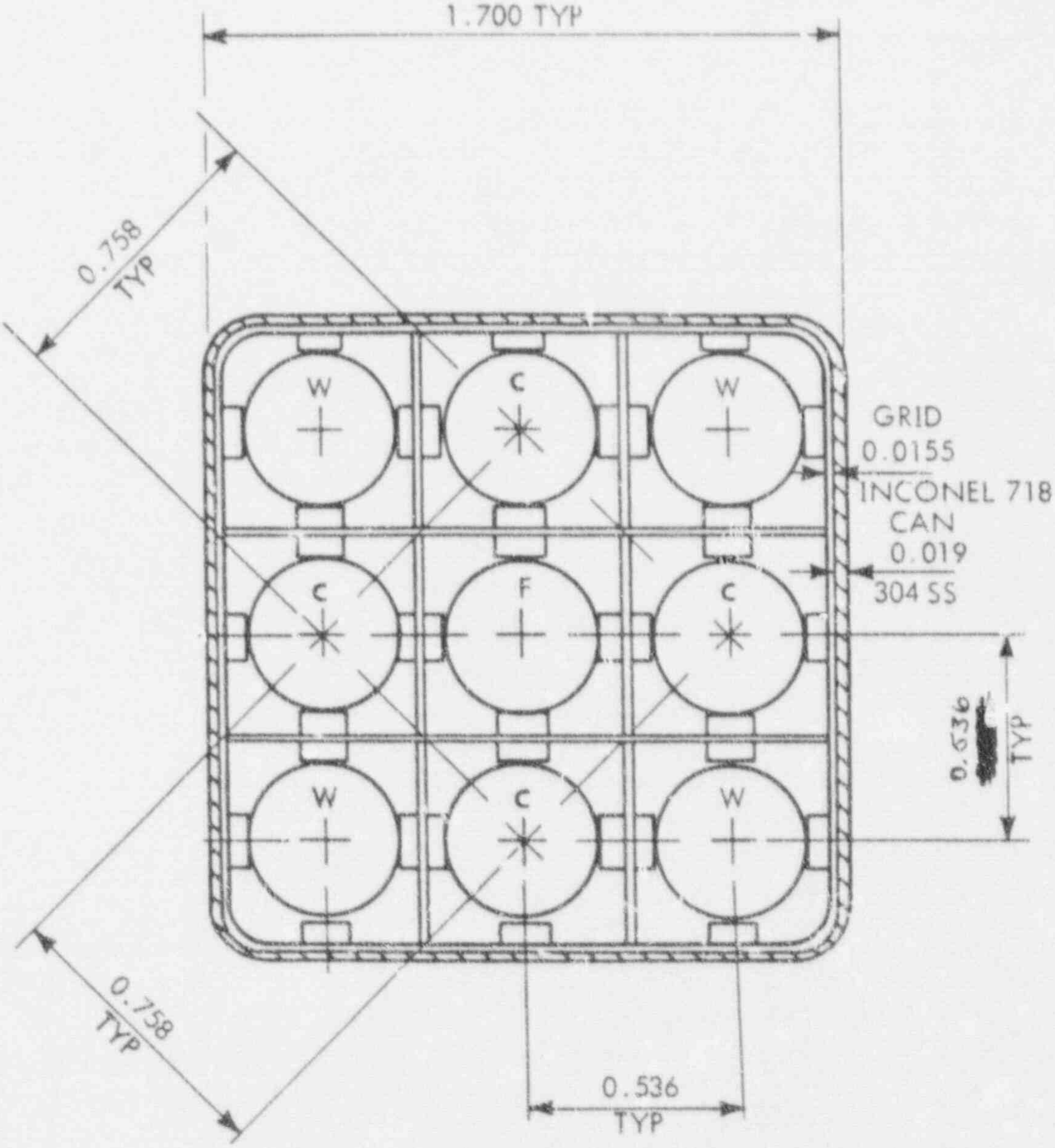
SAXTON REACTOR PLANT
LOOSE-LATTICE PLUTONIUM
FUEL ROD ASSEMBLY

FIGURE 2.1-5



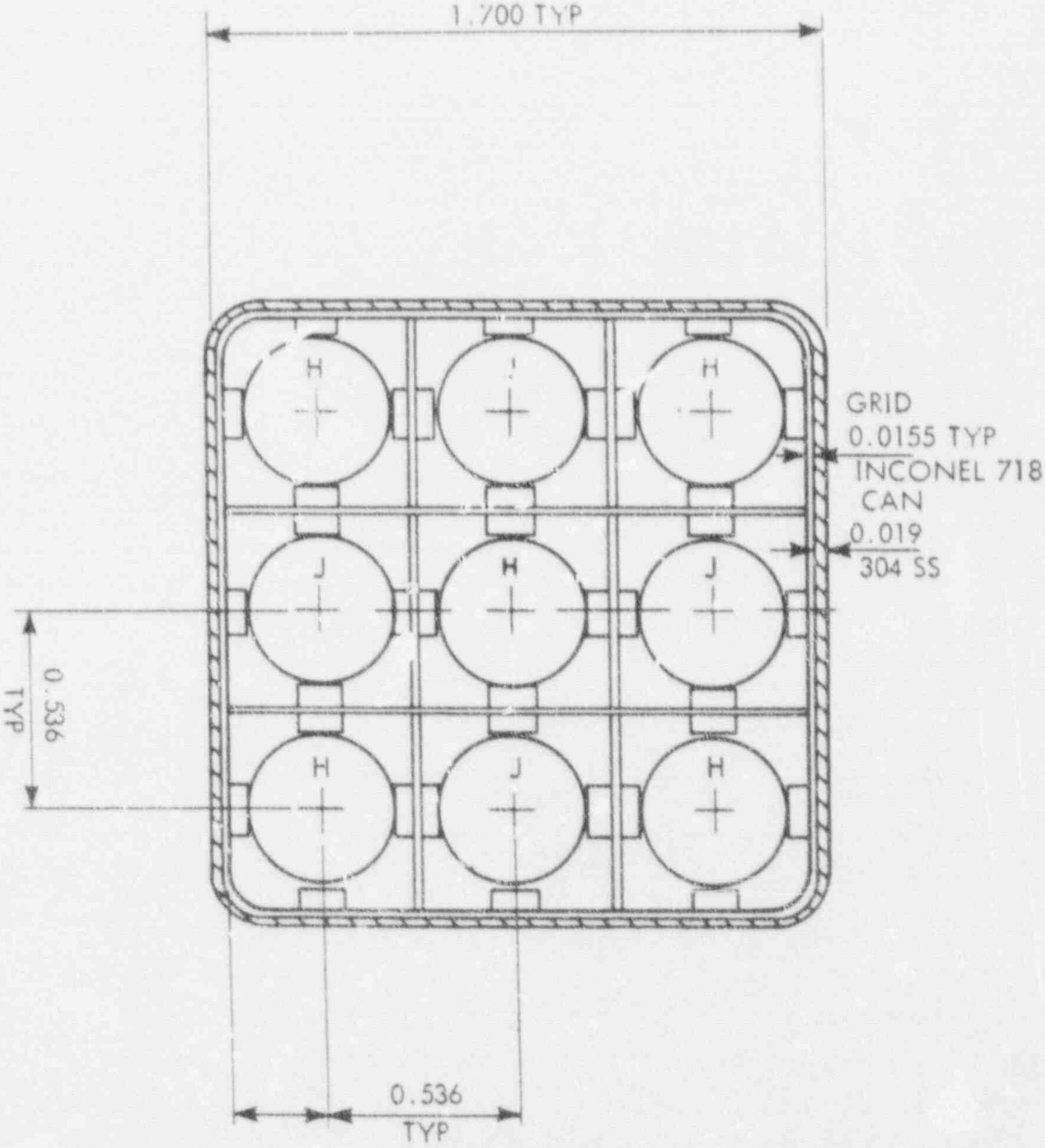
SAXTON REACTOR PLANT
9x9 LOOSE LATTICE
TIE ROD ASSEMBLY

FIGURE 2.1-4



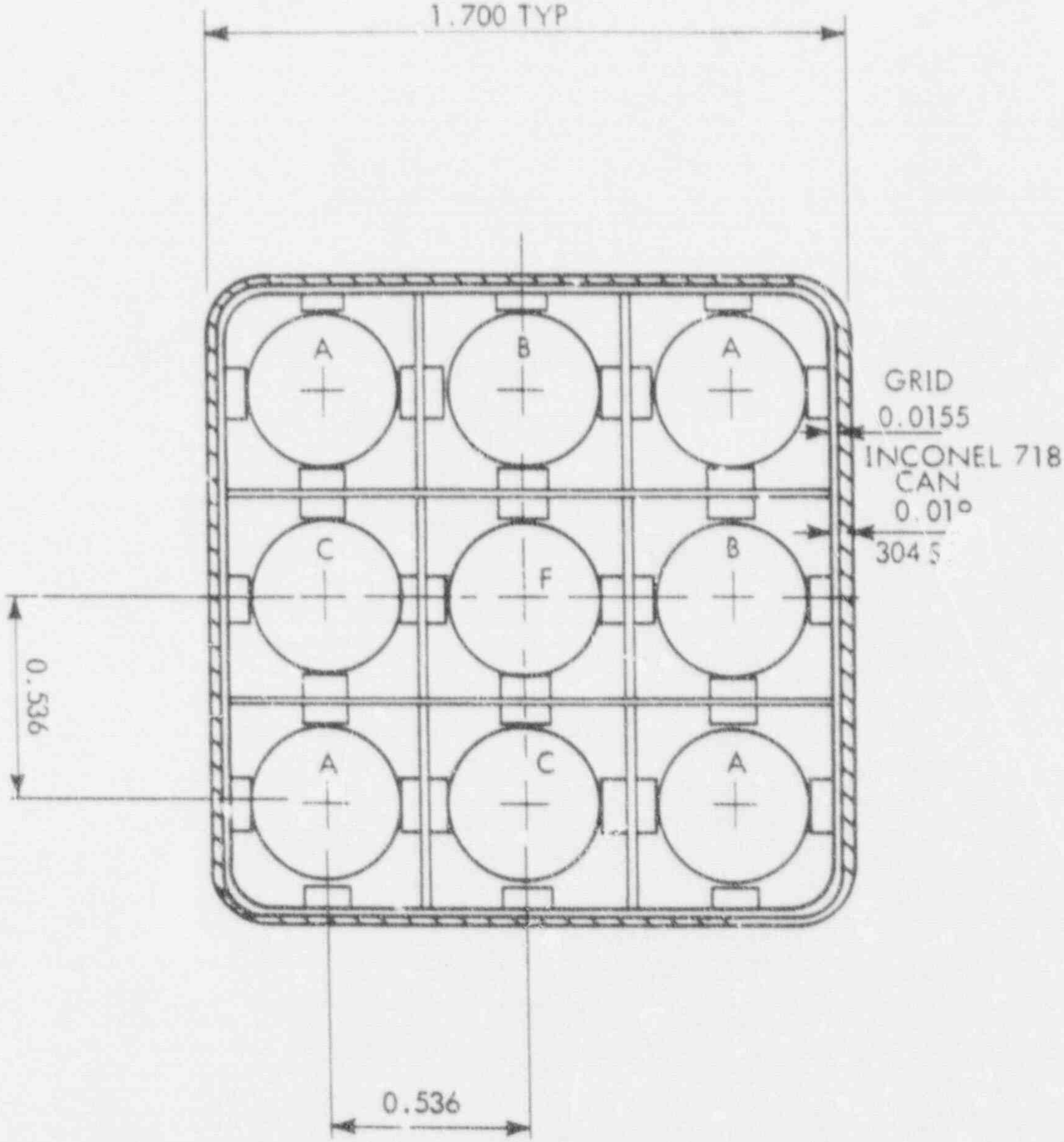
D - HIGH PRESSURE CREEP RODS
E - HYDRIDE TEST RODS
F - FLUX THIMBLE
W - WATER TUBE

Figure 2.1-9 Subassembly 503-4-33



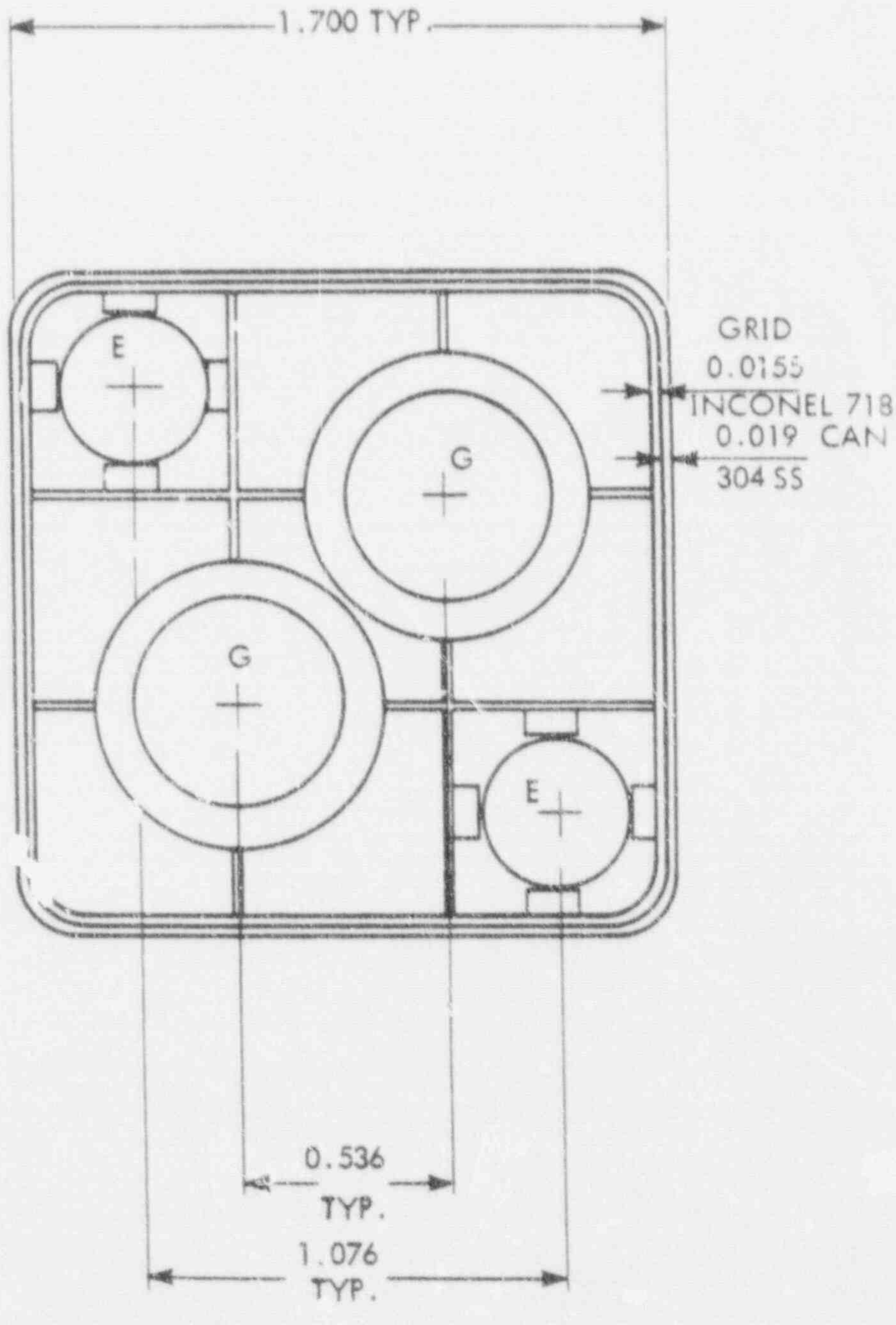
H - STAINLESS STEEL CLAD RODS - 5.7% ENRICHED
J - Zr-4 5.7% ENRICHED CLAD RODS - INTERNALLY PRESSURIZED

Figure 2.1-10 Subassembly 503-4-25



A - Zr-4 CLAD - 12.5% ENRICHED
B - CREEP TEST RODS - 12.5% ENRICHED
C - Zr-4 CLAD-PRESSURIZED - PuO_2-UO_2 RODS - 6.6% ENRICHED

Figure 2.1-11 Subassembly 503-4-32



E - HYDRIDE TEST FUEL RODS
G - MATERIALS COMPATIBILITY TEST RODS

Figure 2.1-12. Subassembly - 503-4-34

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

PHOTOGRAPHIC SCIENCES CORPORATION
770 BASKET ROAD
P.O. BOX 338
WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

PHOTOGRAPHIC SCIENCES CORPORATION
770 BASKET ROAD
P.O. BOX 338
WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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WEBSTER, NEW YORK 14580
(716) 265-1600

2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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(716) 265-1600

2.2 NUCLEAR DESIGN

2.2.1 Analytic Configuration

The burn-ups and enrichments in load-follow and loose-lattice assemblies used in Core III nuclear design are 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 alternate positions in the loose-lattice, water-filled Zircaloy tubes are installed to provide additional moderator while maintaining the desired flow characteristics. Three of these alternate positions in each assembly contain water-filled stainless steel tie rods.

The two UO_2 load-follow assemblies are installed on the flats of the center nine assemblies (See Figure 2.2-1) and contain variations in the U-235 [redacted] enrichment. The purpose of these enrichment variations is to increase the number of rods that operate near the peak power rating of 19.9 kw/ft. 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. As previously stated in the Mechanical Design section (page 2.1-6), the tie rods in the load-follow assemblies contain filler material to control local power distribution in these assemblies. Further, each load-follow assembly contains 4 water-filled Zircaloy tubes and 1 water-filled stainless steel tube as an additional means of improving the assembly power distribution characteristics. In addition, variations in fuel density and pellet dimensions are [redacted] also used to flatten power.

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 linear 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 linear power calculated for each fuel type. In order to maintain the peak linear power 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 linear power for each fuel type. As shown in this figure the peak linear power is maintained constant for approximately the first 1200 hours operation by increasing the core power from 24.9 to 25.8 MWt. At that time the peak linear power in the load-follow fuel has reached its design limit. Thereafter, the core power is increased at a reduced rate holding the peak linear power in the load-follow rods at the design limit. Because of the difference in burnup rate the linear 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 will have operated at their respective design peak linear powers 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 [redacted] values of core power (26.5 MWt), loose-lattice design linear power (24.0 kw/ft), and load-follow design linear power (19.9 Kw/ft) will not all occur simultaneously during the operating of Saxton Core III.

A small margin, 1.5 MWt, in total core power was included to insure that the design peak linear power in each fuel type 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 peak linear power in each fuel type i.e. the design linear power of 24.0 kw/ft in the loose-lattice assemblies and 19.9 kw/ft in the load-follow assemblies. The design core and assembly power distributions are given in Figure 2.3-1. The power characteristics of the fuel rods in the peak loose-lattice and load-follow assemblies are 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. Core III transient analysis physics parameters are 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.

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 rod design and fuel type were included.

The doppler coefficient is governed primarily by the amount of U-238 present. However, since Core II also contained 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% $\Delta k/k$ was determined which is an intermediate value to that of the two previous cores. The most reactive stuck rod was calculated to be worth 6.4% $\Delta k/k$. The bank worth with the maximum worth rod stuck withdrawn is thus 12.0% 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 and Minimum Boron Requirement

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, by 1% $\Delta k/k$.

In computing the maximum reactivity addition on cooldown, it was assumed that a temperature drop to 212°F occurred. The reactivity addition corresponds to the integral of the temperature coefficient curve for zero ppm, from 495°F to 212°F. Note that the temperature coefficient is most negative when there is no boron in the coolant. The difference in the maximum reactivity addition between the HFP and HZP, BOL cases is just the power defect (approximately 1.2% $\Delta k/k$).

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 peak linear powers. 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 CORE III TRANSIENT ANALYSIS PHYSICS PARAMETERS

Parameters	Core III
Delayed Neutron Fraction, β_{eff}	0.00493
Prompt Neutron Life	1.5×10^{-5} sec
Doppler Coefficient	-1.6×10^{-5} $\Delta k/k{}^\circ F$
Moderator Temperature Coefficient at 110°F, 1250 PPM, All Rods Out	$+.73 \times 10^{-4}$ $\Delta k/k/{}^\circ F$
Moderator Temperature Coefficient at H.F.P., 1215 PPM (495°F)	-1.0×10^{-4} $\Delta k/k/{}^\circ F$
Moderator Temperature Coefficient at End of Life, Hot Conditions (495°F, Zero PPM Boron)	-3.0×10^{-4} $\Delta k/k/{}^\circ F$
Pressure Coefficient at 80°F (BOL)	-1.0×10^{-6} $\Delta k/k/psi$
Pressure Coefficient at Hot Conditions (495°F, BOL, 1215 PPM Boron)	$+1.9 \times 10^{-6}$ $\Delta k/k/psi$
Void Coefficients at 80°F (1600 PPM Boron)	+0.08% $\Delta k/k/\%$ Void
Void Coefficients hot operating (495°F, 1215 PPM Boron)	-.25% $\Delta k/k/\%$ Void
Total Control Rod Bank Worth, H.F.P. (495°F, 1215 PPM)	18.4% $\Delta k/k$
Bank Worth With Rod #5 Stuck (495°F, 1215 PPM Boron)	12.0% $\Delta k/k$

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 (at 0 ppm) % ΔK/K	Minimum Boron Concentration Requirement for: Stuck Rod	Stuck Rod Steam Break
Hot Zero Power (BOL)	11.3	12.0	.3	3.0	30	330
Hot Full Power (BOL) Equi Xe	7.3	12.0	-3.7	4.2	0	50
Hot Zero Power 5000 Hrs	1.3	12.0	[redacted] -9.7	3.5	0	0

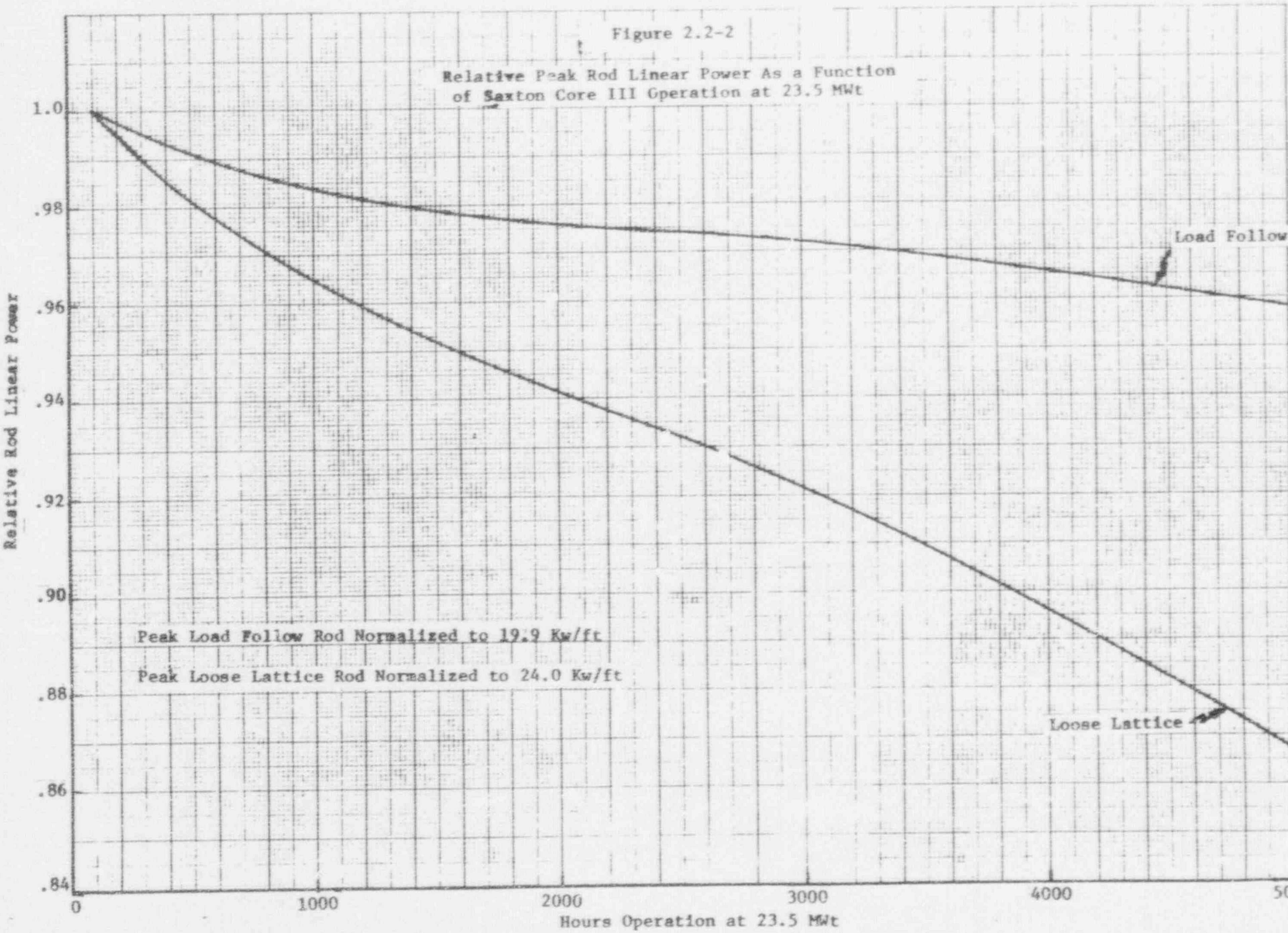


Figure 2.2-2
Relative Peak Rod Linear Power As a Function of Saxton Core III Operation at 23.5 MWt
Relative Rod Linear Power
1.0
.98
.96
.94
.92
.90
.88
.86
.84
0
1000
2000
3000
4000
5000
Hours Operation at 23.5 MWt
Load Follow
Loose Lattice
Peak Load Follow Rod Normalized to 19.9 Kw/ft
Peak Loose Lattice Rod Normalized to 24.0 Kw/ft

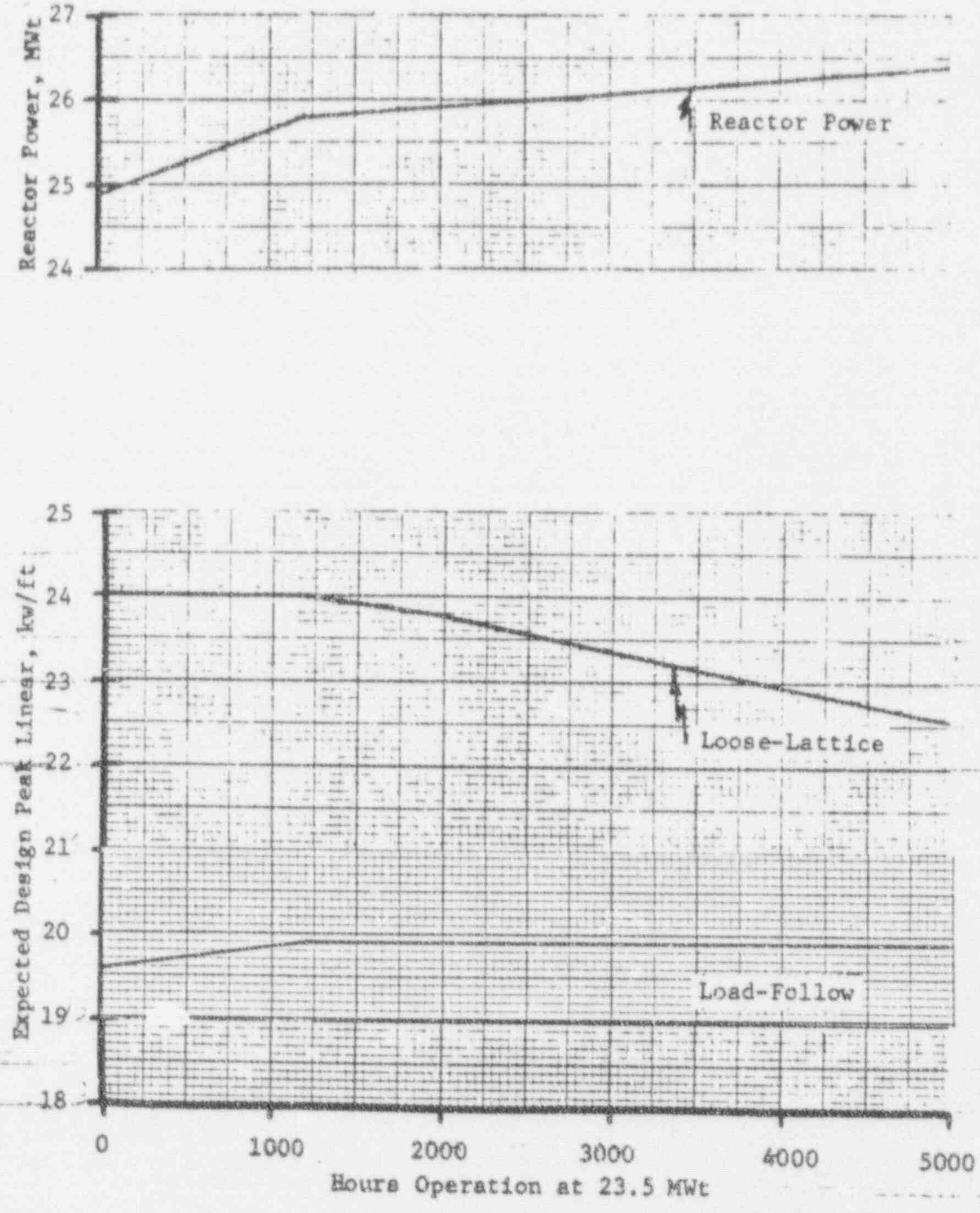


Figure 2.2-3
Anticipated Power Operation
Of Saxton Core III

Revised
3/70

APPENDIX A

Failed Fuel Monitor System

ntroduction

A failed fuel monitor is installed as an experimental system at the Saxton eactor. The general layout, checkout and calibration procedure and anticipated erformance are described below.

Fuel element failure is indicated by the increase of gamma activity resulting rom fission products in the reactor coolant. However, it should be noted that here are other sources of gamma radiation such as:

a. Background

b. N-16 activity

c. Other radioactive isotopes such as corrosion products

escription of System

The system utilizes the pressure drop across the steam generator to circulate reactor coolant in a bypass loop which consists of about 210 feet of stainless steel tubing, a heat exchanger, a radiation monitor, a remote operated flow control valve and a remote reading flow meter (Figure 1).

The radiation monitor consists of a coil of stainless steel tubing which surrounds two GM detectors. To reduce the background radiation the coil is shielded with 4" of lead. The detectors have an operating ange from 1 mr/hr to 30 R/hr.

The heat exchanger maintains reactor coolant below 140°F to prevent damage to the GM detectors. A thermocouple is provided for monitoring the temperature at the detectors.

The remote operated flow control valve and remote reading flow meter permit the reactor coolant flow to be adjusted to provide delay time of about 40 seconds. This delay time provides for sufficient decay of the N-16 activity in the coolant to allow proper operation of the system while the reactor is at full power.

The readout system consists of two independent ratemeter channels and a recorder which can record either of the channels. The two channels and their associated electronics are shown in Figure 2.

Checkout and Cal. ration

The two channels have received a functional checkout and their operating range, counts/minute vs. mr/hr using a Co-60 source, has been determined. The background level in counts/minute for zero power operation has been obtained and a flow rate of 0.42 gpm in the bypass loop has been established as being optimum for the system.

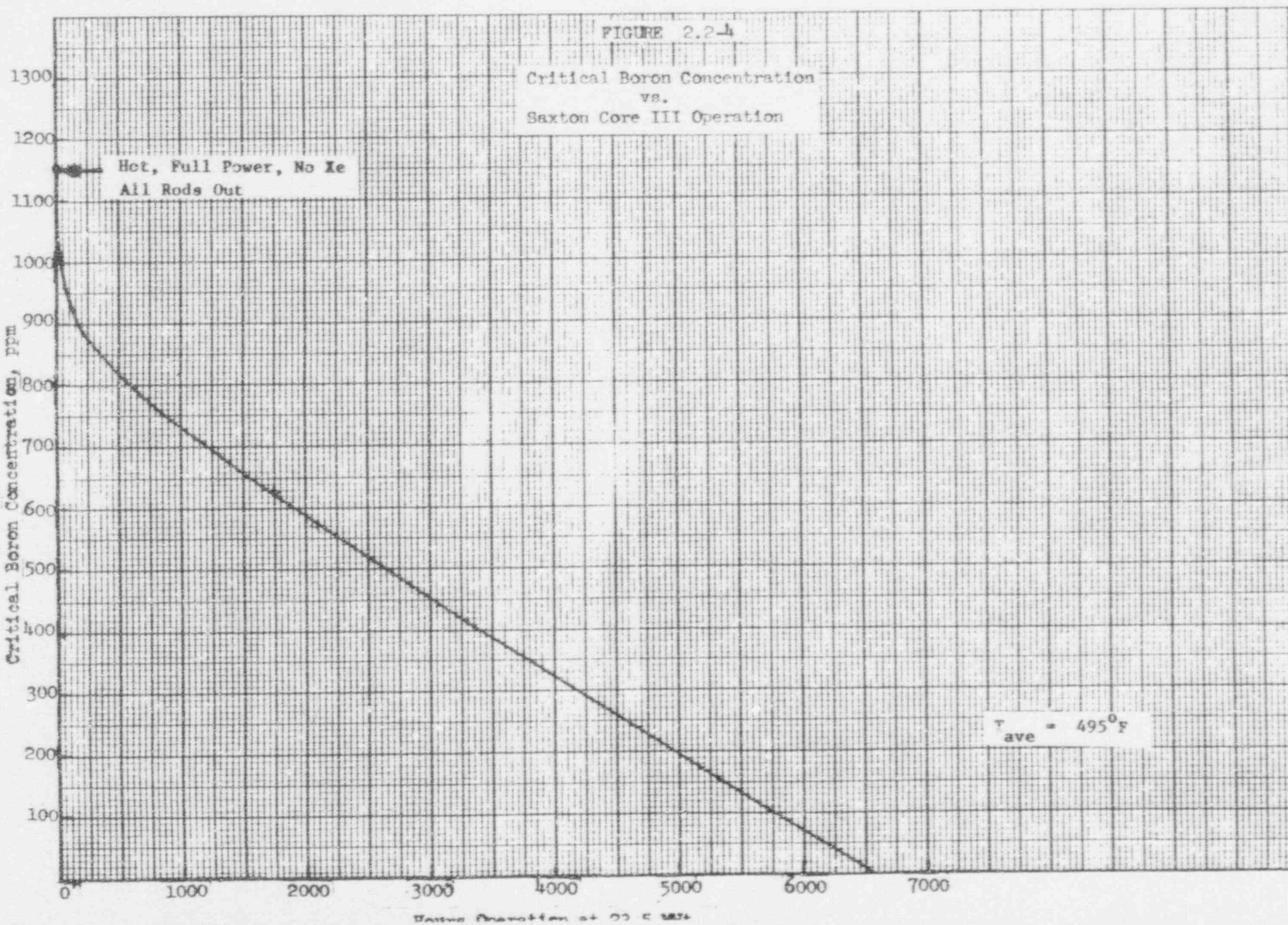


FIGURE 2.2-4

Critical Boron Concentration
vs.
Saxton Core III Operation

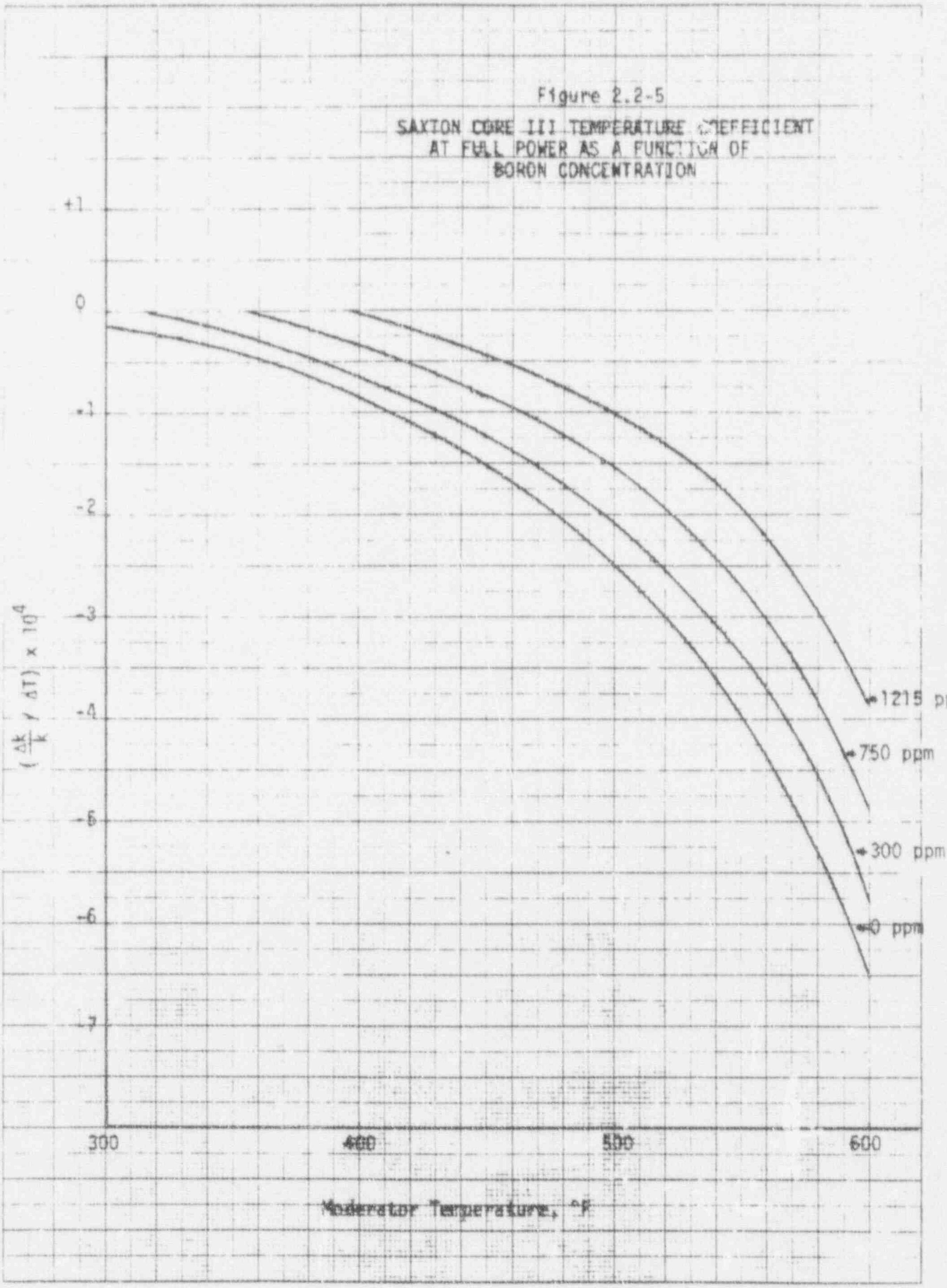


Figure 2.2-5
SAXTON CORE III TEMPERATURE COEFFICIENT AT FULL POWER AS A FUNCTION OF BORON CONCENTRATION
+1
0
-1
-2
-3
-4
-5
-6
-7
$(\frac{\Delta k}{k} / \Delta T) \times 10^4$
1215 pp
750 ppm
300 ppm
0 ppm
300
400
500
600
Moderator Temperature, °F

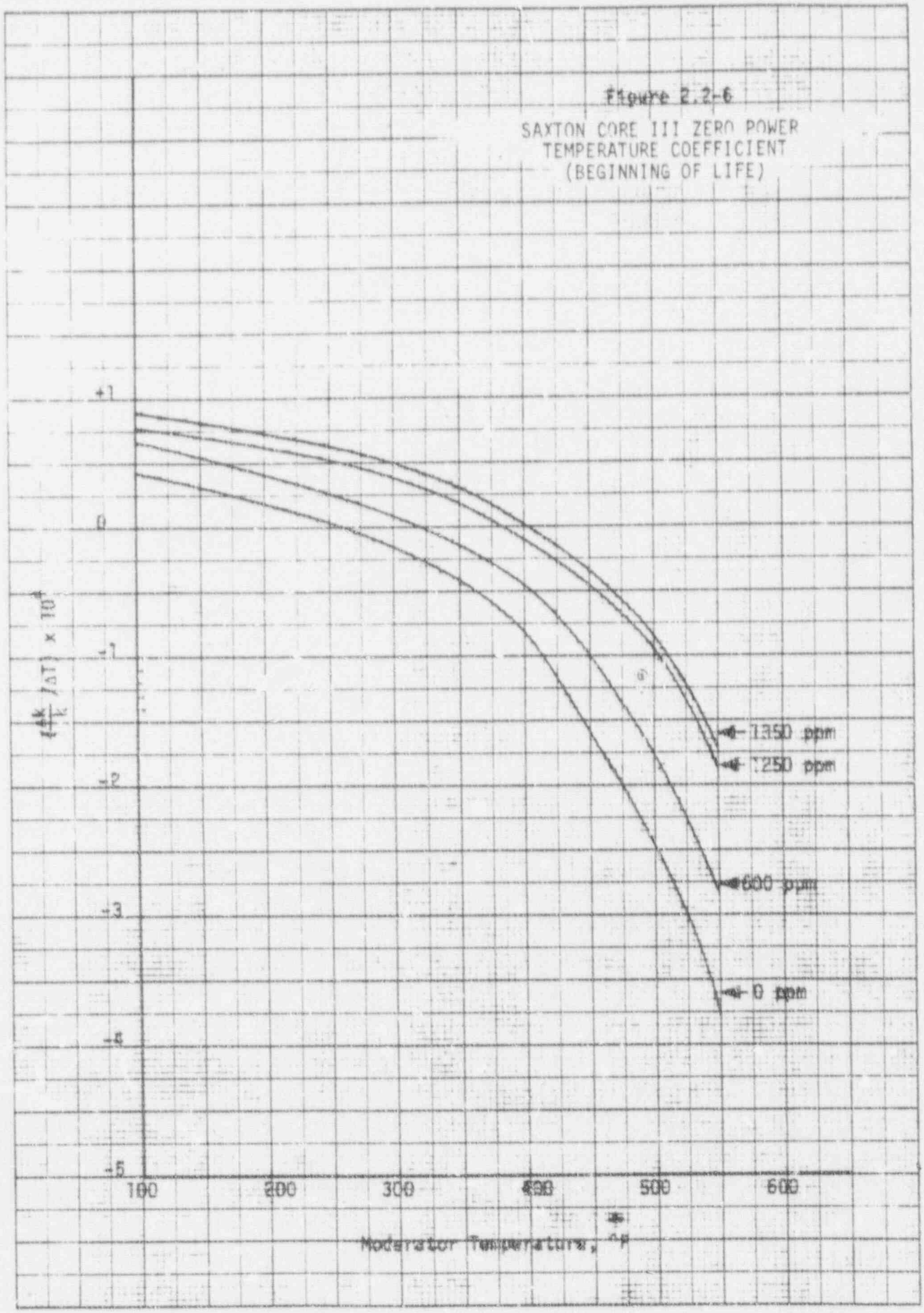


Figure 2.2-6
SAXTON CORE III ZERO POWER
TEMPERATURE COEFFICIENT
(BEGINNING OF LIFE)
+1
0
-1
-2
-3
-4
-5
1350 ppm
1250 ppm
600 ppm
0 ppm
100
200
300
400
500
600
Moderator Temperature, °F

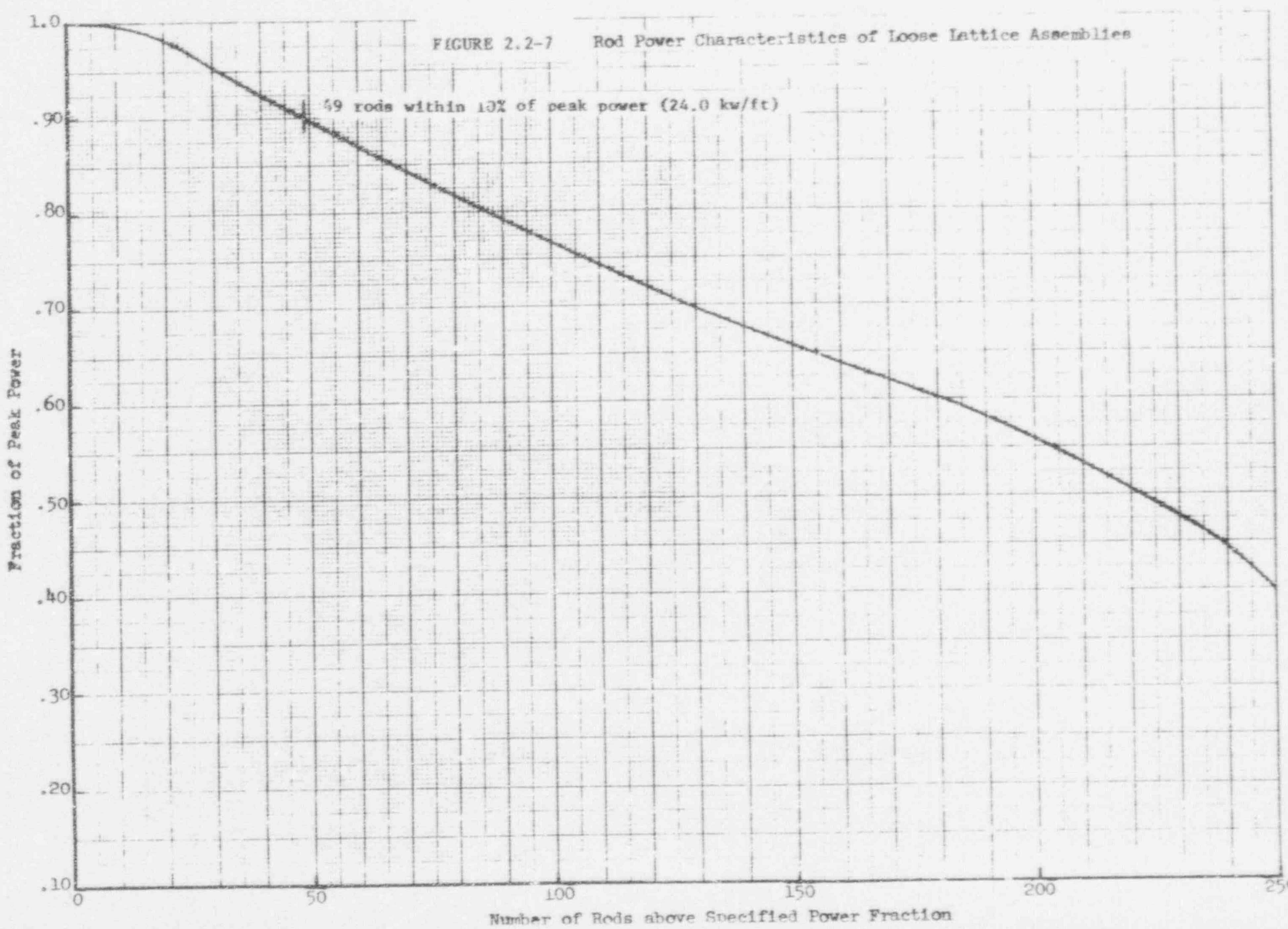
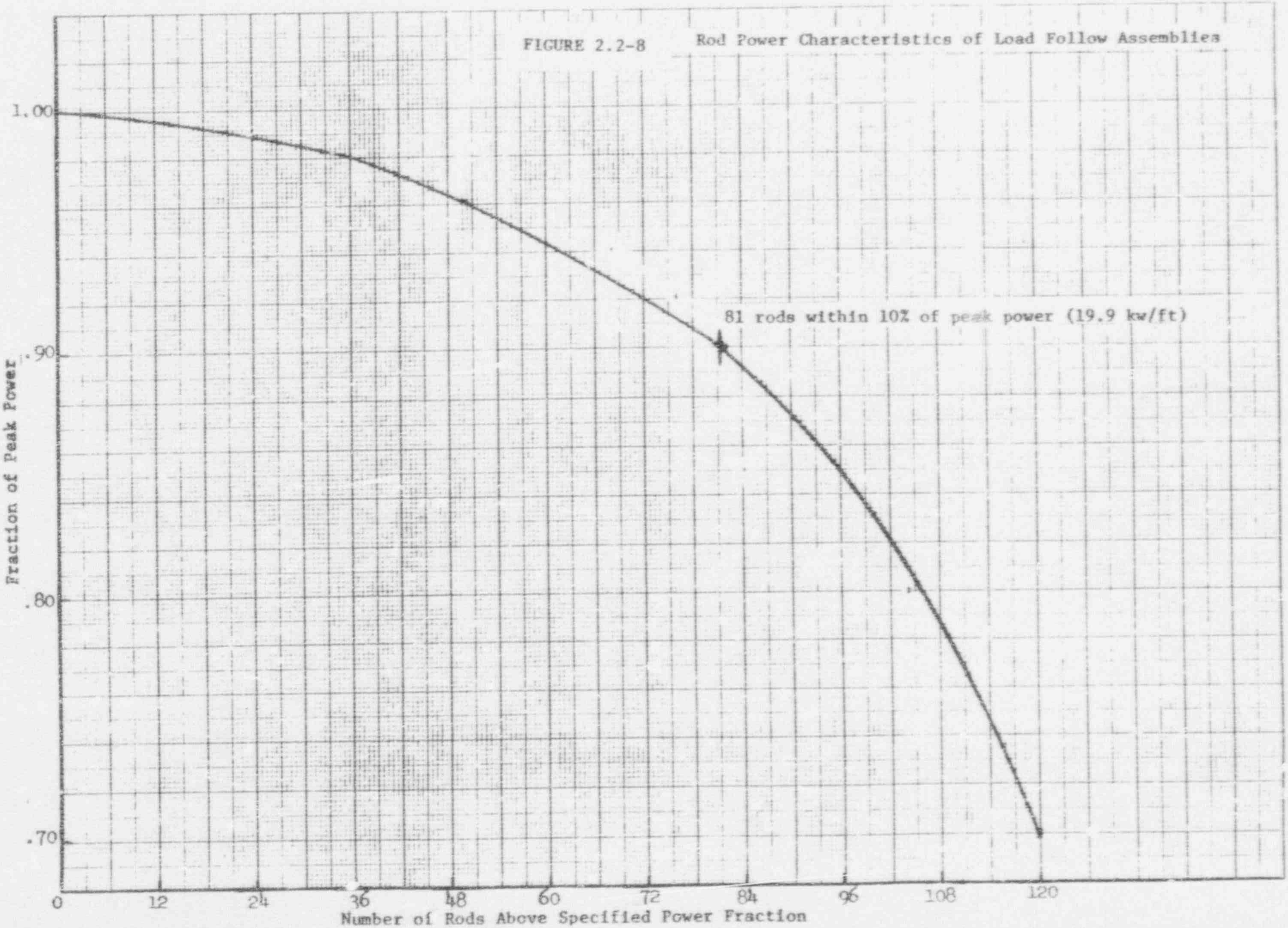


FIGURE 2.2-7 Rod Power Characteristics of Loose Lattice Assemblies
49 rods within 10% of peak power (24.0 kw/ft)
Fraction of Peak Power
1.0
.90
.80
.70
.60
.50
.40
.30
.20
.10
0
50
100
150
200
250
Number of Rods above Specified Power Fraction

FIGURE 2.2-8 Rod Power Characteristics of Load Follow Assemblies



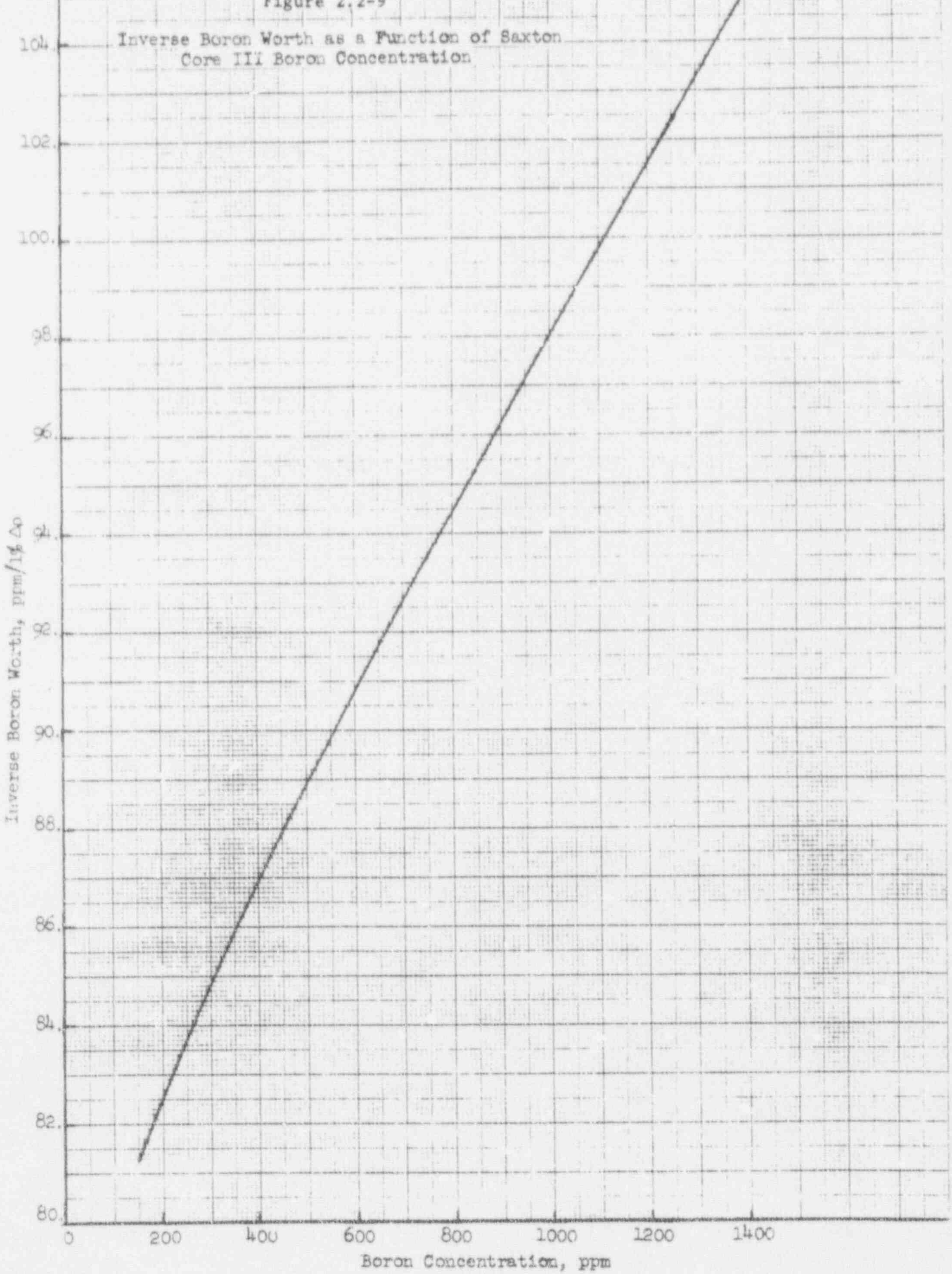


Figure 2.2-9
Inverse Boron Worth as a Function of Saxton
Core III Boron Concentration
104.
102.
100.
98.
96.
94.
92.
90.
88.
86.
84.
82.
80.
Inverse Boron Worth, ppm/1% Δρ
0
200
400
600
800
1000
1200
1400
Boron Concentration, ppm

2.3 THERMAL-HYDRAULIC DESIGN

2.3.1 GENERAL

A detailed evaluation of the thermal and hydraulic characteristics of the proposed Saxton Core III configuration has been conducted. The thermal and hydraulic margins are identified. It is 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 is greater than 1.30 at the maximum reactor control set point conditions.

2.3.2 THERMAL AND HYDRAULIC DESIGN CRITERIA

Acceptable Saxton Core III operating conditions require that several thermal and hydraulic criteria 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.

(2) Center melting of the fuel is not permitted during normal steady state operation or anticipated transient conditions.

Evaluations were conducted at both nominal and overpower conditions using the THINC code.[1]

2.3.3 THERMAL AND HYDRAULIC ANALYSIS

Best estimate X-Y power distributions are shown in Figure 2.3-1. These values are increased by an 8% nuclear uncertainty before being used in the thermal and hydraulic analysis. Figure 2.3-2 shows the detailed rod by rod power distribution for the limiting loose-lattice assembly (D-2) and the limiting load-follow assembly (E-3). These power distributions include an 8% nuclear uncertainty factor.

<u>Hot Channel Factors</u>

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.

---

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

(a) Heat Flux (Fq)

The total hot channel factor is the product of a nuclear hot channel factor describing the nuclear power distribution effects and an engineering hot channel factor to allow for variations from nominal design conditions. The engineering heat flux factor includes the variations in fuel rod diameter and fuel pellet diameter, density, enrichment and eccentricity. Tables 2.3-1 and 2.3-3 summarize the heat flux hot channel factors for Saxton Core III.

(b) Enthalpy Rise (FΔH)

The enthalpy rise in the hot channel is calculated using THINC. For convenience, the total enthalpy rise factor can still be considered as the simple product of several subfactors. In actuality, the interaction of various effects such as power distributions, mixing, and flow redistribution cannot adequately be described by simple factors. Parametric studies were used to determine the subfactor attributed to statistical variations in pellet diameter, density, and enrichment, and rod diameter, pitch and bowing. These variations result in an 8% increase in the enthalpy rise of the hot channel ($1 + 3\sigma = 1.08$).

Possible inlet flow maldistribution effects are taken into account by reducing the flow to the fuel assembly containing the hot channel by 7%, based on experimental data. Flow redistribution, including the effects of local boiling, and mixing are calculated using THINC. The net result of these effects is a decrease in the enthalpy rise of the DNB limiting hot channel by 8%. Table 2.3-1 summarizes the breakdown of engineering hot channel factors in the DNB limiting channel.

Because of unheated rods, the total enthalpy rise cannot be considered as a simple product of nuclear factors and engineering factors. Enthalpies at the exist of the hot channel are calculated by THINC and are reported with the nuclear hot channel factors in Table 2.3-3.

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 chosen 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 presence of the unheated wall affects 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.

The minimum DNBR during overpower conditions is 1.30, as insured by the reactor trip setpoints (See Section 3.1).

Thermal and Hydraulic Design Parameters

Detailed THINC analyses were conducted for the various limiting thermal and hydraulic conditions in the core. The following overall limits apply at nominal steady state operation:

1. 28 MWt
2. 19.9 Kw/ft in load-follow assemblies
3. 24.0 Kw/ft in loose-lattice assemblies

(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 Square Sleeves, and in Line Contact with an Unheated Wall", ASTM 67-WA/HT-29 (1967).

Note that these limits do not occur simultaneously (See Section (1.5). The 24.0 Kw/ft limit occurs at 24.9 MWt at the beginning of life; whereas, the peak fuel rod power in the load-follow assemblies (19.9 Kw/ft at 25.8 MWt) does not occur until later in life (~1200 EFPH).

Table 2.3-2 presents the thermal and hydraulic characteristics for Core III at its design power of 28 MWt. Table 2.3-3 presents the thermal and hydraulic characteristics of the various types of assemblies and sub-assemblies, each at their worst design conditions.

Initially Core III power (24.9 MWt) is limited by the design linear power in the loose-lattice (24.0 Kw/ft) which occurs in the loose-lattice assembly at core location D-2. Further, the Core III hot channel is in the assembly at core location D-2 which has a minimum steady state DNBR=1.81. At this time all other assemblies in the core have lower linear powers and higher DNBR's than the assembly at core location D-2.

After about 1200 EFPH, Core III power is limited by the design linear power in the load-follow assembly (19.9 Kw/ft) and occurs in the load-follow assembly at core location E-3, which has a minimum steady state DNBR=1.75.

That assembly which limits core power changes from loose-lattice to load-follow after about 1200 EFPH, because the peak rod linear power in the loose-lattice burns down at a faster rate than that of the load-follow assemblies. The core hot channel after about 1200 EFPH is also in the load-follow assembly at core location E-3. All other assemblies in the core at this time have lower peak linear powers and higher DNBR's. Core power will not exceed a nominal 28 MWt. Further, that nominal power at which the measured power distribution combined with analysis gives 19.9 Kw/ft in load-follow assemblies, or 24.0 Kw/ft in loose-lattice assemblies will not be exceeded.

The worst condition for the peripheral fuel assemblies occurs at a core power of 28 MWt. The assembly at core location B-3 has the highest peak linear power (14.9 Kw/ft) and a minimum DNBR=3.30. If the subassembly having the worst local flow conditions (Materials Compatability Subassembly) were put in the highest power location (assembly location B-3), the minimum DNBR would be reduced to 2.35.

In summary:

1. The minimum DNBR occurs in the load-follow assembly at core location E-3.

2. The highest peak linear power occurs in the loose-lattice assembly at core location D-2.

3. Peripheral subassemblies have higher DNBR's and lower peak linear powers than the assemblies at core location E-3 or D-2.

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 conditions has been estimated to be 4540°F for the loose-lattice rods and 4650°F for the load-follow rods. At the maximum overpower conditions of 112% power a maximum central fuel temperature of 4880°F is predicted for the loose-lattice assemblies. This is approximately 120°F below 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 fuel 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 120 load-follow fuel rods are expected to operate with some small fraction of their cladding at a 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 (19.9 Kw/ft) the maximum mean cladding temperature would be 720°F. These temperatures were used in the evaluation of mechanical limits.

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 Flow Redistribution and Mixing	0.92
$F_{\Delta H}^E$, Total		1.00

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	2.41 ft^2
Average Velocity Along Fuel Rods	6.28 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	27.0 F
Average in Vessel	493.0 F
Average in Core	493.5 F
Heat Transfer	
Active Heat Transfer Surface Area of Fuel Rods	375.3 ft^2
Average Heat Flux	219,400 Btu/hr-ft^2
Average Thermal Output	6.58 kw/ft
Maximum Clad Surface Temperature at Nominal Pressure	657.4 F

* These thermal and hydraulic design parameters supercede all those previously reported for Core III.

** At 63 cycles

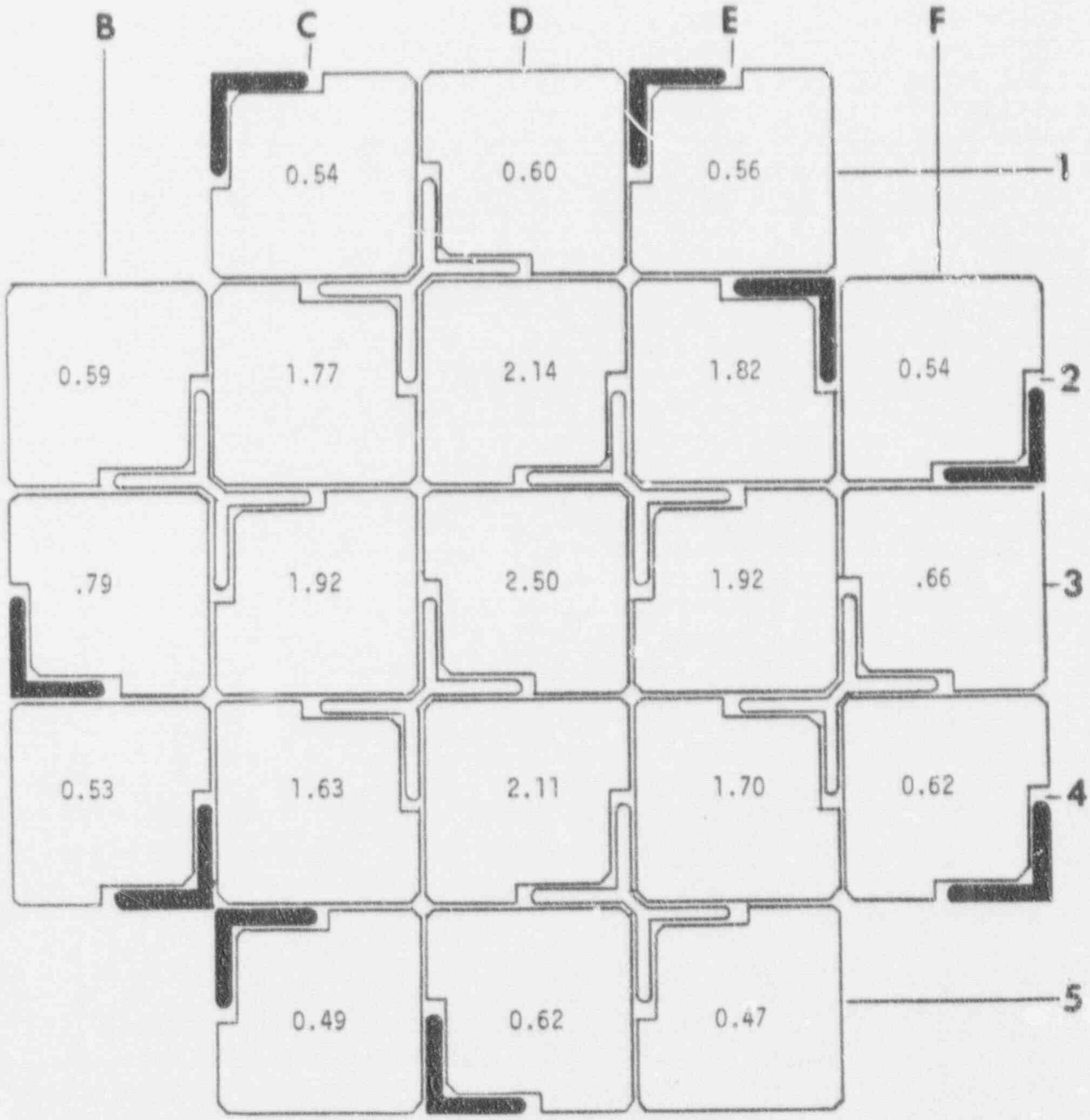
TABLE 2.3-3

THERMAL AND HYDRAULIC DESIGN PARAMETERS

WORST DESIGN CONDITIONS

Center Core Region	Load-Follow Assembly at Core Location E-3	Loose-Lattice Assembly at Core Location D-2	Loose-Lattice Assembly at Core Location D-3	Peripheral Assembly at Core Location F-4 Containing Materials Compatability Subassembly	Peripheral Assembly at Core location B-3	Peripheral Assembly at Core Location B-3 Containing Materials Compatability Subassembly
F_q Heat Flux Hot Channel Factor	3.28	4.10	3.55	1.84	2.26	2.58
$F_{\Delta H}^{NUC}$ Nuclear Radial Factor	2.28	2.95	2.80	1.32	1.62	1.86
F_Z^{NUC} Nuclear Axial Factor	1.38	1.33	1.33	1.33	1.33	1.33
Nominal Outlet Enthalpy in Hot Channel, Btu/lb	542.5	528.1	524.9	510.4	512.7	618.9
Saturation Enthalpy at Minimum Steady State Pressure, Btu/lb	695.0	695.0	695.0	695.0	695.0	695.0
Maximum Heat Flux, Btu/hr-ft^2	662,300	900,200	761,900	402,800	495,200	633,600
Maximum Thermal Output, kw/ft	19.9	24.0	22.9	12.1	14.9	19.0
Minimum W-3 DNB Ratio at 100% Power Nominal Condition	1.75	1.81	2.09	3.44	3.30	1.90
Core Power at which minimum W-3 DNB Ratio occurs, MWt	25.8	24.9	24.9	28.0	28.0	28.0

As a result of more refined calculations, it has been determined that the maximum linear power for the materials compatability subassembly in any Core III peripheral subassembly location will be 19.0 kw/ft rather than the 16.1 kw/ft reported in Change Reports 17 and 18 (See Appendix B) and that the hot channel factors are somewhat higher than those given in Change Report 17.



CORE RADIAL POWER DISTRIBUTION

Figure 2.3-1

DESIGN NUCLEAR RADIAL POWER FACTORS IN
THERMALLY CONTROLLING CORE III ASSEMBLIES

(Values include 8% uncertainty factor)

1.84		1.86		1.88		1.88		1.94
	1.97		2.01		2.02		2.03	
2.10		2.15		2.17		2.17		2.24
	2.24		2.27		2.27		2.24	
		2.36		2.37		2.33		
	2.50		2.50		2.48		2.33	
2.67		2.59		2.56		2.49		
	2.76		2.70		2.52		2.40	
2.95		2.90						

DESIGN PEAK ROD
KW/FT = 24.0

LOOSE-LATTICE ASSEMBLY AT CORE LOCATION D-2

					2.24	1.96	1.80	
	2.16	2.24	2.26	2.21	2.15	2.03		1.64
	2.21	2.23		2.25		2.04	1.87	1.59
	2.24	2.16	2.22	2.25	2.11	2.03	1.86	1.65
	2.23	2.20	2.16		2.06	1.92	1.85	
2.24	2.15	2.09	2.19	2.22	2.09	2.02	1.87	1.65
2.25	2.22	2.20		2.20		1.97	1.85	1.63
2.28		2.22	2.22	2.25	2.13	2.00		1.66
	2.24	2.28	2.22		2.17	2.00	1.75	1.65

DESIGN PEAK ROD
KW/FT = 19.9

LOAD FOLLOW ASSEMBLY AT CORE LOCATION E-3

Figure 2.3-2

2.4 FUEL PERFORMANCE EVALUATION

2.4.1 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 ad 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 rod average burnups and operation at power linear power densities ≤ 16 kw/ft, there is a significant risk of failure of certain of these rods in Core III.[1] 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.

The Core III loose-lattice region has been designed for a peak linear power of 24 kw/ft. Design analyses predict clad strains and internal gas pressures which exceed design limits early in Core III life 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/MTM). Furthermore, the high burnup rods will be limited to operation at lower linear powers 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 should removal of these rods become necessary.

(1) WCAP-3385-52, Saxton Plutonium Program, Mechanical and Thermal-Hydraulic Design of Partial Plutonium Core, December 1965.

2.4.2 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 (except for certain load-follow rods).
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 2.4-1 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 linear powers expected for fuel rods in the seven loose-lattice assemblies. The table also shows the predicted peak linear 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 life 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 linear powers represented and statistical variations in diametral gaps, densities, etc.

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 effecs 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. Since intentional operation to failure is planned for the loose-lattice fuel, thermal-hydraulic analysis of the flow blockage required to reduce the steady-state DNBR at 112% power to a lower limit of 1.3 has been performed. Figure 2.4-1 summarizes these results and indicates that diametral strains would have to be >35%. Therefore, loose-lattice cladding failures are unlikely to result in violation of the DNBR lower limit of 1.3. Cladding strains in failed rods are generally found to be small (<5%) compared to the 16-35% required to produce a steady-state DNBR of less than 1.3.[2]

[2] WCAP-3850-3, Post-Irradiation Examination of CVTR, Fuel Assemblies, August, 1968 (See Appendix D for appropriate pages).

In summary, the Core III irradiation testing is expected to provide valuable information in the validation of fuel element design procedures.

TABLE 2.4-1

SUMMARY OF POWER RATINGS AND BURNUPS EXPECTED
FOR CORE III

Group No.	Expected Peak Linear Power kw/ft	No. of Rods	Peak Initial Burnup, MWD/MTM	Peak Expected EOL Burnup, MWD/MTM
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	57	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
Total		249		
LF-1	17.6-15.8	80	0	19,000-21,000**
LF-2	15.8-14.1	24	0	16,800-19,000**
LF-3	14.1-12.3	12	0	14,700-16,800**
LF-4	12.3-8.8	4	0	10,500-14,700**
Total		120		

* These four rods are in the center removable subassembly.

** These values assume constant exposure at the peak linear power 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.

LL - Loose-Lattice

LF - Load Follow

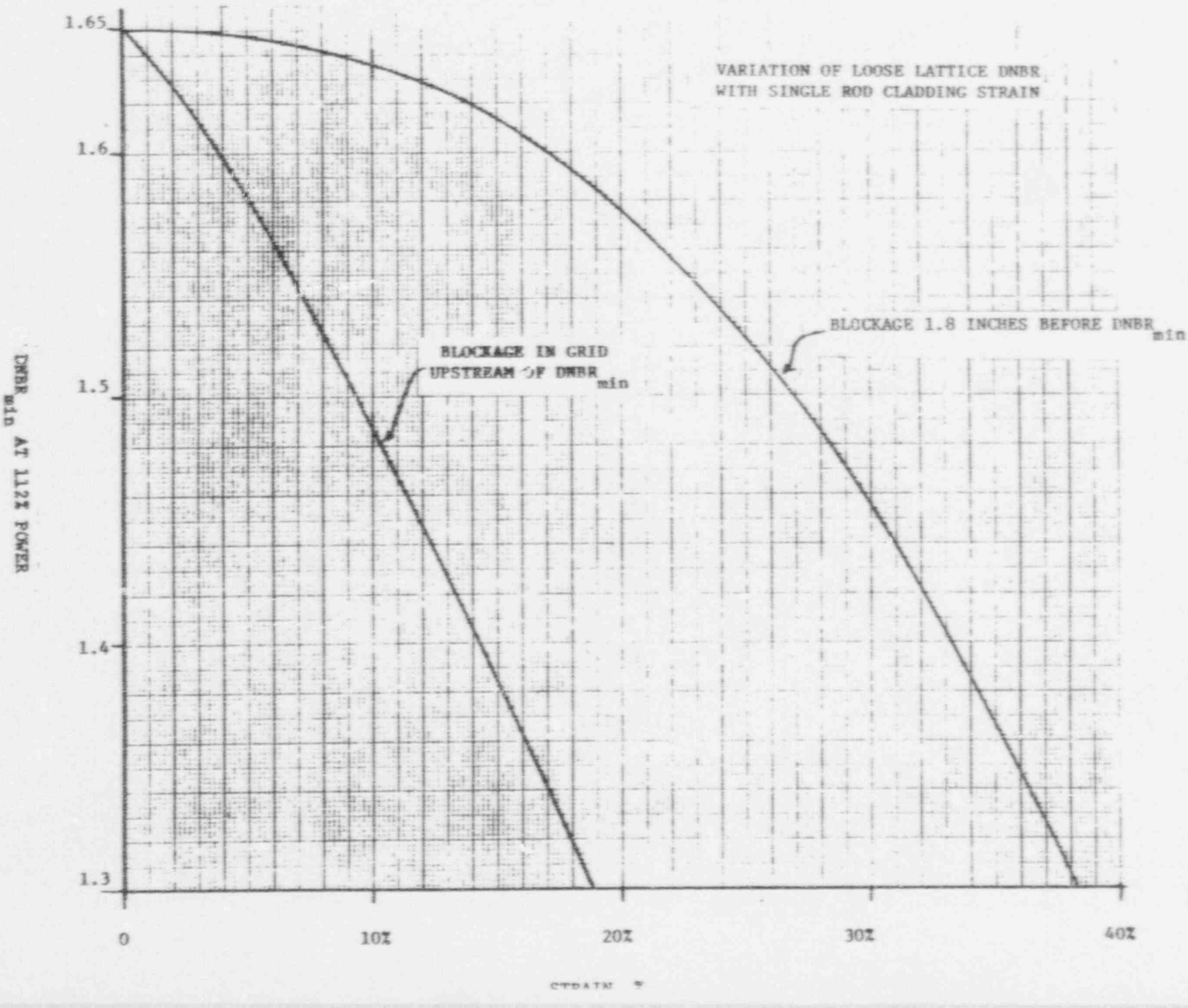

VARIATION OF LOOSE LATTICE DNBR
WITH SINGLE ROD CLADDING STRAIN
BLOCKAGE 1.8 INCHES BEFORE DNBR$_{min}$
BLOCKAGE IN GRID
UPSTREAM OF DNBR$_{min}$
DNBR$_{min}$ AT 112% POWER
1.65
1.6
1.5
1.4
1.3
0
10%
20%
30%
40%

FIGURE 2.4-1

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 pow r density and peak fuel pellet burnup. This relationship has been calculated for the loose-lattice 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 four representative curves showing the design-overpower limits for different initial peak pellet burnups. These curves in combination with the proposed power variations (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.

---

[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 12.5% of the peak pellet power density in Figure 3.1-1 is nuclear and engineering uncertainty. In addition the peak pellet is evaluated at the 112% overpower condition. 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	60 cycles/sec

* The setpoint for reference power level of 28 MWt is 511°F. For initial operation at 24.9 MWt the hot leg trip setpoint will be reduced to 508°F.

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 Section 3.2.

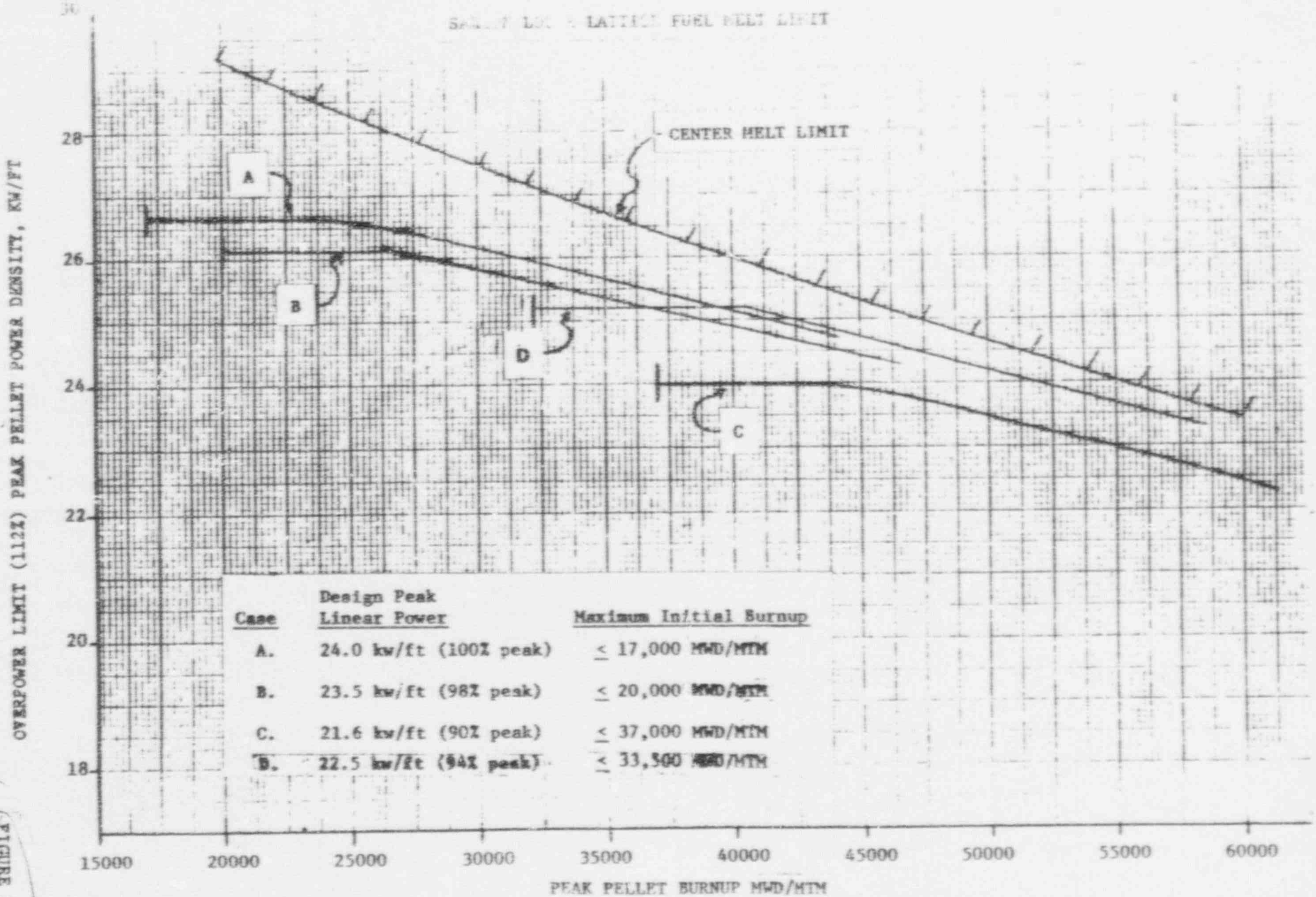
The rod withdrawal accident results are nearly identical to that presented for Core II 35 MWt operation. (See Figure VI-I Appendix E), only assumptions e and f were modified since the maximum overpower reactor trip is 112% for Core III versus 114% for Core II and that the limiting value of the moderator temperature coefficient is -0.5×10^{-4} $\Delta k/°F$ for Core III versus -2.0×10^{-4} $\Delta k/°F$ for Core II.

The reduction in the trip setpoint has the stronger effect, since the reactivity feedback is mostly dependent on the fuel temperature coefficient and only very slightly dependent of the moderator temperature coefficient.

As a net result, the nuclear flux peak is now slightly below 120% and the hot spot heat flux is below the 114% reported for Core II (See Figure VI-I in Appendix E). Thus, the minimum DNB ratio is higher than the 1.31 DNB ratio reported for Core II. This is a direct result of the higher steady state DNB ratio of Core III (1.75) versus Core II (1.52). At the minimum DNB point during the transient, there is still a 3.5% power margin to a DNB ratio of 1.3. Hence, the core is adequately protected against the rod withdrawal accident.

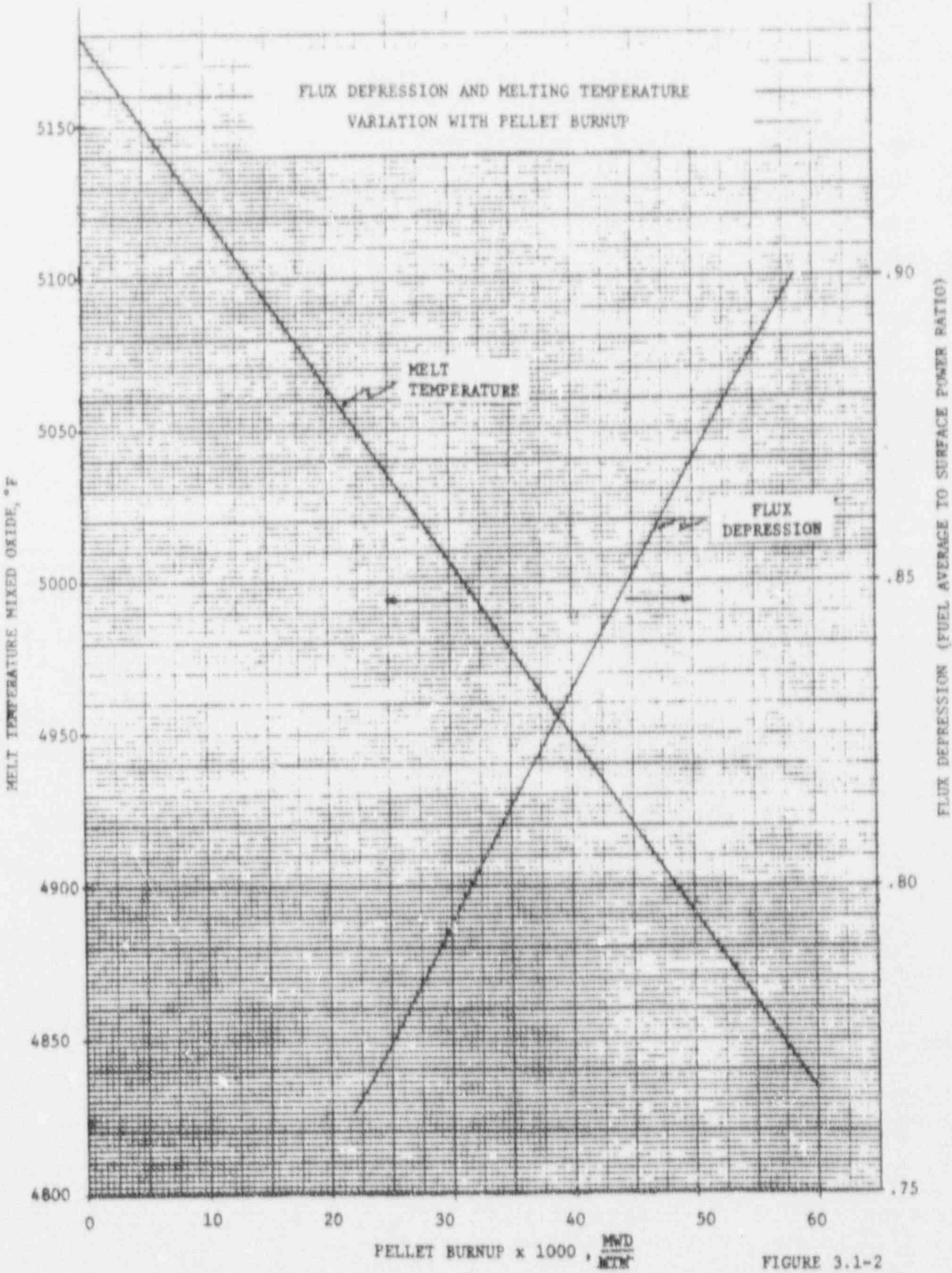
The loss of coolant accident is re-analyzed since the higher fuel temperature and heat flux in the peak power density region of the core during Core III operation would result in increased clad temperature transients.

The steam line break accident has been analyzed previously in Core II 35 MWt Safety Analysis Report, the appropriate pages of which are in Appendix E. Although the linear power density of the peak rods are considerably higher in Core III, the steady state DNB ratio (1.75) for Core III is approximately 15% higher than the Core II (35 MWt) steady state DNB ratio (1.52). As a result, the control and protection system will trip the reactor and terminate the first phase of the steam break accident with even a greater margin of DNB than the one reported for Core II (35 MWt). After the reactor trip is initiated, the principal 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.



LATTICE FUEL MELT LIMIT
CENTER MELT LIMIT
A
B
C
D
OVERPOWER LIMIT (112%) PEAK PELLET POWER DENSITY, KW/FT
PEAK PELLET BURNUP MWD/MTM
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/MTM
B. 23.5 kw/ft (98% peak) ≤ 20,000 MWD/MTM
C. 21.6 kw/ft (90% peak) ≤ 37,000 MWD/MTM
D. 22.5 kw/ft (94% peak) ≤ 33,500 MWD/MTM

FIGURE 3.1-



FLUX DEPRESSION AND MELTING TEMPERATURE
VARIATION WITH PELLET BURNUP
MELT
TEMPERATURE
FLUX
DEPRESSION
MELT TEMPERATURE MIXED OXIDE, °F
FLUX DEPRESSION (FUEL AVERAGE TO SURFACE POWER RATIO)
5150
5100
5050
5000
4950
4900
4850
4800
.90
.85
.80
.75
0
10
20
30
40
50
60
PELLET BURNUP x 1000 , MWD/MTM

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 load-follow 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: -0.5×10^{-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).

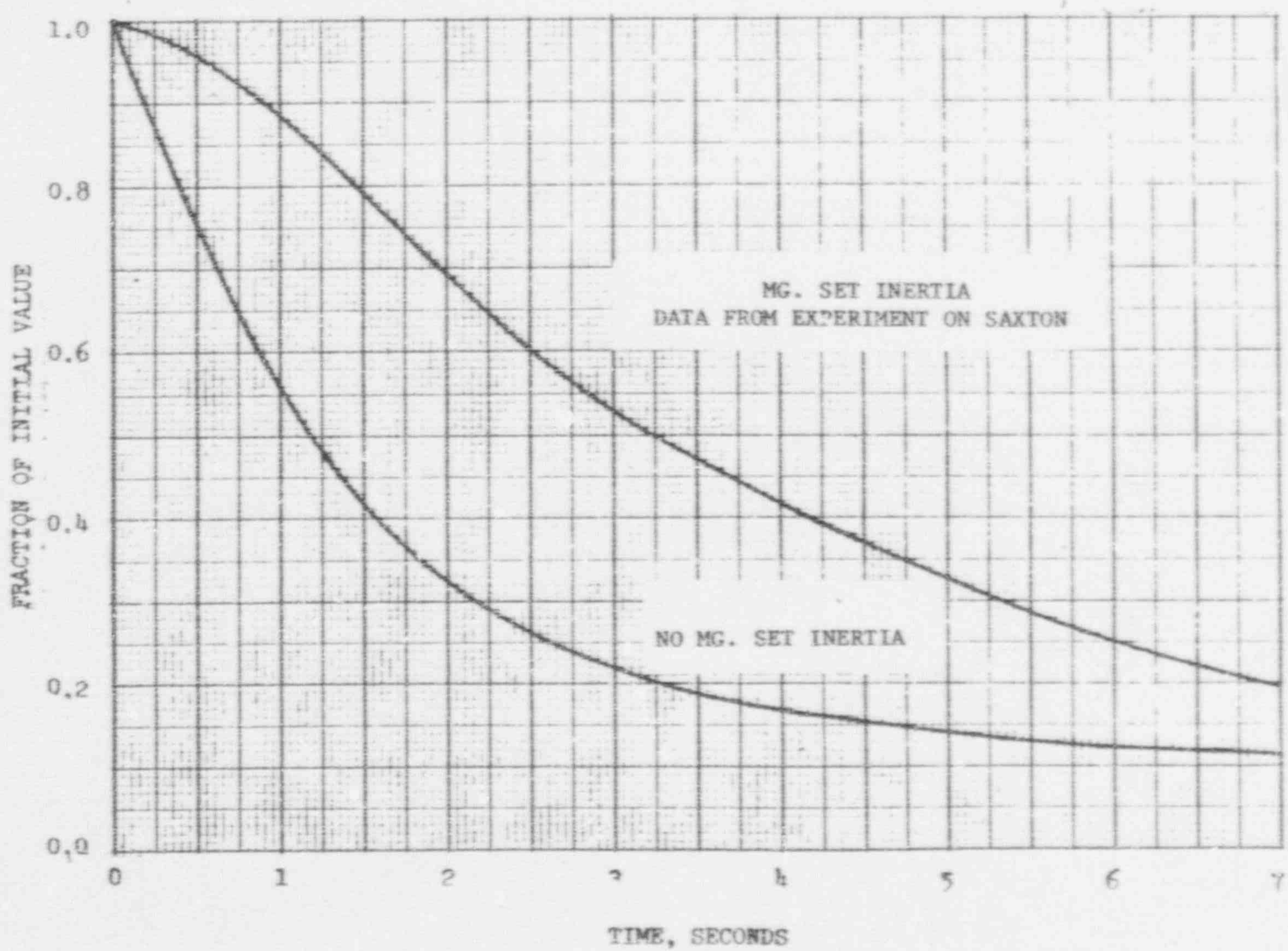
The flow coastdown curves with and without MG set inertia are shown in Figure 3.2-1 and are also given in the Core II Safety Analysis Report. The flow coastdown with inertia is also presented in Figure 3.2-1 and was obtained from recent experimental measurements at the Saxton Plant.

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.

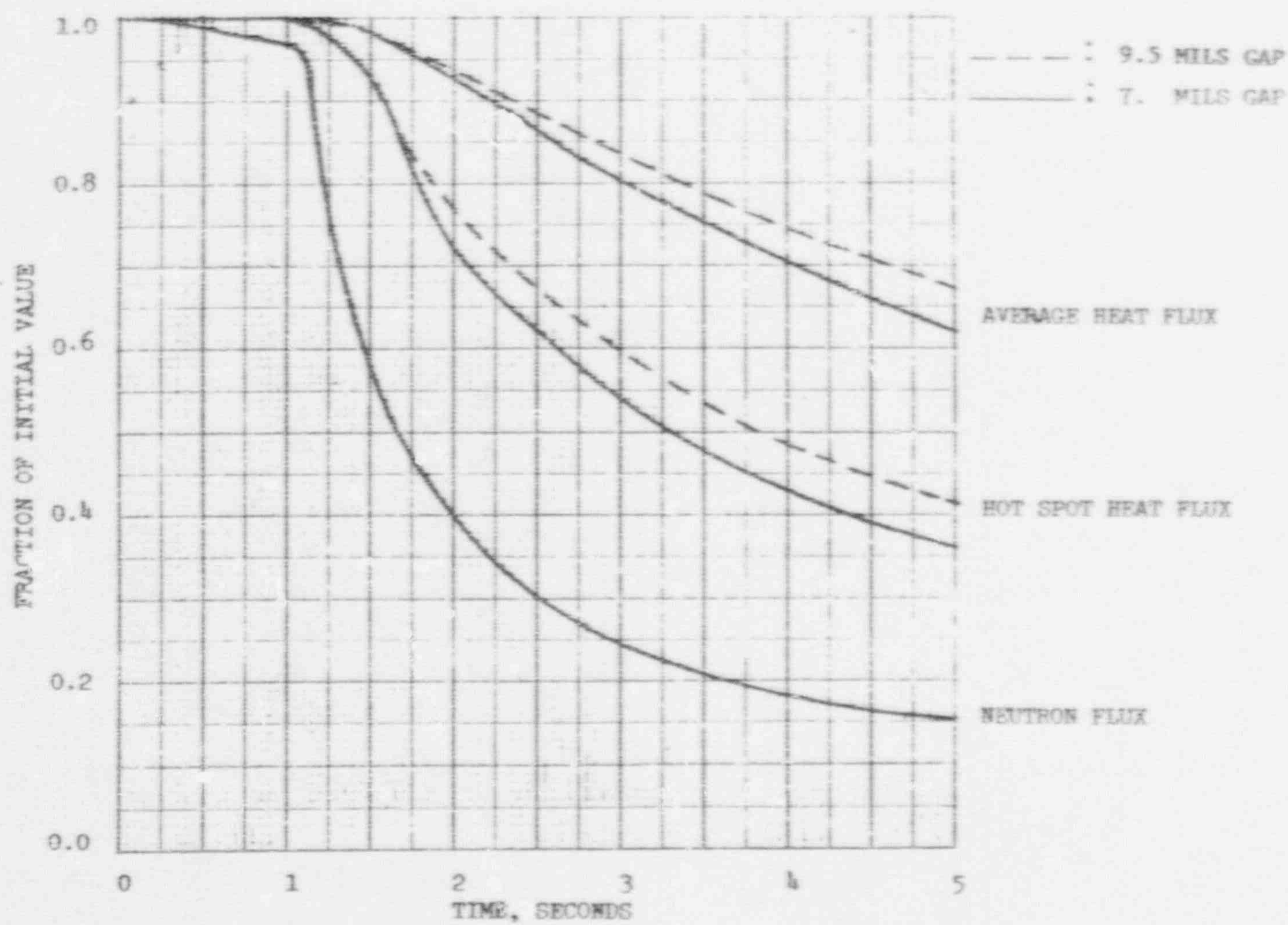
SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT



FLOW COASTDOWN CURVES

FIGURE 3.2-1

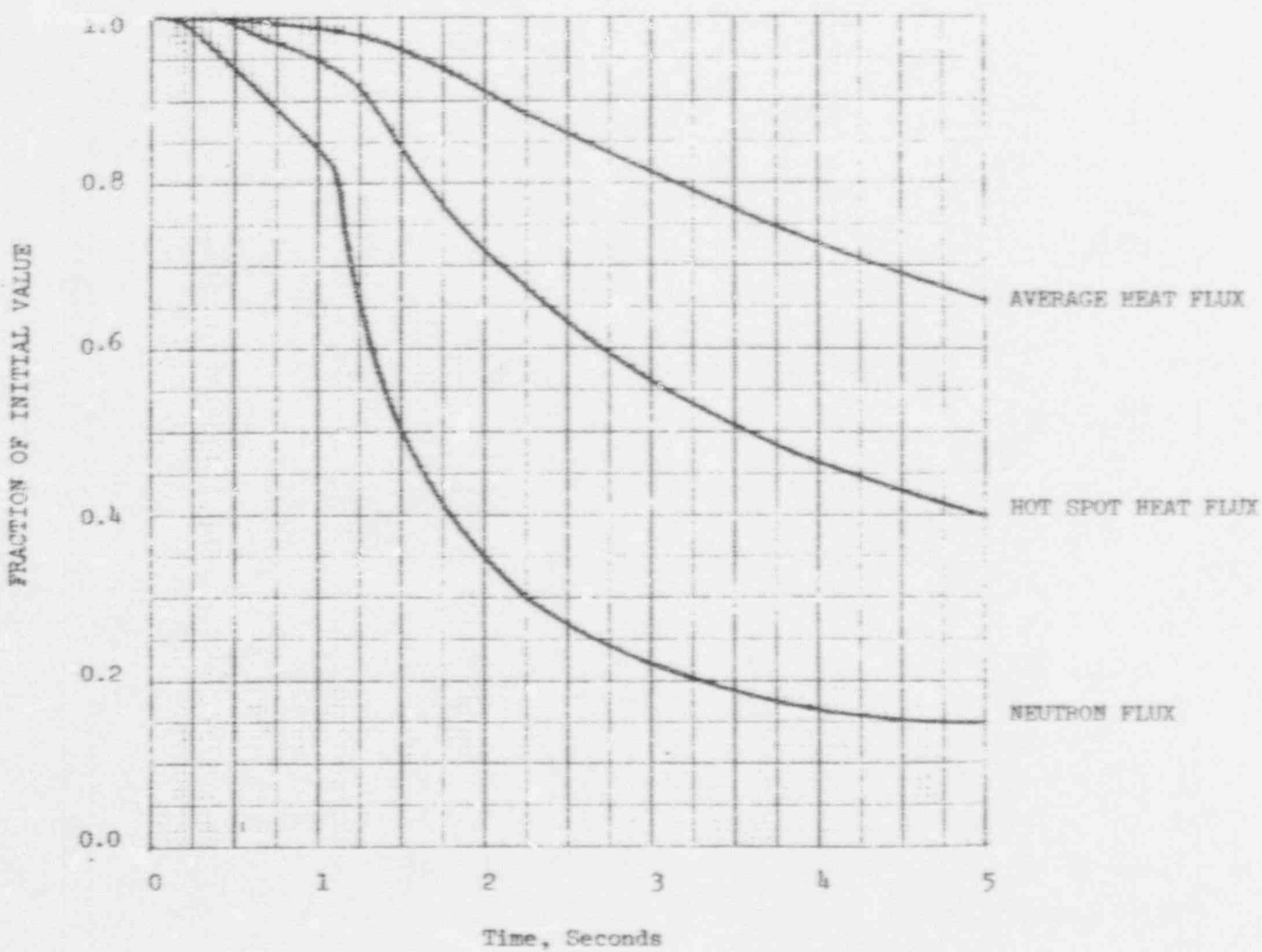
SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5} \Delta K/K/^{\circ}F$
MODERATOR REACTIVITY COEFFICIENT: $-.5 \times 10^{-4} \Delta K/K/^{\circ}F$
TRIP REACTIVITY: 3%



MG SET INERTIA

FIGURE 3.2-2

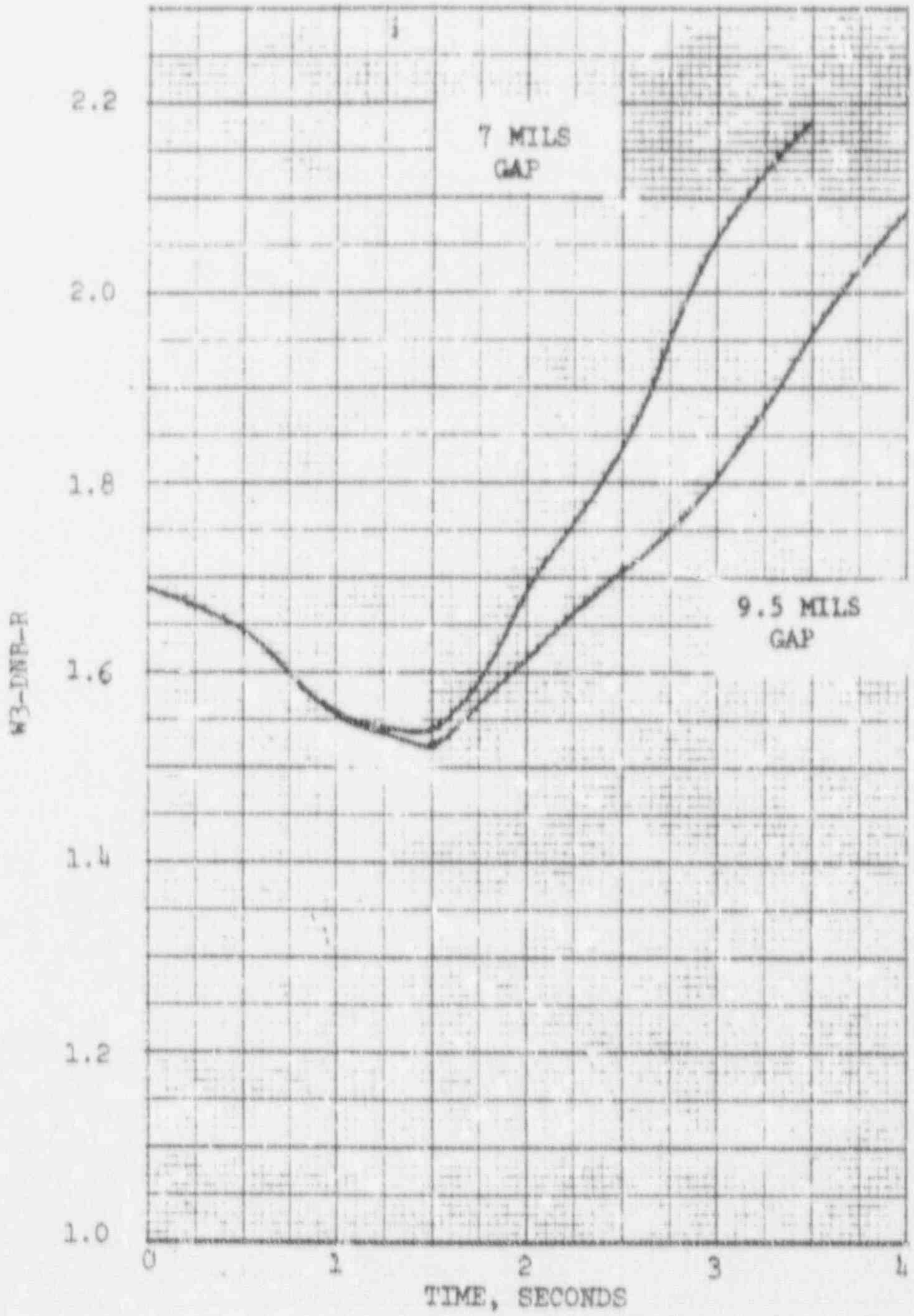
SAXTON 28 MWt OPERATION
LOSS OF FLOW ACCIDENT
TOTAL SCRAM DELAY: 1.1 SECONDS
DOPPLER REACTIVITY COEFFICIENT: $-1. \times 10^{-5}$ $\Delta K/K/^{\circ}F$
MODERATOR REACTIVITY COEFFICIENT: $-.5 \times 10^{-4}$ $\Delta K/K/^{\circ}F$
TRIP REACTIVITY: 3%
FUEL GAP: 9.5 MILS



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: $-.5 \times 10^{-4} \Delta K/K/°F$
TRIP REACTIVITY: 3%

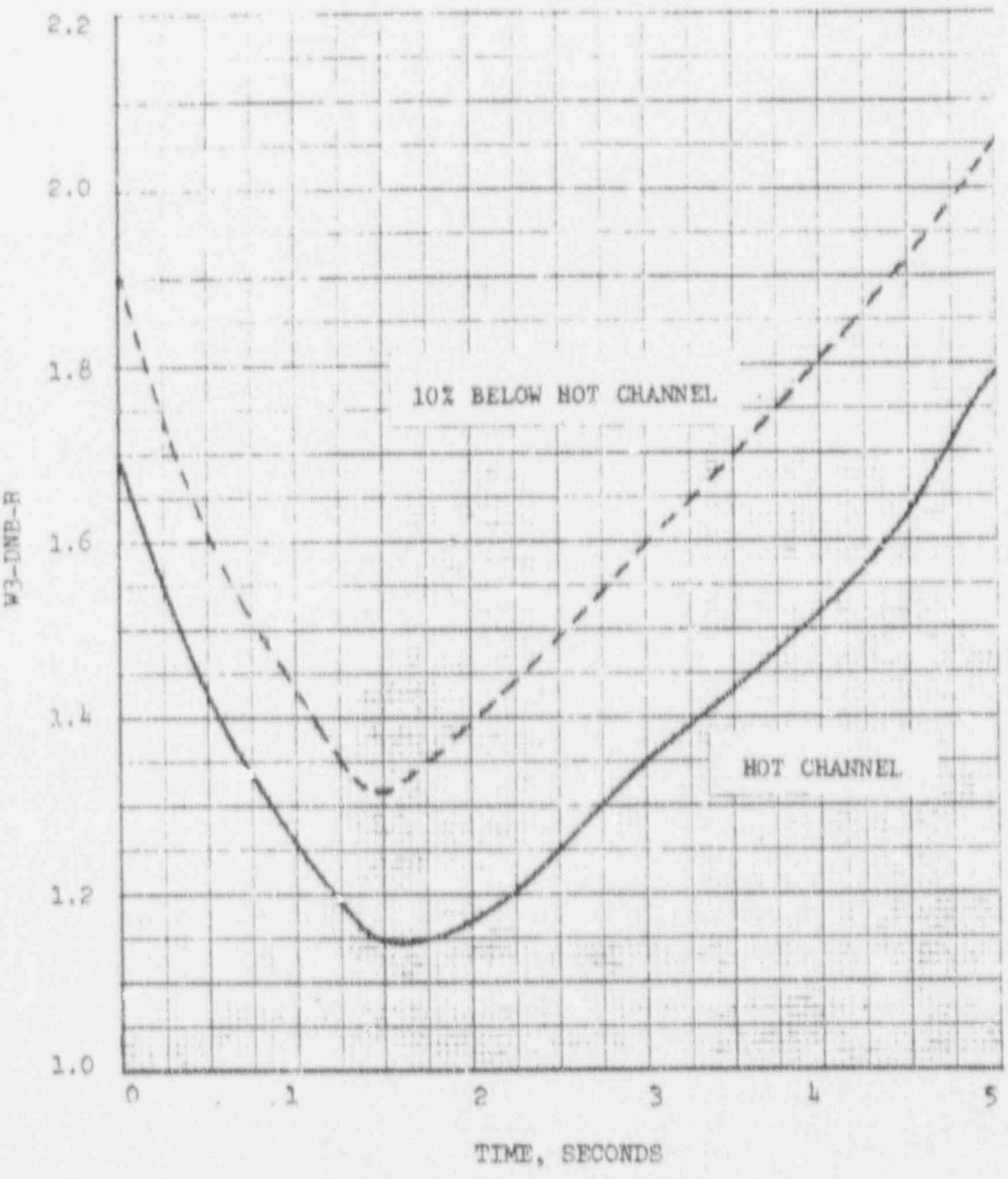


MG SET INERTIA

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: $-.5 \times 10^{-4} \Delta K/K/°F$
TRIP REACTIVITY: 3%
FUEL GAP: 9.5 MILS



NO MG.SET INERTIA

FIGURE 3.2-5

3.3 LOSS OF COOLANT

The loss-of-coolant accident has been re-evaluated for Saxton Core III. The assumptions and analytical techniques used are as follows:

1. The moderator temperature coefficient has been re-evaluated. A less negative coefficient of -0.5×10^{-4} $(\Delta k/k)/°F$ is now used in the analysis rather than the value of -2.0×10^{-4} $(\Delta k/k)/°F$ previously reported.

2. The heat transfer models in the LOCTRA-R2 core thermal analysis code are the same as those described in Section 4 of the report "Saxton Loss-of-Coolant Accident Prevention and Protection" except that stable film boiling immediately following the occurrence of DNB at 0.5 seconds, is no longer assumed. Instead, post DNB heat transfer coefficients during the transition and stable film boiling phases of blowdown are now calculated using an empirical correlation developed by Westinghouse, from steady state heat transfer data. It has been compared with experimentally determined transient data recently obtained as part of the Westinghouse Flashing Heat Transfer research and development program.[1]

   A comparison of the measured heat transfer coefficients obtained from the transient blowdown data and the coefficient calculated with the empirical correlation is presented in Figure 3.3-1. It is concluded that the heat transfer coefficient during the transition and stable film boiling phases of blowdown can be conservatively estimated by the recently developed Westinghouse empirical correlation.

3. The two load follow assemblies contain both Zircaloy and stainless steel clad fuel rods.[2]

4. Reactor power level is 28 MWt.

The specific cases analyzed and a summary of the results are as follows:

Break Size	Total % Core Clad Melt	Total % $Zr-H_2O$ Reaction
Double Ended Break - 1.28 ft^2	8.9	12.5
Intermediate - 0.173 ft^2	1.6	7.5
Surge Line - 0.0375 ft^2	0.0	1.6

Figures 3.3-2 through 3.3-4 show the clad temperature transients for the high power density Zircaloy clad rods located in the Loose-Lattice assemblies. Presented in these figures are the clad temperature transients for rods operating at various fraction of the design peak linear heat rate of 24 kw/ft. Figures 3.3-5 through 3.3-10 show the clad temperature transients for the Zircaloy and stainless steel clad rods located in the load follow assemblies. The design peak linear heat rate in these assemblies is 19.9 kw/ft.

Since the stainless steel clad rods located on the core periphery are operating at approximately 33.1% of the peak linear heat rate of 24 kw/ft, the peak clad temperatures exhibited by these rods would be much lower than those presented in Figures 8 through 10. The clad melt is limited to 8.9% for the double-ended break with no melt occurring for the surge line break.

REFERENCES

1. Farman, R. F., Cermak, J. O., "Post DNB Heat Transfer During Blowdown", WCAP-9005, October, 1968. Westinghouse Proprietary Report.

2. Melehan, J. B., Addendum to Saxton Core III License Application, WCAP-7219, July 16, 1968. Westinghouse Confidential Report.

COMPARISON OF CALCULATED AND MEASURED HEAT TRANSFER COEFFICIENTS

Figure 3.3-1

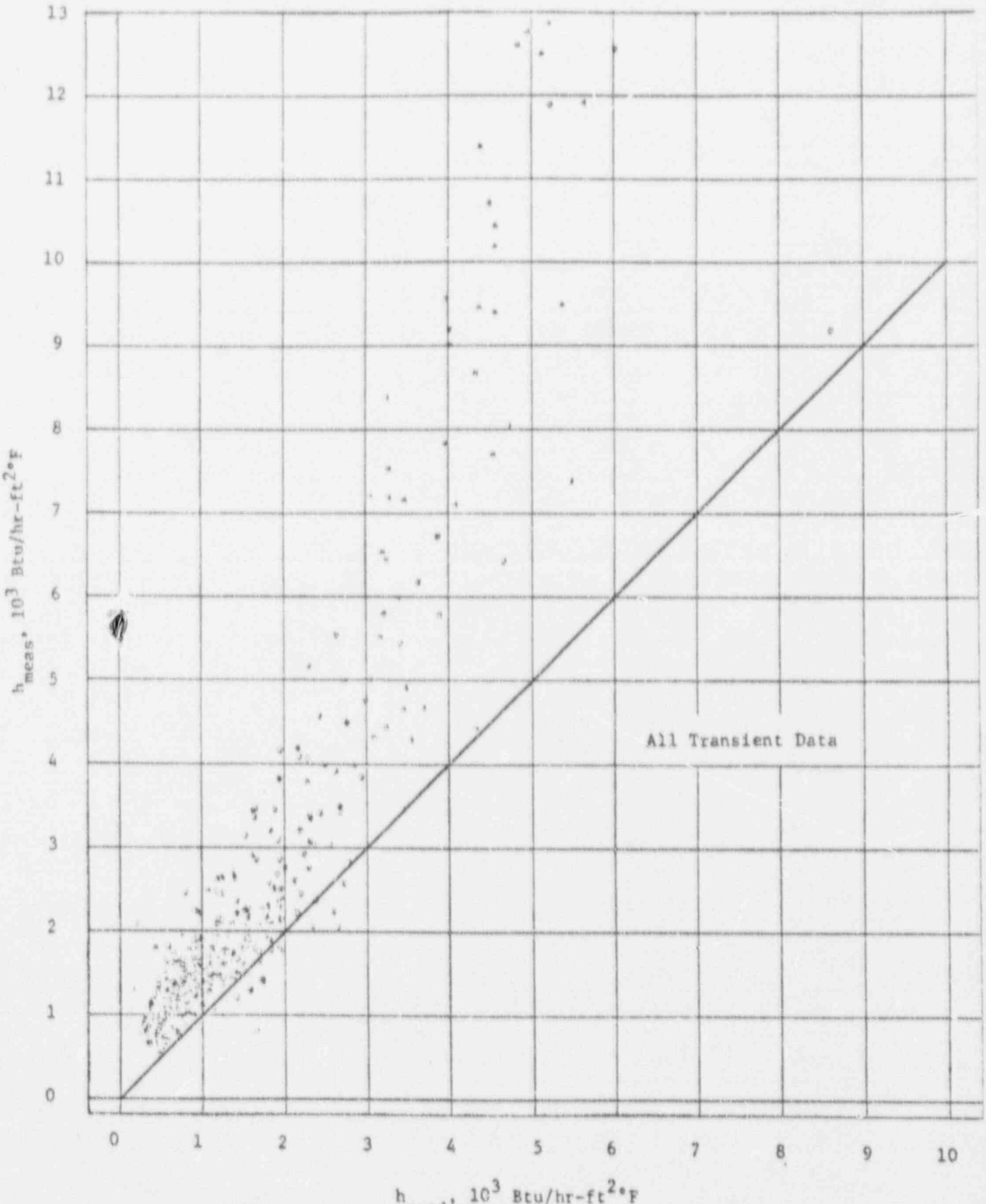


FIGURE 3.3-2

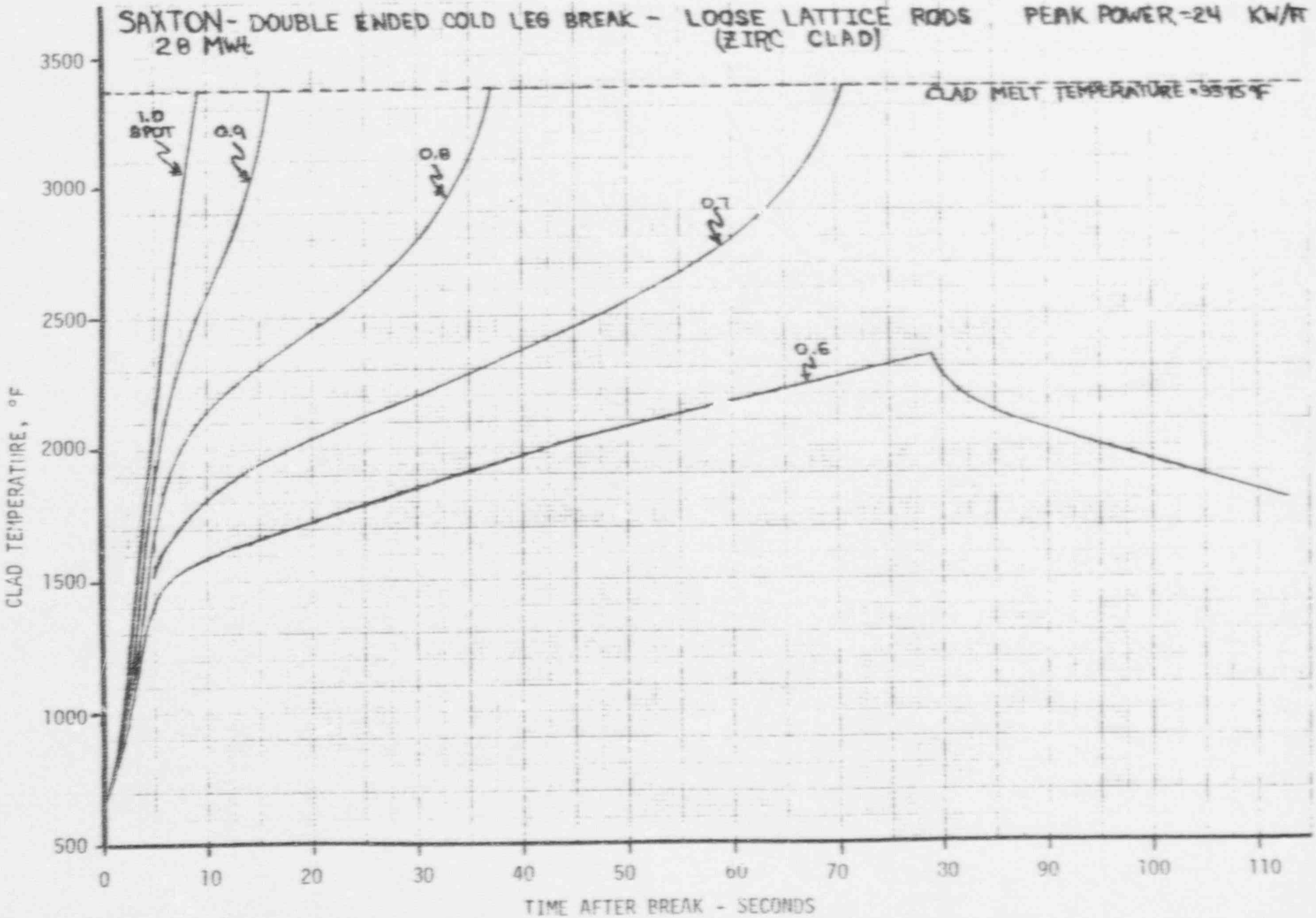
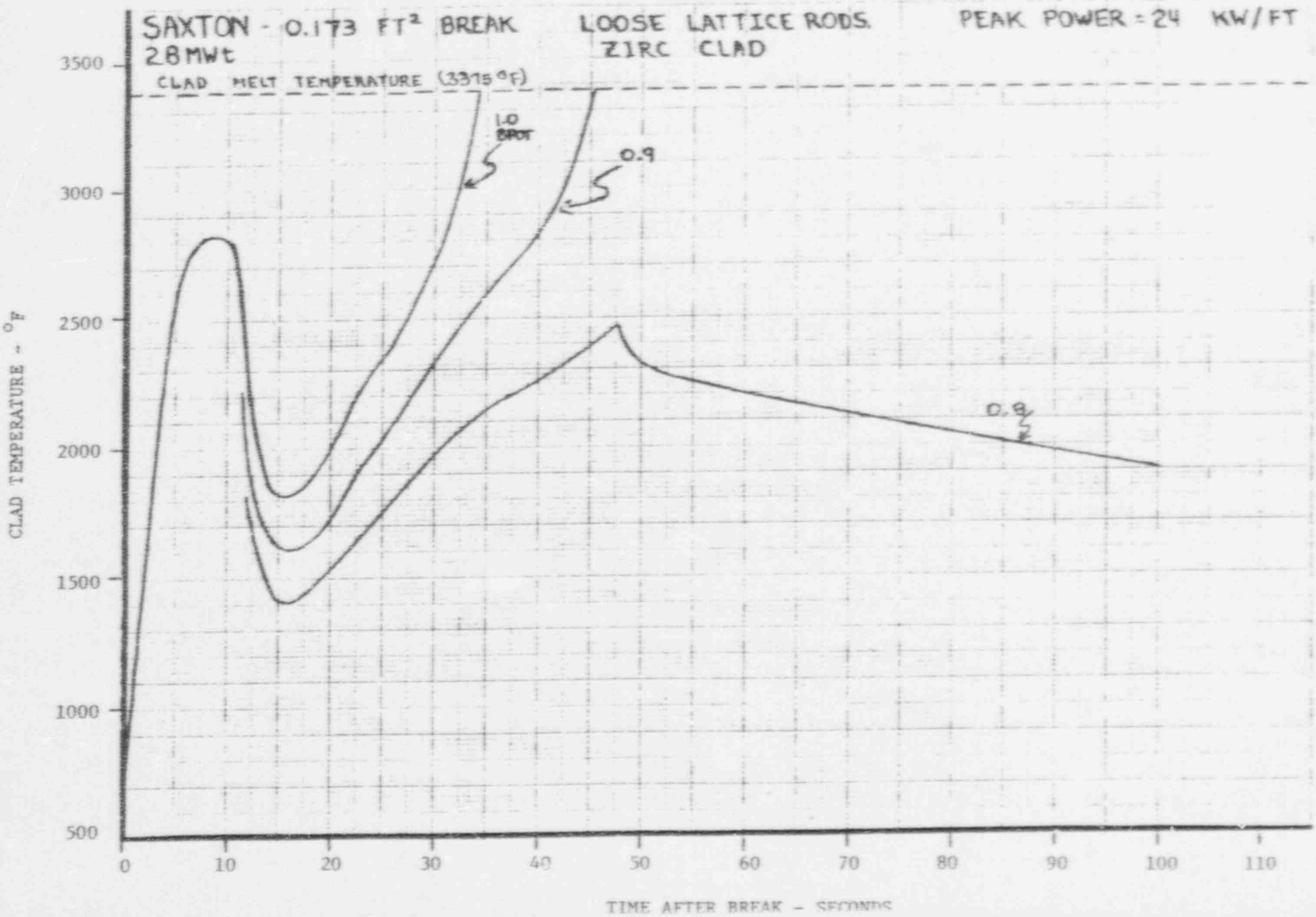
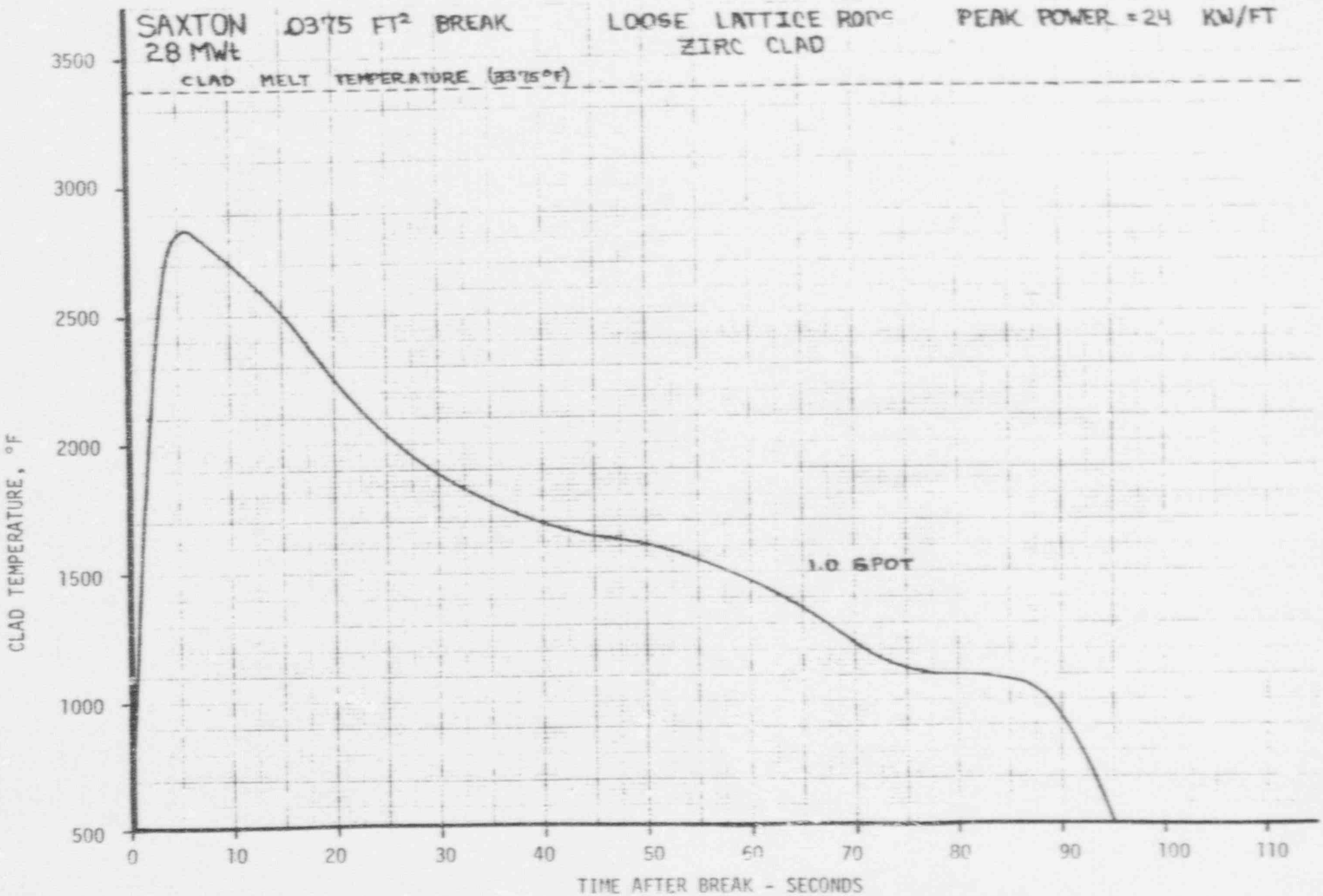

SAXTON- DOUBLE ENDED COLD LEG BREAK - LOOSE LATTICE RODS
(ZIRC CLAD)
PEAK POWER=24 KW/FT
28 MWt
CLAD MELT TEMPERATURE=3375°F
1.0 SPOT
0.9
0.8
0.7
0.6
CLAD TEMPERATURE, °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-3



SAXTON - 0.173 FT² BREAK
28 MWt
LOOSE LATTICE RODS
ZIRC CLAD
PEAK POWER = 24 KW/FT
CLAD MELT TEMPERATURE (3375 °F)
1.0 SPOT
0.9
0.8
CLAD TEMPERATURE - °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-4



SAXTON .0375 FT² BREAK
28 MWt
LOOSE LATTICE RODS
ZIRC CLAD
PEAK POWER = 24 KW/FT
CLAD MELT TEMPERATURE (3375°F)
1.0 SPOT
CLAD TEMPERATURE, °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-5

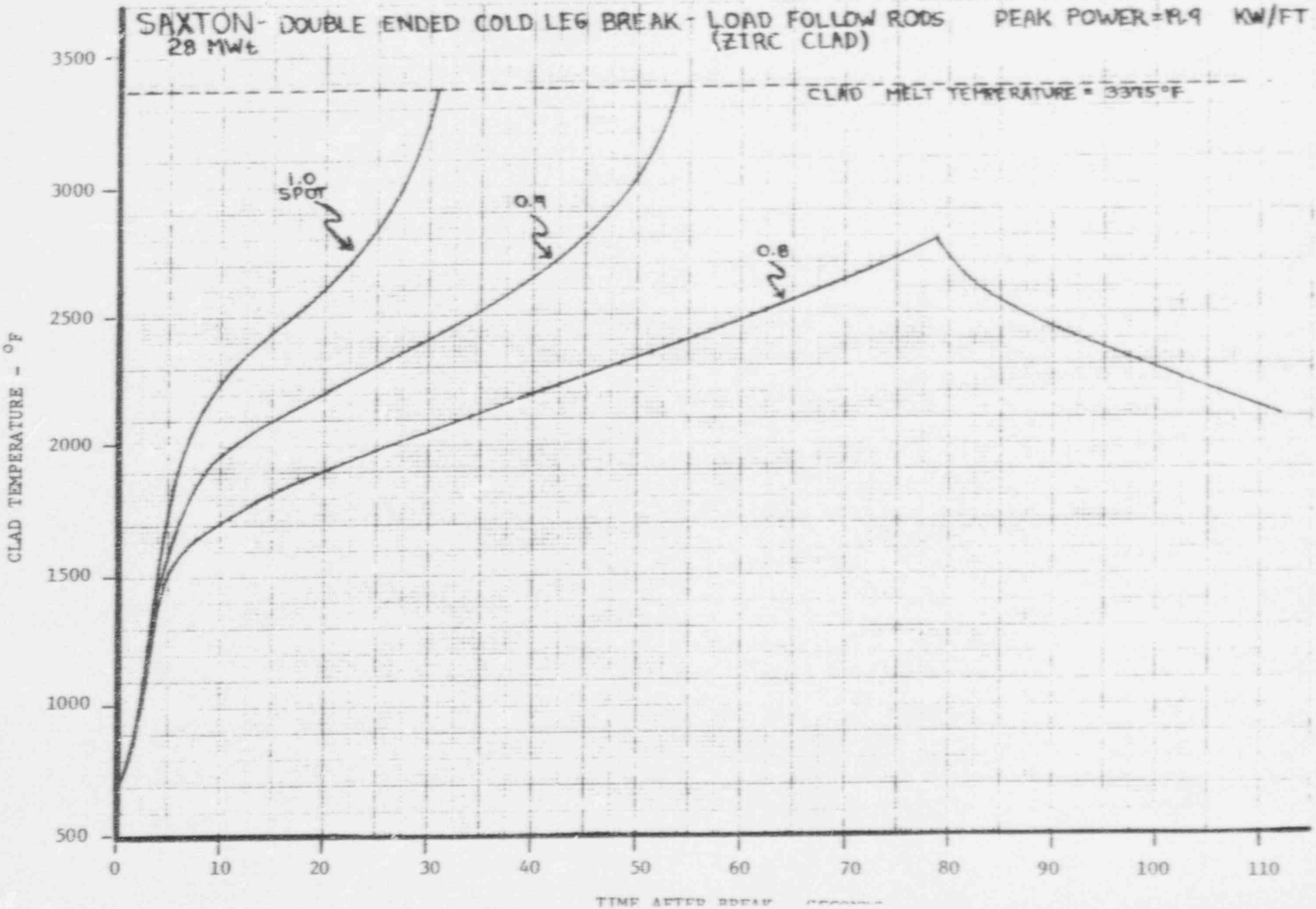
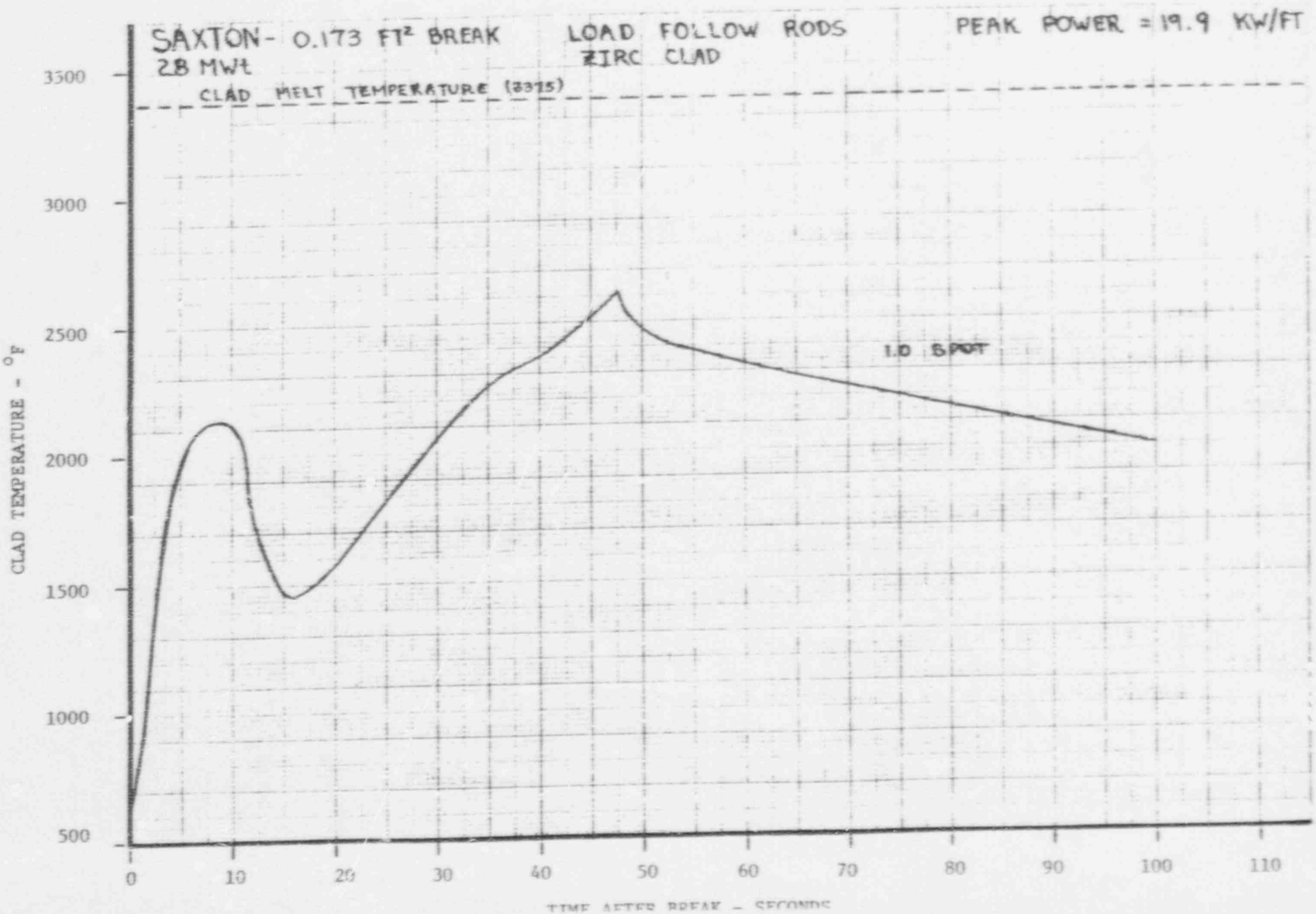

SAXTON- DOUBLE ENDED COLD LEG BREAK - LOAD FOLLOW RODS PEAK POWER=19.9 KW/FT
28 MWt
(ZIRC CLAD)
CLAD MELT TEMPERATURE = 3375°F
1.0 SPOT
0.9
0.8
CLAD TEMPERATURE - °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110

FIGURE 3.3-6



SAXTON- 0.173 FT² BREAK
28 MWt
LOAD FOLLOW RODS
ZIRC CLAD
PEAK POWER = 19.9 KW/FT
CLAD MELT TEMPERATURE (3375)
1.0 SPOT
CLAD TEMPERATURE - °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-8

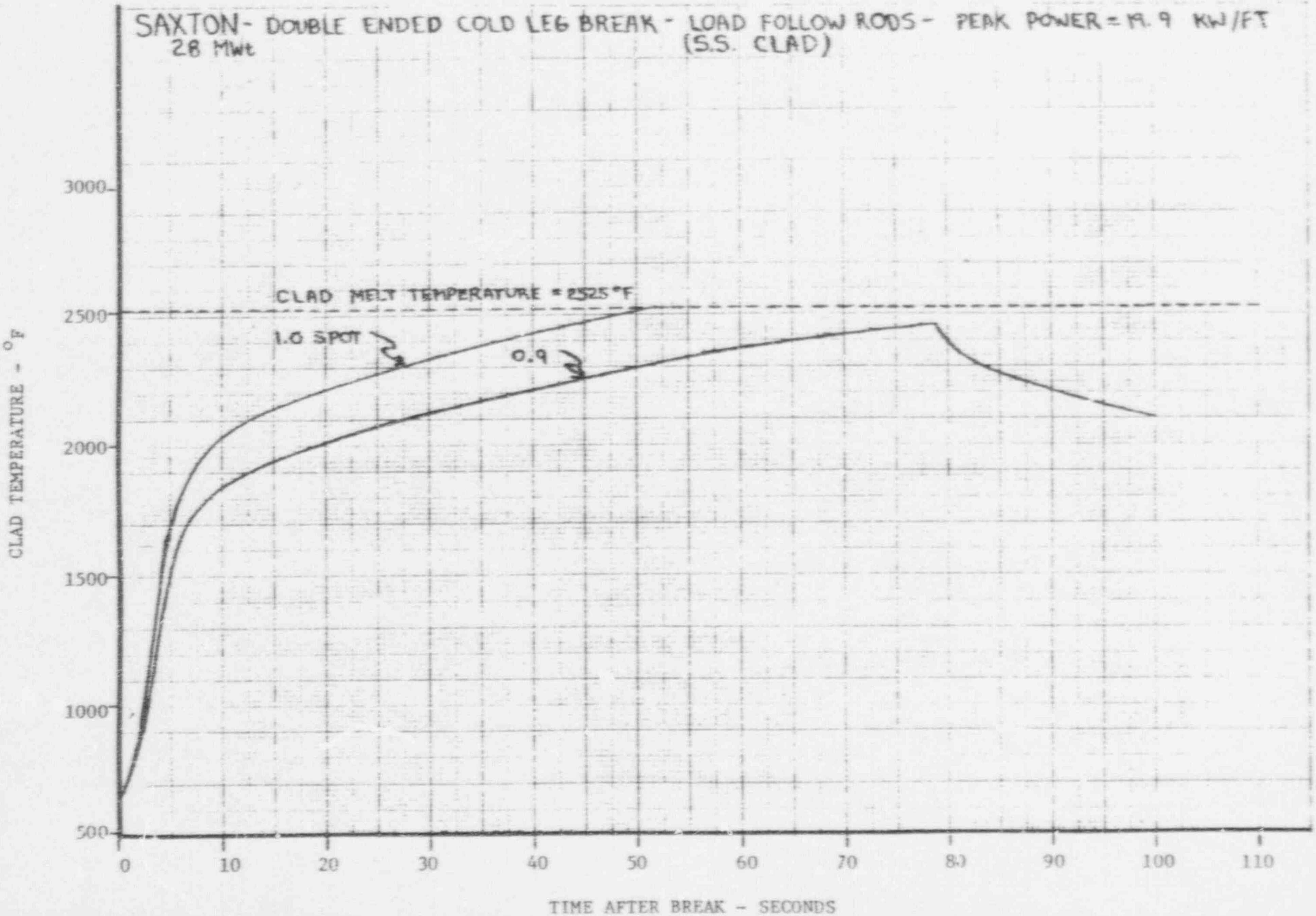

SAXTON- DOUBLE ENDED COLD LEG BREAK - LOAD FOLLOW RODS - PEAK POWER = 19.9 KW/FT
28 MWt
(S.S. CLAD)
CLAD MELT TEMPERATURE = 2525°F
1.0 SPOT
0.9
CLAD TEMPERATURE - °F
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-9

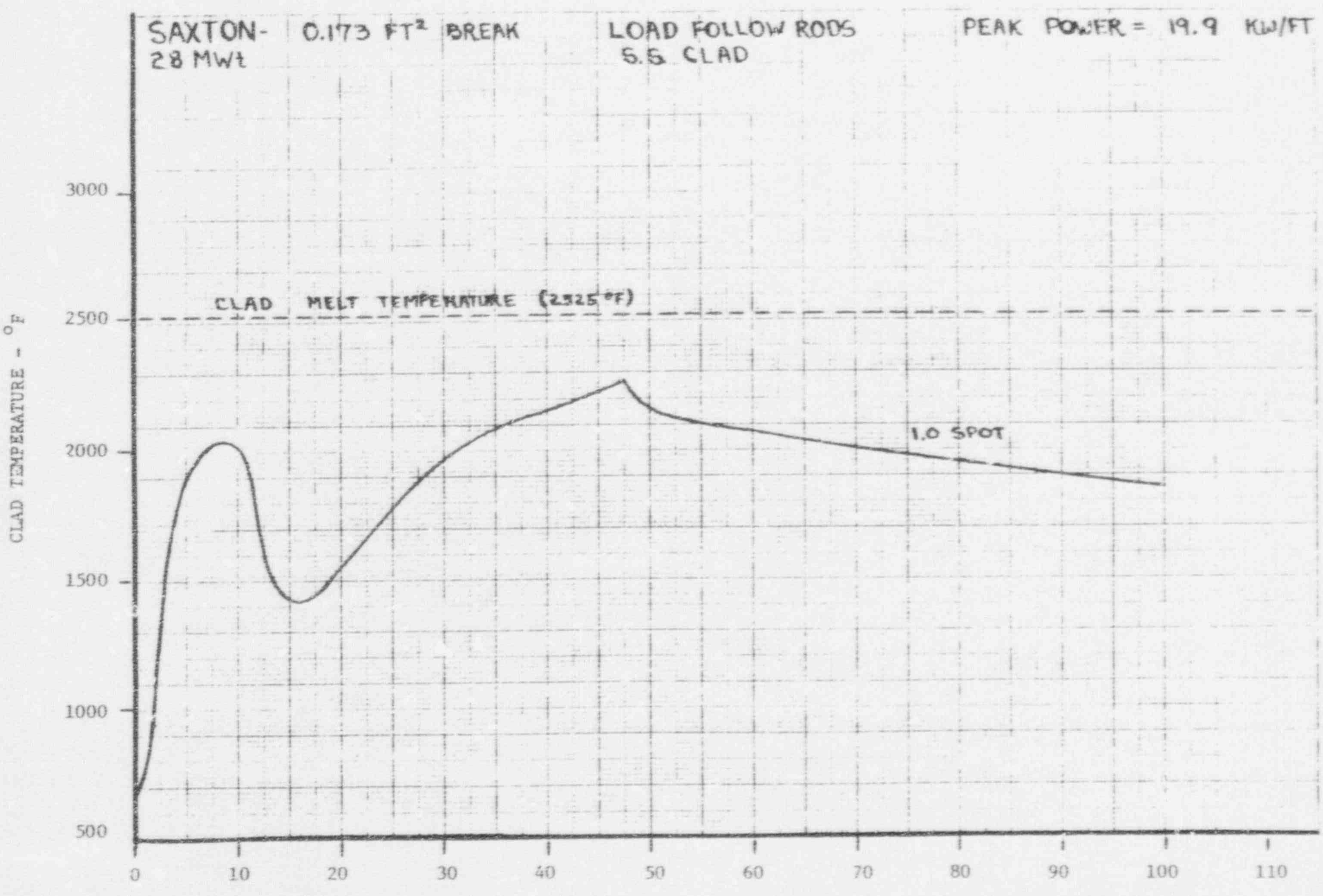

SAXTON-
28 MWt
0.173 FT² BREAK
LOAD FOLLOW RODS
S.S. CLAD
PEAK POWER = 19.9 KW/FT
CLAD MELT TEMPERATURE (2525°F)
1.0 SPOT
CLAD TEMPERATURE - °F
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110

FIGURE 3.3-8

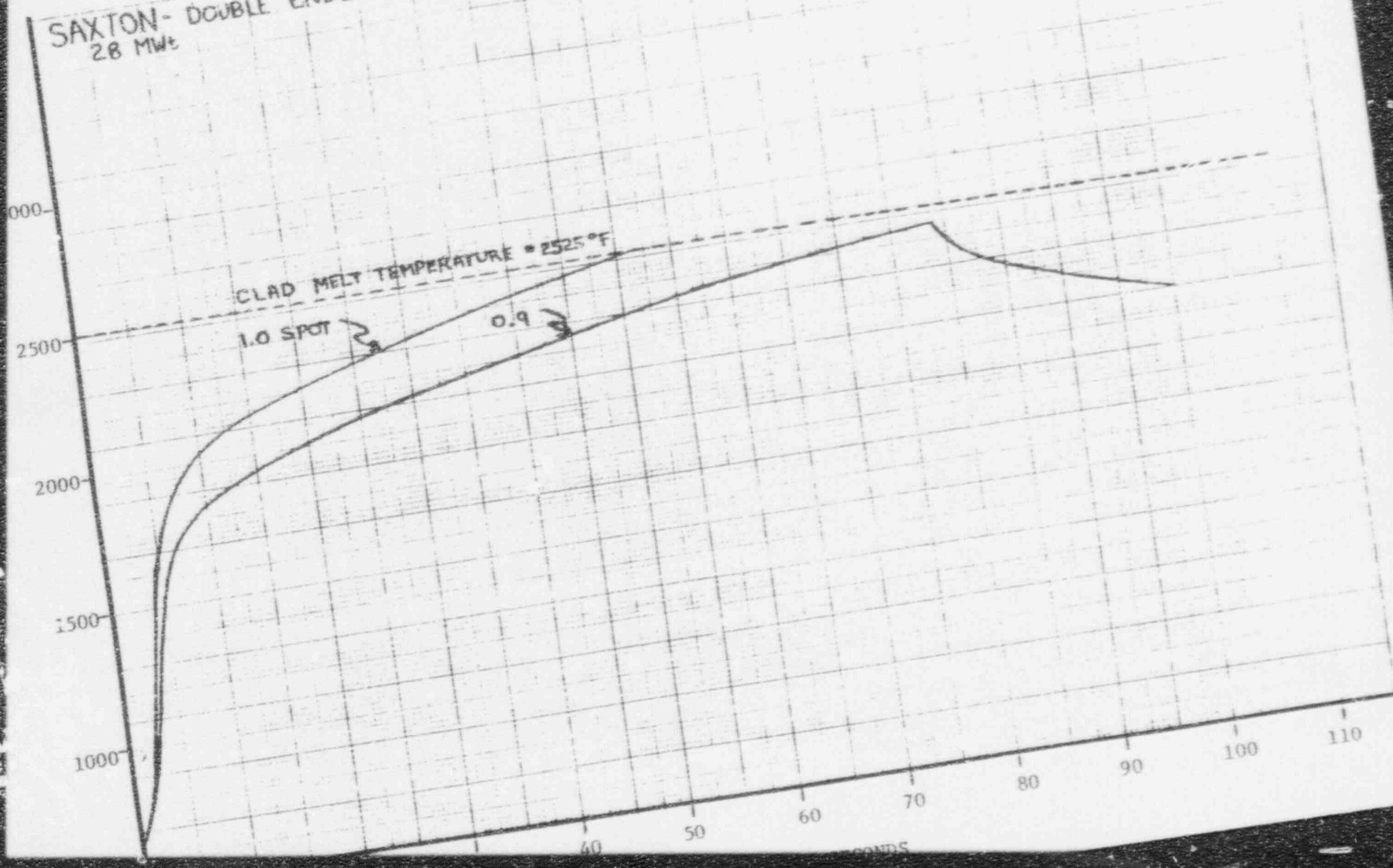
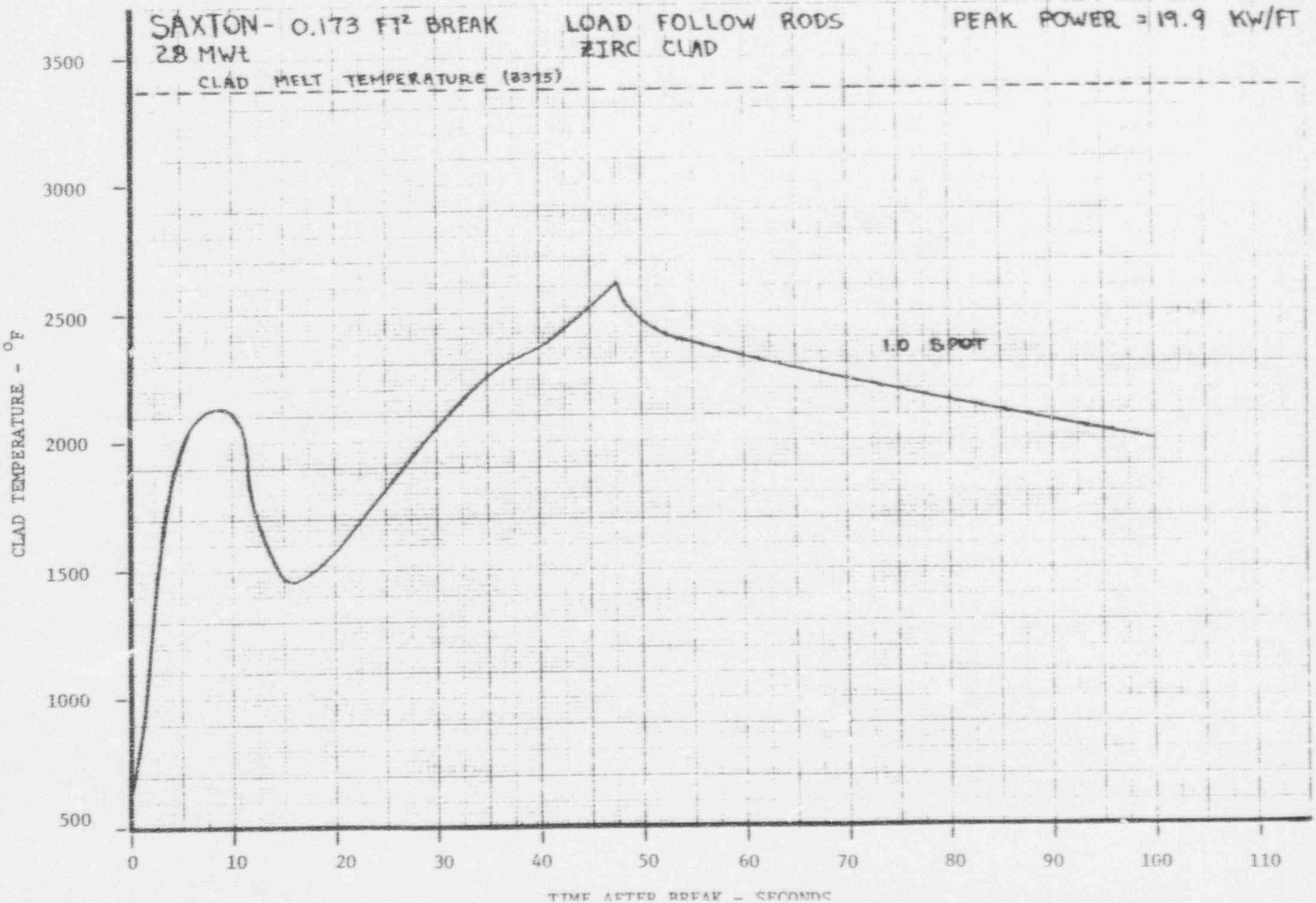

SAXTON - DOUBLE ENDED COLD LEG BREAK LOAD FOLLOW RODS - PEAK POWER = 19.9 KW/FT
(S.S. CLAD)
28 MWt
CLAD MELT TEMPERATURE = 2525°F
1.0 SPOT
0.9
1000
1500
2000
2500
40
50
60
70
80
90
100
110

FIGURE 3.3-6



SAXTON- 0.173 FT² BREAK
28 MWt
LOAD FOLLOW RODS
ZIRC CLAD
PEAK POWER = 19.9 KW/FT
CLAD MELT TEMPERATURE (3375)
1.0 SPOT
CLAD TEMPERATURE - °F
3500
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-8

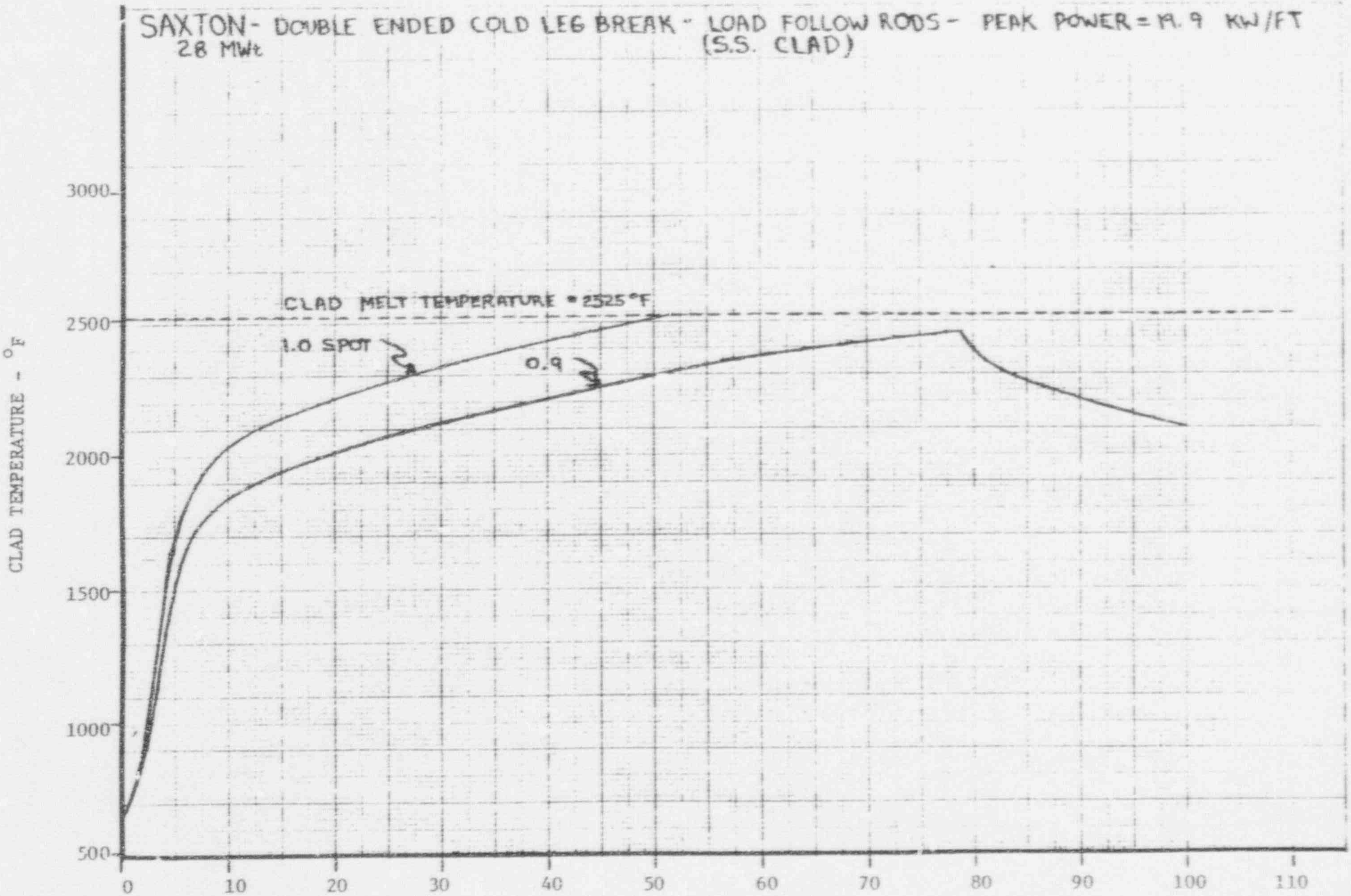
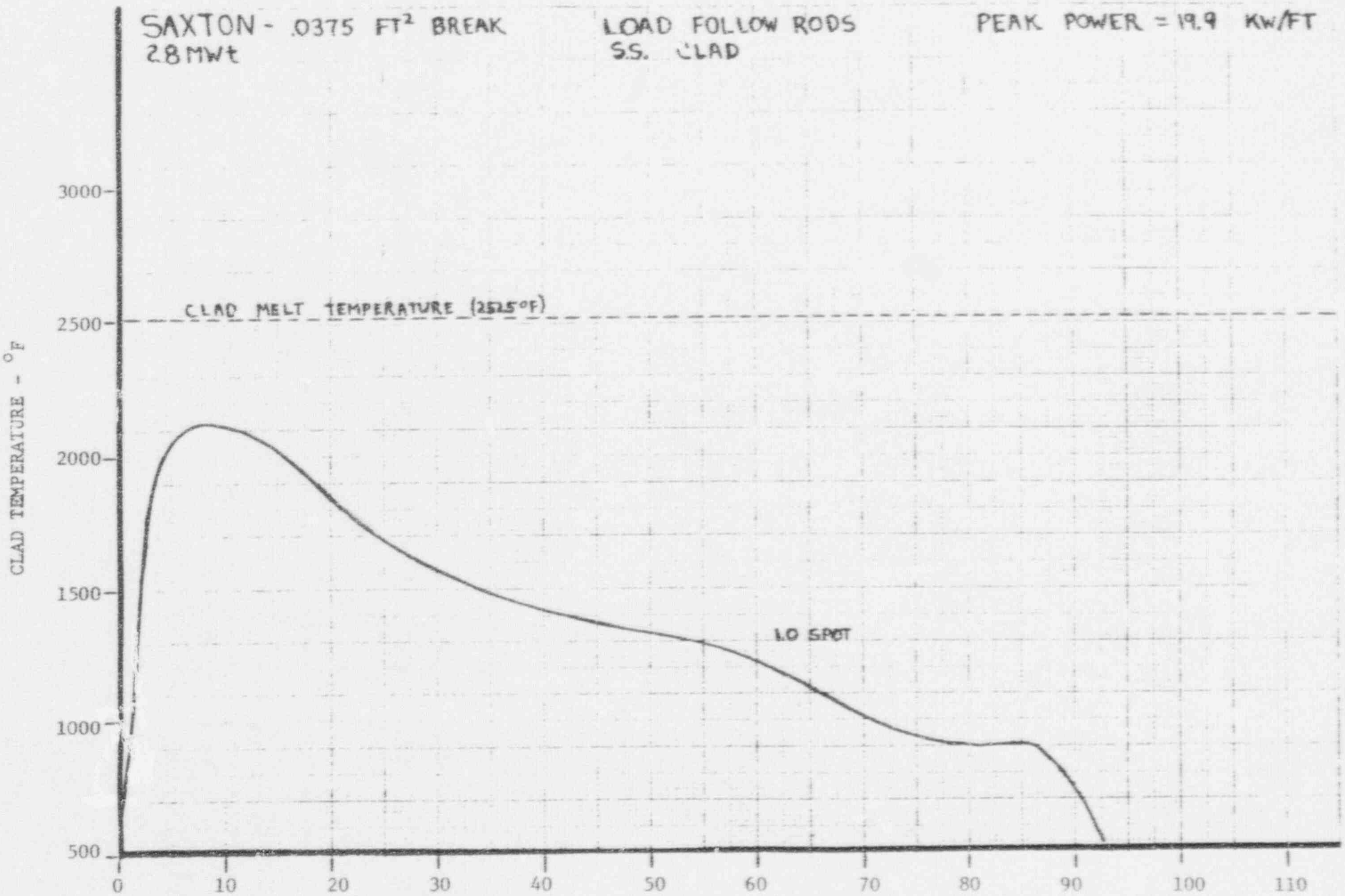

SAXTON- DOUBLE ENDED COLD LEG BREAK - LOAD FOLLOW RODS - PEAK POWER = 19.9 KW/FT
28 MWt
(S.S. CLAD)
CLAD MELT TEMPERATURE = 2525°F
1.0 SPOT
0.9
CLAD TEMPERATURE - °F
3000
2500
2000
1500
1000
500
0
10
20
30
40
50
60
70
80
90
100
110
TIME AFTER BREAK - SECONDS

FIGURE 3.3-10



TIME AFTER BREAK - SECONDS

APPENDIX A

FAILED FUEL MONITOR SYSTEM

APPENDIX A

Failed Fuel Monitor System

Introduction

A failed fuel monitor is installed as an experimental system at the Saxton Reactor. The general layout, checkout and calibration procedure and anticipated performance are described below.

Fuel element failure is indicated by the increase of gamma activity resulting from fission products in the reactor coolant. However, it should be noted that there are other sources of gamma radiation such as:

a. Background

b. N-16 activity

c. Other radioactive isotopes such as corrosion products

Description of System

The system utilizes the pressure drop across the steam generator to circulate reactor coolant in a bypass loop which consists of about 210 feet of stainless steel tubing, a heat exchanger, a radiation monitor, a remote operated flow control valve and a remote reading flow meter (Figure 1).

The radiation monitor consists of a coil of stainless steel tubing which surrounds two GM detectors. To reduce the background radiation the coil is shielded with 4" of lead. The detectors have an operating range from 0.01 mr/hr to about 1 r/hr.

The heat exchanger maintains reactor coolant below 140°F to prevent damage to the GM detectors. A thermocouple is provided for monitoring the temperature at the detectors.

The remote operated flow control valve and remote reading flow meter permit the reactor coolant flow to be adjusted to provide delay time of about 40 seconds. This delay time provides for sufficient decay of the N-16 activity in the coolant to allow proper operation of the system while the reactor is at full power.

The readout system consists of two independent ratemeter channels and a recorder which can record either of the channels. The two channels and their associated electronics are shown in Figure 2.

Checkout and Calibration

The two channels have received a functional checkout and their operating range, counts/minute vs. mr/hr using a Co-60 source, has been determined. The background level in counts/minute for zero power operation has been obtained and a flow rate of 0.42 gpm in the bypass loop has been established as being optimum for the system.

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ystem Characteristics

esign Pressure	Primary	2500	psia
	Secondary	150	psia

aterials — Primary: ½", Type 316 Stainless Steel Tubing

low, primary, maximum — .73 gpm

low Control Valve — remote, air operated fail open

Valve — Hammel-Dahl
Valve Positioner — Masoneilan

eat Exchanger - Sentry Equipment Corp.

Design psi	-	Shell 150 psi,	tube 2485 psi
Design temp	-	" 350°F,	" 680°F
Test psi	-	" 225 psi,	" 4880 psi
Test temp	-	" 70°F,	" 70°F

Tubing: 3/8" O.D. x 0.035 wall

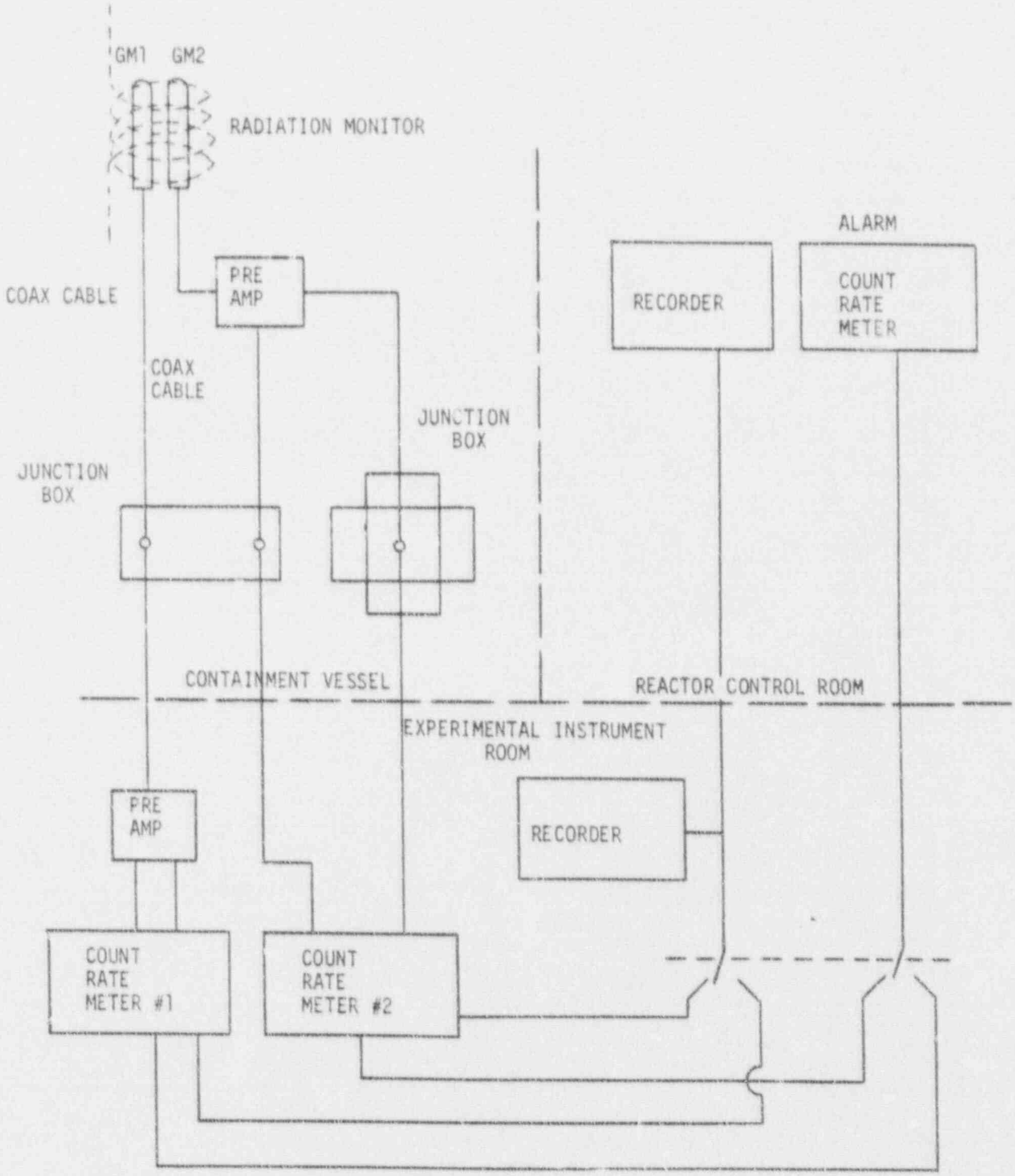
oils:	Radiation Monitor	Experimental
O.D.	3/8 inches	3/8 inches
wall	0.065 "	0.035 "
length	9.5 "	13.5 "
turns	24	34

/P Cell — Foxboro Integral Orifice

Orifice diameter — 0.159"
0-42 gph

.M. Tube - Phillips Model 18509

Operating voltage — 500 to 650 volts
Range — 1 mr/hr to 30 R/hr



FAILED FUEL MONITOR SYSTEM - ELECTRICAL

FIGURE A-2

APPENDIX B

SUMMARY REPORT ON BUCKLING OF SAXTON CORE II

FUEL ASSEMBLIES AND PREVENTION OF BUCKLING IN CORE III

SUMMARY

Buckling of the cans on some Saxton Core II assemblies was observed after Core II operation. The extent of the buckling of the Core II cans and the design modifications used to prevent buckling in Core III are herein reported.

The buckling of the central plutonium assemblies was caused by frictional forces between the grids and fuel rods arising from differential thermal expansion between the stainless steel assembly can and Zircaloy fuel cladding.

The buckling of the peripheral uranium dioxide fueled assemblies was caused by thermal gradients across the assembly and was limited to those assemblies with the Core II type grid design.

The major modifications to the loose lattice assemblies to prevent buckling during Core III operation consist of: (a) reduction of the grid friction loads through resetting of grid springs; and, (b) stiffening of the can structure through the use of full length clips between can halves and replacement of six Zircaloy water tubes by stainless steel water tubes with angle braces welded between the tubes and cans.

To prevent buckling in the load follow assemblies during Core III operation, the assemblies have been modified by reducing the grid friction loads through resetting of grid springs and stiffening of the can structure through: (a) replacement of fuel rods in two corner locations by square stainless steel bars with angle braces welded between the bars and cans; and, (b) angles welded to the inside of the can between fuel rods on the long sides of the can.

None of the buckled assemblies will be reused in Core III.

1.0 SUMMARY OF BUCKLING

1.1 Core II Plutonium Assemblies

Buckling was observed in eight of the nine central plutonium assemblies. The buckling appeared to be of a random nature with no apparent pattern or consistency. However, the four corner assemblies of the square pattern formed by the central nine assemblies appeared, in general, to have the worst buckling. The maximum lateral deflection of the buckles were estimated by visual inspection to be 0.06 to 0.08 inches.

The center span between the second and third grids of the plutonium assemblies experienced the worst buckling with the greatest frequency, the frequency and severity of buckling decreasing towards the end of the assembly. The direction of the buckling (toward or away from the fuel rods) appeared to be completely random and the severity of buckles independent of direction.

Rub marks, which were observed on several assemblies, could be attributed to handling or contact with spacer bars in the spent fuel cask during shipment. However, in at least one case it is definitely concluded, based on the appearance of the marks, that the rubbing occurred in the core and resulted from interference with a control rod assembly.

The single plutonium assembly which did not exhibit buckling contained eighteen stainless steel clad rods. The effect of stainless steel clad fuel rods would be to reduce the friction load exerted by the Zircaloy rods and increase the effective strength of the can (the stainless rods being put into compression as well as the can during differential thermal expansion).

The one plutonium assembly which contained eight stainless steel clad rods was found to have only minor buckling. By analysis, this number of stainless rods is insufficient to prevent buckling of the assembly but the reinforcing effect evidently did reduce the extent of buckling in this assembly.

1.2 Core II UO_2 Assemblies

Buckling was observed on three of the seven Core II design UO_2 fuel assemblies. In this case, however, the buckling was minor in nature and generally was restricted to the upper spans on the sides of the assemblies facing the center of the core. No buckling was observed on any of the Core I design assemblies used in Core II.

A detailed summary of the buckling is given in Table 1 and Figures 1 and 2 show some typical buckles. Figure 3 shows a core cross section indicating buckled assemblies.

TABLE 1

Serial No.	Type	Core Location	Buckling Observed	
503-1-7	SS Clad UO_2 Rods	1D	No	
503-1-19	SS Clad UO_2 Rods	3F	No	
503-1-10	SS Clad UO_2 Rods	5D	No	
503-10-6	SS Clad UO_2 Rods	1C	Yes	Minor
503-10-2	SS Clad UO_2 Rods	2B	No	
503-10-3	SS Clad UO_2 Rods	2F	Yes	Minor
503-10-4	SS Clad UO_2 Rods	4B	No	
503-10-5	SS Clad UO_2 Rods	5E	No	
503-12-2	Zr Clad PuO_2-UO_2 Rods	2C	Yes	
503-12-5	Zr Clad PuO_2-UO_2 Rods	2D	Yes	
503-12-4	Zr Clad PuO_2-UO_2 Rods	2E	Yes	
503-12-3	Zr/SS Clad PuO_2-UO_2 Rods	3C	No	
503-12-6	Zr Clad PuO_2-UO_2 Rods	3E	Yes	
503-12-7	Zr Clad PuO_2-UO_2 Rods	4C	Yes	
503-12-1	Zr/SS Clad PuO_2-UO_2 Rods	4D	Yes	
503-12-8	Zr Clad PuO_2-UO_2 Rods	4E	Yes	
503-7-1	SS Clad UO_2 Rods	1E	No	
503-2-3	SS Clad UO_2 Rods	3B	No	
503-11-1	SS Clad UO_2 Rods	4F	No	
503-11-2	SS Clad UO_2 Rods	5C	Yes	Minor
503-13-1	Zr Clad PuO_2-UO_2 Rods	3D	Yes	

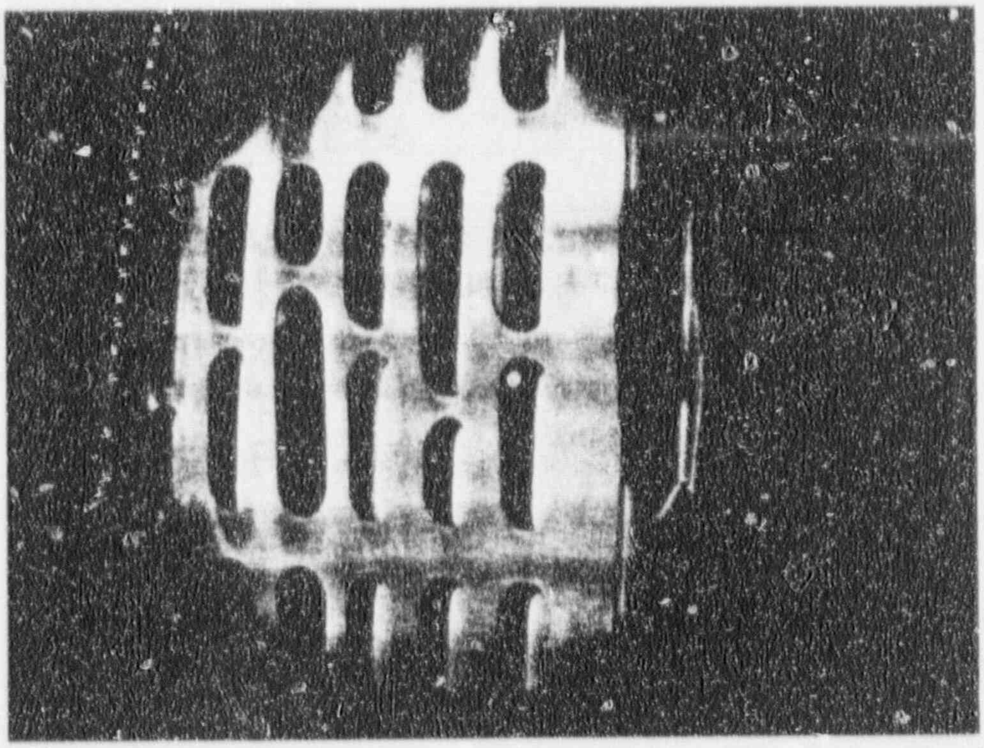


FIGURE 1

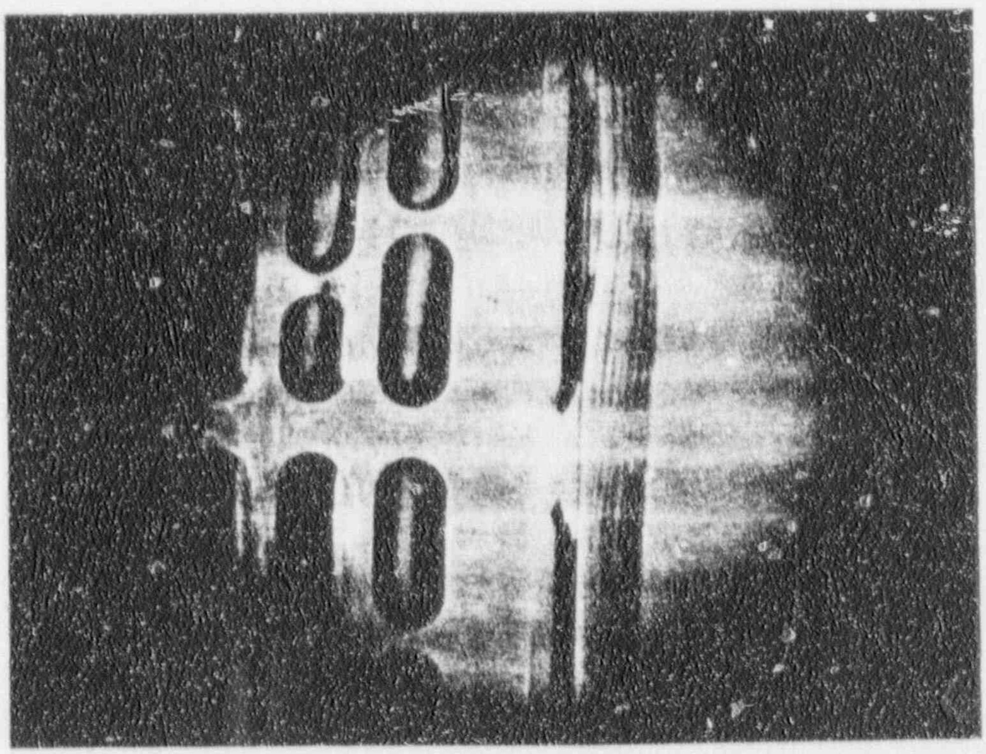


FIGURE 2

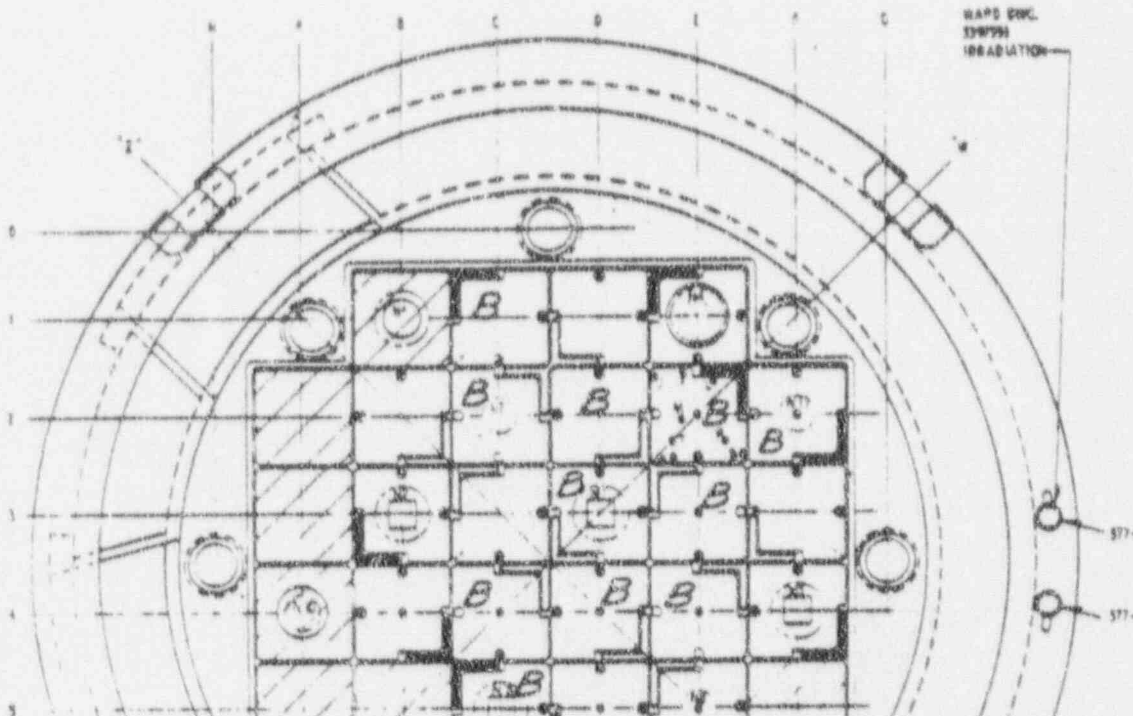


FIGURE 3: Saxton Core Cross-Section Showing Buckled Fuel Assembly Locations "B"

2.0 CAUSE OF BUCKLING

The following possible loading methods were examined as buckling modes for the assemblies:

A. Externally applied loads on the assembly due to:

1. Shipping and handling
2. Interference with reactor internals

B. Loading generated internally to the assembly through:

1. Frictional effects during differential expansion
2. Temperature differentials across the assemblies

2.1 Externally Applied Loads

Evidence of buckling due to these types of loading would have been the collapse of the fuel assembly end spans between the nozzles and end grids. The required end loading would also have had to have been of such magnitude that the top nozzle hold down springs would have collapsed. Examination of the assemblies showed no evidence of either of these conditions.

In addition, analysis showed that the conditions in the reactor during handling operations, which would be necessary to produce this magnitude of loading, could not be realistically predicted. It was concluded, therefore, that the buckling did not result from externally applied loads.

2.2 Internally Generated Loads

2.2.1 Plutonium Assemblies (Zircaloy Cladding)

Calculations show that the buckling of these assemblies occurred due to differential expansion on initial heatup. The buckling, however, would be of an elastic nature at that point, disappearing on subsequent cool down, except for some small amount of permanent set resulting from relaxation due to irradiation. The large buckles developed by a ratchetting mechanism through a number of full temperature cycles of the core, the buckles growing by the additional permanent set occurring with each cycle.

A full temperature cycle is from cold shutdown, through hot operating temperature back to cold shutdown conditions.

Examination of the reactor's thermal history, shows that only five such cycles had occurred prior to the mid-life detailed observation of three fuel assemblies. Estimations of the permanent set indicate that only small deformations would have been present at this stage; this is probably the reason that the buckling was not observed. Subsequent to this observation, eighteen additional cycles occurred which account for the large buckles observed at the end of life.

2.2.2 UO_2 Fueled Assemblies (Stainless Steel Cladding)

Although the UO_2 assemblies which exhibited buckling were of the Core II design with six point contact grid support, the buckling in the assemblies is not attributed to frictional loading. In these assemblies, both the assembly can and fuel cladding are stainless steel. Therefore, any frictional loading in the can caused by differential expansion between the rods and can at operating temperatures would be small and would result from tensile stresses in the can. In addition, buckling was only observed on the hot side of the cans and not randomly distributed around all sides as would be expected with axial friction loads.

It appears, instead, that the observed buckling in these assemblies resulted from a combination of thermal gradients across the assemblies and the resistance to bowing exerted by the rod bundles in the grids used in the Core II design.

The six point contact support used in the Core II grids provide an effective built-in condition for the rods at each grid location and thus resist bowing of the assemblies through restraining moments on the rods. If thermal gradients sufficient to produce bowing in an unrestrained condition were present in the Core II assemblies, the restraint offered by the grids could result in compressive buckling stresses on the hot face of the assemblies. This would not be the case with Core I design assemblies where the grids provide a four point support for the rods and little restraining moment.

Examination of the core temperature distributions based on power distributions during Core II operation showed four assembly positions where thermal gradients would be sufficient to cause buckling in Core II design assemblies and one location which was marginal. Of the four locations, one was occupied by a Core I assembly which exhibited no buckling. Two of the remaining locations were occupied by Core II assemblies which did exhibit buckling.

The Core II assembly occupying the fourth location showed no buckling. In this case, the actual average temperature of coolant flowing through the assembly is in question. From instrumentation in the 3 x 3 test assembly (503-4-29) which was suspended in the 9 x 9 assembly at this location, coolant temperatures approximately 20°F below expected were indicated. Because of channeling effects through the 3 x 3 assembly, the indicated temperature would be basically the discharge temperature from the 3 x 3 assembly and would reflect the effect of deleted fuel rods in the assembly. However, the low temperatures in the 3 x 3 would also tend to reduce coolant temperatures in the 9 x 9 assembly and thus reduce temperature gradients across the assembly since these are directly related to coolant temperature. Although the temperature effects cannot be accurately predicted, it would appear that buckling did not occur because of reduced coolant temperatures.

The last of the three Core II assemblies which exhibited buckling was in the marginal location where the thermal gradients were not sufficiently high to predict buckling. The buckling in this case, however, was very minor and localized and could possibly have resulted from a local weakness in the can (a thin ligament or out of flat condition).

3.0 MODIFICATION OF CORE III ASSEMBLIES TO PREVENT BUCKLING

The center fuel assemblies in Core III were of Core II design and contained Zircaloy water tubes and/or Zircaloy clad fuel. Therefore, they would experience high friction loads due to differential expansion between the stainless steel assembly can and the Zircaloy rods and would be expected to buckle. To prevent buckling during Core III operation, the Core II assembly design required modification.

There were two possible approaches to assembly modification to prevent the buckling:

1. Increase the stiffness of the can sufficiently to withstand imposed loads.
2. Decrease the friction load by reducing the normal force applied to the fuel rods by the grid springs.

A combination of both approaches was used. The spring contact force was reduced to the minimum which would not risk fretting of the fuel rods. However, this minimum contact force would still be sufficient to cause buckling. Therefore, the can was also strengthened.

3.1 Loose Lattice Assemblies

The fuel assembly has been strengthened by replacing six Zircaloy water tubes by six stainless steel ones and spot welding 0.028 inch thick angles between the can and the tubes. The one inch long clips previously used to fasten the two halves of the enclosure have been replaced by full length clips in the spans between grids. A cross-section of a repaired loose lattice assembly is shown in Figure 4.

The grid springs have been reset to give a nominal 6.5 lbs contact force compared to the 15.5 lbs force previously used. The combined effect of the changes produces a safety factor of 1.5 between friction forces generated by the grids and the buckling strength of the cans.

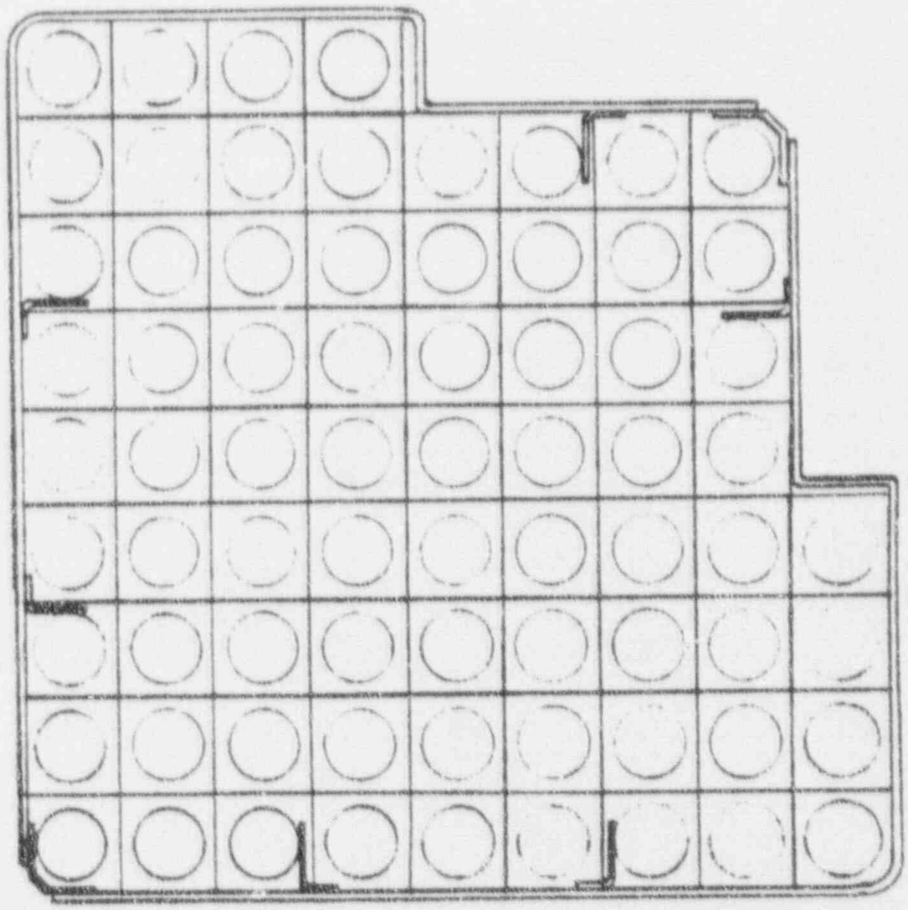


FIGURE 4: Cross-Section of Repaired Loose Lattice Assembly

3.2 Load Follow Assemblies

A slightly different method has been used for the load follow assemblies to obtain the required strength. A solid square stainless steel bar, with two angles spot welded between the bar and the can, is used in place of two stainless steel clad fuel rods in opposite corners of the assembly. A full length angle 0.05 inch thick has been spot welded to the inside of the enclosure skin at the center of each long span between grids. Full length angle clips are also used between the ends of enclosure halves as was done with the loose lattice assemblies. A cross-section of a repaired load follow assembly is shown in Figure 5.

The MAPI assembly to be used in the periphery of the core will be similarly treated.

The same buckling strength factor has been achieved for these assemblies as in the loose lattice.

3.3 Peripheral UO_2 Assemblies

Because of the large thermal gradients across these assemblies in Core III, only Core I type will be used.

An examination of the thermal gradients across the assemblies in the whole core shows that, although no buckling will occur, the worst gradient will produce a bow approximately 0.015 inch over the length of the assembly. This will always be of an elastic nature and will cause no interference problems.

3.4 Thermal Hydraulic Considerations

The modifications used for both the loose lattice and load follow assemblies have not compromised the thermal-hydraulic performance in Core III operation. For both types of assemblies the minimum DNB ratio will not be below the current limit (1.30) specified in the operating license for the Saxton reactor.

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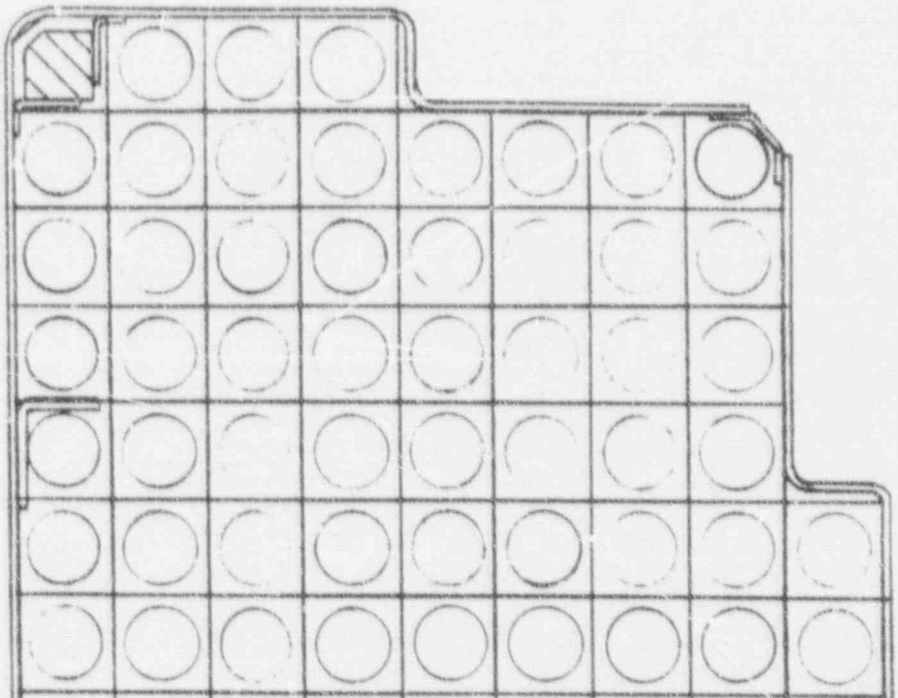


FIGURE 5: Cross-Section of Repaired Load Follow Assembly

With the modifications discussed herein buckling is not anticipated in either the new or irradiated assemblies used in Core III. Assuming buckling did occur, however, up to 0.060 inch inward deflection of the can surfaces can be tolerated without danger of exceeding the minimum allowable DNBR. Deflections of this magnitude in the outward direction could also be tolerated without problems, i.e., binding of control rods would not occur.

The fuel assemblies will be inspected at mid-life for any evidence of can buckling before reinsertion for continued reactor operation. In addition, the control rod scram times are normally checked every six months as a check against gross buckling.

3.5 Nuclear Aspects

The modifications proposed will have no effect on the operation of the load follow assemblies. However, the addition of stainless water tubes in the loose lattice assemblies will reduce the power output of these assemblies by approximately 3%. This may be alleviated slightly by increasing the nominal power output from 26 MWT to approximately 26.3 MWT without seriously affecting any thermal hydraulic or nuclear margins.

3.6 Health and Safety

The modifications to Core III assemblies of Core II design will prevent buckling and present no hazard to the health and safety of the public. The assemblies will be given a detailed examination for buckling at mid core life.

APPENDIX C

CHANGE REPORTS 17 to 23

Change Report No. 17

MATERIALS COMPATIBILITY TEST

1. DESCRIPTION OF CHANGE

A subassembly designated Test Assembly No. 503-4-34 will contain either a primary or an alternate materials compatibility test in which case it will be designated 503-4-34A. The subassembly containing the primary materials compatibility test is illustrated in Figure 1 and the primary test rod in Figure 2. The subassembly containing the alternate materials compatibility test is illustrated in Figure 3 and the alternate test sections are illustrated in Figure 4. The subassembly for the primary test will contain two fuel rods in addition to the two materials compatibility test rods; whereas, the subassembly for the alternate test will contain no fuel rods. The rods which will be with the primary materials compatibility test will be clad with Zircaloy-4 having a nominal wall thickness of 23 mils and contain uranium dioxide (UO_2) pellets, uniformly enriched to 12.5 w/o U-235 and 89.5% to 94% theoretical density.

a. Primary Test Design

The primary design illustrated in Figures 1 and 2 consists of a Zircaloy-4 tube to which are attached 10 stainless steel sleeves, 2.425 inches long, equally spaced over the tube length. The sleeves are mechanically attached to the Zircaloy tube by bulges. Each stainless steel sleeve in the subassembly grid area has welded to it four Inconel spring fingers which hold the tube in place by pressing against a stainless steel cylinder which is welded into each grid of the test subassembly.

b. Alternate Test Design

The alternate design is illustrated in Figures 3 and 4. In this design the Zircaloy tube test sections containing stainless steel sleeves are notched and held between the subassembly grids by the notches. As in the primary test design, the sleeves are mechanically attached to the Zircaloy tube by bulges.

Section B-B of Figure 3 is an illustration of the cross section of the alternate design subassembly showing the arrangement of the test sections in the assembly. The outer two cylindrical cross sections are part of the grid structure while the inner two are test sections.

2. PURPOSE OF CHANGE

The purpose of the change is to allow substantiation of the expectation of negligible crevice corrosion between stainless steel and Zircaloy-4 in an operating PWR environment.

3. SAFETY CONSIDERATIONS

The parameters pertinent to safety considerations for the primary materials compatibility test design are presented in Table I for the assembly containing the tests and two 12.5 w/o U-235 enriched fuel rods. The location of the fuel rods and materials compatibility tests in the assembly are shown in section B-B of Figure 1.

It can be seen in Table I that the calculated values of specific power, heat flux, DNB ratios, and fuel rod and rod surface temperatures are well within safe limits. Further, it is highly unlikely that the materials compatibility test should fail in such a way as to interfere with the heat transfer from the two fueled rods.

It is highly unlikely in the extreme that a failure of an alternate test section could cause an unsafe condition since the subassembly containing the backup alternate test will contain no fuel rods.

4. HEALTH AND SAFETY

It is our conclusion that the health and safety of the public will not be endangered by this change.

TABLE I

THERMAL AND HYDRAULIC PARAMETERS FOR MATERIALS COMPATIBILITY TESTS IN ASSEMBLY CONTAINING TWO 12.5 w/o U-235 ENRICHED RODS

T_{inlet} (28 Mwt) 480°F

Maximum Linear Power Density 16.1 kw/ft

Maximum Surface Heat Flux 540,000 BTU/Hr. ft.^{2}

Hot Channel Factors

F_q	2.43
$F_{\Delta H}$	4.01

Minimum DNB ratios

100% Power (2250 psia T_{inlet} 480°F) 2.45

112% Power (2075 psia T_{inlet} 490°F) 1.93

Mean Clad Temperature at maximum surface heat flux

100% Power 711°F

112% Power 719°F

Clad Stresses*

Maximum <10,000 psi

Clad Strain

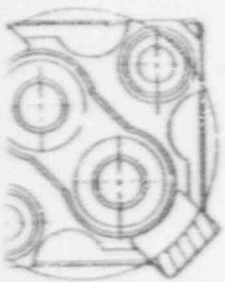
Maximum <0.5%

Fuel Center Temperature

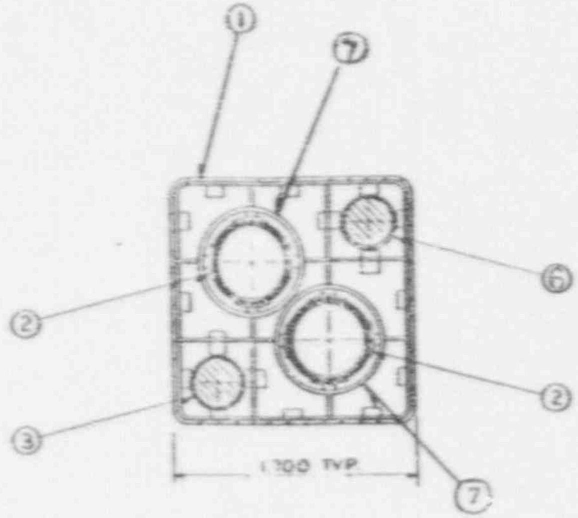
100% Power (16.1 kw/ft) 3750°F

112% Power (18.1 kw/ft) 4120°F

*Pressure tensile stress, hot, operating, end-of-life



CTION A-A



SECTION B-B

Item	Title
1	Enclosure
2	Materials Compatibility Test Rods
3&6	Fuel Rods
4	Top Plate
5	Bottom Plate
7	Stainless Steel Cylinder

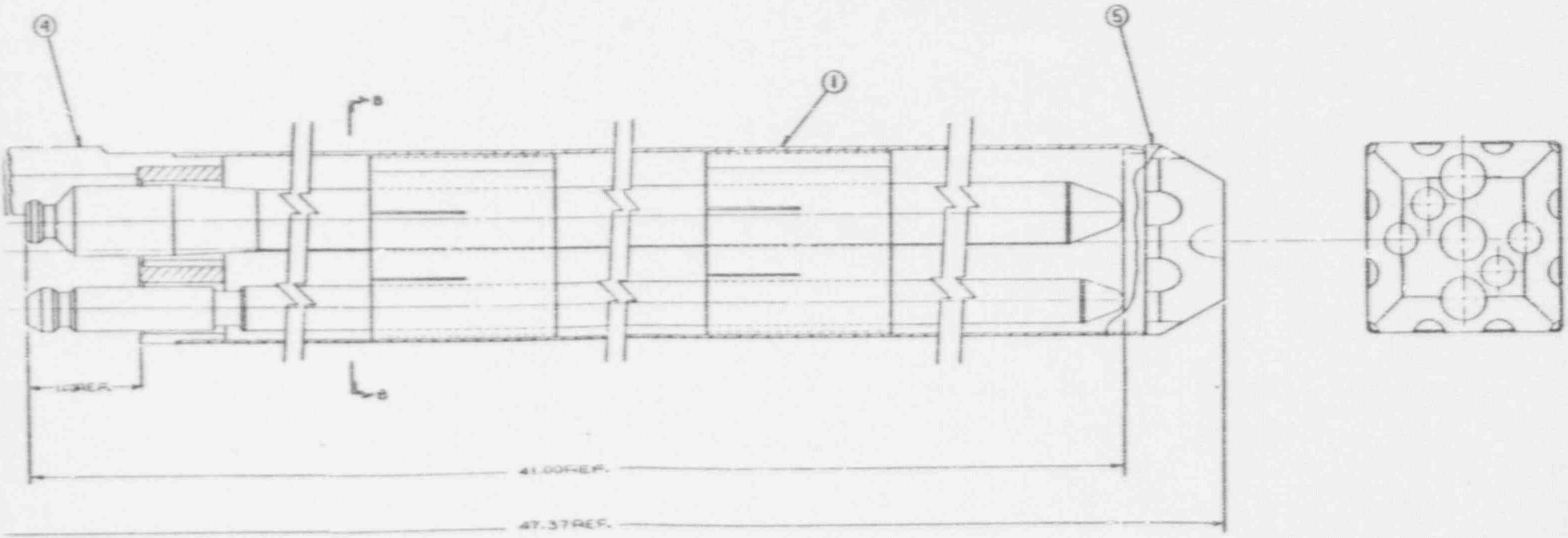
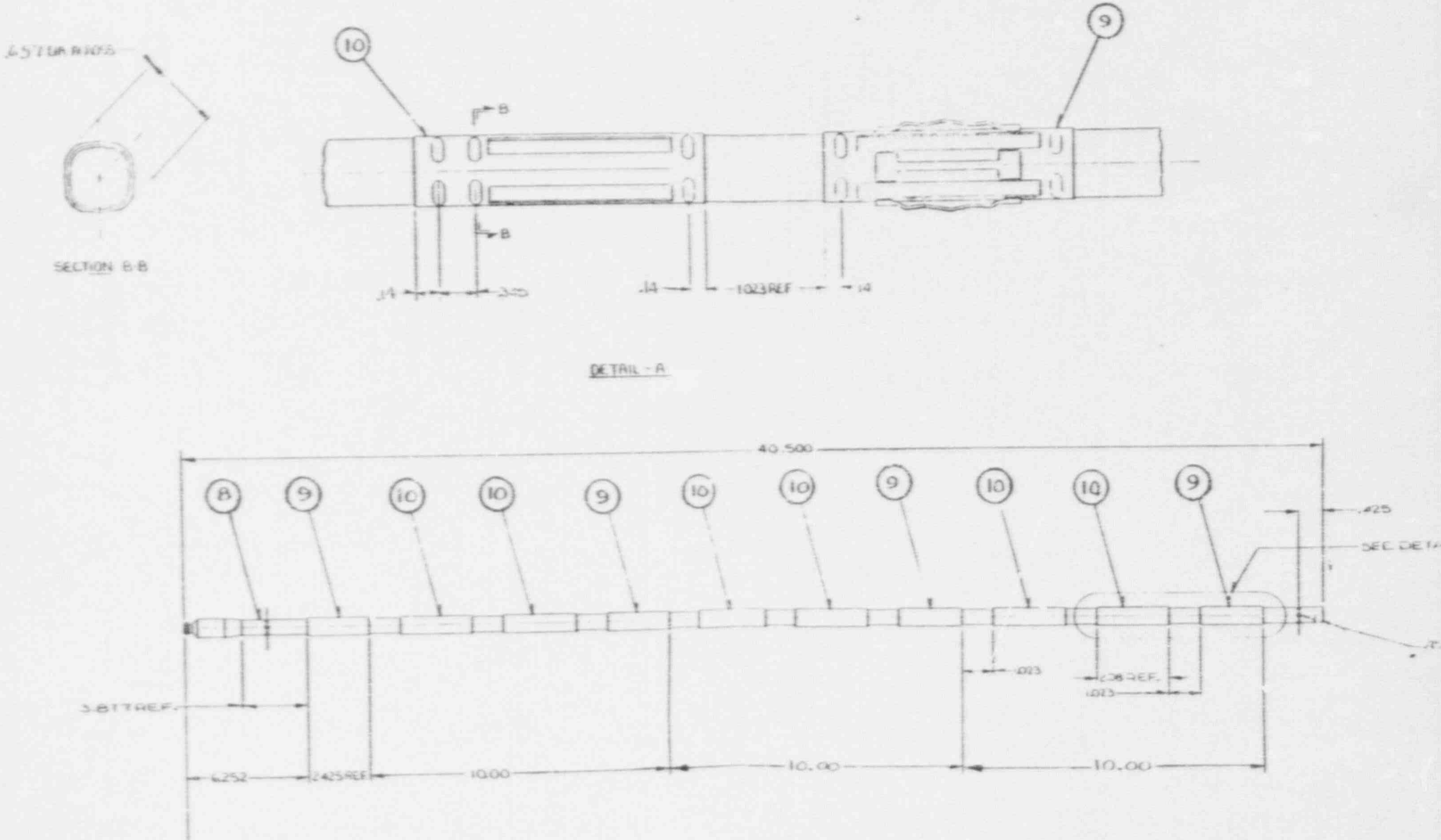


FIGURE 1

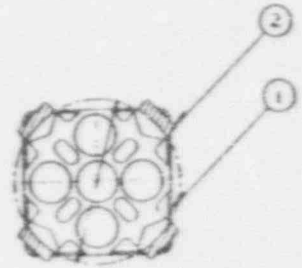
PRIMARY TEST ASSEMBLY



.657 DIA ACROSS
SECTION B-B
10
9
B
B
.14
.345
.14
1.023 REF
.14
DETAIL - A
40.500
8
9
10
10
9
10
10
9
10
10
9
.425
SEE DETAIL
.023
.038 REF.
.023
3.817 REF.
6.252
2.425 REF
10.00
10.00
10.00

FIGURE 2

Item	Title
1	Enclosure
2	Top Plate
3	Bottom Plate
4	Grids



SECTION A-A

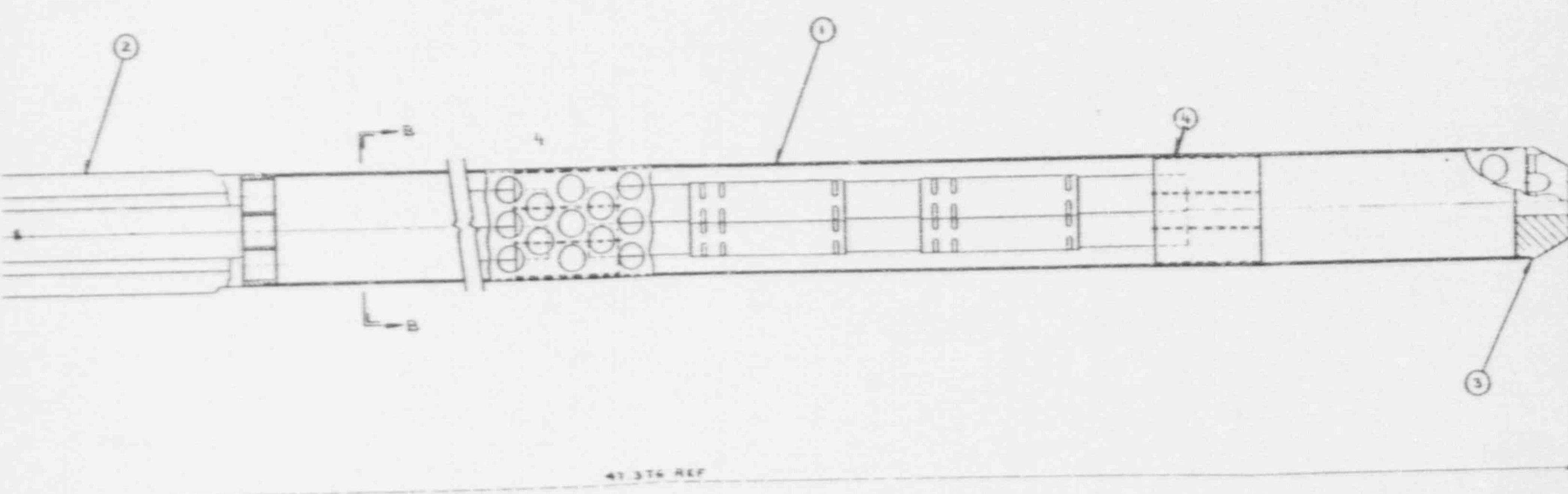
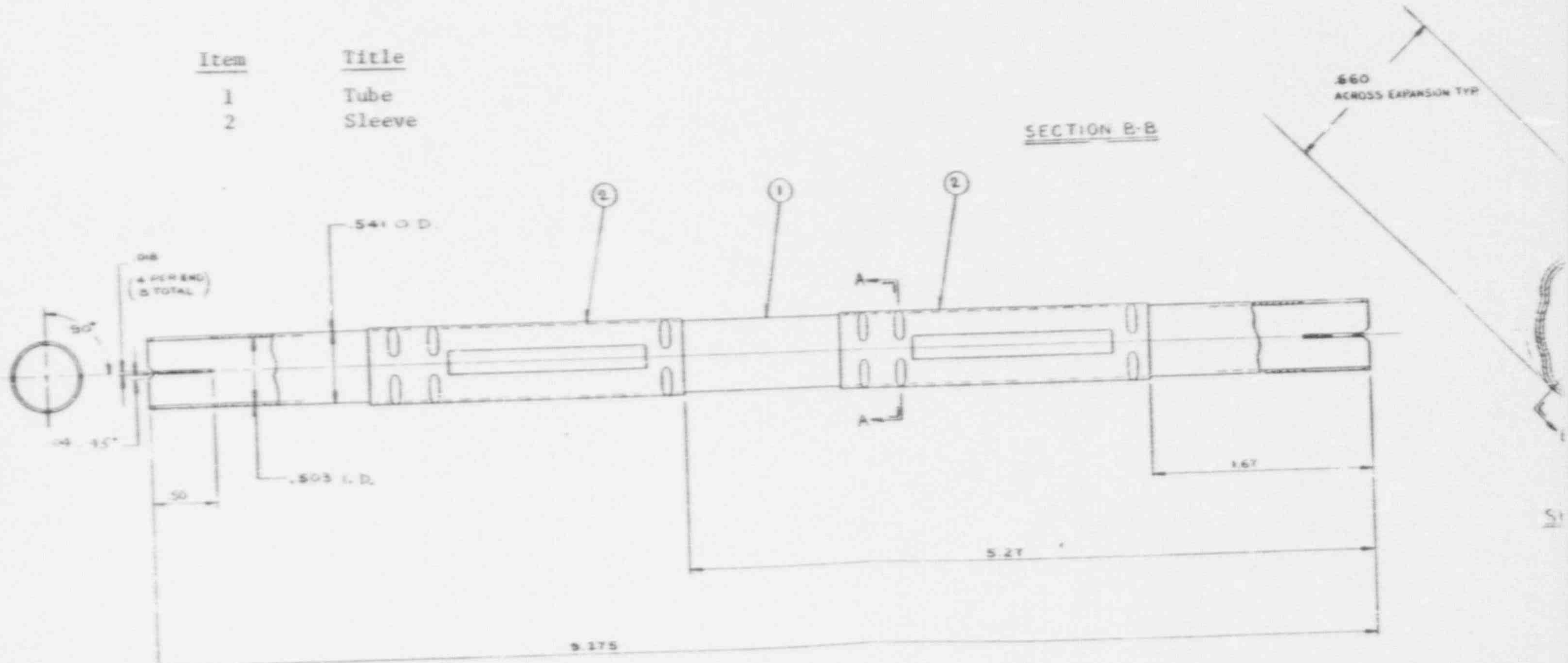


FIGURE 3

Item	Title
1	Tube
2	Sleeve



SECTION B-B

Approval to operate rods similar to these at a maximum calculated operating stress of 22,000 psi has been previously given in Change No. 30. Fuel rods containing 30-75 ppm moisture have previously been successfully operated at Saxton. It is our opinion that this change does not represent any unreviewed safety question.

4. HEALTH AND SAFETY

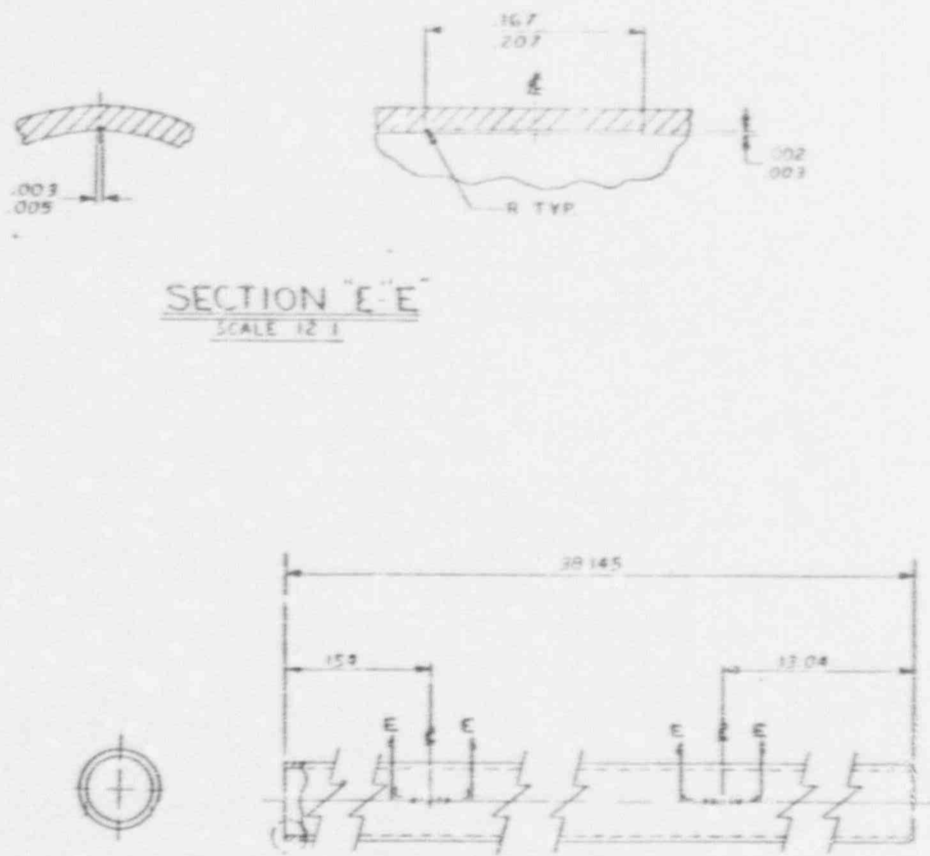
It is our conclusion that the health and safety of the public will not be endangered by this change.

TABLE I

SAFETY PARAMETERS FOR EXPERIMENTAL HYDRIDE RODS

Peak Power kw/ft	Initial Pressure (cold) psia	Fill Gas	Rod Average D_2O ppm	Operating* Stress Level psi
≤16.1	815	80% He-20% Xe	25	<7000
≤16.1	815	80% He-20% Xe	25	<7000
≤16.1	915	He	120	<7000
≤16.1	915	He	120	<7000

*Pressure tensile stress, hot, operating, end-of-life



A. NOTCH (SEE SECTION E-E FOR DIMENSIONS) CENTER TO BE POSITIONED TO THESE DIMENSIONS. LOCAL FLUORIDE CONTAMINATION TO BE LOCATED AT THIS POINT ALSO.

NOTES:

1) NOTCH DIMENSIONS WILL BE VERIFIED BY SECTIONING OF TEST SAMPLES OBTAINED BEFORE AND AFTER FUEL TUBE NOTCHING

FIGURE 1

Change Report No. 19

TEST FUEL RODS - CLAD CREEP BEHAVIOR

1. DESCRIPTION OF CHANGE

Four test fuel rods will be inserted in Saxton Core III peripheral test subassemblies. Two of the test rods will be high pressure creep rods and two low pressure creep rods. The four rods are clad with Zircaloy-4 having a nominal thickness of 23 mils and contain uranium dioxide (UO_2) pellets, uniformly enriched to 12.5 w/o U-235, having less than 30 ppm H_2O and are helium filled. The high pressure creep rods contain 94% theoretical density fuel pellets and have a 10 inch gas plenum where as the low pressure creep rods contain 89.5% theoretical density fuel pellets and have a 2 inch gas plenum.

2. PURPOSE OF CHANGE

The purpose of the change is to assess the effect of irradiation on clad creep behavior.

3. SAFETY CONSIDERATIONS

Parameters pertinent to safety are given in Table I. The maximum calculated operating stress in the rods is 16,000 psi which is much less than previously approved stress levels. The stress level in the low pressure creep rods was not determined since it will be much less than that in the high pressure creep rods, which is well within acceptable limits. Rods similar to the low pressure creep rods were previously approved in Change No. 22, 24 and 29. The high pressure creep rods will be similar to rods previously approved in Change No. 30, the principle difference being that the high pressure creep rods will operate at a 6,000 psi lower stress level than those of Change No. 30.

It is our opinion that this change does not represent any unreviewed safety question.

4. HEALTH AND SAFETY

It is our conclusion that the health and safety of the public will not be endangered by this change.

TABLE I

SAFETY PARAMETERS FOR CREEP TEST RODS

Test Rod Type	Peak Power kw/ft	Initial Pressure cold psia	Operating Stress Level* psi
High Pressure Creep	<16.1	1915	<16,000
Low Pressure Creep	<16.1	290	negligible compressive stress

*Pressure tensile stress, hot, operating, end-of-life

Change Report No. 20

CORROSION TEST OF ZIRCALOY-4 SPOT WELDS AND BI-METALLIC SANDWICH

1. DESCRIPTION OF CHANGE

Test sections containing sandwich spot welds as shown in Figure 1 will be suspended ver'!-ally in four removable water filled tubes as shown in Figure 2. .he tubes containing the spot weld tests will replace four standard water tubes in two of the central nine Core III assemblies. The location of the four removable water tubes which contain the tests are shown in the following table.

Assembly	Core Location	Tube Location in Assembly*
503-17-8	C-3	E-1 and A-5
503-17-9	E-4	E-1 and A-5

The following table is a tabulation of the parts shown in Figures 1 and 2.

Part No.	Title	Material
Figure 1		
1	Top Eng Plug	Zircaloy-4
2	Retainer	Zircaloy-4
3	Pin	Zircaloy-4
4	Retainer	Zircaloy-4
5	Retainer	Zircaloy-4
6	Retainer	Zircaloy-4
7	Clip	Stainless Steel 304L
8	Clip	Inconel 718
Figure 2		
9	Water Tube	Zircaloy-4
10	Bottom End Plug	Zircaloy-4

*See Figure 3 for explanation of assembly location designation.

(

2. PURPOSE OF CHANGE

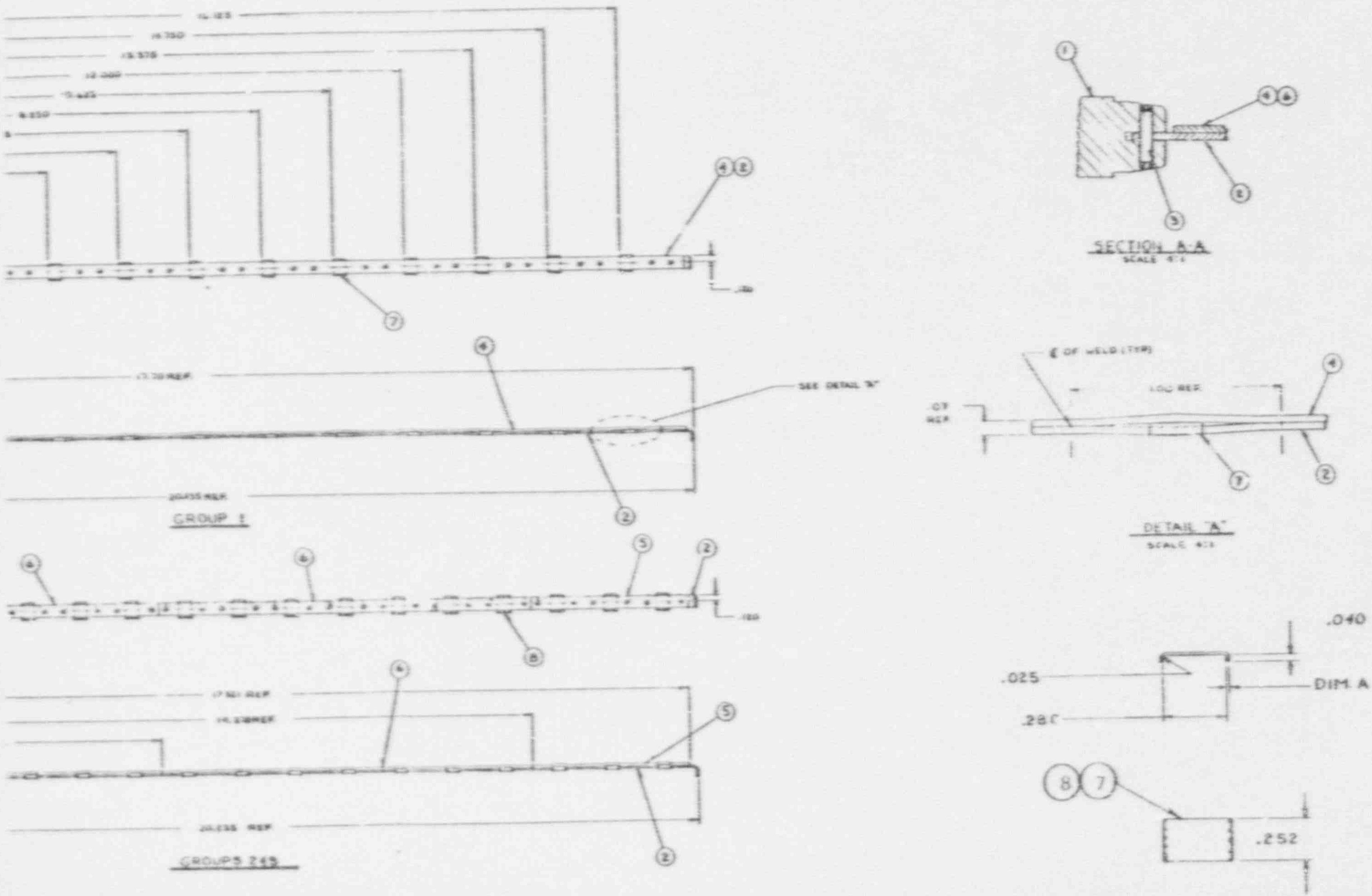
The purpose of the change is to permit the evaluation of corrosion and hydriding effects on spot welded joints in an operating pressurized water reactor environment.

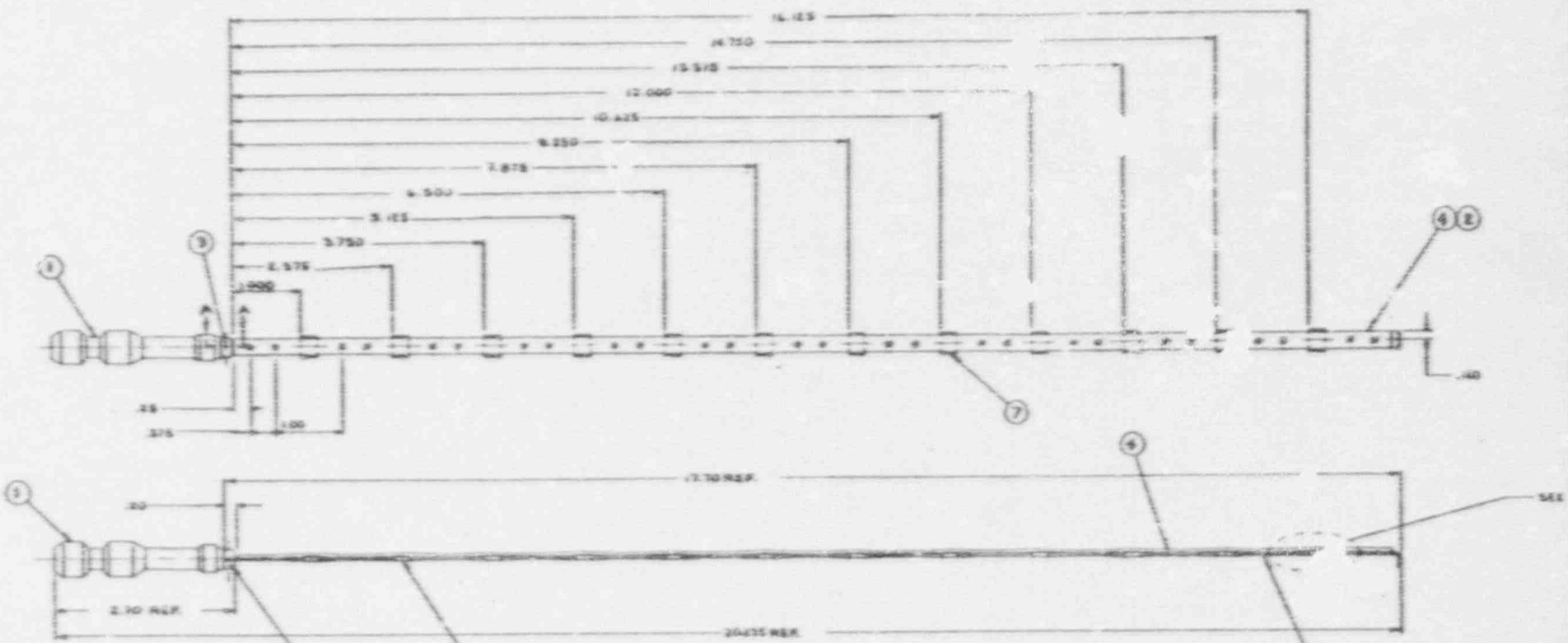
3. SAFETY CONSIDERATIONS

The test sections containing bi-metallic sandwich and spot welds will be suspended vertically in the water tube by being welded to the upper end plug. Coolant access will be through 0.125 inch and 0.062 inch diameter bleed holes in the upper and lower sections of the tube respectively. The bleed holes are such that it is highly improbable that material large enough to be detrimental to the reactor operation could get from inside the tube into the coolant. It is our opinion that this change does not represent any unreviewed safety question.

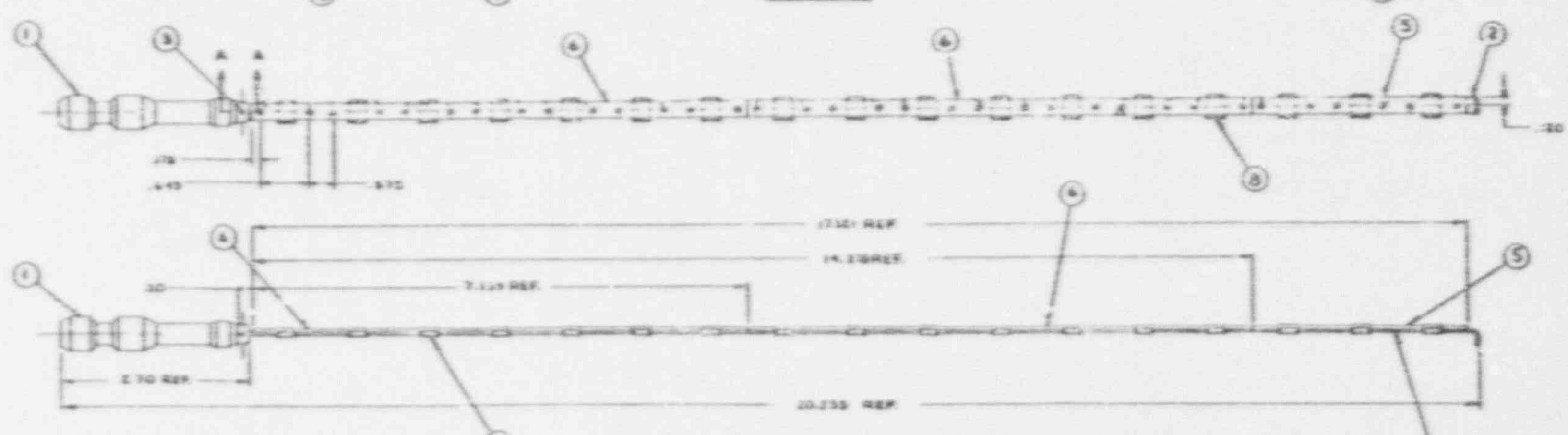
4. HEALTH AND SAFETY

It is our conclusion that the health and safety of the public will not be endangered by this change.


SECTION A-A
DETAIL "A"
GROUP 1
GROUPS 2&3
SEE DETAIL "A"
.040
.025
DIM. A
.280
.252



GROUP 1



GROUPS 2&3

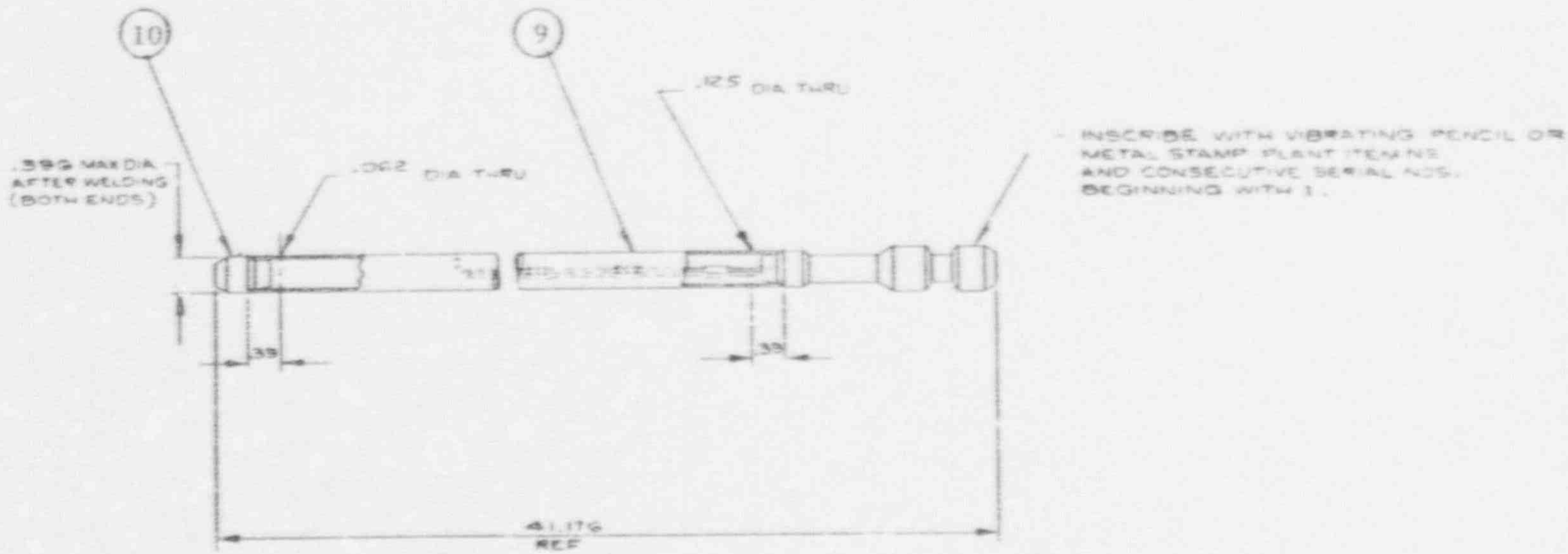


FIGURE 2

Assembly-Removable Water Tube Containing Test Section

Fuel rods at restricted coolant channels were replaced by solid Zircaloy bars to maintain thermal-hydraulic safety margins and to minimize the disturbance of the power distribution in surrounding fuel rods.

The following table is a tabulation of the thermal and hydraulic design parameters which still pertain to Saxton Core III operation.

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

	Loose Lattice Assembly	Load Follow Assembly
Center Core Region	UO_2-PuO_2	UO_2
Fq 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

It is our opinion that this change does not represent any unreviewed safety question.

4. HEALTH AND SAFETY

It is our conclusion that the health and safety of the public will not be endangered by this change.

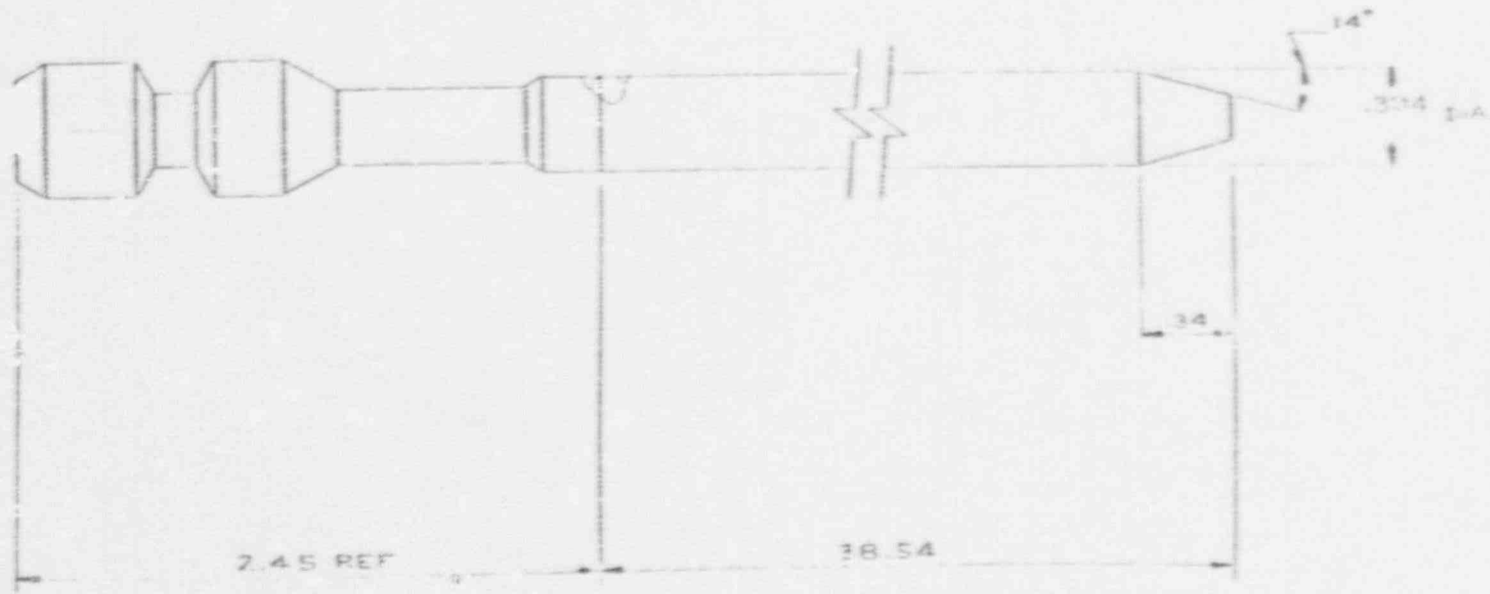


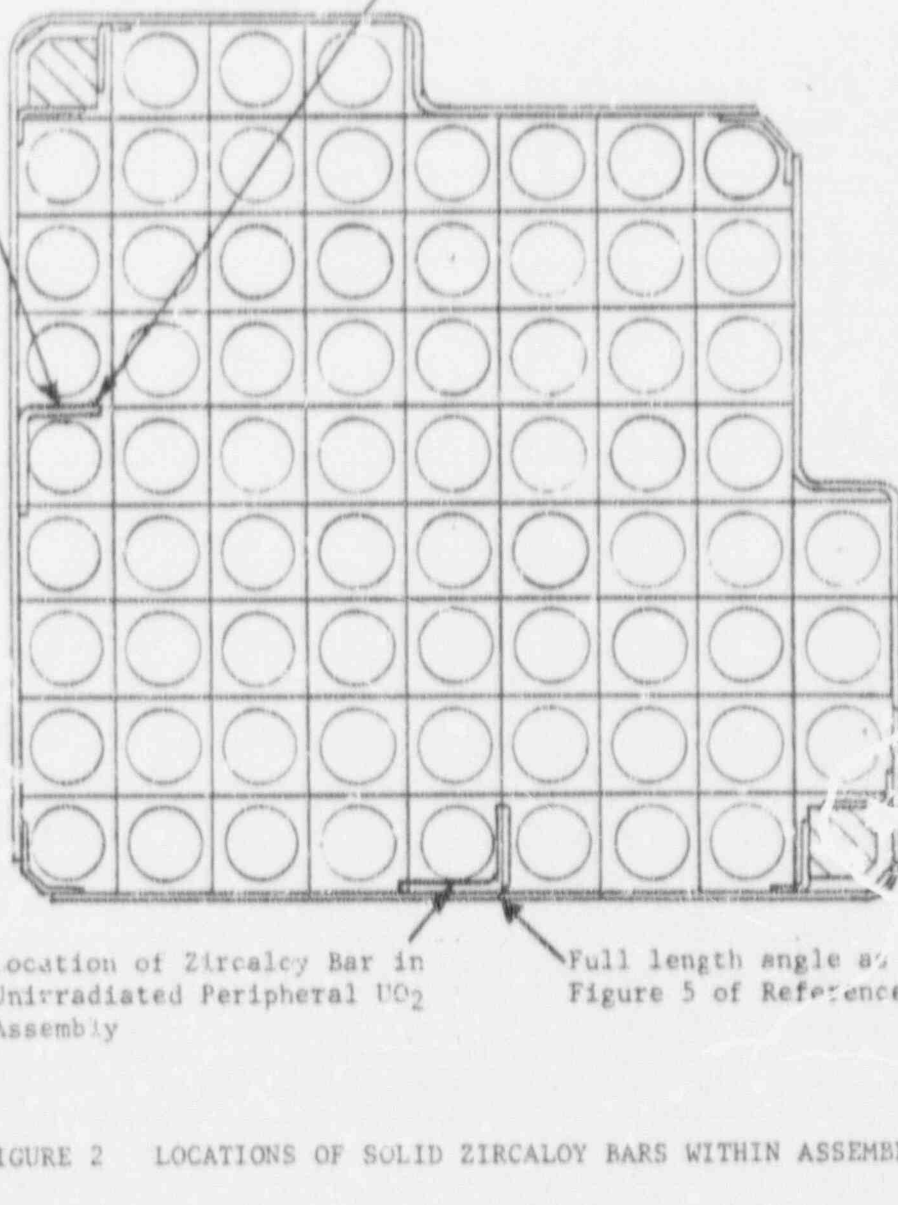
FIGURE 1

SOLID ZIRCALOY BAR

WESTINGHOUSE ELECTRIC CORPORATION

Full length angle as shown in Figure 5 of Reference (1)

Location of Zircaloy Bar in Unirradiated Peripheral UO_2 Assembly and Load Follow Assembly



Location of Zircaloy Bar in Unirradiated Peripheral UO_2 Assembly

Full length angle as shown in Figure 5 of Reference (1)

FIGURE 2 LOCATIONS OF SOLID ZIRCALOY BARS WITHIN ASSEMBLIES

(1) Summary Report on Buckling of Saxton Core II Fuel Assemblies and Prevention of Buckling in Core III.

APPENDIX D

PARTS OF WCAP-3850-3 WHICH DESCRIBE EXPERIMENTALLY OBSERVED STRAIN IN CVTR FUEL RODS

The ridging described in the following pages is at most a few mils while blisters of up to 8 mils were observed. This corresponds to <5%.

3.4.1.1.2 Gamma Scanning, Fuel Rods 53.861 and 53.862

Failed fuel rod 53.861 and sister rod 53.862 were scanned for gross gamma activity in the Transfer Building pool after wire spacers had been removed. The results, averaged over scans taken 90° apart, are given in Figures 3-3 and 3-4. Peak-to-average activity was 1.17 for rod 53.861 and 1.16 for rod 53.862. Integrated activity for rod 53.862 was about 5.5 percent lower than the failed rod, which can be accounted for by the combined uncertainties of the instrumentation (±5 percent accuracy) and hot channel factors (for example, the flux skew factor, F_{ϕ}, is 1.025 from critical experiments*).

3.4.1.1.3 Profilometry, Length Measurements

Typical profilometer traces are shown in Figures 3-5 and 3-6 for rods 53.861 and 53.862. The scan for the latter rod shows no anomalies; average diameters over the fuel bearing portion of the rod fall between 0.4871 and 0.4883 inch. Fuel rod ovality (as estimated by comparing adjacent minima and maxima on the helical trace) varies between 0.2 and 2.0 mils. At no point along the rod do the measured diameters lie outside fabrication specifications of 0.486/0.489 inch.

Failed rod 53.861 is characterized (Figure 3-5) by clad ridging, in contrast to the relatively smooth trace from its sister rod. The height of the ridges is nominally 1.0 - 1.5 mils, and the spacing between ridges is approximately the length of a fuel pellet, 0.650 inch. Areas of ridging generally are concentrated within 25 to 58 inches from the bottom of the rod. Average diameters fall within 0.4870 and 0.4885 inch, but localized high ovality (up to 3.2 mils) or ridging result in extremes of 0.4860 and 0.4902 inch.

The measured post-irradiation length of rod 53.862 was 103.094 inches. A pre-irradiation measurement is not available for this rod, although the rod had been identified as being within fabrication specifications (102.851/102.967 inches). Using the nominal dimension, the length increase during irradiation was 0.12 percent. No length measurement was made on failed rod 53.861 because of the defect at the upper end plug.

*CVNA-287, CVTR Fuel Management, pp. 125-134

3.4.1.1.4 Helium Leak Testing

The bottom (~100 inch) section of fuel rod 53.861 beneath the defect was leak tested with the apparatus described in Section 3.3.2 and Figure 3-1. No secondary leaks were detected in the fuel tube or lower end plug. The minimum detectable leak with the helium test is estimated as 4×10^{-8} cc/sec. (STP).

Fission gas collection and analysis from sister rod 53.862 appeared normal and no further leak testing was considered necessary.

3.4.1.2 Special Test Assembly G-4

3.4.1.2.1 Visual Examination, Failed Rod 83.831

Axial locations and orientations (with respect to the wire spacer hole in the lower end plug) of the blisters on the failed rod are shown in Figure 3-7. Photomacrographs of the blisters at 14 and 16-1/2 inches from the bottom were presented in WCAP-3850-2, p. 42, Dec. 21, 1967. The 14-inch blister was initially observed as defective, but the 16-1/2-inch blister may have been damaged during post-irradiation handling. As measured by the profilometer, the 14-inch blister (largest) was ~7.8 mils high and extended axially over approximately 600 mils of the clad.

Crud deposition was very light and uniform on the failed region and probably was not a factor. A large ridge (Section 3.4.1.2.3) was observed 83 inches from the top of the rod. In addition, there were indications of other ridged areas on the rod and sister rod 83.832, but the small magnitude of the individual ridges made identification impossible, even with the aid of the profilometer scans.

3.4.1.2.2 Gamma Scanning, Fuel Rods 83.831 and 83.832

Failed fuel rod 83.831 and sister rod 83.832 were scanned for gross gamma activity in the Transfer Building pool. The axial distributions are presented in Figures 3-8 and 3-9. Peak-to-average activity is 1.15 and 1.16 for rods 83.831 and 83.832, respectively. The integrated activity of rod 83.832 is about 2.7 percent higher. The curves shown here have not been normalized for decay with the G-3 rods (Figures 3-3 and 3-4) and the discrepancy in activity levels reflects the differing power and decay histories of the two assemblies (Table 3-1).

3.4.1.2.3 Profilometry, Length Measurements

The surface profile of failed rod 83.831 is characterized by blisters and ridging. Figures 3-10a and 3-10b show the trace in the vicinity of large clad blisters at 16-1/2 and 17-1/2 inches from the bottom, and also show a typical ridging pattern at ∿44 inches. Regions of regular ridging (∿1 mil ridge height) exist over most of the fuel-bearing zone of this rod, along with several larger (∿2 mils), discrete ridges. A particularly prominent ridge (∿4 mils) was observed at the approximate top of the fuel column, 83 inches from the bottom. Average clad diameters (excluding local blistered and ridged regions) range between 0.4884 and 0.4907 inch. This rod is characterized by areas of high ovality, particularly in the plenum region (∿83 to 100 inches) where the ovality exceeds 4 mils.

Sister rod 83.832 also showed areas of regular (∿1 mil) ridging and larger, discrete ridges similar to the failed rod. Again, high ovality (≤5 mils) occurs in the plenum zone, and within the bottom ∿12 inches of the rod (2 - 4 mils). Average rod diameters vary between 0.4866 and 0.4878 inch over the fuel-bearing region.

Post-irradiation length measurements (±0.002 inch) for the G-4 rods were: rod 83.831 (failed), 103.160 inch, rod 83.832, 103.149 inch. No explicit pre-irradiation lengths were measured for these rods, but taking the nominal length of 102.909 inch, the length increases are 0.24 percent for rod 83.831 and 0.23 percent for rod 83.832. Taking the maximum length allowed under the fabrication specifications, the increase for these rods is 0.18 percent or greater.

3.4.1.2.4 Helium Leak Testing

The defective region of rod 83.831 (9 - 18 inches) was sectioned out and the remaining bottom and top pieces were tested for secondary leaks. No leaks were found.

APPENDIX E

ROD WITHDRAWAL AND STEAM LINE BREAK ACCIDENT ANALYSIS FOR SAXTON CORE II 35 MWt OPERATION

VI. ACCIDENT ANALYSIS

A. GENERAL

The increased power level (35 MWt as compared to the 23.5 MWt licensed operating power) and the corresponding changes in operating conditions and instrument settings require that some of the incidents previously reported (in the Saxton Final Safeguards Report and in the Safeguards Report for the Saxton Reactor Partial Plutonium Core II) be re-evaluated. Information in Section 502 of the Final Safeguards Report relating to the possible causes of incidents and the safeguards provided applies to the incidents analyzed for this report and will not be repeated.

B. REACTIVITY INCIDENTS

1. Rod Withdrawal Incident

An uncontrolled rod withdrawal is assumed to occur from an electrical or mechanical failure in the nuclear instrumentation and control systems or by operator error. In this unlikely event, an electrical interlock ensures that only one of the two control rod groups would be withdrawn. Assuming that the most reactive control rod group is withdrawn at its maximum rate (1.5 inches per minute) in its maximum worth region, the reactivity addition rate is limited to less than 7.2×10^{-5} Δk/sec. In Section VI of the Safeguards Report for the Saxton Reactor Partial Plutonium Core II, rod withdrawal transients are presented for cold and hot subcritical, and full power operation initial conditions. These analyses were based on a conservative high insertion rate of 2.5×10^{-4} Δk/sec and illustrate the effectiveness of the overpower trip in terminating a rod withdrawal transient. Startup from the hot or cold subcritical condition is of less consequence than rod withdrawal from power, so rod withdrawal from power is re-analyzed. The principal change affecting the transients starting from subcritical is the increase in the overpower trip setpoint from 115% of 23.5 MWt to 107% of 35 MWt.

The additional energy generated and, hence, the increase in fuel temperature in reaching the higher trip level is not significant. It should be noted that the analyses presented for rod withdrawal from subcritical in Section VI of the Safeguards Report for the Saxton Reactor Partial Plutonium Core II are highly conservative in that no credit is taken for either the startup rate trip (set at 2 decades/minute) or for the reduction in overpower trip setpoint (to 5 MWt) during zero power operation.

The transient for rod withdrawal from power is shown in Figure VI-1 based on the following conservative conditions:

a) Initial power level is 103% of the nominal (35 MWt) power to allow for calorimetric errors.

b) Initial primary coolant pressure is at its minimum value of 2150 psia to allow for instrument errors.

Initial coolant inlet temperature is at its maximum value of 485°F allowing for instrument error.

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

e) Minimum expected absolute value of negative moderator temperature coefficient: -2.0×10^{-4} $\Delta k/°F$.

f) Reactor trip initiation due to overpower at 114% of the 35 MWt power (7% over the 107% reactor trip setpoint to allow for instrumentation errors).

With the nuclear flux and hot spot heat flux peak at 120% and 114% of their nominal full power values respectively, the minimum DNB ratio (calculated using the W-3 correlation) is 1.31 and occurs 3.7 seconds after initiation of the incident. This indicates that the power level reactor trip protection would prevent core damage in the improbable event of an uncontrolled rod withdrawal.

2. Steam Line Break Incident

Rupture of a secondary plant steam line is reflected into the primary system as a step load increase and results in a decreasing coolant inlet temperature. The negative moderator temperature coefficient causes reactivity and power to increase. For large breaks, the control capability of the plant is exceeded and the reactor protection system will automatically initiate a reactor trip by either the overpower or low pressure condition. Following trip, heat extraction exceeds heat generation and the coolant temperature decreases further. Due to the negative moderator temperature coefficient, the shutdown margin is reduced until heat removal is terminated.

The consequences of a steam line break for the proposed 35 MWt operating conditions are not changed from those reported previously in Reference (1), (2) and (3). The lower secondary temperature, 420°F for 35 MWt operation as compared to 490° for 23.5 MWt operation) will result in a smaller steam mass flow rate through the break and will reduce the rate of cooldown in the primary system. Since the average coolant temperature is reduced to 500°F for the proposed operating conditions, the moderator temperature coefficient is less negative than that assumed in the reference analyses. Both of these effects will result in a less severe reactivity transient. The effect of the increased power level is to increase decay heat following reactor trip which reduces slightly the primary system cooldown. The operating conditions for 35 MWt result in less reduction in shutdown margin during blowdown of the steam generator contents.

C. MECHANICAL INCIDENTS

1. Loss of Flow Incident

A loss of coolant flow could result from loss of electric power to the reactor coolant pump motor or from mechanical failure in the pump motor or coupling between the pump motor and pump. A mechanical failure causing sudden seizure of the pump motor is not considered credible. Following loss of coolant flow, coolant temperature will increase because of reactor trip circuit delays which allow continued power generation while flow coastdown is occurring. If the heat generation is not terminated rapidly enough to prevent DNB, clad failure can result.

Power generation during a loss of coolant flow incident is terminated by an automatic reactor trip initiated from either a low voltage signal on the reactor coolant pump bus, a low frequency signal on the MG set control or from a low flow signal.

In the event of a loss of the station 480 V auxiliary electrical system, reactor coolant pump coastdown is extended (see Section IV) by automatic switching which transfers the field supply of the generator to the station battery. This extended coastdown

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(1) Saxton-Final Safeguards Report

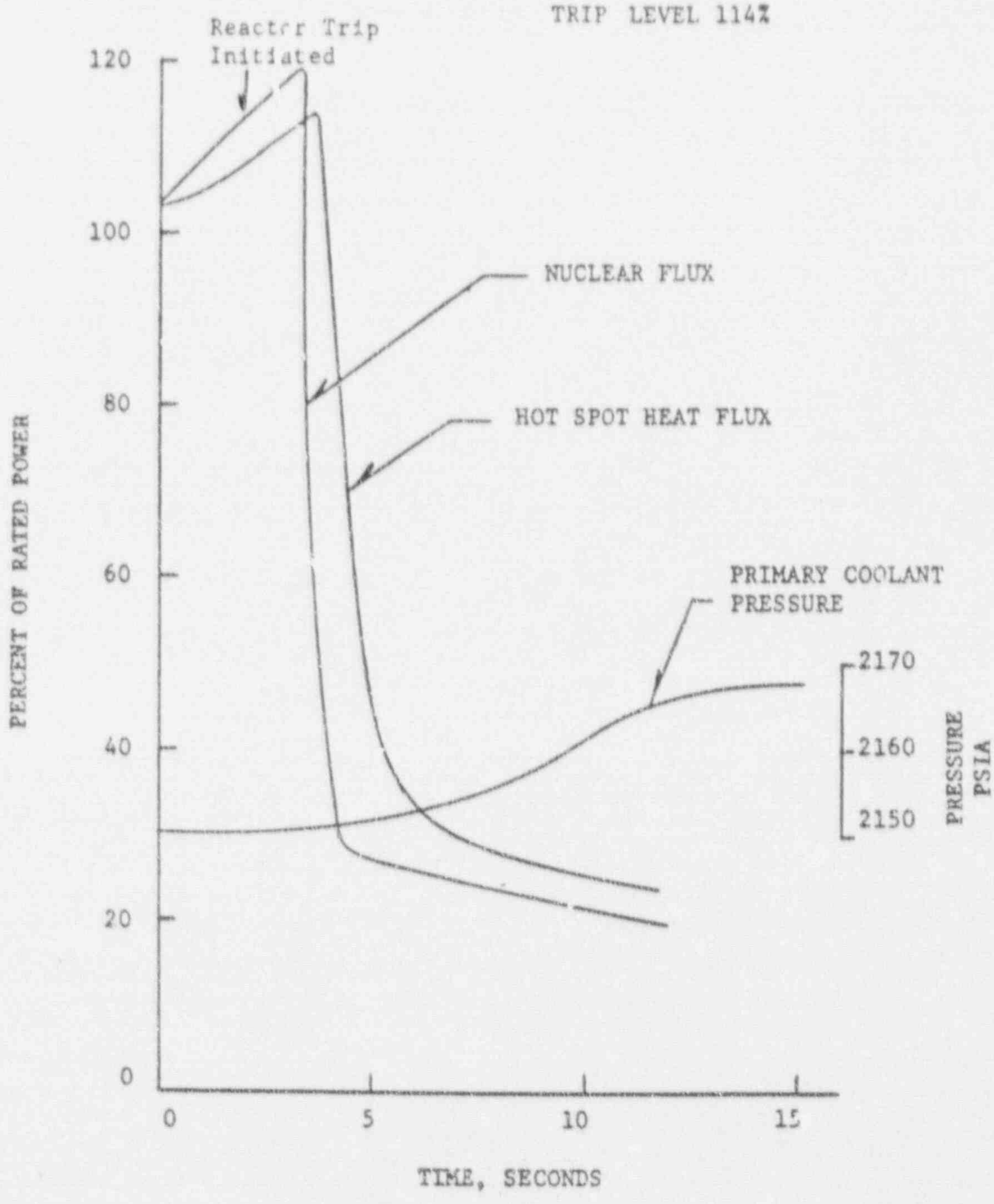
(2) Safeguards Report for Phase I of Saxton Nuclear Experimental Corporation Five-Year Research and Development Program (December 1961).

(3) Saxton Nuclear Experiment Corporation - Safeguards Report for the Saxton Reactor Partial Plutonium Core II (March 1965).

FUEL TEMPERATURE COEFFICIENT = 1.0×10^{-5} $\Delta k/°F$

MODERATOR TEMPERATURE COEFFICIENT = -2.0×10^{-4} $\Delta k/°F$

TRIP LEVEL 114%



CONTINUOUS ROD WITHDRAWAL

(REACTIVITY INSERTION RATE = 2.5×10^{-4} Δk/SECONDS)

FIGURE VI-1